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see Tech spec

SUBJECT: Forwards marked-up Tech Specs.Tech Specs,representing util policy decision to utilize 24 month (from startup to startup) refueling outages.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

November 20, 1984
(NMP2L 0253)

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Schwencer:

Re: Nine Mile Point Unit 2
Docket No. 50-410

Enclosed please find the marked up copy of the Nine Mile Point Unit 2 Technical Specifications. The standard Boiling Water Reactor 5 Technical Specifications have been marked up to reflect the Unit 2 unique design conditions and operating philosophy. The Technical Specifications represent an eight month intensive effort by General Electric, Stone & Webster, Niagara Mohawk Licensing, Operations and Site Technical groups to develop accurate technical specifications. This document is based upon a quality process which includes references to key project documents (such as calculations) to assure its accuracy.

Additionally, we plan to have our independent Compliance & Verification team verify the Technical Specifications. This review will provide further assurance that the Technical Specifications reflect the design and FSAR commitments. These later reviews by the Compliance & Verification team will be complete prior to fuel loading.

Also, these Technical Specifications represent the Niagara Mohawk policy decision to utilize 24 month (from startup to startup) refueling outages. Based on our practice, this represents an economic fuel cycle for Nine Mile Point Unit 2. We will be prepared to discuss this issue at our first

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Encl To: M. Haughey (Advanced 11/29/84)
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Technical Specifications review meeting with the Commission staff. These Technical Specifications will be updated periodically to show revised design conditions based upon the NRC's review of the FSAR. For example, design changes, if any, as a result of the resolution of open issues or new requirements such as anticipated transients without scram will be addressed in revisions. Changes to this standard Boiling Water Technical Specifications are noted by italicized words.

Very truly yours,

CVMangan

C. V. Mangan
Vice President
Nuclear Engineering & Licensing

NLR:ja
Enclosure
xc: Project File (2)
R. Gramm, NRC Resident Inspector

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U.S. NRC
DOCKET 50-410

TECHNICAL SPECIFICATIONS AND BASES

FOR

NINE MILE POINT NUCLEAR STATION UNIT 2 OPERATION AT 3323 THERMAL MEGAWATTS

NIAGARA MOHAWK POWER CORPORATION
SYRACUSE, NEW YORK

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

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

1.7 CONTAINMENT LEAK TEST PROGRAM

The comprehensive testing of the primary containment system that includes type A, B and C tests and secondary containment in-leakage tests.

a. Containment System

The additional barriers, after the reactor coolant pressure boundary that are designed, to prevent the release of quantities of radioactive material that would have a significant effect on the health of the public. It includes:

i) the primary containment structure, including access openings and penetrations;
ii) containment isolation valves, pipes, closed systems, and other components used to effect isolation of the containment atmosphere from the outside environs;
iii) those systems or portion of systems that by their functions extend the primary containment structure boundary to include their system boundary;
iv) the secondary containment function, including access openings and penetrations; and
v) the Reactor Building Protection System which consists of the Reactor Building Ventilation and Standby Gas Treatment Systems.

b. Primary Containment

The structure (drywell and suppression chamber) that encloses the major components of the reactor coolant pressure boundary, as defined in 10CFR50.2, designed to contain accident pressure and serve as a leakage barrier against the uncontrolled release of radioactivity to the environment.

c. Secondary Containment

A secondary fission product control function existing when a structure (the reactor building and aux. bays) completely surrounds the primary containment and is held at a pressure of 0.25 inches (water), below adjacent regions under all wind conditions.
design

d. Type A Test

Test to measure the containment system overall integrated leakage rate under conditions representing design basis accident containment pressure and alignments. These tests are performed at periodic intervals. Also called the integrated leak rate test (ILRT).

e. Type B Test

Pneumatic test to detect and measure local leakage through the following containment penetrations:

i) Those whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, or electrical penetrations fitted with flexible metal seal assemblies.

1.7 e. ii) Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

f. Type C Test

Pneumatic or hydraulic test intended to mearure primary containment system isolation valve leakage rates. Such valves are:

i) Containment isolation valves; includes those valves that are closed or that close automatically upon receipt of an isolation signal in response to controls intended to effect containment isolation or that operate under post-accident conditions to effect containment isolation.

ii) Valve systems designed to operate subsequent to a design basis accident, which may become a part of the containment isolation system barrier due to post-accident operation.

g. Upper Confidence Limit (UCL)

A calculated value constructed from sample data with the intention of placing a statistical upper bound on the true leakage rate. This is calculated at 95 percent probability.

CORE ALTERATION

1.9 ~~1.7~~ CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, ~~sources, incore instruments~~ or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position. Control rod movement with the control rod drive hydraulic system is not considered to be a core alteration.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

CRITICAL POWER RATIO

1.11 ~~1.8~~ The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the ~~(GEXL)~~ correlations to cause some point in the assembly to experience boiling transition, divided by the actual fuel assembly operating power.

DOSE EQUIVALENT I-131

1.12 ~~1.9~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, ~~I-133,~~ I-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."

1.8 CONTINUOUS FIRE WATCH

In visual contact of area under surveillance due to inoperable fire protection equipment.

~~Ē-AVERAGE DISINTEGRATION ENERGY~~

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 ~~1.11~~ The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 ~~1.12~~ The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to (complete suppression of the electric arc between the fully open contacts) ~~(energization)~~ of the recirculation pump circuit breaker ~~(trip coil)~~ from (initial movement) ~~(when the monitored parameter exceeds its trip setpoint at the channel sensor)~~ of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

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

1.15 FIRE SUPPRESSION WATER SYSTEM

A Fire Suppression Water System shall consist of: a water supply system, fixed extinguishing systms of both automatic sprinklers and sprays, and manual fire fighting equipment consisting of standpipe risers with hose connections and hose reels.

1.16 FIRE WATCH PATROL

At least each hour an area with inoperable Fire Protection Equipment shall be inspected for abnormal conditions.

(FRACTION OF LIMITING POWER DENSITY

1.17 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by (the specified LHGR limit for that bundle type) ~~(13.4 kw/ft).~~

(FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

1.19 ~~1.13~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

1.20 *GASEOUS RADWASTE TREATMENT SYSTEM*

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting main condenser offgas and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.21 ~~1.14~~ IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

1.22 ~~1.15~~ The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.23 ~~1.16~~ A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.24 ~~1.17~~ LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.25 ~~1.18~~ A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, ie., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.26 *MAJOR REFUELING OUTAGE*

For the purpose of designating frequency of testing and surveillance, a major refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed until the next regularly scheduled outage. (Not to exceed 24 months).

1.27 MEMBER(S) OF THE PUBLIC

Member(s) of the public shall include persons who are not occupationally associated with the Nine Mile Point Nuclear Station. This category does not include employees of Niagara Mohawk Power Corporation, the New York State Power Authority, its contractors or vendors who are occupationally associated with Nine Mile Point Unit 2. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with Nine Mile Point Unit 2.

1.28 MILK SAMPLING LOCATION

A milk sampling location is that location where 10 or more head of milk animals are avialable for the collection of milk samples.

MINIMUM CRITICAL POWER RATIO

1.29 ~~1.20~~ The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core ~~(for each class of fuel)~~.

1.30 OFFSITE DOSE CALCULATION MANUAL (ODCM)

~~1.3~~ The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

OPERABLE - OPERABILITY

1.31 ~~1.21~~ A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) ( ) ~~and when~~ all necessary attendant instrumentation, controls, electrical power[,] cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATING CYCLE

1.32 An operating cycle is that portion of Station operation between reactor startups following each major refueling outage. (Not to exceed 24 months).

OPERATIONAL CONDITION - CONDITION

1.33 ~~1.22~~ An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

(Implicit in this definition shall be the assumption that)

[Sources, except as noted in Section 3.0.5]

PHYSICS TESTS

1.34 ~~1.23~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.35 ~~1.24~~ PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.36 ~~1.25~~ PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

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

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

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

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

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

1.37 PROCESS CONTROL PROGRAM

~~1.30~~ The PROCESS CONTROL PROGRAM (PCP) shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

1.38 PURGE - PURGING

Purge or purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. (The purge is completed when the oxygen concentration exceeds 19.5 percent.)

RATED THERMAL POWER

1.39 ~~1.26~~ RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~(2436)~~ 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.40 ~~1.27~~ REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE OCCURRENCE

1.41 ~~1.28~~ A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification~~s 6.9.1.8 and 6.9.1.9.~~ 6.6.

ROD DENSITY

1.42 ~~1.29~~ ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.43 ~~1.30~~ SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

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

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

1.43 SECONDARY CONTAINMENT INTEGRITY (Cont.)

d. ~~(At least one)~~ ~~(The)~~ door in each access to the secondary containment is closed. ~~(except for normal entry and exit).~~

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

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

1.45 SITE BOUNDARY

The site boundary shall be that line around the Nine Mile Point Nuclear Station beyond which the land is neither owned, leased, nor otherwise controlled by Niagara Mohawk Power Corporation or the New York Power Authority.

1.46 SOLIDIFICATION

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

1.47 SOURCE CHECK

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

1.44 SHUTDOWN MARGIN

~~1.31~~ SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

1.48 STAGGERED TEST BASIS

~~1.32~~ A STAGGERED TEST BASIS shall consist of:

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

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

1.49 THERMAL POWER

~~1.33~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

1.50 TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

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

b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response times may be measuried by any series of sequential overlapping or total steps such that both entire response time components are measured.

UNIDENTIFIED LEAKAGE

1.51 ~~1.36~~ UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

1.52 UNRESTRICTED AREA

The unrestricted area shall be any area at or beyond the site boundary access to which is not controlled by Niagara Mohawk Power Corporation or the New York Power Authority for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes. That area outside the restricted area (10CFR 20.3(a)(14)) but within the site boundary will be controlled by the owner as required.

1.53 VENTILATION EXHAUST TREATMENT SYSTEM

~~1.19~~ A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

1.54 VENTING

~~1.20~~ VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL CONDITIONS

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

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

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

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

**See Special Test Exceptions 3.10.1 and 3.10.3.

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

###Trip may be bypassed when placing the reactor mode switch in the shutdown position if all control rods are fully inserted.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

[*Safety Limit Critical Power Ratio* (SLCPR)]

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

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

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS - *SCRAM*

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ {120}/{125} divisions of full scale	≤ {122}/{125} divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ {15}% of RATED THERMAL POWER	≤ {20}% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.66 W+{51}%, with a maximum of	≤ 0.66 W+{54}%, with a maximum of
2) High Flow Clamped	≤ {113.5}% of RATED THERMAL POWER	≤ {115.5}% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, ~~(118)%~~	≤ {118}% of RATED THERMAL POWER	≤ {120}% of RATED THERMAL POWER
d. Inoperative	NA	NA
~~(e. Downscale~~	~~≥ (5)% of RATED THERMAL POWER~~	~~≥ (3)% of RATED THERMAL POWER~~
3. Reactor Vessel Steam Dome Pressure - High	≤ (~~1045~~ *1037*) psig	≤ (~~1065~~ *1057*) psig
4. Reactor Vessel Water Level - Low, Level 3	≥ (~~12.5~~ *159.3*) inches above instrument zero*	≥ (~~11.0~~ *157.8*) inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ (6)% closed	≤ (7)% closed
6. Main Steam Line Radiation - High	≤ (~~2.5~~ *3.0*) x full power background	≤ (~~3.0~~ *3.6*) x full power background
7. ~~(Primary Containment)~~ (Drywell) Pressure - High	≤ (~~1.69~~ *1.68*) psig	≤ (~~1.89~~ *1.88*) psig
~~8. Scram Discharge Volume Water Level - High~~	~~≤ (36)% of full scale~~	~~≤ (39)% of full scale~~
9. Turbine Stop Valve - Closure	≤ (5)% closed	≤ (7)% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ (~~500~~ *530*) psig	≥ {*465*} psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

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

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS - SCRAM

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Scram Discharge Volume Water Level - High		
a. Level Transmitters	≤ (*)% of full scale	≤ (*)% of full scale
b. Float Switches	≤ 25 gallons	≤ 25 gallons

* To be determined during preliminary testing.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

2.1 SAFETY LIMITS

[*Safety Limit Critical Power Ratio* (SLCPR)]

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than ~~(1.06)~~ [ ] MCPR greater than [~~(1.06)~~] represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the ~~(GEXL)~~ correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

SAFETY LIMITS

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

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

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

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

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the ~~(GEXL)~~ correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical ~~(764)~~ assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

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

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Admendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

c. *GE STARS*

Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	{1.76}
Feedwater Temperature	{0.76}
Reactor Pressure	{0.5}
Core Inlet Temperature	{0.2}
Core Total Flow	{2.5}
Channel Flow Area	{3.0}
Friction Factor Multiplier	{10.0}
Channel Friction Factor Multiplier	{5.0}
TIP Readings	{6.3}
R Factor	{1.5}
Critical Power	{3.6}

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

Bases Table B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

THERMAL POWER	{3323} MW
Core Flow	{108.5} Mlb/hr
Dome Pressure	{1010.4} psig
Channel Flow Area	{0.1089} ft^2
R-Factor	High enrichment - {1.043} Medium enrichment - {1.039) Low enrichment - {1.030}

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code *1971* Edition, including Addenda through *winter* 19*72*, which permits a maximum pressure transient of (110)%, (1375) psig, of design pressure, (1250) psig. The Safety Limit of (1325) psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to (1375) psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the ~~(USAS Piping Code, Section B31.1) (and the)~~ (ASME Boiler and Pressure Vessel Code, (*1977*) Edition, including Addenda through *summer* 19*77* for the reactor recirculation piping), which permits a maximum pressure transient of (120)%, ~~(1380) psig~~, of design ~~pressure, (1150) psig for suction piping and (1250) psig for discharge piping. The pressure Safety Limit is selected to be the~~ lowest ~~trans~~ient overpressure allowed by the (applicable codes). *(1)*

2.1.4 REACTOR VESSEL WATER LEVEL

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

(1) The design pressures are 1250 psig for suction piping, 1650 psig for discharge piping to the exit of the discharge block valve, and 1550 psig for the remainder of the discharge piping to the vessel nozzles. The pressure Safety Limit is selected to be the lowest transient over pressure allowed by the applicable codes.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is [~~equal to or less than the drift allowance assumed~~] for each trip in the safety analyses. *The trip setpoints and allowable values also contain additional margin for instrument accuracy and calibration capability.*

1. Intermediate Range Monitor, Neutron Flux - High

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

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section *15.4* ~~(15.1.12)~~ of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately (1)% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to (21)% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of (170) cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

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

[an allowance for instrument drift specifically allocated]

LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

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

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale ~~(118)~~% setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of ~~(6) ±(1)~~ seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

0.6

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when ~~(the design TOTAL PEAKING FACTOR is exceeded)~~ (MFLPD is greater than or equal to FRTP).

3. Reactor Vessel Steam Dome Pressure-High

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

are *load rejection or* *turbine control valve fast closure, and*

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint ~~(h~~as been used in transient analyses dealing with coolant inventory decrease. ~~The results report~~ed in Section ~~15 show that scram~~ and ~~isolation of all process lines,~~ except ~~main steam,~~ at th~~is level~~ adequat~~ely protects the fuel and~~ the pressure bar~~rier, because~~ ~~MCHFR is greater than 1.0 in all cases, and system pressure does not~~ reach ~~the safety valve settings~~. The scram setting~~)~~ was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

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

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. ~~(No credit was~~ taken for operation of this trip in the acc~~ident~~ analyses; ~~however,~~ its functiona~~l capability at the~~ spec~~ified trip setting is required by this~~ specificati~~on to~~ enhance the overall ~~reliabilit~~y ~~of the Reactor Protection System.)~~

7. {Primary Containment} ~~(Drywell)~~ Pressure-High

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of { } *approximately 25* gallons of water.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of ~~(10)~~ *5*% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient. ~~(assuming the turbine bypass valves (fail to) operate)~~.

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

The turbine control valve fast closure trip *with or without* anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than ~~(~~30~~)~~ milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a ~~faster~~ *slower* closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section ~~(15.1.0)~~ *(15.2.2)* of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

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

12. Manual Scram

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

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

3.0.4 ~~3.0.3~~ When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

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

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

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.5 ~~3.0.4~~ Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.3 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant systems(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of specification 3.0.4. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in a condition stated in the individual specification.

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

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

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

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

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

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

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

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within ~~one hour~~ 72 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

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

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% delta k/k:

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

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

3/4.1.3. CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

1. Within one hour:

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

b) Disarm the associated directional control valves** *hydraulically by closing the drive water and exhaust water isolation valves.*

c) Comply with Surveillance Requirement 4.1.1.c.

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

2. Restore the inoperable control rod to OPERABLE status within ~~48~~ 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

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

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

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

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

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

a) Electrically, or

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

   a) Electrically, or

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

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

3. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

4.1.3.1.2 When above the ~~(preset power level)~~ {low power setpoint} of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

   2. Open when the scram signal is reset.

b. Proper ~~(float)~~ *and* ~~(~~level sensor~~)~~ response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level ~~(ΔP level measuring system)~~ instrumentation ~~(after each scram from a pressurized condition)~~ ~~(~~at least once per 31 days~~)~~.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

b. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
~~(46)~~ 45	~~(0.375)~~ .43
~~(36)~~ 39	~~(1.096)~~ .86
~~(26)~~ 25	~~(2.000)~~ 1.93
~~(6)~~ 05	~~(4.000)~~ 3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
~~(46)~~ 45	~~(0.398)~~ 0.45
~~(36)~~ 39	~~(1.169)~~ 0.92
~~(26)~~ 25	~~(2.120)~~ 2.05
~~(6)~~ 05	~~(4.380)~~ 3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

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

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

b. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

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

   a) Restore the inoperable accumulator to OPERABLE status, or

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

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

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

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

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

      1) Electrically, or

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

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

b. In OPERATIONAL CONDITION 5*:

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

   a) Electrically, or

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

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

c. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

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

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is ≥940 ~~(910) +(30)~~, ~~-(0)~~ psig unless the control rod is inserted and directional valves disarmed or scrammed.

b. At least once per operating cycle ~~18 months~~ by:

1. Performance of a:

   a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

   b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of (940) +(30), -(0) psig on decreasing pressure.

2. ~~(Verifying)~~ (Measuring and recording the time ~~for up to 10 minutes)~~ that (each individual) accumulator check valve maintains (the associated) accumulator pressure above the alarm set point ~~(for greater than or equal to 10 minutes)~~ with no control rod drive ~~pump operating.~~ charging water pressure.

   , starting at normal system operating pressure,

REACTIVITY CONTROL SYSTEMS

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

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

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

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

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

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

      a) Electrically, or

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

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

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

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

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

      a) Electrically, or

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

c. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

CONTROL ROD POSITION INDICATION ~~(Optional, solid state RSCS)~~

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

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

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

3. When THERMAL POWER is:

a) Within the ~~(preset power level)~~ (low power setpoint) of the RSCS:

1) Declare the control rod inoperable, and

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

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

1) Electrically, or

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

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

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

c. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

b. The provisions of Specification 3.0.~~4~~5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

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

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

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

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

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

ROD SEQUENCE CONTROL SYSTEM (Optional, Banked Position Type)

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Performance of a {self-test}: ~~{system diagnostic function}:~~

   1. Within 8 hours prior to each reactor startup, and

   2. Prior to movement of a control rod after rod inhibit mode automatic initiation when reducing THERMAL POWER.

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

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

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

*See Special Test Exception 3.10.2
#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With one RBM channel inoperable:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

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

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

b. In OPERATIONAL CONDITION 5*:

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

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

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

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

1. The temperature of the sodium pentaborate solution *in the storage tank* is ~~within the limits of Figure 3.1.5-1.~~ *≥70°F on Figure 3.1.5-2.*

2. The available volume of sodium pentaborate solution is ~~greater than or equal to ( ) gallons.~~ *within the limits of figure 3.1.5-1.*

3. The heat tracing circuit is OPERABLE by determining the temperature of the {pump suction piping} to be greater than or equal to {70}°F. *on Figure 3.1.5-2.*

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining that the available net weight of sodium pentaborate is greater than or equal to {5680} lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

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

c. Demonstrating that, when tested {pursuant to Specification 4.0.5} ~~(at least once per 92 days)~~, the minimum flow requirement of {41.2}gpm per pump at a pressure of greater than or equal to {1220} psig is met.

d. At least once per operating cycle ~~18 months~~ during shutdown by:

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

{2. Demonstrating that the pump relief valve setpoint is less than or equal to ~~{1600}~~ 1387 with no back pressure psig and verifying that the relief valve does not actuate during recirculation to the test tank.}

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

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

---

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F ~~the limit of Figure 3.1.5-1~~.

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

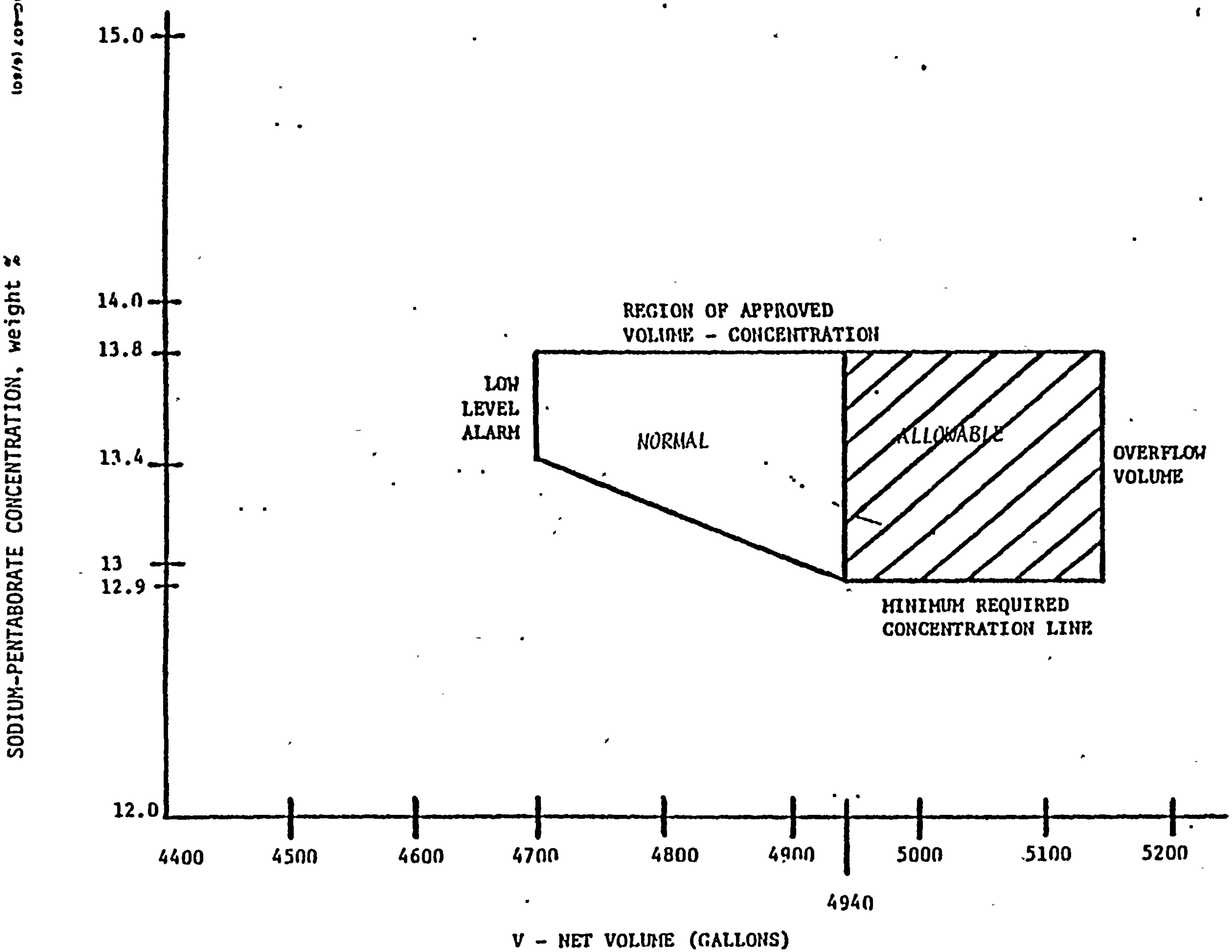
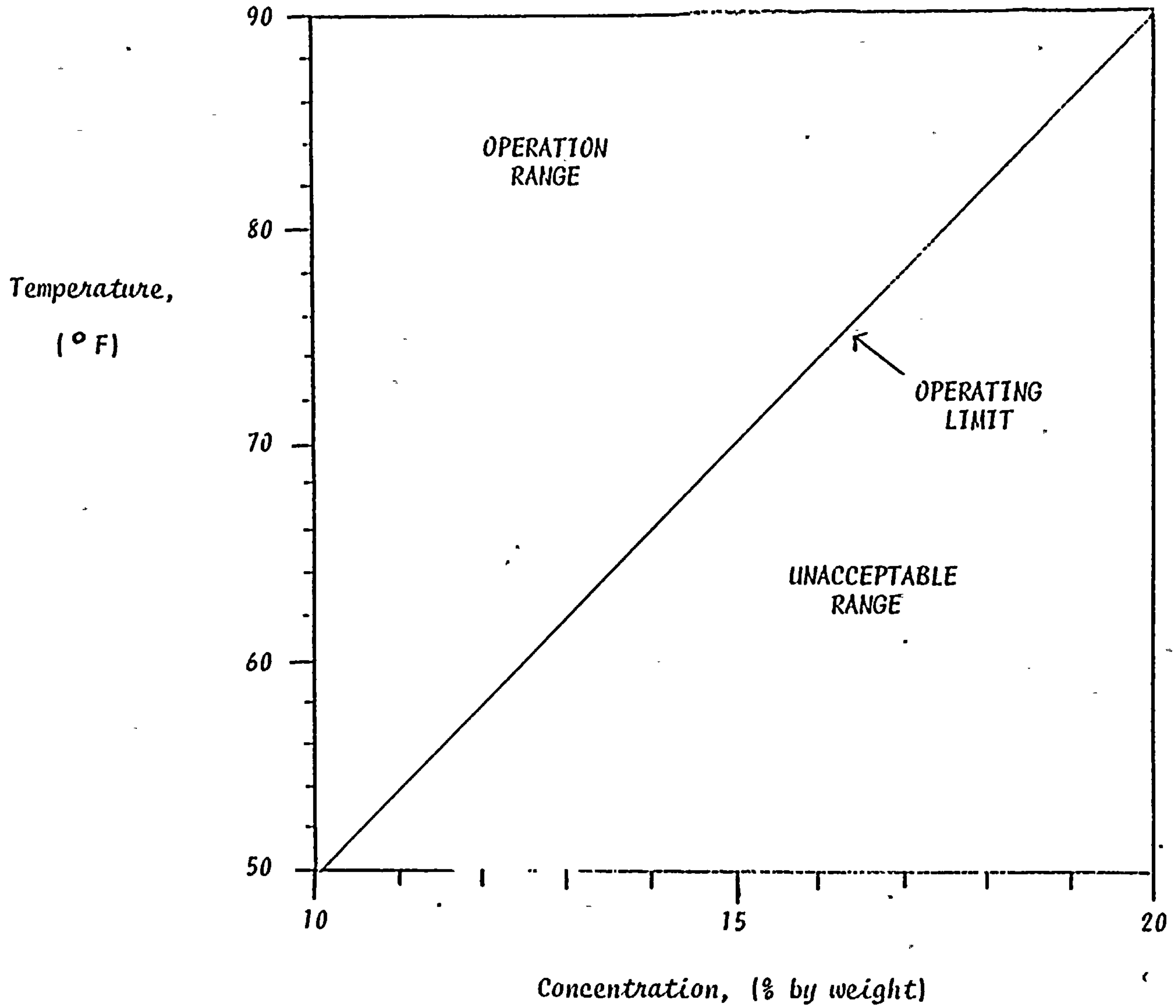


FIGURE 3.1.5-1 SODIUM-PENTABORATE TANK VOLUME vs CONCENTRATION REQUIREMENTS



SODIUM PENTABORATE SOLUTION

TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-2

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

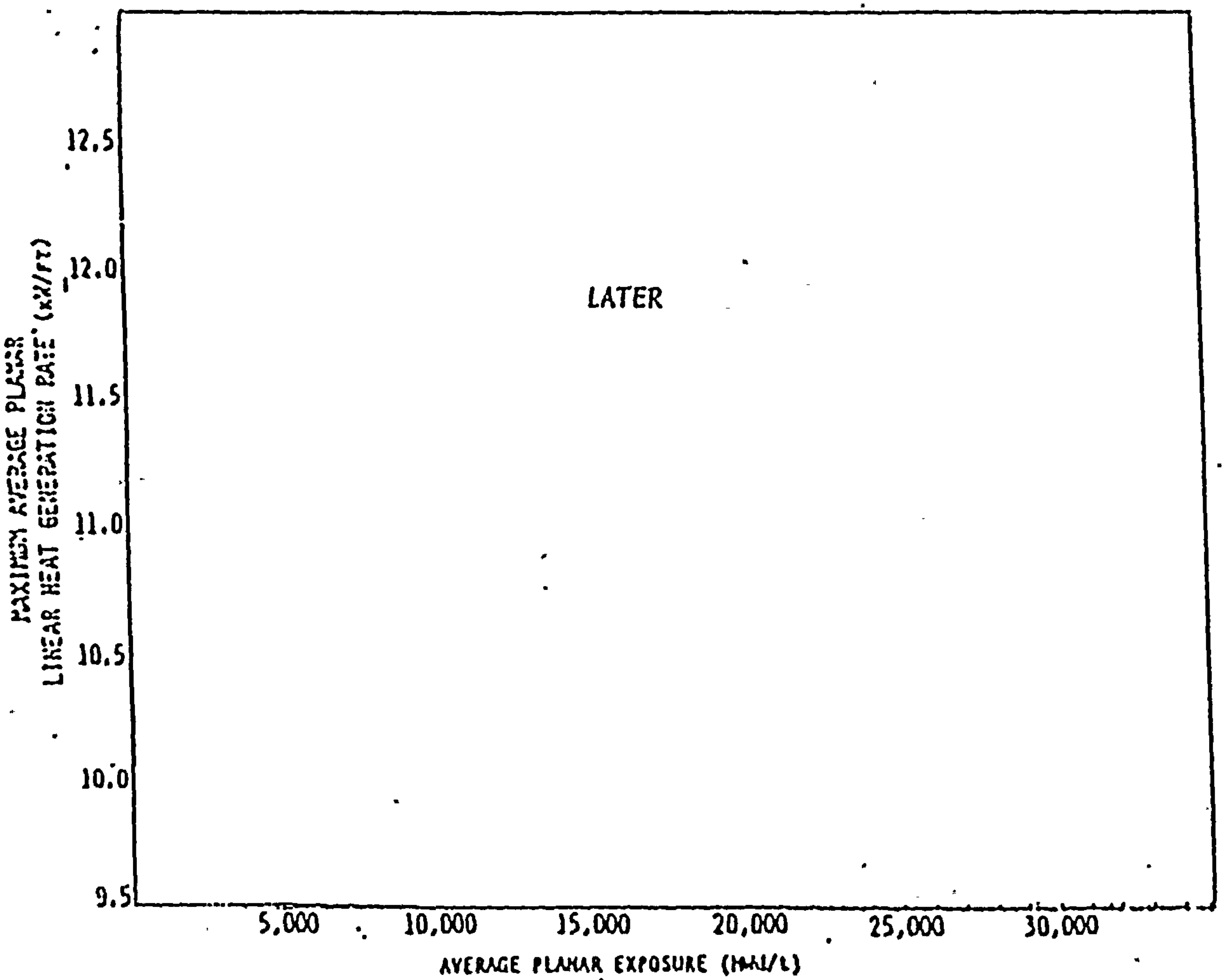
SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES 8CR183, 8CR233 AND 8CR711

Figure 3.2.1-1

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of ~~(108.5)~~ 108.5 million lbs/hr.
T = Lowest value of the ratio ~~of (design TPF, (2.43) for (8 x 8) fuel, divided by the MTPF obtained for any class of fuel in the core)~~ (FRACTION OF RATED THERMAL POWER divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY). ~~T is always less than or equal to 1.0.~~ *T is applied only if less than or equal to 1.0*

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

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value~~(*)~~ within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next ~~4~~ 6 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

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

3/4.2.3 MINIMUM CRITICAL POWER RATIO ~~(Optional ODYN Option A)~~

LIMITING CONDITION FOR OPERATION

(1.24)

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit times the K_f ~~shown in from Figure 3.2.3-1, (provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2), with MCPR for 8 x 8 fuel = (1.29).~~ of Figure 3.2.3-1

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

ACTION:

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

~~1. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus (2000) MWD/t, with MCPR for (8 x 8) fuel = (1.27).~~

~~2. EOC minus (2000) MWD/t to EOC, with MCPR for 8x8 and 8x8R fuel = (1.27).)~~

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

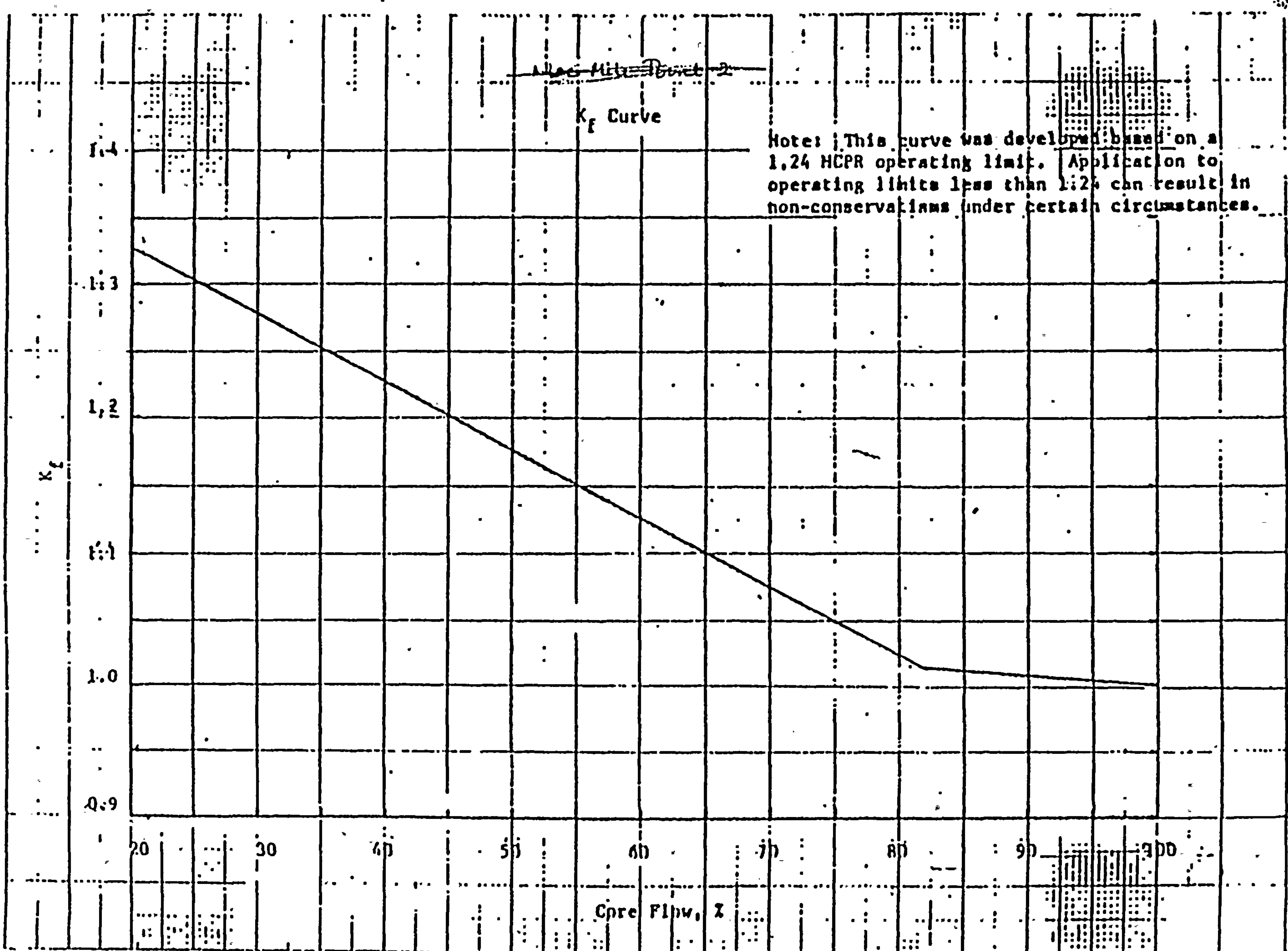


FIGURE 3.2.3-1 K_f AS A FUNCTION OF PERCENT OF CORE FLOW

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed {13.4} kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than {25}% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION - *SCRAM*

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification ~~3.0.4~~ *3.0.5* are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

operating cycle

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per ~~18 months~~. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times ~~18 months~~ where N is the total number of redundant channels in a specific reactor trip system. *(24 months)*

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

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

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION - SCRAM

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

NMP-UNIT 2
~~GE-STS (BWR/5)~~

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION - *SCRAM*

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

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

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

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

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

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

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to ~~< (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER~~ [ ], within 2 hours.

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

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

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

[*less than the automatic bypass setpoint*]

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

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

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

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than ~~(11)~~ (14) LPRM inputs to an APRM channel.

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

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

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

(g) Also actuates the standby gas treatment system.

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

(i) This function shall be automatically bypassed when turbine first stage pressure ~~is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.~~ [ ]

(j) Also actuates the EOC-RPT system.

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

[is less than or equal to 20% of turbine first stage pressure in psia, at valves wide open turbine throttle steam flow, equivalent to Thermal Power less than 30% of Rated Thermal Power. To allow for instrumentation accuracy, calibration and drift a setpoint of 17.0% of turbine first stage pressure in psig is used.]

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES - SCRAM

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

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

**Not including simulated thermal power time constant, 6 ±0.6 seconds.
#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS - SCRAM

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

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

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

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

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

(h) This calibration shall consist of (verifying) (the adjustment as required, of) the 6 ± ~~1~~ *0.06* second simulated thermal power time constant.)

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

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

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

(l) Once every operating cycle, perform a channel functional test with the mode switch in startup.

INSTRUMENTATION

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per ~~18 months~~ *operating cycle*. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 1~~8 months~~ *(not to exceed 24 months)*, where N is the total number of redundant channels in a specific isolation trip system.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION GROUP(S) OPERATED BY SIGNAL (i)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION SIGNALS				
a. Reactor Vessel Water Level	12***	(f)	(g)	NA
1. Low, Low, Low [Level 1]	1	2	1,2,3	20
2. Low Low [Level 2]	2,3,6,7,8,9(b)(e),12	2	1,2,3#	20
3. Low [Level 3]	,5	2	1,2,3#	20
b. Drywell Pressure High	3,8(b)(e),9,11(d),12	2	1,2,3	20
c. Main Steam Line				
1. Radiation - High	1,2(c)	2	1,2,3	21
2. Pressure - Low	1	2	1	23
3. Flow - High	1	2	1,2,3	21
d. Main Steam Line Tunnel				
1. Temperature - High	1	2	1,2,3	21
2. ΔTemperature - High	1	2	1,2,3	21
3. Turbine Bldg. MSL (High Space Temperature) Enclosure	1	1	1,2,3	21
e. Condenser Vacuum Low	1	2	1,2*,3*	21
f. RHR Equipment Area				
1. Temperature High	5,10	1	1,2,3	28
2. ΔTemperature - High	5,10	1	1,2,3	28
g. Reactor Vessel Pressure High	5	1	1,2,3	28
h. SBGT Exhaust - High Radiation	9	1	During Purge Operations	29

TABLE 3.3.2-1 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION GROUP(S) OPERATED BY SIGNAL (i)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION SIGNALS (Cont.)				
i. RWCU Equipment				
1. ΔFlow - High	6,7	1	1,2,3	22
2. Nonregenerative Heat Exchanger Outlet Temperature - High	6	1	1,2,3	22
3. Standby Liquid Control, SLCS, Initiation	6,7[f]	1	1,2,3	22
j. RWCU Equipment Area				
1. Temperature - High	6,7	1	1,2,3	22
2. ΔTemperature - High	6,7	1	1,2,3	22
k. Remote Manual Isolation Pushbutton [NSSS]	2,9,11,12	(2)/group	1,2,3	(24)
	6,7,10[h]	(1)/group	1,2,3	(26)
	1,5,8,13	(1)/valve	1,2,3	(25)(26)
2. RCIC ISOLATION SIGNALS				
a. RCIC Steam Supply Pressure - Low	(10), 11[f]	2	1,2,3	22
b. RCIC Turbine Exhaust Diaphragm Pressure - High	10	2	1,2,3	22
c. RCIC Equipment Area				
1. Temperature - High	10	2	1,2,3	22
2. ΔTemperature	10	2	1,2,3	22

TABLE 3.3.2-1 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION GROUP(S) OPERATED BY SIGNAL (i)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. RCIC ISOLATION SIGNALS				
e. RCIC Steam Line Tunnel				
1. Temperature - High	10	1	1,2,3	22
2. ΔTemperature - High	10	1	1,2,3	22
f. Manual Isolation Push Button [RCIC]	10,11,13	1/System	1,2,3	24
g. Drywell Pressure - High	11	2	1,2,3	22
h. RHR/RCIC Flow - High	10	1	1,2,3	28
3. SECONDARY CONTAINMENT ISOL. SIGNALS (ONLY)				
a. Reactor Building Above the Refuel Floor Exhaust Radiation High	(b)(e)	1	1,2,3,5 and **	27
b. Reactor Building Below the Refuel Floor Exhaust Radiation High	(b)(e)	1	1,2,3,5 and **	27
4. HIGH PRESSURE CORE SPRAY				
a. Hi Drywell Pressure ##	14	##	##	##
b. Reactor Water Level (Level 2)##	14	##	##	##

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

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

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

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

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

ACTION 24 - Restore the manual isolation pushbutton function 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.

ACTION 25 - Restore the manual isolation pushbutton function to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

ACTION 27 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

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

NOTES

* When any turbine stop valve is greater than 90% open and/or when the key locked bypass switch is open.

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

*** Signal from LPCS initiation circuitry applicable operating conditions and surveillance as per Table 4.3.3.1 and 4.3.3.2.

(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.

(b) Also actuates the standby gas treatment system.

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

(d) Only used in conjunction with Low RCIC Steam supply pressure to isolate 2ICS*MOV148 and 2ICS*MOV164.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Manual initiation of SLCS Pump 2SLS*P1B closed a 2WCS*MOV102 and manual initiation of SLCS pump 2SLS*P1A closes 2WCS*MOV 112.

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

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

(i) Refer to Table 3.6.3-1 for applicable valves in each isolation group.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION SIGNALS		
a. Reactor Vessel Water Level		
1. Low, Low, Low [Level 1]	$\geq$ 17.8 inches	$\geq$ 10.8 inches
2. Low Low [Level 2]	$\geq$ 108.8 inches	$\geq$ 101.8 inches
3. Low [Level3]	$\geq$ 159.3 inches	$\geq$ 157.8 inches
b. Drywell Pressure High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Main Steam Line		
1. Radiation - High	$\leq$ 3 X Full Power Bkg.	$\leq$ 3.6 X Full Power Bkgd.
2. Pressure - Low	$\geq$ 765 psig	$\geq$ 736 psig
3. Flow - High	$\leq$ 103 psid	$\leq$ 109.5 psid
d. Main Steam Line Tunnel		
1. Temperature - High	$\leq$ 140°F†	†
2. ΔTemperature - High	$\leq$ 50°F†	†
3. Turbine Bldg. Leads Enclosure	$\leq$ 140°F†	†
e. Condenser Vacuum Low	$\geq$ 8.5 in Hg vacuum	$\geq$ 7.6 inches Hg vacuum
f. RHR Equipment Area		
1. Temperature High	$\leq$ 200°F†	†
2. ΔTemperature - High	$\leq$ 100°F†	†
g. Reactor Vessel Pressure High	$\leq$ 128 psig	$\leq$ 148 psig
h. SBGT Exhaust - High Radiation	$\leq 1.6 \times 10^{-2}$ μCi/cc†	$\leq 2.0 \times 10^{-2}$ μCi/cc

TABLE 3.3.2-2 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION SIGNALS		
i. RWCU Equipment		
1. ΔFlow - High	$\leq$ 150.5 gpm	$\leq$ 165 gpm
2 Δ Flow High Timer	$\leq$ 45 sec	$\leq$ 47 sec
3. Nonregenerative Heat Exchanger Outlet Temperature - High	$\leq$ 140°F	N/A
4. Standby Liquid Control, SLCS, Initiation	N/A	N/A
j. RWCU Equipment Area		
1. Temperature - High	$\leq$ 140°F†	†
2. ΔTemperature - High	$\leq$ 50°F	†
k. Manual Isolation Pushbutton [NSSS]	N/A	N/A
2. RCIC ISOLATION SIGNALS		
a. RCIC Steam Line Flow High Timer	$\geq$ 3 sec	$\leq$ 13 sec
b. RCIC Steam Supply Pressure - Low	$\geq$ 60 psig	$\geq$ 55 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig
d. RCIC Equipment Area		
1. Temperature - High	$\geq$ 175°F $\leq$ 225°F†	†
2. ΔTemperature - High	$\leq$ 100°F†	†

TABLE 3.3.2-2 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. RCIC ISOLATION SIGNALS		
e. RCIC Steam Line Tunnel		
1. Temperature - High	$\leq$ 200°F†	†
2. ΔTemperature - High	$\leq$ 100°F†	†
f. Manual Isolation Push Button [RCIC]	N/A	N/A
g. Drywell Pressure - High*	*	*
h. RHR/RCIC Flow - High	$\leq$ 96 in H_2O	$\leq$ 104.5 in H_2O
3. SECONDARY CONTAINMENT ISOL. SIGNALS (ONLY)		
a. Reactor Building Above the Refuel Floor Exhaust Radiation High	$\leq 1.7 \times 10^{-3}$ μc/cc	$\leq 2.1 \times 10^{-3}$ μCi/cc
b. Reactor Building Below the Refuel Floor Exhaust Radiation High	$\leq 1.7 \times 10^{-3}$ μc/cc	$\leq 2.1 \times 10^{-3}$ μCi/cc

* Trip setpoint located in Table 3.3.3-2
† Preliminary setpoint - actual setpoint to be determined during start-up test program.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

FUNCTION	RESPONSE TIME (Seconds)#
Main Steam Line	
1) Reactor vessel water level - level 1	≤ 1.0
2) Flow - High	≤ 0.5

Isolation System instrumentation response time.

TABLE 3.3.2-4

ISOLATION GROUPS	ISOLATION SIGNALS
1	Z,X,C,D,E,P,T,R,RM,AA
2	B,C,Z,RM
3	B,F,Z,RM
4	A,L,M,Z,F,RM
5	A,L,M,Z,RM
6	B,U,J,S,W,Z,RM
7	B,J,U,S,Z,RM
8	B,F,Z,RM
9	B,F,Y,Z,RM
10	K,M,H,Z,RM
11	H* & F*,RM
12	G,RM
13	RM
14	N,RM

*Both signals must be coincidental to cause isolation

TABLE 3.3.2-5

KEY TO ISOLATION SIGNALS:

A = Low reactor vessel water level 3

B = Low reactor vessel water level 2

C = High main steam line radiation

D = High main steam line flow

E = High main steam line tunnel area ambient temperature

F = High drywell pressure

G = Low pressure ECCS initiation signal

H = Low RCIC steam supply pressure

J = High reactor water cleanup system equipment area differential and ambient temperatures

K = Reactor core isolation cooling high pipe routing area temperature and equipment area temperature, ~~low steam supply~~ pressure, High steam line differential pressure, high turbine exhaust diaphragm pressure

L = High reactor vessel pressure

M = High residual heat removal system equipment area differential and ambient temperatures

N = HPCS initiation signal

P = Low main steam line turbine inlet pressure

R = Low main condenser vacuum

S = Standby liquid control system actuated

T = High main steam line tunnel differential ~~and ambient~~ temperatures

U = High reactor water cleanup system differential flow

W = High reactor water cleanup system nonregenerative heat exchanger outlet temperature

X = Low reactor water level, level 1

Y = Standby gas treatment exhaust hi rad

LC = Locked closed

RM = Remote manual switch from control room

LMC = Locked closed - position indicator

Z = Manual isolation

AA= Turbine Building high space temperature

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION - SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION
1. PRIMARY CONTAINMENT ISOLATION SIGNALS			
a. Reactor Vessel Water Level			
1. Low, Low, Low [Level 1]	S	M	R(c)
2. Low Low [Level 2]	S	M	R(c)
3. Low [Level3]	S	M	R(c)
b. Drywell Pressure High	S	M	R(c)
c. Main Steam Line			
1. Radiation - High	S	M	R
2. Pressure - Low	S	M	R(c)
3. Flow - High	S	M	R(c)
d. Main Steam Line Tunnel			
1. Temperature - High	S	M	R(d)
2. ΔTemperature - High	S	M	R(d)
3. Turbine Bldg. Leads Enclosure	S	M	R(d)
e. Condenser Vacuum Low	S	M	R(c)
f. RHR Equipment Area			
1. Temperature High	NA	M	R(d)
2. ΔTemperature - High	NA	M	R(d)
g. Reactor Vessel Pressure High	S	M	R(d)
h. SBGT Exhaust - High Radiation	S	SA(e)	R

TABLE 4.3.2.1-1 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION-SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION
1. PRIMARY CONTAINMENT ISOLATION SIGNALS			
i. RWCU Equipment			
1. ΔFlow - High	S	M	R(c)
2 Δ Flow High Timer	N/A	M	R
3. Nonregenerative Heat Exchanger Outlet Temperature - High	S	M	R
4. Standby Liquid Control, SLCS, Initiation	N/A	M(b)	N/A
j. RWCU Equipment Area			
1. Temperature - High	S	M	R(d)
2. ΔTemperature - High	S	M	R(d)
k. Manual Isolation Pushbutton [NSSS]	N/A	M(a)	N/A
2. RCIC ISOLATION SIGNALS			
a. RCIC Steam Line Flow High, Timer	N/A	M	R
b. RCIC Steam Supply Pressure - Low	S	M	R(c)
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R(c)
d. RCIC Equipment Area			
1. Temperature - High	S	M	R(d)
2. ΔTemperature - High	S	M	R(d)

TABLE 4.3.2.1-1 (Cont.)

ISOLATION ACTUATION INSTRUMENTATION -SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION
2. RCIC ISOLATION SIGNALS			
e. RCIC Steam Line Tunnel			
1. Temperature - High	S	M	R(d)
2. ΔTemperature - High	S	M	R(d)
f. Manual Isolation Push Button [RCIC]	N/A	M	N/A
g. Drywell Pressue - High*	*	*	*
h. RHR/RCIC Flow - High	S	M	R(c)
3. SECONDARY CONTAINMENT ISOL: SIGNALS (ONLY)			
a. Reactor Building ABOVE the Refuel Floor Exhaust Radiation - High	S	SA(e)	R
b. Reactor Building BELOW the Refuel Floor Exhaust Radiation - High	S	SA(e)	R

* Signals from LPCS and RHR initiation signals; see Table 4.3.3.1-1

(a) Manual Isolation pushbuttons are tested at least once per operating cycle during shutdown. All other circuitry associated with Manual Isolation shall receive a Channel Functional Test at least once per 31 days as part of the circuitry required to be tested for the Automatic System Isolation.

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

(c) Calibrate trip unit once per 31 days.

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

(e) Source check required once per 31 days.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

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

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

2. 72 hours. *if HPCS or RCIC are inoperable.*

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per ~~18 months~~. *operating cycle.* Each test shall include at least one channel per trip system such that all channels are tested at least once every N times ~~18 months~~ where N is the total number of redundant channels in a specific ECCS trip system.

(~~not to exceed~~ 24 months)

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[b]	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
~~c. LPCS Pump Discharge Flow-Low (Bypass)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~31~~
~~d. Reactor Vessel Pressure-Low (LPCS Permissive)~~	~~(1)~~	1, 2, 3,	32
c. LPCS Injection Valve Permissive	1	4*, 5*	33
~~e. Reactor Vessel Pressure-Low (LPCI Permissive)~~	~~(1)~~	1, 2, 3,	32
d. LPCI Injection Valve Permissive	1	4*, 5*	33
e. ~~f.~~ LPCI Pump A Start Time Delay Relay Normal Power	~~(~~1~~)~~	1, 2, 3, 4*, 5*	32
~~g. LPCI Pump A Discharge Flow-Low (Bypass)~~	(1)	~~1, 2, 3, 4*, 5*~~	~~31~~
~~h. Division I Bus Power Monitor~~##	~~(2)~~	~~1, 2, 3, 4*, 5*~~	~~(34)~~
f. ~~i.~~ Manual Initiation	~~(~~1~~)~~/~~(~~system~~)~~	1, 2, 3, 4*, 5*	~~(~~35~~)~~
~~j.~~ LPCI Pump A Start Time Relay Emer. Power#	1	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[b]	1, 2, 3	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. ADS Timer	~~(~~1~~)~~	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	~~(~~1~~)~~	1, 2, 3	32
e. LPCS Pump Discharge Pressure-High (Permissive)	~~(~~2~~)~~	1, 2, 3	32
f. LPCI Pump A Discharge Pressure-High (Permissive)	(2)	1, 2, 3	32
g. Manual Initiation	~~(1)/(valve)~~ 2/System	1, 2, 3	~~(~~35~~)~~
~~h.~~ LPCS Pump Start Time Delay Normal Power	1	1,2,3,4*,5*	32
LPCS Pump Start Time Delay Emer. Power	1	1,2,3,4*,5*	32

TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION 2 TRIP SYSTEM			
1. RHR ~~B~~ B&C (LPCI MODE)			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30
~~c. Reactor Vessel Pressure Low (LPCI Permissive)~~	~~(1)~~/valve	1, 2, 3,	32
c. LPCI Injection Valve Permissive	1/valve	4*, 5*	33
d. LPCI Pump (B) Start Time Delay Relay Normal Power	(1)	1, 2, 3, 4*, 5*	32
~~e. LPCI Pump Discharge Flow Low (Bypass)~~	~~(1)/(pump)~~	~~1, 2, 3, 4*, 5*~~	~~31~~
~~(f. Division II Bus Power Monitor##~~	~~(2)~~	~~1, 2, 3, 4*, 5*~~	~~(34))~~
e. ~~g.~~ Manual Initiation	~~(1)/(system)~~	1, 2, 3, 4*, 5*	~~(35)~~
~~h.~~			
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3	30
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30
c. ADS Timer	~~(1)~~	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	~~(1)~~	1, 2, 3	32
e. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	1 ~~(2)~~/~~(pump)~~	1, 2, 3	32
f. Manual Initiation	2/system ~~(1)/(valve)~~	1, 2, 3	~~(35)~~
g.			
LPCI Pump (C) Start Time Delay Relay Normal Power	1	1,2,3,4*,5*	32
LPCI Pump (B) Start Time Delay Relay Emer. Power	1	1,2,3,4*,5*	32
LPCI Pump (c) Start Time Delay Relay Emer. Power	1	1,2,3,4*,5*	32

TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - ~~(~~Low, Low, Level 2~~)~~	4(b)	1, 2, 3, 4*, 5*	36
b. Drywell Pressure - High	4(b)	1, 2, 3	36
c. Reactor Vessel Water Level-High, Level ~~(~~8~~)~~	4 ~~(2)~~(c)	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tank Level-Low	~~(~~2~~)~~(d)	1, 2, 3, 4*, 5*	37
e. Suppression Pool Water Level-High	(2)(d)	1, 2, 3, 4*, 5*	37
~~f. Pump Discharge Pressure-High (Bypass)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~31~~
~~g. HPCS System Flow Rate-Low (Permissive)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~31~~
~~(h. Division III Bus Power Monitor~~ ##	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~34)~~
f. ~~i.~~ Manual Initiation	~~(~~1~~)~~/~~(~~system~~)~~	1, 2, 3, 4*, 5*	~~(~~35~~)~~
~~j.~~			

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3 ~~+~~/bus	2 ~~+~~/bus	2 ~~+~~/bus	1, 2, 3, 4**, 5**	38
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)##	3/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	~~39~~ 38

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump ~~discharge~~ injection valve only. ~~on 2-out-of-2 logic.~~
(d) Provides signal to HPCS pump suction valves only.
* When the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~(~~100~~)~~ psig.
\#\# Alarm only.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

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

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

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

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

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

~~(ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, verify bus power availability at least once per 12 hours or declare the associated ECCS inoperable.)~~

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

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

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

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

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

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

*The provisions of Specification 3.0.~~4~~ *5* are not applicable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level	17.8 ≥ ~~(130)~~ inches*	10.8 ≥ ~~(137)~~ inches
b. Drywell Pressure - High	≤ ~~(1.69)~~ psig 1.68	≤ ~~(1.89)~~ psig 1.88
~~c. LPCS Pump Discharge Flow-Low~~	~~≥ (640) gpm~~	~~≥ (520) gpm~~
~~d. Reactor Vessel Pressure-Low~~ c. LPCS Injection Valve Permissive	≤ ~~≥ (700) psig~~, psid (decreasing)	≤ ~~≥ (725) psig~~, psid (decreasing)
~~e. Reactor Vessel Pressure-Low~~ d. LPCI Injection Valve Permissive	≤ ~~≥ (650) psig~~, (decreasing)	≤ ~~≥ (720) psig~~, psid (decreasing)
~~f.~~ e. LPCI Pump A Start Time Delay Relay normal power	≤ ~~(5)~~ seconds	≤ ~~(~~ 6.0 ~~)~~ seconds
~~g. LPCI Pump A Discharge Flow-Low~~	~~≥ (550) gpm~~	~~≥ (——) gpm~~
~~(h. Division I Bus Power Monitor~~	~~≥ (——) (volts)~~	~~≥ (——) (volts))~~
~~i.~~ f. Manual Initiation	NA	NA
j. ______	______	______
LPCI Pump A Start Time Delay Relay Emer. Power	≤ 1 second	≤ 1.25 seconds
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	17.8 ≥ ~~(130)~~ inches*	10.8 ≥ ~~(137)~~ inches
b. Drywell Pressure - High	≤ ~~(1.69)~~ psgi 1.68 psig	≤ ~~(1.89)~~ psig 1.88
c. ADS Timer	~~(≥ 90,)~~ ≥ ~~≤~~ (105) seconds ~~(≥ 90,)~~	≤ ~~(117)~~ seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ ~~(12.5)~~ inches* 159.3	≥ ~~(11)~~ inches 157.8
e. LPCS Pump Discharge Pressure-High	145 ≥ ~~(146)~~ psig, increasing	≥ ~~(136)~~ psig, increasing 125
f. LPCI Pump A Discharge Pressure-High	125 ≥ ~~(114)~~ psig, increasing	≥ ~~(106)~~ psig, increasing 115
g. Manual Initiation	NA	NA
~~h. ______~~	______	______
LPCS Pump Time Delay Relay Normal Power	≤ 10 sec.	≤ 11 sec.
LPCS Pump Time Delay Relay Emer. Power	≤ 6 sec.	≤ 7 sec.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ 17.8 inches*	≥ 10.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. LPCI Injection Valve Permissive	≤ 650 psig (decreasing)	≤ 720 psig (decreasing)
d. LPCI Pump (B) Start Time Delay Relay Normal Power	≤ 5 seconds	≤ 6 seconds
LPCI Pump (C) Start Time Delay Relay NOrmal Power	≤ 10 seconds	≤ 11 seconds
LPCI Pump (B) Start Time Delay Relay Emer. Power	≤ 1 seconds	≤ 1.25 seconds
LPCI Pump (c) Start Time Delay Relay Emer. Power	≤ 6 seconds	≤ 7 seconds
e. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ 17.8 inches*	≥ 10.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≥ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level - Low, Level 3	≥ 159.3 inches*	≥ 157.8 inches
e. LPCI Pump (B and C) Discharge Pressure - High	≥ 125 psig, increasing	≥ 115 psig, increasing
f. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ 108.8 inches*	≥ 101.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 202.3 inches*	≤ 209.3 inches
d. Condensate Storage Tank Level - Low	≥ 94 inches**	≥ 91 inches**
e. Suppression Pool Water Level - High	≤ 200'9.59" #	≤ Y=201'1.1" #
f. Manual Initiation	NA	NA
D. LOSS OF POWER		
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage (##))	a. 4.16 kv Basis 3212.86 volts b. ≤ 3 sec. time delay	3212.86 volts ≤ 3 sec. time delay
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3607.76voltsvolts b. ≤ 30 sec. trip ###	3607.76volts volts ≤ 30 sec. trip ###

* See Bases Figure B 3/4 3-1

** Minimum CST level to allow time to switch to pool suction, measured from top of HPCS, CST suction nozzle

\# Pool high level - elevation

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

\### Trip under accident conditions.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	~~≤ (40)~~ ≤ 37
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	37
~~a. Pumps A and B~~	≤ ~~(45)~~
~~b. Pump C~~	~~≤ (40)~~
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ {27}
5. LOSS OF POWER	NA

*

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

d. LPCI Pump (B) Start Time Delay Relay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
LPCI Pump (C) Start Time Delay Relay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
LPCI Pump (B) Start Time Delay Relay Emer. Power	NA	M	R	1, 2, 3, 4*, 5*
LPCI Pump (C) Start Time Delay Relay Emer. Power	NA	M	R	1, 2, 3, 4*, 5*

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

* When the system is required to be OPERABLE, after being manually realigned, as applicable, per Specification 3.5.2.

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

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

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

* When the system is required to be OPERABLE, after being manually realigned, as applicable, per Specification 3.5.2.

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

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

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

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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

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

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ - ~~(38)~~ 108.8 inches*	≥ ~~( )~~ 101.8 inches
2. Reactor Vessel Pressure - High	≤ ~~(1120)~~ 1050 psig	≤ ~~( )~~ 1065 psig

See Bases Figures B 3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

* *Calibrate trip unit once per 31 days*

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

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

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or ~~(take the ACTION required by Specification 3.2.3)~~ ~~(~~reduce THERMAL POWER to less than ~~(~~30~~)~~% of RATED THERMAL POWER within the next 6 hours~~)~~.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or ~~(take the ACTION required by Specification 3.2.3)~~ ~~(~~reduce THERMAL POWER to less than ~~(~~30~~)~~% of RATED THERMAL POWER within the next 6 hours~~)~~.

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

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per ~~18 months~~ *operating cycle*. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per ~~36~~ *48* months. ~~(The time allotted for breaker arc suppression, ( ) ms, shall be verified by test at least once per 60 months.)~~

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure	2[b]
2. Turbine Control Valve-Fast Closure	2[b]

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

[b] This function shall be automatically bypassed when turbine first stage pressure is less than or equal to ~~(---) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.~~
20% of turbine first stage pressure in psia, at valves wide open turbine throttle steam flow, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration and drift, a setpoint of 17.0% of turbine first stage pressure in psig is used.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$ {5}% closed	$\leq$ {7}% closed
2. Turbine Control Valve-Fast Closure	$\geq$ (~~500~~) psig 530	$\geq$ (~~414~~) psig 465

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milleseconds)
1. Turbine Stop Valve-Closure	≤ ~~(100)~~ 190
2. Turbine Control Valve-Fast Closure	≤ ~~(100)~~ 190

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	M~~(*)~~	R
2. Turbine Control Valve-Fast Closure	M~~(*)~~	{R}

~~(*Including trip system logic testing.)~~

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than (~~100~~) 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per ~~18 months~~. *operating cycle.*

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	ACTION
a. Reactor Vessel Water Level - (Low Low, Level 2)	2	50
b. Reactor Vessel Water Level - High, Level (8)	2~~(b)~~	~~51~~ 50
c. Condensate Storage Tank Water Level - Low	(2)~~(c)~~(b)	~~52~~ 51
~~d. Suppression Pool Water Level - High~~	(2)(c)	52
~~e.~~ d. Manual Initiation	(1)/~~(system)~~~~(d)~~(c)	~~(53)~~ 52

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition. ~~provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.~~
~~(b) One trip system with two-out-of-two logic.~~
~~(c)~~ (b) One trip system with one-out-of-two logic.
~~(d)~~ (c) One trip system with one channel.

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

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

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

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

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

ACTION ~~52~~ 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION ~~53~~ 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within ~~(8)~~ hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥ ~~(38)~~ 108.8 inches*	≥ ~~( )~~ 101.8 inches
b. Reactor Vessel Water Level - High, Level (8)	≤ (202.3) inches*	≤ (209.3) inches
c. Condensate Storage Tank Level - Low	≥ (94) inches	≥ (91) inches
~~d. Suppression Pool Water Level - High~~	~~≤ ( ) inches~~	~~≤ ( ) inches~~
d. ~~e.~~ Manual Initiation	NA	NA

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

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - {Low Low, Level 2}	S	M[b]	R ~~(b)~~
b. Reactor Vessel Water Level - High, Level (8)	S	M[b]	R[b]
c. Condensate Storage Tank Level - Low	{S}	M[b]	{R}[b]
d. ~~Suppression Pool Water Level - High~~	~~(S)~~	~~M~~	~~(R)~~
~~e.~~ d. Manual Initiation	NA	~~(~~M[a] ~~) & (R)~~	NA

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

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

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

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

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

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

ACTION 61 - With the number of OPERABLE Channels:

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

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

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

NOTES

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

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

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

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

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

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

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

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + {40}%	≤ 0.66 W + {43}%
b. Inoperative	NA	NA
c. Downscale	≥ {5}% of RATED THERMAL POWER	≥ {3}% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale	≤ 0.66 W + {42}%*	≤ 0.66 W + {45}%*
b. Inoperative	NA 4	NA
c. Downscale	≥ {5}% of RATED THERMAL POWER	≥ {3}% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ {12}% of RATED THERMAL POWER	≤ {14}% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA 1	NA 1.6
b. Upscale	≤ {2 x 10^5} cps	≤ (~~5~~ x 10^5) cps
c. Inoperative	NA	NA 1.8
d. Downscale	≥ {3} cps	≥ ~~(2)~~ cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ {108/125} of full scale	≤ {110/125} of full scale
c. Inoperative	NA	NA
d. Downscale	≥ {5/125} of full scale	≥ {3/125} of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 25 ~~(18)~~ gallons	≤ 25 ~~(18)~~ gallons
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW UNIT		
a. Upscale	≤ {108 % rated flow ~~/125} divisions of full scale~~	≤ (111 % rated flow ~~/125} divisions of full scale~~
b. Inoperative	NA	NA
c. {Comparator} ~~{Downscale}~~	≤ {10}% flow deviation	≤ {11}% flow deviation

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

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. REACTOR MODE SWITCH		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	~~S/U(b)(e)~~,M(c)	Q	1*
b. Inoperative	NA	S/U(b)~~(e)~~,M(c)	NA	1*
c. Downscale	NA	S/U(b)(e),M(c)	Q	1*
2. APRM				
a. Flow Biased Neutron Flux Upscale	~~(NA)~~	S/U(b),M W	~~(Q)~~ W(d)(e) SA	1
b. Inoperative	NA	S/U(b),M W(b)	NA	~~1,~~ 2, 5
c. Downscale	(NA)	S/U(b),M W	(Q) W(b) SA	1
d. Neutron Flux - Upscale, ~~Startup~~ Setdown	(NA)	S/U(b),M W(b)	(Q) W(b)	2, 5
Inop	NA	W	NA	1
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(b)	NA	2, 5
b. Upscale	NA	S/U(b),W(b)	Q	2, 5
c. Inoperative	NA	S/U(b),W(b)	NA	2, 5
d. Downscale	NA	S/U(b),W(b)	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(b)	NA	2, 5
b. Upscale	NA	S/U(b),W(b)	Q	2, 5
c. Inoperative	NA	S/U(b),W(b)	NA	2, 5
d. Downscale	NA	S/U(b),W(b)	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	(M)(Q)	R	1, 2, 5**
b. Scram Trip Bypass	NA	M	NA	~~(1, 2,)~~ 3, 4, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1
c. (Comparator)-(Downscale)	NA	-S/U(b),M	Q	1
7. REACTOR MODE SWITCH				
a. Shutdown Mode	NA	R	NA	3, 4
b. Refuel Mode	NA	R	NA	5

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

c. Includes reactor manual control multiplexing system input.

* With THERMAL POWER ≥ ~~(30)~~% of RATED THERMAL POWER.

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

d. *This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM Channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.*

e. *This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.*

f. *This calibration shall consist of verifying the trip setpoint only.*

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

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

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

c. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNEL(S) OPERABLE	APPLICABLE CONDITIONS	SETPOINT**	ACTION
1. Control Room Air Intake (a)	1	1,2,3,5 and *	$\leq 1.3 \times 10^{-5}$ μCi/cc	70
2. Reactor Building Above Refuel Floor (b)(c)	1	1,2,3, and *	$\leq 1.7 \times 10^{-3}$ μCi/cc	71
3. Reactor Building Below Refuel Floor (b)(c)	1	1,2,3 and *	$\leq 1.7 \times 10^{-3}$ μCi/cc	71
4. Drywell Atmosphere				
a. Gas	1	At all times	$\leq 1.0 \times 10^{-4}$ μCi/cc***	72
b. Particulate	1	At all times	$\leq 5.7 \times 10^{-7}$ μCi/cc***	72
5. Service Water Discharge - RHR Heat Exchanger	1/heat exchanger	(d)	$\leq 3.1 \times 10^{-4}$ μCi/cc***	76
6. Main Condenser Air Ejector Off-Gas Pre-Treatment	1(e)	1,2	$\leq 5.7 \times 10^{1}$ μCi/cc	73
7. Standby Gas Treatment	1 (g)	During containment purging	$\leq 1.6 \times 10^{-2}$ μCi/cc	74
8. Area Monitors				
a) New Fuel Storage Vault Area (criticality)	1	(f)	$\leq 1.0 \times 10^{2}$ mR/hr***	75
b) Main Control Room	1	At all times	$\leq 2.5 \times 10^{-1}$ mR/hr***	75

(a) Initiates Control Room emergency filtration with channel at high setpoint.
(b) Normal Reactor Building ventilation isolation and standby gas treatment initiation at high setpoint. (Channel is gas detector only.)
(c) See section 3.6.5.3
(d) With RHR heat exchangers in operation.
(e) Isolates system with channel at high setpoint.
(f) With fuel in the new fuel storage vault.
* When irradiated fuel is being handled in secondary containment and during CORE ALTERATIONS, and operations with a potential for draining the reactor vessel (with fuel in the vessel).
** Above measured background.
*** Alarm only.

(g) Isolates ~~system~~ primary containment purge on high radiation. With channel at high setpoint, clear personnel from vicinity of all purge line valves and continue purging. Observe main stack effluent monitor alarm status to ensure that offsite dose criteria are not exceeded.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 70 - With less than the required channel operable, initiate and maintain operation of Control Room emergency filtration system within one hour.

ACTION 71 - With less than the required channel operable, initiate and maintain isolation of Reactor Building ventilation and operation of standby gas treatment within one hour.

ACTION 72 - With less than the required channel operable, obtain and analyze at least one grab sample of the monitored parameter once per 24 hours. See Section 3.4.3.1.

ACTION 73 - With the number of channels OPERABLE less than the required by the Minimum Channels OPERABLE requirement, releases via this pathway may continue provided grab samples are taken and analyzed once per 12 hours.

ACTION 74 - Close the affected system isolation valves within one hour or initiate the pre-planned alternate method of monitoring.

ACTION 75 - With less than the required monitor operable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 76 - With less than the required channel operable, obtain and analyze a grab sample of the monitored parameter once per 12 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	SOURCE CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Control Room Air Intake	S	M	SA	R
2. Reactor Building Above Refuel Floor *	S	M	SA	R
3. Reactor Building Below Refuel Floor *	S	M	SA	R
4. Drywell Atmosphere	S	M	SA	R
5. Service Water Discharge - RHR Heat Exchanger	S	M	SA	R
6. Main Condenser Air Ejector Off-Gas Pre-Treatment	S	M	SA	R
7. Standby Gas Treatment System Post-Treatment	S	M	SA	R
8. Area Monitors				
(a) New Fuel Storage Vault Area (Criticality)	S	M	SA	R
(b) Main Control Room	S	M	SA	R

* Gas detector only.

SEISMIC MONITORING INSTRUMENTATION(*)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, ~~in lieu of any other report required by Specification 6.9.1, prepare and~~ submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.2-1.

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to (0.01) g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. ~~In lieu of any other report required by Specification 6.9.1~~, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

~~(*This specification not required for additional units at a common site provided at least one unit has seismic instrumentation and corresponding technical specifications meeting the recommendations of Regulatory Guide 1.12, April 1974.)~~

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

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

(a) With reactor control room indication and annunciation.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs			
a. Reactor Bldg. Mat El. 175'-0	M	SA	R
b. Reactor Bldg. Refueling Fl. El. 353'-10	M	SA	R
c. Control Bldg Mat El. 214'-0	M	SA	R
~~d.~~	~~M~~	~~SA~~	~~R~~
2. Triaxial Peak Accelerographs			
a. Diesel Gen Bldg Service Water Piping	NA	NA	R
b. Prim. Cont High Pr. Core Spray Piping	NA	NA	R
c. Prim. Cont Recirc Pump Motor	NA	NA	R
~~d.~~	~~NA~~	~~NA~~	~~R~~
3. Triaxial Seismic Switches			
a. Reactor Bldg Mat El 175'-0	M(a)	SA	R
~~b.~~	~~M~~	~~SA~~	~~R~~
~~c.~~	~~M~~	~~SA~~	~~R~~
4. Triaxial Response-Spectrum Recorders			
a. Reactor Bldg Mat El 175'-0	M	SA	R
b. Prim Cont RHR Piping Pene El 294-6"	NA	NA ~~SA~~	R
c. Reactor Bldg Refueling Fl. El 353'-10"	NA	NA	R
d. Control Bldg Mat El 214'-0	NA	NA	R

(a) Except seismic trigger.

Delete

INSTRUMENTATION

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

Delete

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. (30) ft. and (200) ft.	1 each
b. Wind Direction	
1. Elev. (30) ft. and (200) ft.	1 each
c. Air Temperature Difference	
1. Elev. (30/200) ft.	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. (30) ft. and (200) ft.	D	SA
b. Wind Direction		
1. Elev. (30) ft. and (200) ft.	D	SA
c. Air Temperature Difference		
1. Elev. (30/200) ft.	D	SA

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

b. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

b. *Once per operating cycle, proper transfer of control to remote shutdown panel shall be verified except for RCIC turbine control circuit.*

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	2CES*PNL405	1
2. Reactor Vessel Water Level (2)	2CES*PNL405	1
3. *Suppression Pool Water Level*	2CES*PNL405	1
3. Safety/Relif Valve Position, (2) valves		1/(valve)
4 *Suppression Pool Water Temp. (2)*	2CES*PNL405	1
4. Suppression Chamber Water Level		1
5. *RHS H_X Discharge Flow*	2CES*PNL405	1
5. Suppression Chamber Water Temperature		1
6. *Service Water Flow to RHS Ht. Exch. (A,B)*	2CES*PNL405	1
6. Suppression Chamber Air Temperature		1
7. *RHS Ht. Exch. Service Water Outlet Temp. (A,B)*	2CES*PNL405	1
7. Drywell Pressure		1
8. *RCIC Turbine Speed*	2CES*PNL405	1
8. Drywell Temperature		1
9. *RCIC Pump Discharge Flow*	2CES*PNL405	1
9. RHR System Flow		1
10. *RHS Heat Exchanger In/Out Temp. (A,B)*	2CES*PNL405	1
10. RHR Service Water System Flow		1
11. *RHS Discharge to Radwaste Temp.*	2CES*PNL405	1
11. RHR Service Water Temperature		1
12. *Service Water Pump Discharge Flow (A,B,C,D,E&F)*	2CES*PNL405	1
12. RCIC Pump Speed		1
13. *Reactor Vessel shell flg temp*	2CES*PNL405	1
13. RCIC Pump Speed		1
14. *Reactor Vessel Bottom Head Temp*	2CES*PNL405	1
15. *Condensate Storage Tank Level (A,B)*	2CES*PNL405	1
16. *ADS Acc. tank no. 32 pressure*	2CES*PNL405	1
17. *ADS Acc. tank no. 33 pressure*	2CES*PNL405	1
18. *ADS Acc. tank no. 38 pressure*	2CES*PNL405	1
19. *ADS Acc. tank no. 35 pressure*	2CES*PNL405	1

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	READOUT LOCATION
1. Reactor Vessel Pressure	M	R	2CES*PNL405
2. Reactor Vessel Water Level (2)	M	R	2CES*PNL405
3. *Suppression Pool Water Level*	M	R	2CES*PNL405
3. Safety/Relief Valve Position	M	NA	
4. *Suppression Pool Water Temp. (2)*	M	R	2CES*PNL405
4. Suppression Chamber Water Level	M	R	
5. *RHS H_X Discharge Flow*	M	R	2CES*PNL405
5. Suppression Chamber Water Temperature	M	R	
6. *Service Water Flow to RHS Ht. Exch. (A,B)*	M	R	2CES*PNL405
6. Suppression Chamber Air Temperature	M	R	
7. *RHS Ht. Exch. Service Water Outlet Temp. (A,B)*	M	R	2CES*PNL405
7. Drywell Pressure	M	R	
8. *RCIC Turbine Speed*	NA	R	2CES*PNL405
8. Drywell Temperature	M	R	
9. *RCIC Pump Discharge Flow*	NA	R	2CES*PNL405
9. RHR System Flow	M	R	
10. *RHS Heat Exchanger In/Out Temp. (A,B)*	M	R	2CES*PNL405
10. RHR Service Water System Flow	M	R	
11. *RHS Discharge to Radwaste Temp.*	M	R	2CES*PNL405
11. RHR Service Water Temperature	M	R	
12. *Service Water Pump Discharge Flow (A,B,C,D,E,&F)*	M	R	2CES*PNL405
12. RCIC System Flow	M	R	
13. *Reactor Vessel shell flg temp*	M	R	2CES*PNL405
13. RCIC Turbine Speed	M	R	
14. *Reactor Vessel Bottom Head Temp*	M	R	2CES*PNL405
14. ________	____	____	
15. *Condensate Storage Tank Level (A,B)*	M	R	2CES*PNL405
16. *ADS Acc. tank no. 32 pressure*	M	R	2CES*PNL405
17. *ADS Acc. tank no. 33 pressure*	M	R	2CES*PNL405
18. *ADS Acc. tank no. 38 pressure*	M	R	2CES*PNL405
19. *ADS Acc. tank no. 35 pressure*	M	R	2CES*PNL405

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2; *(except as noted on Table 3.3.7.5-1)*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3. .5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Reactor Vessel Pressure *	2	1	80
2. Reactor Vessel Water Level *	2	1	80
3. Suppression ~~Chamber~~ Pool Water Level *	2	1	80
4. Suppression ~~Chamber~~ Pool Water Temperature *	(4)(8), 1/sector	(4)(8), 1/sector	80
5. Suppression Chamber Air Temperature *	2	1	80
6. Drywell Pressure *	2	1	80
7. Drywell Air Temperature	2	1	80
8. Drywell Oxygen Concentration	2	1	80
9. Drywell Hydrogen Concentration Analyzer and Monitor	2	1	80
10. Safety/Relief Valve Position Indicators	(2)/valve	1/valve	80
~~11. In-Core Thermocouples.~~	~~(4)/core quadrant~~	~~(2)/core quandrant~~	~~80~~
11. Drywell High Range Radiation Monitors	2	1	81
~~12. Primary Containment Gross Radiation Monitors~~	~~2~~	~~1~~	~~81~~
12. Main Stack Gaseous Radiation Effluent Monitor (a)	1	1	83
~~13. Containment Ventilation Monitor(#)~~	~~(1)~~	~~1~~	~~81~~
13. Radwaste/Reactor Building Vent Gaseous Radiation Effluent Monitor (a)	1	1	83
~~14. Off-gas and Rad Waste Building Ventilation Monitor(#)~~	~~(1)~~	~~1~~	~~81~~
14. Service Water Discharge - RHR Heat Exchanger Radiation Monitor	1/heat exch.	1/heat exch.	84
~~15. Fuel Handling Area Monitor(#)~~	(1)	1	81
15. Service Water Effluent Radiation Monitor	1	1	84
~~16. Turbine Building Ventilation Monitor(#)~~	~~(1)~~	~~1~~	~~81~~
16. Vital Area Radiation Monitors			
17. ________________	(1)	1	81

(#Noble gas monitors.) (a) For all OPERATIONAL CONDITIONS and when irradiated fuel is being handled in secondary containment.

*[illegible] in the Control Room

Table 3.3.7.5-1 (Cont.)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
16. Vital Area Radiation Monitors			
a. Radwaste Sample Room	1	1	81
b. Radwaste Control Room	1	1	81
c. Main Control Room	1	1	81
d. Relay and Computer Room	1	1	81
e. Turbine Building Gaseous Effluent Monitor Location	1	1	81
f. Main Stack Gaseous Effluent Monitor Location	1	1	81
17. Refuel Platform High Radiation Monitor	**	1	82

** Required when handling irradiated fuel or irradiated components in the fuel pool.

Table 3.3.7.5-1 (Continued)
ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

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

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

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

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

~~b. In lieu of another report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

ACTION 82 - [ ] *suspend movement of irradiated components in the fuel pool or reactor cavity, or, initiate the preplanned alternate method of monitoring the appropriate parameter.*

ACTION 83 - *With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken and analyzed at least once per 12 hours.*

ACTION 84 - *With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcuries/ml.*

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Suppression Chamber Water Level	M	R
4. Suppression Chamber Water Temperature	M	R
5. Suppression Chamber Air Temperature	M	R
6. Primary Containment Pressure	M	R
7. Drywell Air Temperature	M	R
8. Drywell Oxygen Concentration	M	R
9. Drywell Hydrogen Concentration Analyzer and Monitor	~~(M)(NA)~~	Q*
10. Safety/Relief Valve Position Indicators	M	R
11. In-Core Thermocouples	M	R
~~11. In-Core Thermocouples~~	~~M~~	~~R~~
11. Drywell High Range Radiation Monitors	M	R
~~12. Primary Containment Gross Radiation Monitors~~	~~M~~	~~R~~
12. Main Stack Gaseous Effluent Radiation Monitor #	M	R
~~13. Containment Ventilation Monitor(#)~~	M	R
13. Radwaste/Reactor Building Gaseous Effluent Radiation Monitor #	M	R
~~14. Off-gas and Rad Waste Building Ventilation Monitor(#)~~	M	R
14. Service Water Discharge - RHR Heat Exchanger Radiation Monitor	M	R
~~15. Fuel Handling Area Monitor(#)~~	M	R
15. Service Water Effluent Radiation Monitor	M	R
~~16. Turbine Building Ventilation Monitor(#)~~	M	R
~~17.~~ ____________	M	R

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

~~(# Noble gas monitors.)~~
Noble gas monitor only.

Table 4.3.7.5-1 (Cont.)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
16. Vital Area Radiation Monitors		
a. Radwaste Sample Room	M	R
b. Radwaste Control Room	M	R
c. Main Control Room	M	R
d. Relay and Computer Room	M	R
e. Turbine Building Gaseous Effluent Monitor Location	M	R
f. Main Stack Gaseous Effluent Monitor Location	M	R
17. Refuel Platform High Radiation Monitor	*	R

* Immediately prior to handling irradiated fuel or irradiated components in the fuel pool and once per month thereafter.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

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

a. In OPERATIONAL CONDITION 2*, three

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. Performance of a:

1. CHANNEL CHECK at least once per:

   a) 12 hours in CONDITION 2*, and

   b) 24 hours in CONDITION 3 or 4.

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

b. Performance of a CHANNEL FUNCTIONAL TEST:

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

2. At least once per 31 days.

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

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

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or ~~(TPF)~~ ~~(MFLPD)~~.

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.10-1 shall be OPERABLE.

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

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.10-1:

a) Within 1 hour, establish a FIRE WATCH PATROL to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

b) Restore the minimum number of instruments to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2

c) The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

*This does not include detector sensitivity check.

TABLE 3.3.7.10-1

FIRE DETECTION INSTRUMENTATION

*Based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

INSTRUMENT LOCATION (Illustrative) (**)		MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE
A. Reactor Bldg./Aux. Bays				
201 SW	El. 175'-0"			15(16)
202 SW	"			7(7)
203 SW	"			6(6)
204 SW	"			6(6)
205 NZ	"			7(7)
206 SW	"			8(8)
207 SW	"			7(7)
208 SW	"			11(10)
212 SW	"	12(13)		31(34)
213 SW	"	18(20)		32(35)
211 SW	El. 196'-0"			20(22)
214 SW	"			20(22)
221 SW	El. 215'-0"			26(28)
222 SW	"			36(39)
223 SW	"			36(39)
224 SW	"			23(25)
231 SW	El. 240'-0"			28(31)
232 SW	"	5(5)		29(32)
238 SW	"	1(1)		29(32)
239 SW	"			27(29)
243 SW	El. 261'-0"	5(5)		35(38)
245 SW	"	2(2)		34(37)
252 SW	El. 289'-0"	4(4)		36(39)
255 SW	"	4(4)		30(33)
261 NZ	El. 306'-0"	13(14)		
262 NZ	"			24(26)
271 SW	El. 328'-10"			18(19)
272 SW	"			18(19)
273 SW	"			14(15)
274 SW	"			18(19)
281 NZ	El. 353'-10"			77(85)

TABLE 3.3.7.10-1

(Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative) (**)		MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE

* based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

B. Control Building (Zones)

Zone	Elevation	HEAT	FLAME	SMOKE
305 NW	El. 214'-0"			4(4)
306 NW	"			9(10)
307 NW	"			1(1)
308 NW	"			1(1)
309 NW	"			5(5)
310 NZ	"			4(4)
311 NZ	"			3(3)
312 NZ	"			9(10)
321 NW	El. 237'-0"			4(4)
322 NW	"			13(14)
323 NW	"			14(15)
324 NW	"			4(4)
325 NW	"			3(3)
326 NW	"			3(3)
327 NW	"			4(4)
331 NW	El. 261'-0"			18(20)
332 NW	"			5(5)
333 XL	"			7(7)
334 NZ	"			4(4)
335 NZ	"			4(4)
336 XL	"			7(7)
337 NW	"			5(5)
338 NZ	"			4(4)
339 NZ	"			1(1)
340 NZ	"			2(2)
341 NZ	"			2(2)
342 XL	"			4(4)
351 NZ	El. 288'-6"			16(17)
352 NW	"			4(4)
353 SG	"	48(48)		24(24)
354 SG	"	48(48)		48(48)
355 NZ	"			(LTR)
356 NZ	"			18(20)
357 XG	"			8(8)
358 XG	"			4(4)
359 NW	"			5(5)
360 NZ	"			10(11)
362 SG	"	40(40)		20(20)

TABLE 3.3.7.10-1

(Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative) (**)			MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE

* based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

			HEAT	FLAME	SMOKE
B. (cont.)	371 NW	El. 306'-0"			4(4)
	372 NZ	"			(LTR)
	373 NZ	"			24(26)
	374 SG	"	40(40)		20(20)
	375 SG	"	40(40)		20(20)
	376 SG	"			10(11)
	377 NW	"			3(3)
	378 NZ	"			9(9)
	380 NZ	"			18(20)
	387 SG	"	35(35)		28(28)
C.	Diesel Generator Building				
	401 NZ	El. 261'-0"			9(9)
	402 SW	"			6(6)
	403 SW	"			6(6)
	404 SW	"			6(6)
D.	Electrical Tunnels				
	301 NW	El. 215'-0"			25(27)
	302 NW	"			10(11)
	303 NW	"			3(3)
	304 NW	"			11(12)
	236 NZ	El. 237'-0"			8(8)
	237 NZ	"			9(9)
E.	Service Water pump pits				
	806 NZ	El. 224'-0"			6(6)
	807 NZ	"			6(6)
F.	Fire Pump rooms				
	804 NW	El. 261'-0"			8(8)
	805 NZ	"			2(2)

(**) List all detectors in areas required to insure the OPERABILITY of safety related equipment and indicate instruments which automatically actuate fire suppression system.

Delete NA

INSTRUMENTATION

CHLORINE (AND AMMONIA) DETECTION SYSTEM (Optional)

LIMITING CONDITION FOR OPERATION

3.3.7.8 Two independent chlorine (and ammonia) detection system subsystems shall be OPERABLE with their (alarm)/(trip) setpoints adjusted to actuate at a:

a. Chlorine concentration of less than or equal to (5) ppm, and

b. Ammonia concentration of less than or equal to ( ) ppm.

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one chlorine (and/or one ammonia) detection subsystem inoperable, restore the inoperable detection subsystem to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the (isolation) mode of operation.

b. With both chlorine (and/or ammonia) detection subsystems inoperable, within one hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the (isolation) mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine (and ammonia) detection system subsystems shall be demonstrated OPERABLE by performance of

a. CHANNEL CHECK at least once per 12 hours,

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

c. CHANNEL CALIBRATION at least once per 18 months.

Delete NA

INSTRUMENTATION

CHLORIDE INTRUSION MONITORS (Optional)

LIMITING CONDITION FOR OPERATION

3.3.7.9 The chloride intrusion monitor channels shown in Table 3.3.7.9-1 shall be OPERABLE with alarm setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.7.9-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION

a. With a chloride intrusion monitor channel trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.3.7.9-2, declare the monitor inoperable until the monitor is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than the Minimum OPERABLE Channels for up to two functional units, sample the parameter monitored by the inoperable channel(s) of the functional unit(s) at least once per 4 hours; restore at least the Minimum OPERABLE Channels for at least two functional units to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

c. With the number of OPERABLE Channels less than the Minimum OPERABLE Channels for more than two functional units, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.9 Each of the above required chloride intrusion monitors shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.9-1.

Delete NA

TABLE 3.3.7.9-1

CHLORIDE INTRUSION MONITORS

FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS
1. Chloride detectors in the condenser hotwell outlet headers	(4)
2. Chloride detectors in the condensate pump discharge	1
3. Chloride detector in the inlet to the deep bed demineralizer	1

Delete NA

TABLE 3.3.7.9-2

CHLORIDE INTRUSION MONITORS SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. Chloride detectors in the condenser hotwell outlet headers	≤ (1.0) μmhos/cm
2. Chloride detectors in the condensate pump discharge	≤ (0.3) μmhos/cm (2.0 μmhos/cm for wide range monitor)
4. Chloride detector in the inlet to the deep bed demineralizer	≤ (0.3) μmhos/cm

Delete NA

TABLE 4.3.7.9-1

CHLORIDE INTRUSION MONITORS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Chloride detectors in the condenser hotwell outlet headers	D	M	R
2. Chloride detectors in the condensate pump discharge	D	M	R
3. Chloride detector in the inlet to the deep bed demineralizer	D	M	R

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.11 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. CHANNEL CHECK at least once per 24 hours,

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

~~c.~~ *e.* CHANNEL CALIBRATION at least once per ~~18 months.~~ *operating cycle.*

b. Once every 7 days listen to the Audio output.

d. Once every 92 days verify the background noise.

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INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12-1

3.3.7.12 ~~3.3.3.10~~ The radioactive liquid effluent monitoring instrumentation channels shown in Table ~~3.3-12~~ shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times *other than when the line is valved out and locked.*

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table ~~3.3-12~~ 3.3.7.12-1. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.~~3~~ 4, 3.0.~~4~~ 5, and ~~6.9.1.9~~ 6.6.1.a.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 ~~4.3.3.10~~ Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table ~~4.3-12.~~ 4.3.7.12-1.

(add to STS Section 3/4.3.7)

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TABLE ~~3.3-12~~ 3.3.7.12-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	~~(1)~~ 1	128
~~b. Steam Generator Blowdown Effluent Line~~	(1)	29
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line A	~~(1)~~ 1	130
b. Service Water Effluent Line B	1	130
~~b. Component Cooling Water System Effluent Line~~	(1)	~~30~~
c. Cooling Tower Blowdown Line	1	130
~~c. Turbine Building (Floor Drains) Sumps Effluent Line~~	~~(1)~~	30
~~3.~~ CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
~~a. Steam Generator Blowdown Effluent Line~~	(1)	29
~~b. Turbine Building Sumps Effluent Line~~	(1)	30
3. ~~4.~~ FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	~~(1)~~ 1	131
b. Service Water Effluent Line A		
~~b. Steam Generator Blowdown Effluent Line~~	~~(1)~~ 1	131
c. Service Water Effluent Line B		
~~c. Discharge Canal~~	~~(1)~~ 1	131
d. Cooling Tower Blowdown Line		
~~d. Turbine Building Sumps Effluent Line~~	~~(1)~~ 1	131

TABLE ~~3.3-12~~ 3.3.7.12-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5.	RADIOACTIVITY RECORDERS*		
	a. ~~Liquid Radwaste Effluent Line~~	~~(1)~~	~~33~~
	b. Steam Generator Blowdown Effluent Line	(1)	33
4. ~~6.~~	TANK LEVEL INDICATING DEVICES**	1	132
	~~a.~~ ________	~~(1)~~	~~32~~
	~~b.~~ ________	~~(1)~~	~~32~~
	~~c.~~ ________	~~(1)~~	~~32~~
	~~d.~~ ________	~~(1)~~	~~32~~

*Required only if alarm/trip set point is based on recorder-controller.

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

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3.3.7.12-1
TABLE ~~3.3-12~~ (Continued)

TABLE NOTATION

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

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

~~ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml:~~

~~a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram DOSE EQUIVALENT I-131.~~

~~b. At least once per 24 hours when the specific activity of~~ the secondary coolant is less than or equal to 0.01 microcurie/gram DOSE EQUIVALENT I-131.

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

ACTION 131 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated ~~at least~~ once per 4 hours during actual releases. Pump performance curves generated in situ may be used to estimate flow.

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

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~~TABLE 3.3-12 (Continued)~~

~~TABLE NOTATION~~

~~ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided the gross radioactivity level is determined at least once per 4 hours during actual releases.~~

4.3.7.12-1
TABLE ~~4.3-12~~

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R~~(3)~~(c)	~~Q(1)~~ SA(a,b)
~~b. Steam Generator Blowdown Effluent Line~~	D	M	~~R(3)~~	~~Q(1)~~
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water ~~System~~ Effluent Line A	D	M	R~~(3)~~(c)	Q~~(2)~~ SA(b)
b. Service Water Effluent Line B ~~b. Component Cooling Water System Effluent Line~~	D	M	R~~(3)~~(c)	Q~~(2)~~ SA(b)
c. Cooling Tower Blowdown Line ~~c. Turbine Building (Floor Drains) Sumps Effluent Line~~	D	M	R~~(3)~~(c)	~~Q(1)~~ SA(b)
~~3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR~~				
~~a. Steam Generator Blowdown Effluent Line~~	D	N.A.	R	Q
~~b. Turbine Building Sumps Effluent Line~~	D	N.A.	R	Q

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4.3.7.12-1

TABLE ~~4.3-12~~ (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
3. ~~4.~~ FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D~~(4)~~ (d)	N.A.	R	Q
b. *Service Water Effluent Line A* ~~b. Steam Generator Blowdown Effluent Line~~	D~~(4)~~ (d)	N.A.	R	Q
c. *Service Water Effluent Line B* ~~c. Discharge Canal~~	D~~(4)~~ (d)	N.A.	R	Q
d. *Cooling Tower Blowdown Line* ~~d. Turbine Building Sumps Effluent Line~~	D~~(4)~~ (d)	N.A.	R	Q
~~5. RADIOACTIVITY RECORDERS*~~				
~~a. Liquid Radwaste Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
4. ~~6.~~ TANK LEVEL INDICATING DEVICES*	*D***	*N.A.*	*R*	*Q*
~~a.~~ ________	~~D**~~	~~N.A.~~	~~R~~	~~Q~~
b. ________	~~D**~~	~~N.A.~~	~~R~~	~~Q~~
c. ________	~~D**~~	~~N.A.~~	~~R~~	~~Q~~
d. ________	~~D**~~	~~N.A.~~	~~R~~	~~Q~~

*See footnote~~s~~ on page ~~3/4 3-73~~: *Table 3.3.7.12-1*

**During liquid additions to the tank.

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4.3.7.12-1

TABLE 4.3-12 (Continued)

TABLE NOTATION

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

~~1.~~ instrument indicates measured levels above the alarm/trip setpoint.

~~2. Circuit failure.~~

~~3. Instrument indicates a downscale failure.~~

~~4. Instrument controls not set in operate mode.~~

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

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

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

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

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

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.13 ~~3.3.3.11~~ The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.13-1 ~~3.3-13~~ shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table ~~3.3-13~~ 3.3.7.13-1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.13-1 ~~3.3-13~~. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.~~3~~4, 3.0.~~4~~5, and ~~6.9.1.9.b~~ 6.6.1.a are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.13 ~~4.3.3.11~~ Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table ~~4.3-13~~ 4.3.7.13-1.

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TABLE ~~3.3-13~~ 3.3.7.13-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. ~~WASTE~~ OFF GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	~~(1)~~ 1	* *	~~35~~ 137
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
b. ~~d.~~ ~~Effluent~~ System Flow Rate Measuring Device	(1)	* *	136
c. ~~e.~~ Sampler Flow Rate Measuring Device	(1)	* *	136
2. ~~2A.~~ ~~WASTE~~ OFF GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM ~~(for systems designed to withstand the effects of a hydrogen explosion)~~			
a. Hydrogen Monitor Train A or	(1) 1	**	139
b. Hydrogen ~~or Oxygen~~ Monitor Train B	~~(1)~~ 1	**	139
~~2B. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)~~			
~~a. Hydrogen Monitor~~	~~(2)~~	~~**~~	~~40~~
~~b. Hydrogen or Oxygen Monitor~~	~~(2)~~	~~**~~	~~40~~

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TABLE ~~3.3-13~~ 3.3.7.13-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~3. CONDENSER EVACUATION SYSTEM~~			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36
4. VENT HEADER SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36
3. STANDBY GAS TREATMENT			
~~5.~~ CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	~~(1)~~ 1	***	138
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41

TABLE ~~3.3-13~~ 3.3.7.13-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. STANDBY GAS TREATMENT (continued) ~~5. CONTAINMENT PURGE SYSTEM (Continued)~~			
d. Flow Rate Monitor	~~(1)~~ 1	***	136
e. Sampler Flow Rate Monitor	~~(1)~~ 1	***	136
4. ~~6.~~ ~~AUXILIARY BUILDING VENTILATION SYSTEM~~ MAIN STACK EFFLUENT			
a. Noble Gas Activity Monitor	~~(1)~~ 1	*	137
b. Iodine Sampler	~~(1)~~ 1	*	141
c. Particulate Sampler	~~(1)~~ 1	*	141
d. Flow Rate Monitor	~~(1)~~ 1	*	136
e. Sampler Flow Rate Monitor	~~(1)~~ 1	*	136
5. RADWASTE/REACTOR BUILDING VENT EFFLUENT ~~7. FUEL STORAGE AREA VENTILATION SYSTEM~~			
a. Noble Gas Activity Monitor	~~(1)~~ 1	*	137
b. Iodine Sampler	~~(1)~~ 1	*	141
c. Particulate Sampler	~~(1)~~ 1	*	141
d. Flow Rate Monitor	~~(1)~~ 1	*	136
e. Sampler Flow Rate Monitor	~~(1)~~ 1	*	136

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~8.~~ RADWASTE AREA VENTILATION SYSTEM			
~~a.~~ Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d. Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~e. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~9. STEAM GENERATOR BLOWDOWN VENT SYSTEM~~			
~~a. Noble Gas Activity Monitor~~	~~(1)~~	~~*~~	~~37~~
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d. Flow Rate Monitor~~	~~(1)~~	*	36
~~e. Sampler Flow Rate Monitor~~	(1)	*	36

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3.3.7.13-1
TABLE ~~3.3-13~~ (Continued)

TABLE NOTATION

*** *When SECONDARY CONTAINMENT is required.*

* At all times.

** During ~~waste~~ *off* gas ~~holdup~~ system operation. ~~(treatment for primary system offgases)~~.

~~ACTION 35 -~~ With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION *1*36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION *1*37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken ~~at~~ *and analyzed at* least once per 12 hours. ~~and these samples are analyzed for gross activity within 24 hours.~~

ACTION *1*38 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.~~

ACTION *1*39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this ~~waste~~ *off* gas ~~holdup system~~ *train* may continue provided grab samples are collected at least once per 24 hours and analyzed within the following 4 hours, and proper function of the recombiner is assured by monitoring recombiner temperature in accordance with approved procedures. *monitoring is performed by channel at entrance to 75 sec holdup pipe or,*

~~ACTION~~ *1*40 - With the number of channnels OPERABLE one ~~less than required~~ by ~~the Minimum Channels~~ OPERABLE requirement, ~~operation of this system may continue~~ for up to 14 days ~~provided grab samples are taken~~ and analyzed daily. ~~With both channels inoperable~~ operation may continue for up to 14 days ~~provided grab samples are~~ taken and analyzed (1) every 4 hours during degassing ~~operations~~, and (2) daily during other operations.

ACTION *1*41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

Close the affected system isolation valves within one hour or initiate the pre-planned alternate method of monitoring.

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TABLE ~~4.3-13~~ 4.3.7.13-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	CONDITIONS ~~MODES~~ IN WHICH SURVEILLANCE REQUIRED
1. OFF WASTE GAS ~~HOLDUP~~ SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	~~P~~ D	~~P~~ M	R~~(3)~~ (c)	SA(a,b) ~~Q(1)~~	* *
~~b.~~ Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
b. ~~d.~~ ~~Effluent~~ System Flow Rate Measuring Device	P	N.A.	R	~~Q~~SA	* *
c. ~~e.~~ Sampler Flow Rate Monitor	D	N.A.	R	~~Q~~SA	* *
2. ~~2A.~~ OFF WASTE GAS ~~HOLDUP~~ SYSTEM EXPLOSIVE GAS MONITORING SYSTEM ~~(for systems designed to withstand the effects of a hydrogen explosion)~~					
a. Hydrogen Monitor Train A	D	N.A.	Q~~(4)~~ (d)	M	**
b. Hydrogen or ~~or Oxygen~~ Monitor Train B	D	N.A.	Q~~(4) or Q(5)~~ (d)	M	**
2B. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Hydrogen or Oxygen Monitor	D	N.A.	Q(4) or Q(5)	M	**

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NMP-UNIT 2

~~TABLE 4.3-13 (Continued)~~

~~RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

~~INSTRUMENT~~	~~CHANNEL CHECK~~	~~SOURCE CHECK~~	~~CHANNEL CALIBRATION~~	~~CHANNEL FUNCTIONAL TEST~~	~~MODES IN WHICH SURVEILLANCE REQUIRED~~
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. VENT HEADER SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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NMP-UNIT 2 ~~PWR-STS-1~~

4.3.7.13-1
TABLE ~~4.3-13~~ (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	CONDITIONS ~~MODES~~ IN WHICH SURVEILLANCE REQUIRED
3. ~~5.~~ STANDBY GAS TREATMENT ~~CONTAINMENT PURGE SYSTEM~~					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	~~P~~ M	R~~(3)~~ (c)	SA (a,b) ~~Q(1)~~	***
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
b. ~~d.~~ Flow Rate Monitor	D	N.A.	R	~~Q~~ SA	***
c. ~~e.~~ Sampler Flow Rate Monitor	D	N.A.	R	~~Q~~ SA	***
4. ~~6.~~ MAIN STACK EFFLUENT ~~AUXILIARY BUILDING VENTILATION SYSTEM~~					
a. Noble Gas Actvity Monitor	D	M	R~~(3)~~ (c)	Q~~(2)~~ (b)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE ~~4.3-13~~ 4.3.7.13-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	CONDITIONS MODES IN WHICH SURVEILLANCE REQUIRED
5. REACTOR/RADWASTE BUILDING VENT EFFLUENT					
7. FUEL STORAGE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R~~(3)~~(c)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
8. RADWASTE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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~~TABLE 4.3-13 (Continued)~~

~~RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

INSTRUMENT	~~CHANNEL CHECK~~	~~SOURCE CHECK~~	~~CHANNEL CALIBRATION~~	~~CHANNEL FUNCTIONAL TEST~~	~~MODES IN WHICH SURVEILLANCE REQUIRED~~
~~9.~~ STEAM GENERATOR BLOWDOWN ~~VENT SYSTEM~~					
~~a.~~ ~~Noble Gas~~ Activity Monitor	D	M	R(3)	Q(2)	*
~~b. Iodine Sampler~~	W	N.A.	N.A.	N.A	*
~~c. Particulate~~ Sampler	W	N.A.	N.A.	N.A.	*
d. ~~Flow Rate Monitor~~	D	N.A.	R	Q	*
~~e. Sampler~~ Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE ~~4.3-13~~ 4.3.7.13-1 (Continued)

TABLE NOTATION

*** *When SECONDARY CONTAINMENT INTEGRITY is required*

* At all times ~~other than when the line is valved out and locked.~~

** During ~~waste~~ *off* gas ~~holdup~~ system operation (treatment for primary system offgases).

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

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

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

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

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

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

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

~~(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:~~

~~1. One volume percent oxygen, balance nitrogen, and~~

~~2. Four volume percent oxygen, balance nitrogen.~~

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

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

a. At least once per 7 days by:

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

a) For the overspeed protection control system; *and*

~~1) Four high pressure turbine control valves, and~~

~~2) Four low pressure turbine interceptor valves~~

~~3)~~ For the electrical overspeed trip system and the mechanical overspeed trip system;

1) Four high pressure turbine ~~throttle~~ stop valves,

2) Four high pressure turbine reheat stop valves,

2) ~~3)~~ Four high pressure turbine control valves, and

3) ~~4)~~ ~~Four~~ *Six* low pressure turbine *[combined stop and]* intercept~~or~~ valves.

SURVEILLANCE REQUIREMENTS (Continued)

[a position vs. time plot]

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

operating cycle

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

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

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

~~b. For the suppression pool (and drywell) spray system:~~

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

~~2. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, declare the associated system inoperable.~~

~~c.~~ b. For the feedwater system/main turbine trip system:

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
~~1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM~~		
a. Drywell Pressure-High	1	1, 2, 3
b. Containment Pressure-High	1	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	1	1, 2, 3
d. Timers		
1) System A	1	1, 2, 3
2) System B	1	1, 2, 3
~~2.~~ 1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level (8)	3	1

TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
~~1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM~~		
~~a. Drywell Pressure-High~~	~~≤ (1.69) psig~~	~~≤ (1.89) psig~~
~~b. Containment Pressure-High~~	~~≤ (35) psig~~	~~≤ ( ) psig~~
~~c. Reactor Vessel Water Level - Low Low Low, Level 1~~	~~≥ ( ) psig~~	~~≥ ( ) psig~~
~~d. Timers~~		
~~1) System A~~	~~≤ (12) minutes~~	~~≤ (13.2) minutes~~
~~2) System B~~	~~≤ (14) minutes~~	~~≤ (15.4) minutes~~
~~2.~~ 1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level ~~(8)~~	≤ ~~(54.5)~~ 202.3 inches*	≤ ~~(56.0)~~ 203.8 inches

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

TABLE 4.3.9.1-1 (Continued)

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
~~1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM~~				
~~a. Drywell Pressure-High~~	~~(NA)~~	~~(M)~~	~~(Q)~~	~~1, 2, 3~~
~~b. Containment Pressure-High~~	~~(NA)~~	~~(M)~~	~~(Q)~~	~~1, 2, 3~~
~~c. Reactor Vessel Water Level-Low Low Low, Level 1~~	~~(NA)~~	~~(M)~~	~~(Q)~~	~~1, 2, 3~~
~~d. Timers~~	~~(NA)~~	~~(M)~~	~~(R)~~	~~1, 2, 3~~
~~2.~~ 1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level ~~(8)~~	~~(NA)~~	~~(M)~~	~~(R)~~	1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation loops are operating at the same flow control valve position.

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

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

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

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

b. *With the safety/relief valve function inoperable for more than one of the above required 17 safety/relief valves, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least ~~(11)~~ *17* {of the following} reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 *safety-relief valves @ 1148 psig, ±1%*
4 ~~(3)~~ safety-relief valves @ {1175} psig ±1%
4 ~~(3)~~ safety-relief valves @ {1185} psig ±1%
4 ~~(3)~~ safety-relief valves @ {1195} psig ±1%
4 ~~(2)~~ safety-relief valves @ {1205} psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety*/relief* valve function of one or more of the above required *17* safety/relief valves inoperable, *restore the valve to OPERABLE status within 72 hours or* be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. ~~b.~~ With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than *110* ~~(95)~~°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) ~~within 2 minutes or~~ if suppression pool average water temperature is *110* ~~(105)~~°F or greater, *and* place the reactor mode switch in the Shutdown position.

d. ~~c.~~ With one or more safety/relief valve ~~(tail pipe pressure switches)~~ {acoustic monitors} inoperable, restore the inoperable ~~(switch(es))~~ {monitor~~(s)~~} to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The ~~(tail-pipe pressure switch)~~ {acoustic monitor} for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be ~~((20) ± (5) psig) ( )~~ *1% of rated flow per SPOS* by performance of a:

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

b. CHANNEL CALIBRATION at least once per ~~18 months~~ *operating cycle*. (**)

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

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

Delete NA

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

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

Valve No.	Setpoint* (psig) ± 1% Open	Close
(F051D)	(1033)	(926)
(F051B)	(1073)	(936)
(F047D)	(1113)	(946)
(F047B)	(1113)	(946)
(F051A)	(1113)	(946)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 ~~The~~ *Two of the* following *three* reactor coolant system leakage detection systems shall be OPERABLE:

a. ~~The~~ *A* primary containment atmosphere ~~(gaseous or particulate)~~ *gaseous* radioactivity monitoring system,

~~b. The primary containment sump flow monitoring system, and~~

~~c. Either the (primary containment air coolers condensate flow rate monitoring system) or the primary containment atmosphere (gaseous or particulate) radioactivity monitoring system.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST ~~at least once per 31 days~~ *semi-annually* and a CHANNEL CALIBRATION at least once per ~~18 months.~~ *operating cycle.*

b. ~~Primary containment sump flow~~ *Drywell floor and equipment drain tank fill rate* monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per ~~18 months.~~ *oprating cycle.*

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

b. *A primary containment atmosphere particulate radioactivity monitoring system.*

c. *The drywell floor and equipment drain tank fill rate monitoring system.*

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No known PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

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

~~d. 1 gpm leakage at a reactor coolant system pressure of (950) ±(10) psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.~~

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the primary containment atmospheric ~~{particulate}~~ ~~{and}~~ ~~{gaseous}~~ radioactivity at least once per ~~{4}~~ (12) hours,

b. Monitoring the ~~primary containment sump flow~~ *drywell floor and equipment drain tank fill* rate at least once per ~~{4}~~ {12} hours,

~~c. Monitoring the primary containment air coolers condensate flow rate or the (gaseous) (particulate) radioactivity at least once per (4) (12) hours, and~~

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

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

a. At least once per ~~18 months~~, *operating cycle*, and *any time the reactor is shut down*

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

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

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

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

b. CHANNEL CALIBRATION at least once per ~~18 months~~. *operating cycle.*

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM	MAXIMUM ALLOWABLE LEAKAGE
2CSH*MOV107	HPCS	≤ 1.0 GPM
2CSH*AOV108	HPCS	≤ 1.0 GPM
2CSL*MOV104	LPCS	≤ 1.0 GPM
2CSL*AOV101	LPCS	≤ 1.0 GPM
2ICS*AOV156	RCIC	≤ 1.0 GPM
2ICS*AOV157	RCIC	≤ 1.0 GPM
2RHS*MOV112	RHR	≤ 1.0 GPM
2RHS*MOV113	RHR	≤ 1.0 GPM
2RHS*MOV104	RHR	≤ 1.0 GPM
2RHS*MOV40 a&b	RHR	≤ 1.0 GPM
2RHS*MOV67 a&b	RHR	≤ 1.0 GPM
2RHS*MOV24 a,b&c	RHR-(LPCI)	≤ 1.0 GPM
2RHS*AOV16 a,b&c	RHR-(LPCI)	≤ 1.0 GPM
2RHS*AOV39 a&b	RHR	≤ 1.0 GPM

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS

INSTRUMENT VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
2RHS*PIS7 a,b&c	Residual Heat Removal	480
2CSL*PS108	Low Pressure Core Spray	575

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.~~4~~5 are not applicable.

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

3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN as rapidly as practical within the cooldown rate limit.

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

c. At all other times:

1. With the conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours.

2. With the chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 24 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

3. The provisions of Specification 3.0.~~3~~4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

a. ~~b.~~ Analyzing a sample of the reactor coolant:

1. Chlorides ~~at least~~ once per:

   a) 72 hours, and

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

2. Conductivity ~~at least~~ once per 72 hours.

3. pH ~~at least~~ once per~~:~~

   ~~a) 72 hours, and~~

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

b. ~~c.~~ Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, for up to 31 days, obtaining an in-line conductivity measurement ~~at least~~ once per:

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

2. 24 hours at all other times.

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

1. 7 days, and

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

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

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

* *Samples for analysis will normally be obtained from station sample points. For conditions requiring isolation of all normal sample points, but with periodic or continuous make-up, samples of make-up water may be substituted for reactor water samples.*

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

~~b. Less than or equal to 100/Ē microcuries per gram.~~

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

ACTION:

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

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 12 ~~six~~-month period, prepare and submit a Special Report to the Commission pursuant to Specification ~~6.9.2~~ 6.9.3 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

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

~~3. Greater than 100/Ē microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.~~

b. In OPERABLE CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 ~~or greater than 100/Ē microcuries per gram~~, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification ~~6.9.1~~ 6.6.1a. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. In OPERATIONAL CONDITION 1 or 2, with:

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

1. ~~2.~~ The off-gas level, before the charcoal adsorbers ~~at the SJAE~~, increased by more than ~~(10,000)~~ 15,000 microcuries per second in one hour during steady state operation at release rates less than ~~(75,000)~~ microcuries per second, or

2. ~~3.~~ The off-gas level, before the charcoal adsorbers ~~at the SJAE~~, increased by more than ~~(15%)~~ 20% in one hour during steady state operation at release rates greater than ~~(75,000)~~ microcuries per second,

perform the sampling and analysis requirements of Item 4b and Item 5b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.~~2~~3 ~~at least once per~~ within 92 days containing the results of the specific activity analysis together with the ~~below~~ following additional information for each occurrence.

Additional Information

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

SURVEILLANCE REQUIREMENTS

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

~~*Not applicable during the startup test program.~~

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross ~~Beta and~~ Gamma Activity Determination	~~At least~~ Once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	~~At least~~ Once per 31 days	1
3. *Isotopic Analysis* ~~Radiochemical for Ē Determination~~	~~At least~~ Once per 6 months*	1
4. Isotopic Analysis for Iodine	a) ~~At least~~ Once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION 3.4.5.b.	1#, 2#, 3#, 4#
	b) ~~At least~~ One sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION 3.4.5.c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	*a)* ~~At least~~ Once per 31 days	1
	b) One sample, between 2 and 6 hours following the change in off-gas level, as required by ACTION 3.4.5.c.	

*~~Sample to be taken~~ after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was ~~last subcritical for 48 hours or longer~~.

#Until the specific activity of the primary coolant system is restored to within its limits.

a. *The maximum rate of change of reactor vessel steam space coolant temperature during normal heatup or cooldown shall be limited to 100°F in any 1 hour.*

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

~~a. A maximum heatup of (100)°F in any one-hour period,~~

~~b. A maximum cooldown of (100)°F in any one-hour period~~,

b. ~~c.~~ A maximum temperature change of less than or equal to ~~(10)~~ *20*°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

c. ~~d.~~ The reactor vessel flange and head flange *metal* temperature *shall be maintained* greater than or equal to ~~(~~70~~)~~°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor ~~coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A' or B and B', as applicable, at least once per 30 minutes.~~ *vessel steam space pressure and metal temperature of the reactor vessel flange surfaces, bottom outside surface and bottom head inside surface, as measured by the bottom head drain temperature shall be determined to be to the right of the appropriate region defined by Figures 3.4.6.1-1a, b, c at least once per 30 minutes.*

SURVEILLANCE REQUIREMENTS (Continued)

~~4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.~~

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined ~~to determine reactor pressure vessel fluence~~ as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. ~~The results of these fluence determinations shall be used to update the curves of Figure 3.4.6.1-1.~~

4.4.6.1.~~4~~ 5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to (70)°F:

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

1. ≤ ~~100~~ 90°F, at least once per 12 hours.

2. ≤ (80)°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs, *except that 10% of the bolting studs may be fully tensioned to below 70°F.*

4.4.6.1.2 The rate of change at the reactor steam space coolant temperature during heat-up and cooldown shall be determined to be less than or equal to 100°F in any 1 hour at least once per 30 minutes.

4.4.6.1.4 The reactor flux wire specimens shall be removed at the first normal outage after 1 year and before the end of 2 years of initial operation and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify the Figure B3/4.4.6-1. The results of these fluence determinations, in conjunction with Bases Figure B3/4.4.6.2, shall be used to adjust, if needed, the curves of Figure 3.4.6.1-1.

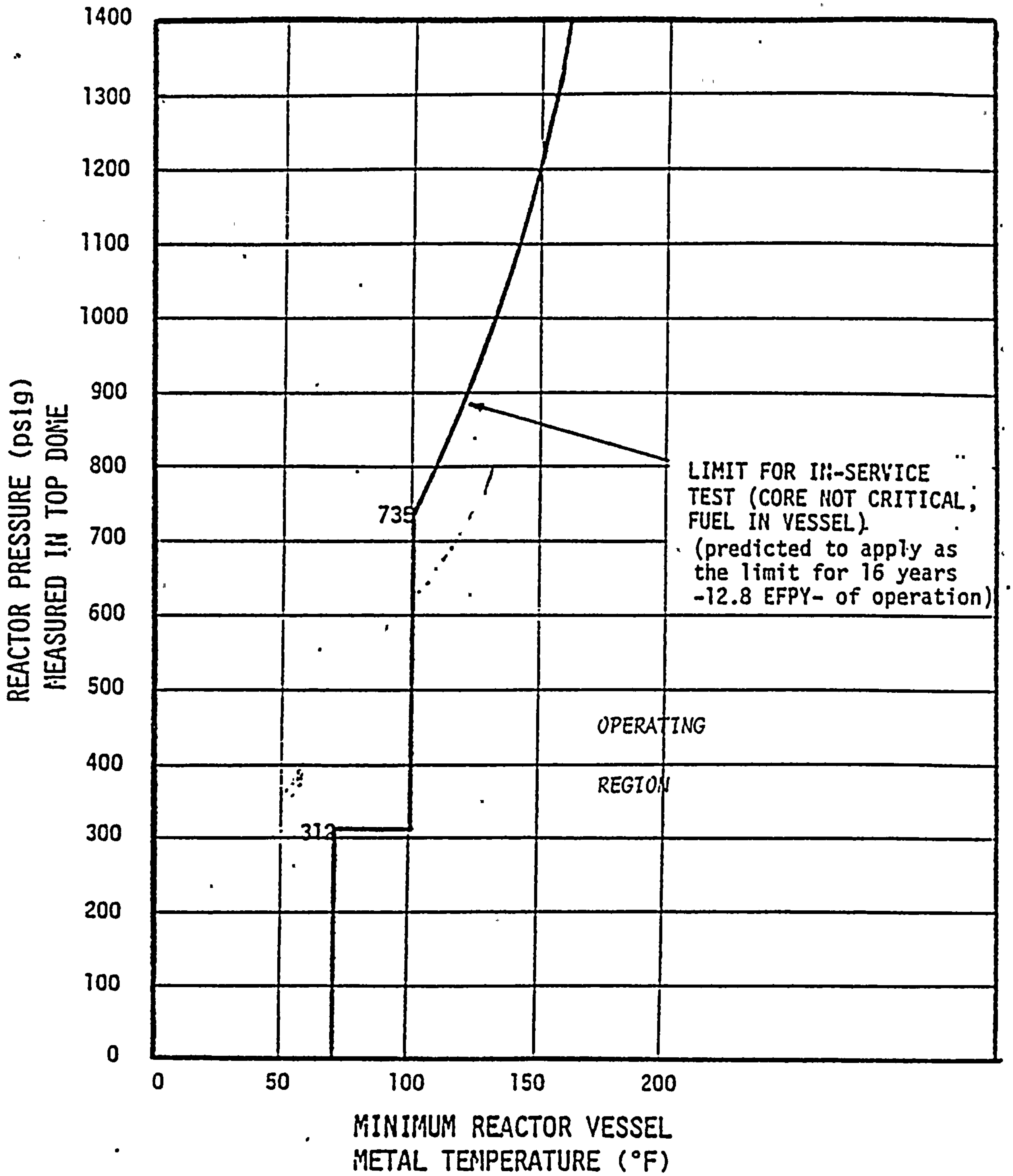

FIGURE 3.4.6.1-1a MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION DURING HYDROSTATIC TESTING (REACTOR CORE NOT CRITICAL)

LIMIT FOR IN-SERVICE TEST
(CORE NOT CRITICAL, FUEL IN VESSEL)

PRESSURE (psig)	TEMPERATURE (F)
0	70
312	70
313	100
735	100
750	103.5
800	109
850	114.5
900	120
950	125.5
1000	131
1050	135.5
1100	140
1150	144
1200	148
1250	151.5
1300	155
1350	158.8
1400	161

MINIMUM REACTOR VESSEL METAL TEMPERATURE FOR PRESSURIZATION
DURING HEATUP OR COOLDOWN (REACTOR NOT CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO 16 YEARS (12.8 EFFECTIVE FULL POWER YEARS) OF CORE OPERATION

TABLE 3.4.6.1-1a

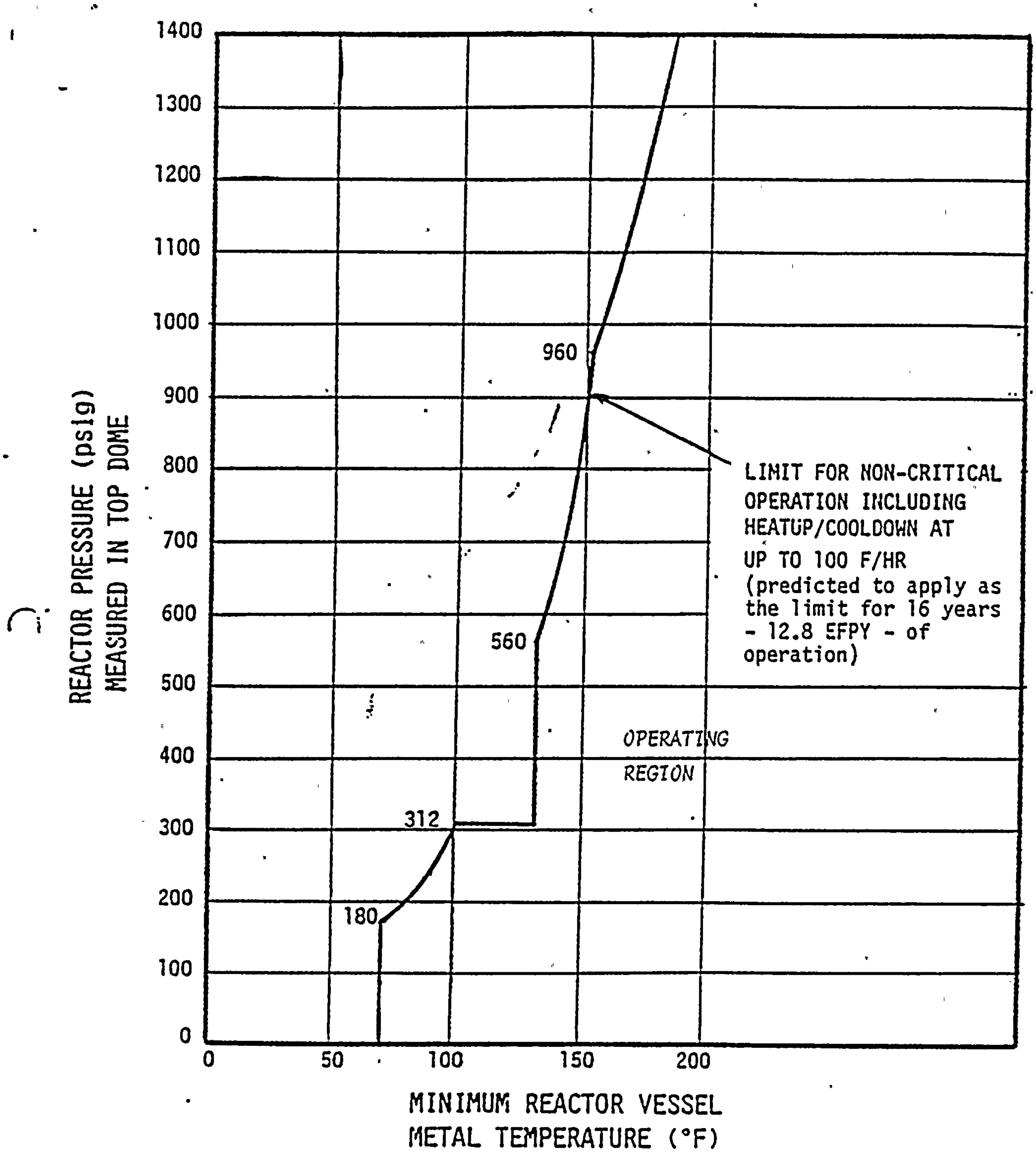

FIGURE 3.4.6.1-1b MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION DURING HYDROSTATIC TESTING (REACTOR CORE NOT CRITICAL)

LIMIT FOR NON-CRITICAL OPERATION
INCLUDING HEARUP/COOLDOWN AT
UP TO 100F/HR

PRESSURE (psig)	TEMPERATURE (F)
0	70
90	70
180	70
240	90
312	100
313	130
560	130
600	133
650	138.5
700	142
750	145
800	147
850	148.5
900	150
960	152
1000	157
1050	161.5
1100	165
1150	168.5
1200	172
1250	175.5
1300	179
1350	182
1400	185

MINIMUM REACTOR VESSEL METAL TEMPERATURE FOR PRESSURIZATION DURING HEATUP OR COOLDOWN (REACTOR NOT CRITICAL) (HEATING OR COOLING RATE 100F/HR) FOR UP TO 16 YEARS (12.8 EFFECTIVE FULL POWER YEARS) OF CORE OPERATION.

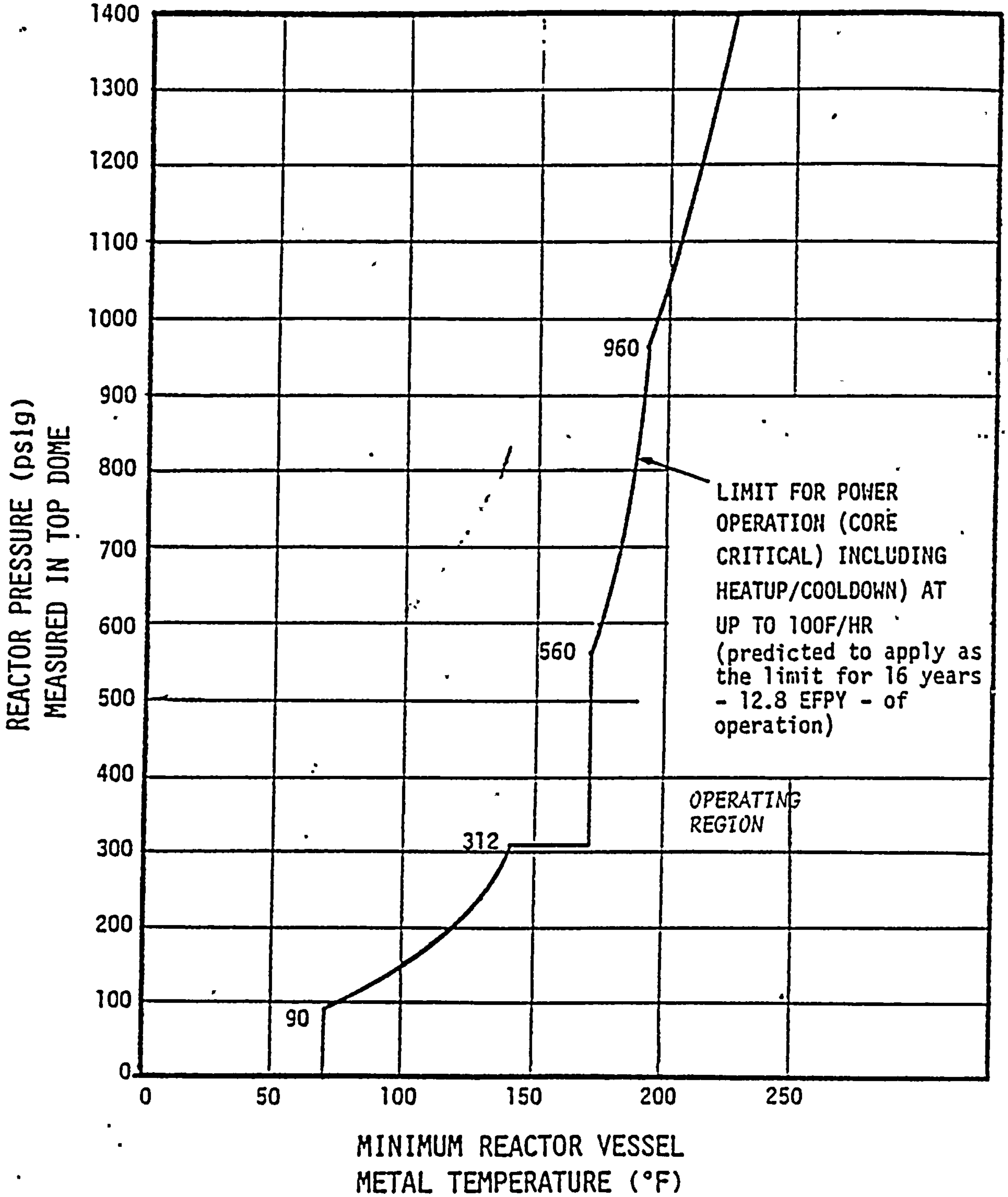

FIGURE 3.4.6.1-1c MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION DURING HYDROSTATIC TESTING (REACTOR CORE NOT CRITICAL)

LIMIT FOR POWER OPERATION (CORE CRITICAL)
INCLUDING HEATUP/COOLDOWN AT UP TO 100F/HR

PRESSURE (psig)	TEMPERATURE (F)
0	70
90	70
180	110
240	130
312	140
313	170
560	170
600	173
650	178.5
700	182
750	185
800	187
850	188.5
900	190
960	192
1000	197
1050	201
1100	205
1150	208.5
1200	212
1250	215.5
1300	219
1350	222
1400	225

MINIMUM REACTOR VESSEL METAL TEMPERATURE FOR PRESSURIZATION
DURING HEATUP OR COOLDOWN (REACTOR NOT CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO 16 YEARS (12.8 EFFECTIVE FULL POWER YEARS) OF
CORE OPERATION

TABLE 3.4.6.1-1c

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ ¼T	WITHDRAWAL TIME (EFPY)
1	3°	0.41	one fourth service life
2	177°	0.41	three fourth service life
3	183°	0.41	standby

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than ~~(1045)~~ 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding ~~(1045)~~ 1020 psig, reduce the pressure to less than ~~(1045)~~ 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than ~~(1045)~~ 1020 psig at least once per 12 hours.

*Not applicable during anticipated transients.

REACTOR COOLANT SYSTEM

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

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

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

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

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

b. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between ~~(3)~~ and ~~(5)~~ seconds when tested pursuant to Specification 4.0.5.

REACTOR COOLANT SYSTEM

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

d. The provisions of Specification 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. ECCS division 1 consisting of:

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

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

3. ~~(At least)~~ ~~(7)~~ OPERABLE ADS valves.

b. ECCS division 2 consisting of:

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

2. ~~(At least)~~ ~~(7)~~ OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel., ~~including a Division 1 or 2 service water system operable.~~

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

---

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

#See Special Test Exception 3.10.6.

##*A&B LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when reactor vessel is less than the RHR cut-in permissive setpoint.*

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

EMERGENCY CORE COOLING SYSTEMS

LIMTING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

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

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

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

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

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

~~g. With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise, declare the associated ECCS inoperable.~~

f. ~~h.~~ In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification ~~6.9.2~~ 6.4 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the useage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

1. LPCS pump develops a flow of at least 6350 gpm with a pump differential pressure greater than or equal to (**) psid.

2. LPCI pump develops a flow of at least 7450 gpm with a pump differential pressure greater than or equal to (**) psid.

3. HPCS pump develops a flow of at least 6350 gpm with a pump differential pressure greater than or equal to (**) psid.

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

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

---

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

** To be determined during pre-op testing.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

2. At least once per operating cycle:

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

   b. Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig and observing that either:

      1. The SRV Discharge Acoustic Monitoring System responds accordingly, or

      2. The SRV Discharge Line Temperature Monitoring System responds accordingly.

   c. Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying low a alarm setpoint of 186 ± 2 psig decreasing pressure.

   d. Perform a leak rate test on each accumulator not to exceed 1 SCHF.

---

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

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

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

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

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

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

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

1. From the suppression chamber, or

2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least 283,000 ~~(150,000)~~ available gallons of water, equivalent to a level of [58.5%]

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

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

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

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

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

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

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 SUPPRESSION ~~CHAMBER~~ [POOL]

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression ~~chamber~~ [pool] shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least ~~(142,160)~~ [145,495] ft^3, equivalent to a level of ~~(25'10").~~ [23'-5 5/8"]

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least ~~(,)~~ [145,495] ft^3, equivalent to a level of ( ), except that the suppression ~~chamber~~ [pool] level may be less than the limit or may be drained provided that:

1. No operations are performed that have a potential for draining the reactor vessel,
2. The reactor mode switch is locked in the Shutdown or Refuel position,
3. The condensate storage tank contains at least ~~(150,000)~~ [283,000 gals] available gallons of water, equivalent to a level of [58.5%] and
4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

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

ACTION:

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

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

~~#See~~ [pool] Specification 3.6.2.1 for pressure suppression requirements.

*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 3.9.8 and 3.9.9.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool ~~chamber~~ water pool level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression ~~chamber~~ water level to be greater than or equal to 23'-5 5/8" ~~(26'10") or (14' 0"), as applicable~~, at least once per 12 hours by (an alternate method).

d. With both suppression pool ~~chamber~~ water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression ~~chamber~~ water level to be greater than or equal to 23'-5 5/8" ~~(26'10") or (14' 0"), as applicable~~, at least once per 12 hours by (at least one alternate pool method).

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool ~~chamber~~ shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, ~~as applicable:~~

1. 23'-5 5/8" ~~(26'10")~~ at least once per 24 hours.

~~2. ( ) at least once per (12) hours.~~

b. ~~(At least)~~ two suppression pool ~~chamber~~ water level instrumentation channels OPERABLE ~~with the~~ low water level alarm setpoint ~~at greater than or equal to ( ) (or ( ), as applicable,)~~ by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

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

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

b. Verify footnote conditions * to be satisfied.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at ~~Pa, (40.4)~~ P_t 40.0 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

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

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

*See Special Test Exception 3.10.1

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

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary Containment leakage rate shall be limited to the following conditions when the reactor coolant system temperature is $\geq$ 200°F:

a. Type A Test

(1) The overall integrated leakage rate L_t shall not exceed 1.10 weight percent of the contained air per 24 hours at the test pressure (P_t) of 40 psig.

(2) The maximum allowable operational leakage rate L_{to} which shall be met prior to power operation following a Type A test (either as measured or following repairs and retest) shall not exceed 0.75 L_t (0.825 weight percent per day) at the test pressure of 40 psig.

(3) The maximum allowable operational (Type A) leakage rate may be prorated between 0.75 L_t and L_t in proportion to the time elapsed since the last Type A test. This leakage rate may be used in lieu of 0.75 L_t when establishing the acceptability of Type B and C tests performed between Type A tests.

(4) When adding the leakage rate measured during a Type C test to the results of a Type A test, the lower leakage rate of the two isolation valves in a line shall be used. to determine the overall integrated leak rate

b. Type B & Type C Tests

(1) If the total leakage rates of testable penetrations, isolation valves, airlock seals and testable containment components as adjusted to a test pressure of 40 psig (P_t) is greater than 90% L_{to}, repairs and retests shall be performed to correct the condition.

(2) When system pressure (Psys) on the opposite side of the isolation valve under test cannot be reduced to atmospheric pressure, then test pressure shall not be less than P_t (40) + Psys.

(3) Leakage shall be less than or equal to 6.0 SCFH for any one main steam line isolation valve when tested at P_t (40) psig.

(4) Leakage less than or equal to 1.875 SCFH for one main steam drain line isolation valve when tested at P_t (40) psig.

(5) Leakage less than or equal to 0.234 SCFH for any one post-accident sampling system gas sample or return line block valve when tested at P_t (40) psig.

(6) Air lock leakage rate of less than or equal to .05 L_a is acceptable at P_t of 40 psig.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION (Cont.)

APPLICABILITY: Applies to the allowable leakage rate of the primary containment system whenever primary containment is required per 3.6.1.1.

ACTION AND RESTORATION:

a. Integrated Primary Containment Leakage Rate - Type A Test

(1) Integrated leak rate tests shall be performed at the test pressure (P_t) of 40 psig. Containment pressure shall not be permitted to decrease more than one (1) psi below P_t.

(2) Type B and C tests should be completed prior to each Type A test. Type B and C leakages not accounted for in the Type A test shall be added to the upper confidence limit (UCL) to estimate the overall integrated leakage rate.

(3) The measured leakage rates corresponding to specification 3.6.1.2.b.3 thru 3.6.1.2.b.5 shall not be added to the results of the type A test when determining the primary containment overall integrated leakage rate.

(4) If the leakage rate exceeds the acceptance criterion, corrective action shall be required. If, during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with the satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local leakage test shall be performed before and after the repair of each isolated leakage path. The sum of the post repaired local leakage rates and the UCL shall be less than 75 percent of the maximum allowable leakage rate, L_t (40). Local leakage rates shall not be subtracted from the Type A test results to determine the acceptability of a test.

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

(6) A Type A test shall last a minimum of eight hours after stabilization. (Initial Test shall be for a full 24 hours to establish baseline statistics.) A verification test shall be performed following each Type A test. The verification test provides a method for assuring that systematic error or bias is given adequate consideration. During the verification test, containment perssure may decrease more than one (1) psi below P_t.

(7) Start-up of the plant cannot occur until a successful Type A Test is conducted.

ACTION AND RESTORATION: (Cont.)

b. Type B and Type C Tests

(1) Leakage repair to testable penetrations, isolation valves, airlock seals and testable containment components shall be performed when necessary to provide an overall successful Type A test result and to ensure total leakage rates do not exceed those specified by 3.6.2.b.1 thru 3.6.2.b.5; otherwise, shut down and repair to allowable standards as follows:

With one primary containment air lock door inoperable:

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

(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 at least once per 31 days.

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

(d) The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.2 a. Type A Testing

Frequency

(1) Three Type A tests shall be conducted during each ten year service interval. The first test shall be performed 30-50 months after the beginning of each ten year service interval; the second test shall be 70-90 months after the beginning of each interval; and the third test shall be coincident with the ten year schedule inservice inspection shutdown.

(2) Retesting

(a) If a Type A test fails to meet the acceptance criteria of 3.6.1.2.a, a Corrective Action Plan that focuses attention on the cause of the problem shall be developed and implemented. A Type A test that meets the requirements of is required prior to plant start-up. A report of the Corrective Action following the failed Type A test shall be submitted to the NRC for approval together with the Containment Leak Test Report.

4.6.1.2 a. Type A Testing (Cont.)

Frequency (Cont.)

(2) (b) If any periodic Type A test fails to meet the acceptance criteria of 3.6.1.2.a, the test schedule applicable to subsequent Type A tests will be reviewed and approved by the NRC.

(c) (i) If two consecutive periodic Type A tests (not including an immediate retest under (a)) fail to meet the acceptance criteria of 3.6.1.2.a or 3.6.1.2.b, not withstanding the periodic retest schedule of (b), a Type A test must be performed at each refueling outage unless alternative leak test requirements are acceptable to the NRC. This testing shall be performed until two consecutive periodic Type A tests (not including an immediate retest under (a)) meet the acceptance criteria of 3.6.1.2.a or 3.6.1.2.b, after which time the retest schedule specified in the retest schedule specified in (b) should be resumed.

(ii) If a Type A test fails solely due to a specifically identified and subsequently corrected Type B or C tested leak path, a Corrective Action Plan shall be developed and submitted to the NRC for approval showing increased Type B or C testing frequency, as appropriate, in place of the additional Type A tests required under (i) above.

4.6.1.2 b. Local Leak Rate - Type B and Type C Tests

(1) Primary containment testable penetrations and isolation valves (see LCO Tables and), except as noted, shall be tested at a pressure of 40 psig (Pac) each major refueling outage, not to exceed two years, except bolted double-gasketed seals which shall be tested whenever the seal is closed after being opened and at least at each refueling outage not to exceed a two year interval.

(2) Personnel airlocks shall be tested in accordance with the following:

(a) The airlocks shall be tested at a test pressure of 40 psig within 24 hours after primary containment integrity is required following a refueling outage or maintenance outage requiring drywell access.

(b) Airlocks opened during periods when primary containment integrity is required shall be tested within three days after being opened. For airlock doors open more frequently than once every three days, the airlocks shall be tested at least once every three days.

(c) The airlocks shall be tested every six months following a refueling or maintenance outage at a test pressure of psig

(3) Containment components (i.e., RHR heat exchangers) not included in (1) and (2) above which required leak repairs following any integrated leakage rate test in order to meet the allowable leakage rate, L_t shall be subjected to local leak tests at a pressure of 40.0 psig at each refueling outage.

4.6.1.2.b Local Leak Rate - Type B and Type C Tests (Cont.)

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

(5) Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_t of 40.0 psig, at intervals no greater than once per 3 years.

(6) Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_t of 40 psig and the seal system capacity is adequate to maintain system pressure for at least 30 days.

(7) Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.3 and 4.6.1.8.4.

(8) The provisions of Specification 4.0.2 are not applicable to 24 month or 40 $\pm$ 10 month surveillance intervals.

4.6.1.2.c Other Surveillance Items

(1) Operational Leak Rate Monitoring

(a) When the primary containment is inerted, the containment shall be monitored for gross leakage by weekly review of the inerting system make-up requirements.

(b) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

(2) Inspection

The accessible interior surfaces of the primary containment shall be visually inspected each operating cycle for evidence of deterioration.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY (Prestressed Concrete Containment with Ungrouted Tendons)

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 Primary Containment Tendons. The structural integrity of the primary containment tendons shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter, by:

a. Determining that a representative sample* of at least (15) tendons, (5) vertical and (10) hoop, each have a lift off force of between ____ (minimum) and ____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount ______ log t and the minimum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount _____ log t, where t is the time interval in years from initial tensioning of the tendon to the current testing date. This test shall include an unloading cycle in which each of these tendons is detensioned to determined if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to their observed lift off force, ± 3%. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously. If the lift off force of any one tendon in the total sample population is out of the predicted bounds, less than minimum or greater than maximum, an adjacent tendon on each side of the defective tendon shall also be checked for lift off force.

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group, vertical and hoop, may be kept unchanged after the initial selection.

SURVEILLANCE REQUIREMENTS (Continued)

Primary Containment Tendons (Continued)

If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. This single tendon shall be restored to the required level of integrity. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons checked for lift off force and change in elongation during subsequent tests may be reduced to a representative sample of at least 9 tendons, 3 vertical and 3 hoop.

b. Removing one wire or strand from each vertical and hoop tendon checked for a lift off force and determining over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. There are no changes in physical appearance of the sheathing filler grease.

3. A minimum tensile strength value of ____ psi guaranteed ultimate strength of the tendon material for at least three wire or strand samples, one from each end and one at mid-length, cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

4.6.1.5.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.5.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests while the containment is at its maximum test pressure.

4.6.1.5.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of these surfaces. The inspection shall be performed prior to the Type A Containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.4 Reports Any abnormal degradation of the containment structure detected during the above require tests and inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the tendon condition; the condition of the concrete, especially at tendon anchorages; the inspection procedure; the tolerances on cracking; and the corrective actions taken.

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between ~~(2.0)~~ - .5 and ~~+(2.0)~~ + .75 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed ~~(110)~~ 150°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than ~~(110)~~ 150°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the (arithmetical) average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth	E.P. No.
a.	306'-9"	354°	2CMS*TE 101
b.	294'-5"	117°	2CMS*TE 102
c.	283'-0"	58°	2CMS*TE 103
d.	268'-0"	203°	2CMS*TE 104
e.	255'-6"	326°	2CMS*TE 105
f.	244'-0"	295°	2CMS*TE 106
g.	306'-9"	189°	2CMS*TE 116
h.	296'-4"	323°	2CMS*TE 117
i.	282'-6"	243°	2CMS*TE 118
j.	262'-3"	28°	2CMS*TE 119
k.	253'-11"	169°	2CMS*TE 120
l.	244'-0"	110°	2CMS*TE 121

LIMITING CONDITION FOR OPERATION

3.6.1.8 The [*primary containment*] ~~drywell and suppression chamber (6) inch~~ purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each [*14" & 12"*] ~~(20) inch~~ purge [*supply and exhaust valve*] ~~valve~~ shall be ~~sealed~~ closed.

b. Each [*14" & 12" supply and exhaust*] ~~(6) inch~~ purge valve may be open for purge system operation ~~for inerting, deinerting and pressure control~~ *with such operation limited to 90 hours per 365 days (inerting, deinerting and pressure control).*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

~~a. With a (20) inch drywell and suppression chamber purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal the 20 inch valve(s) or otherwise isolate the penetration within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

a. ~~b.~~ With a [*any primary containment purge supply*] ~~(6) inch drywell and suppression chamber purge supply and/or~~ exhaust isolation valve(s) inoperable or open (for more than (90) hours per 365 days) for other than inerting, deinerting or pressure control, close the open ~~(6) inch~~ valve~~(s)~~ or otherwise isolate the penetration(s) within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~(c.~~ With a *any primary containment purge supply* ~~drywell and suppression chamber purge supply~~ and/or exhaust isolation valve~~(s)~~ with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirements 4.6.1.8.3 and/or 4.6.1.8.4, restore the inoperable valve~~(s)~~ to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.)

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Each *primary containment* ~~(20) inch drywell and suppression chamber~~ purge supply and exhaust isolation valve shall be verified to be ~~sealed~~ closed at least once per 31 days.

~~(~~4.6.1.8.2 At least once per 6 months on a STAGGERED TEST BASIS each ~~sealed closed (20) inch drywell and suppression chamber~~ [*14" and 12"*] purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.05) L_a when pressurized to P_a.~~)~~

Delete NA

CONTAINMENT SYSTEMS

ELECTRICAL

PRIMARY CONTAINMENT PENETRATION PRESSURIZATION SYSTEM ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.6.1.9 The primary containment penetration pressurization system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment penetration pressurization system inoperable, restore the system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9 The primary containment penetration pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a, (16.5) psig, and has adequate capacity to maintain system pressure for at least 30 days.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between ~~(148,000)~~ 154,794 ft^3 and ~~(142,160)~~ 145,495 ft^3, equivalent to a level between ~~(27'-10")~~ 24'11 5/8" and ~~(26'10")~~ 23'5 5/8", ~~and a~~

2. Maximum average temperature of ~~(95)~~ 90°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

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

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

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

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable A/√K design ~~valve~~ value of ~~(0.03)~~ .059 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than ~~(95)~~ 90°F, restore the average temperature to less than or equal to ~~(95)~~ 90°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than (105)°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than ~~(95)~~ 90°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than:

   a) ~~(95)~~ 90°F for more than 24 hours and THERMAL POWER greater than (1)% of RATED THERMAL POWER, or

   b) (110)°F,

   place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

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

#See Specification 3.5.3 for ECCS requirements.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable and/or with one suppression pool water temperature instrumentation channel in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.

d. With both suppression chamber water level instrumentation channels inoperable and/or with both suppression pool water temperature instrumentation channels in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water level channel and one inoperable temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

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

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to ~~(95)~~ 90°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to (105)°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to ~~(95)~~ 90°F, by verifying:

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

   b) THERMAL POWER to be less than or equal to (1)% of RATED THERMAL POWER after suppression chamber average water temperature has exceeded ~~(95)~~ 90°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 90 ~~(95)~~°F, by verifying suppression chamber average water temperature less than or equal to (120)°F.

c. By verifying (at least) two suppression chamber water level instrumentation (channels)(divisions, with 2 channels per division) and at least sixteen suppression pool water temperature instrumentation channels, at least one pair in each suppression pool sector, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

with the water level and temperature alarm setpoint for:

1. High water level ≤ ~~(—)~~, *24'11 5/8"*
2. Low water level ≥ ~~(—), and~~ *23'5 5/8", and*
3. High water temperature ≤ ~~(—)~~ °F. *90*

(d. At least once per ~~18 months~~ *operating cycle* by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of ~~(5)~~ *3* psi and verifying that the $A/\sqrt{K}$ calculated from 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.)

or

(d. By conducting drywell-to-suppression chamber bypass leak ~~tests and verifying~~ that the $A/\sqrt{k}$ calculated ~~from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass~~ leak tests are ~~conducted:~~

1. ~~At least once per 18 months at an initial differential pressure of (1.5) psi, and~~
2. ~~At the first refueling outage and then on the schedule required for Type A overall integrated Containment Leakage Rate tests by Specification 4.6.1.2.a, at an initial differential pressure of (5) psi,~~

~~except that, if the first two (5) psi leak tests performed up to that time result in:~~

1. ~~A calculated $A/\sqrt{k}$ within the specified limit, and~~
2. ~~The $A/\sqrt{k}$ calculated from the leak tests at (1.5) psi is ≤ 20% of the specified limit,~~

~~then the leak tests at (5) psi may be discontinued.~~

CONTAINMENT SYSTEMS

SUPPRESSION POOL ~~(AND DRYWELL)~~ SPRAY

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR~~SW~~ heat exchanger and the suppression pool ~~(and drywell)~~ spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool ~~(and/~~or drywell~~)~~ spray loop inoperable, restore the inoperable loop to OPERABLE status within (72 hours) (7 days) or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least [400] ~~(300)~~ gpm on recirculation flow through the RHR heat exchanger, test line and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

(c. At least once per operating cycle ~~18 months~~ by performance of a system functional test which includes manual ~~simulated automatic~~ actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position.)

(d. 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.)

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

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump; and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least 7450 ~~(7,700)~~ gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

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

CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: ~~(As shown in Table 3.6.3-1.)~~ (OPERATIONAL CONDITIONS 1, 2 and 3 ~~(and **))~~

ACTION:

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

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

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

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

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

~~(Otherwise, in Operational Condition,** suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.)~~

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

~~(~~b. At least once per ~~18 months~~ operating cycle by removing ~~(~~at least one of ~~(the)~~ explosive squib~~(s)~~ from ~~(~~at least one of ~~(the)~~ explosive valve~~(s)~~ ~~(~~, such that each explosive squib in each explosive valve will be tested at least once per 36 months,~~)~~ and initiating the 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 and operating life, as applicable~~)~~.

TABLE 3.6.3-1

VALVE FUNCTION AND NUMBER	ISOLATION GROUPS (a)(b)	ISOLATION SIGNALS	MAXIMUM CLOSING TIME (SECONDS)
AUTOMATIC ISOLATION VALVES			
2MSS*HYV6 A, B, C, D	1	Z,X,C,D,E,P,T,R,RM	3 to 5
2MSS*HYV7 A, B, C, D	1	Z,X,C,D,E,P,T,R,RM	3 to 5
2MSS*MOV208	1	Z,X,C,D,E,P,T,R,RM	15
2MSS*MOV111	1	Z,X,C,D,E,P,T,R,RM	15
2MSS*MOV112	1	Z,X,C,D,E,P,T,R,RM	15
2RHS*MOV33 A, B	12	G,RM	35
2RHS*MOV104	5	A,L,M,Z,RM	50
2RHS*MOV40 A, B	5	A,L,M,Z,RM	40
2RHS*MOV67 A, B	5	A,L,M,Z,RM	15
2RHS*MOV112	5	A,L,M,Z,RM	45
2RHS*MOV113	5	A,L,M,Z,RM	45
2CSH*MOV111	14	N,RM	60
2ICS*MOV164	11	H * & F *,RM	15
2CCP*MOV94 A, B	8	B,F,Z,RM	30
2CCP*MOV17 A, B	8	B,F,Z,RM	30
2CCP*MOV16 A, B	8	B,F,Z,RM	30
2CCP*MOV15 A, B	8	B,F,Z,RM	30
2DFR*MOV120	8	B,F,Z,RM	45
2DFR*MOV121	8	B,F,Z,RM	45
2DER*MOV119	8	B,F,Z,RM	35
2DER*MOV120	8	B,F,Z,RM	35
2RCS*SOV104	2	B,F,Z,RM	N/A
2RCS*SOV105	2	B,F,Z,RM	N/A
2FPW*SOV218	8	B,F,Z,RM	N/A
2FPW*SOV219	8	B,F,Z,RM	N/A
2FPW*SOV220	8	B,F,Z,RM	N/A
2FPW*SOV221	8	B,F,Z,RM	N/A
2DFR*MOV139	8	B,F,Z,RM	25
2DFR*MOV140	8	B,F,Z,RM	25
2DER*MOV130	8	B,F,Z,RM	15
2DER*MOV131	8	B,F,Z,RM	15

TABLE 3.6.3-1 (Cont.)

VALVE FUNCTION AND NUMBER	ISOLATION GROUPS (a)(b)	ISOLATION SIGNALS	MAXIMUM CLOSING TIME (SECONDS)
2CCP*MOV265	8	B,F,Z,RM	60
2CCP*MOV273	8	B,F,Z,RM	60
2CCP*MOV122	8	B,F,Z,RM	60
2CCP*MOV124	8	B,F,Z,RM	60
2CPS*AOV104	9	B,F,Y,Z,RM	5
2CPS*AOV105	9	B,F,Y,Z,RM	5
2CPS*AOV106	9	B,F,Y,Z,RM	5
2CPS*AOV107	9	B,F,Y,Z,RM	5
2CPS*AOV108	9	B,F,Y,Z,RM	5
2CPS*AOV109	9	B,F,Y,Z,RM	5
2CPS*AOV110	9	B,F,Y,Z,RM	5
2CPS*AOV111	9	B,F,Y,Z,RM	5
2IAS*SOV164	8	B,F,Z,RM	N/A
2IAS*SOV165	8	B,F,Z,RM	N/A
2IAS*SOV166	8	B,F,Z,RM	N/A
2IAS*SOV184	8	B,F,Z,RM	N/A
2HCS*MOV1 A, B	8	B,F,Z,RM	30
2HCS*MOV2 A, B	8	B,F,Z,RM	30
2HCS*MOV3 A, B	8	B,F,Z,RM	30
2HCS*MOV4 A, B	8	B,F,Z,RM	30
2HCS*MOV5 A, B	8	B,F,Z,RM	30
2HCS*MOV6 A, B	8	B,F,Z,RM	30
2CPS*SOV119	9	B,F,Z,RM	N/A
2CPS*SOV120	9	B,F,Z,RM	N/A
2CPS*SOV121	9	B,F,Z,RM	N/A
2CPS*SOV122	9	B,F,Z,RM	N/A
2CMS*SOV24 A, B, C, D	8	B,F,Z,RM	N/A
2CMS*SOV26 A, B, C, D	8	B,F,Z,RM	N/A
2CMS*SOV32 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV33 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV34 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV35 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV60 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV61 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV62 A, B,	8	B,F,Z,RM	N/A
2CMS*SOV63 A, B,	8	B,F,Z,RM	N/A
2LMS*SOV152	8	B,F,Z,RM	N/A
2LMS*SOV153	8	B,F,Z,RM	N/A
2LMS*SOV156	8	B,F,Z,RM	N/A
2LMS*SOV157	8	B,F,Z,RM	N/A

TABLE 3.6.3-1 (Cont.)

VALVE FUNCTION AND NUMBER	ISOLATION GROUPS (a)(b)(c)	ISOLATION SIGNALS	MAXIMUM CLOSING TIME (SECONDS)
2IAS*SOV167	8	B,F,Z,RM	N/A
2IAS*SOV168	8	B,F,Z,RM	N/A
2IAS*SOV180	8	B,F,Z,RM	N/A
2IAS*SOV185	8	B,F,Z,RM	N/A
2RCS*SOV65 A, B	8	B,F,Z,RM	N/A
2RCS*SOV66 A, B	8	B,F,Z,RM	N/A
2RCS*SOV67 A, B	8	B,F,Z,RM	N/A
2RCS*SOV68 A, B	8	B,F,Z,RM	N/A
2RCS*SOV79 A, B	8	B,F,Z,RM	N/A
2RCS*SOV80 A, B	8	B,F,Z,RM	N/A
2RCS*SOV81 A, B	8	B,F,Z,RM	N/A
2RCS*SOV82 A, B	8	B,F,Z,RM	N/A
2ICS*MOV121	10	K,M,H,Z,RM	14
2ICS*MOV128	10	K,M,H,Z,RM	14
2ICS*MOV170	10	K,M,H,Z,RM	20
2WCS*MOV102	7	B,J,U,S,Z,RM	14
2WCS*MOV112	6	B,U,J,S,W,Z,RM	14
2ICS*MOV148	11	H* & F*,RM	15
TIP BALL VALVES (SOV's) (A, B, C, D, E)	3	B,F,Z,RM	N/A
TIP PURGE VALVE (SOV) C51-J004A-E	3	B,F,Z,RM	N/A

TABLE 3.6.3-1 (Cont.)

VALVE FUNCTION AND NUMBER	ISOLATION GROUPS (a)(b)	ISOLATION SIGNALS	MAXIMUM CLOSING TIME (SECONDS)
B. REMOTE MANUAL ISOLATION VALVES			
	VALVE GROUP A, B		
2RMS*MOV15 A, B	13	RM	130
2RHS*MOV 1, A, B, C	13	RM	60
2RHS*MOV30, A, B	13	RM	130
2RHS*MOV25, A, B	13	RM	130
2RHS*MOV24, A, B, C	13	RM	30
2CSH*MOV118	13	RM	30
2CSH*MOV105	13	RM	10
2CSH*MOV107	13	RM	45
2CSL*MOV112	13	RM	130
2CSL*MOV104	13	RM	60
2ICS*MOV136	13	RM	50
2ICS*MOV143	13	RM	10
2ICS*MOV122	13	RM	90
2ICS*MOV126	13	RM	20
TIP SHEAR (EXPLOSIVE)	13	RM	N/A
VALVES A, B, C, D, E	13	RM	N/A
2FWS*MOV21 A & B	13	RM	80
2WCS*MOV200	13	RM	55
2RHS*MOV26 A, B[c]	13	RM	15
2RHS*MOV27 A, B[c]	13	RM	15

(a) See Specification 3.3.2, Table 3.3.2-1, (and Specification 3.3.6.1, Table 3.3.6.1-1) for isolation signal(s) that operates each valve group.
(b) May be opened on an intermittent basis under administrative control.
(c) These valves are the RHR Heat Exchangers vent lines isolation valves. The vent line connects to the RHR SRV Discharge Header before it penetrates the primary containment. The position indicators for these valves are provided in the Control Room for remote manual isolation. Not subject to type C test.

VALVE FUNCTION AND NUMBER

C. MANUAL ISOLATION VALVES

2SAS*HCV160
2SAS*HCV161
2SAS*HCV162
2SAS*HCV163

2AAS*HCV134
2AAS*HCV135
2AAS*HCV136
2AAS*HCV137

2FWS*AOV23 A, B(e)
2FWS*V12 A, B

2RHS*AOV16 A, B, C(e)
2RHS*AOV39 A, B, C(e)

2CSH*AOV108(e)

2CSL*AOV101(e)

2ICS*AOV156(e)
2ICS*AOV157(e)

2SLS*V10
2SLS*MOV5 A, B(d)

TIP PURGE CHECK VALVES

2GSN*V168
2GSN*V169
2GSN*V170

2IAS*V448
2IAS*V449

2SSC*V203
2SSC*V204

2RCS*V59 A, B
2RCS*V60 A, B
2RCS*V90 A, B-25-1

2RHS*V19(a)(c)
2RHS*V20(a)(c)
2RHS*V117(a)(c)
2RHS*V118(a)(c)

VALVE FUNCTION AND NUMBER

D. OTHER ISOLATION VALVES

SAFETY/RELIEF VALVES [(a)]

2RHS*RV20 A, B, C
2RHS*RV61 A, B, C
2RHS*RV108
2RHS*RV110
2RHS*RV139
2RHS*RV152
2RHS*RV56 A, B
2RHS*SV34 A, B
2RHS*SV62 A, B

2CSL*RV105
2CSL*RV123

2CSH*RV113
2CSH*RV114

EXCESS FLOW CHECK VALVES[(b)]
REACTOR INSTRUMENTATION LINES

2ISC*EFV1
2ISC*EFV2
2ISC*EFV3
2ISC*EFV4
2ISC*EFV5
2ISC*EFV6
2ISC*EFV7
2ISC*EFV8
2ISC*EFV10
2ISC*EFV11
2ISC*EFV13
2ISC*EFV14
2ISC*EFV15
2ISC*EFV17
2ISC*EFV18
2ISC*EFV20
2ISC*EFV21
2ISC*EFV22

TABLE 3.6.3-1 (Cont.)

VALVE FUNCTION AND NUMBER
2ISC*EFV23
2ISC*EFV24
2ISC*EFV25
2ISC*EFV26
2ISC*EFV27
2ISC*EFV28
2ISC*EFV29
2ISC*EFV30
2ISC*EFV31
2ISC*EFV32
2ISC*EFV33
2ISC*EFV34
2ISC*EFV35
2ISC*EFV36
2ISC*EFV37
2ISC*EFV38
2ISC*EFV39
2ISC*EFV40
2ISC*EFV41
2ISC*EFV42

(a) Not subject to Type C leakage tests.
(b) Subject to type A test. Type C test not required.
(c) These valves are simple check valves, located on the vacuum breaker lines for RHR safety relief valves (SRV's) discharge headers. The SRV discharge header, which terminated under pool water, has no containment isolation valves.
(d) 2SLS MOV5 A & B are globe stop check valves. These valves close upon reverse flow. The motor operator is provided to remote manually close the valve from the control room.
(e) These valves are testable check valves. They close upon reverse flow. The air operator on each valve is provided only for periodic testing of the valve.

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more vacuum breakers in one pair of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours. Restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With ~~(one)~~ ~~(the)~~ position ~~indicator~~ *indication* of any suppression chamber - drywell vacuum breaker inoperable:

1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, ~~(and)~~(or).

~~(2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by (conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter).)~~

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

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days observe ~~by verifying (both) (the)~~ position indicator~~(s)~~ OPERABLE ~~by observing expected valve movement~~ during the cycling test.

3. At least once per operating cycle ~~18 months~~ by;

   a) Verifying the opening setpoint, from the closed position, to be ~~less than or~~ equal to ~~(0.5)~~ 0.25 psid, and

   b) Verifying ~~(both)~~ (the) position indicator~~(s)~~ OPERABLE by performance of a CHANNEL CALIBRATION.

3/4.6.5 ~~SECONDARY CONTAINMENT~~ REACTOR BUILDING

REACTOR BUILDING ~~SECONDARY CONTAINMENT~~ INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 REACTOR BUILDING ~~SECONDARY CONTAINMENT~~ INTEGRITY shall be maintained.

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

ACTION:

Without REACTOR BUILDING ~~SECONDARY CONTAINMENT~~ INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore REACTOR BUILDING ~~SECONDARY CONTAINMENT~~ INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the reactor building ~~secondary containment~~, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 REACTOR BUILDING ~~SECONDARY CONTAINMENT~~ INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the reactor building ~~secondary containment~~ is less than or equal to (0.25) inches of vacuum water gauge.

b. Verifying at least once per 31 days that:

1. All reactor building ~~secondary containment~~ equipment hatches ~~and blowout panels~~ are closed and sealed.

2. (At least one) ~~(The)~~ door in each access to the reactor building ~~secondary containment~~ is closed (, except for routine entry and exit).

3. All reactor building ~~secondary containment~~ penetrations not capable of being closed by OPERABLE reactor building ~~secondary containment~~ automatic isolation dampers~~/valves~~ and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers~~/valves~~ secured in position.

c. At least once per ~~18 months:~~ *operating cycle:*

1. Verifying that one standby gas treatment subsystem will draw down the reactor building ~~secondary containment~~ to greater than or equal to (0.25) inches of vacuum water gauge in less than or equal to ~~(120)~~ 90 seconds, and

2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to (0.25) inches of vacuum water gauge in the reactor building ~~secondary containment~~ at a flow rate not exceeding *greater than (LTR) cfm and* ~~(2300)~~ 3500 CFM.

*When irradiated fuel is being handled in the reactor building ~~secondary containment~~ and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

REACTOR BUILDING
~~SECONDARY CONTAINMENT~~ AUTOMATIC ISOLATION (DAMPERS)~~(VALVES)~~

LIMITING CONDITION FOR OPERATION

3.6.5.2 The reactor building ~~secondary containment~~ ventilation system automatic isolation ~~(dampers)(valves)~~ shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

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

ACTION:

With one or more of the reactor building ~~secondary containment~~ ventilation system automatic isolation ~~(dampers)(valves)~~ shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation ~~(damper)(valve)~~ OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable ~~(damper)(valves)~~(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated ~~(damper)(valve)~~ secured in the isolation position, or

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

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

Otherwise, in Operational Condition *, suspend handling of irradiated fuel in the ~~secondary containment~~ reactor building, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.~~3~~ 4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each reactor building ~~secondary containment~~ ventilation system automatic isolation ~~(damper)(valve)~~ shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the (damper)~~(valve)~~ to service after maintenance, repair or replacement work is performed on the (damper)~~(valve)~~ or its associated actuator, control or power circuit by cycling the (damper)~~(valve)~~ through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per operating cycle ~~18 months~~ by verifying that on a containment isolation test signal each isolation (damper)~~(valve)~~ actuates to its isolation position.

c. By verifying the isolation time to be within its limit when tested pursuant to Specification 4.0.5.

*When irradiated fuel is being handled in the reactor building ~~secondary containment~~ and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.6.5.2-1

REACTOR BUILDING
~~SECONDARY CONTAINMENT~~ VENTILATION SYSTEM AUTOMATIC ISOLATION (DAMPERS) ~~(VALVES)~~

(DAMPER) ~~(VALVE)~~ FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply (Damper) ~~(Valve)~~ (2HVR*AOD1A ~~)~~ or B)	~~(5)~~ 6
~~2. Reactor Building Ventilation Supply (Damper)(Valve) ( )~~	~~(5)~~
2. ~~3.~~ Reactor Building Ventilation Exhaust (Damper) ~~(Valve)~~ (2HVR*AOD9A ~~)~~ or B)	(5)
3. ~~4.~~ Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (2HVR*AOD10A ~~)~~ or B)	(5)
4. ~~5.~~ Reactor Bldg. Ventilation Test Damper (2HVR*AOD39A or B)	~~( )~~ 10
5. ~~6.~~	~~( )~~

(Ref: Spec. NMP-2 - P413T)

CONTAINMENT SYSTEMS

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

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

   2. In Operational Condition *, *reactor building* suspend handling of irradiated fuel in the ~~secondary containment~~, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.~~3~~ *4* are not applicable.

b. With both standby gas treatment subsystems inoperable in Operational Condition *, suspend handling of irradiated fuel in the ~~secondary containment~~ *reactor building*, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.~~3~~*4*. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters ~~OPERABLE~~, *on, in order to reduce the buildup of moisture on the adsorbers and HEPA filters.*

*When irradiated fuel is being handled in the ~~secondary containment~~ *reactor building* and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel *(with fuel in vessel)*.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per operating cycle or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place Visual (external only), HEPA, and Adsorber testing acceptance criteria by removal of greater than or equal to 99.95% of the DOP or halogenated hydrocarbon gas using the test procedures of ANSI N510-1980, Sections 5, 10 and 12, with the system flow rate at 3500 cfm ± 20%:

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Appendix A, ANSI N509-1980 removes greater than or equal to 99.825% of the methyl iodide when tested in accordance with ANSI N510-1980 at 80°C and 95% RH.

3. Verifying a subsystem flow rate of 3500 cfm ± 20% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Appendix A ANSI N509-1980 removes greater than or equal to 99.825% of the methyl iodide when tested in accordance with ANSI N510-1980 at 80°C and 95% RH.

d. At least once per operating cycle by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.5 inches Water Gauge while operating the filter train at a flow rate of 3500 cfm ± 20%.

2. Verifying that the filter train starts and isolation valves open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic initiation signal.

3. Verifying that the decay heat removal isolation valves are opened.

4. Verifying that the heaters dissipate 20 ± 2.0 kw when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to ~~(99.95) (*)~~% of the DOP when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of ~~(2300)~~ 3500 cfm ± ~~10%.~~ 20%.

f. After each complete or partial replacement of a charcoal adsorber from the charcoal bed ~~bank~~ by verifying that the charcoal adsorbers remove greater than ~~(99.95) (*)~~% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of ~~(2300)~~ 3500 cfm ± ~~10%.~~ 20%.

~~(*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99%, when a filter efficiency of 90% is assumed.)~~

CONTAINMENT SYSTEMS

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

~~a. At least once per 6 months by verifying during a recombiner system functional test that the minimum (heater sheath) temperature increases to greater than or equal to (700)°F within (90) minutes. (Upon reaching (700)°F, increase the power setting to maximum power for (2) minutes and verify that the power meter reads greater than or equal to (60) kW.) (Maintain ≥ (700)°F for at least (4) hours.)~~

b. At least once per ~~18 months by:~~

   1. Performing a CHANNEL CALIBRATION of all (control room) recombiner (operating) instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to (10,000) ohms.

   (3. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials, etc.)

c. By measuring the system leakage rate:

   1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

   2. By measuring the leakage rate of the system outside of the containment isolation valves at Pa, ~~(40.4)~~ psig, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2. *40.00*

a. At least once per 12 months by verifying during a recombiner system functional test that the minimum outlet gas has temperature increases to greater than or equal to 700°F within 90 minutes. Maintain ≥1150°F for at least 4 hours.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.4 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume., *based on noncondensible gases.*

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.4 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*See Special Test Exception 3.10.5.

Delete NA

CONTAINMENT SYSTEMS

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one Reactor Building - suppression chamber vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one Reactor Building - suppression chamber vacuum breaker open, close the vacuum breaker within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by (an alternate means). Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Reactor Building - suppression chamber vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days by:

   a) Cycling each vacuum breaker through at least one (complete) (test) cycle (of full travel).

   b) Verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

2. At least once per 18 months by:

   a) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of (0.5) psid.

   b) Visual inspection.

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

Delete NA

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by:

1. Starting the blower(s) from the control room and operating the blower(s) for at least 15 minutes.
2. Cycling each (air dilution valve) through at least one complete cycle of full travel.
3. Energizing the heaters and verifying (a temperature rise of greater than or equal to ( )°F within ( ) minutes) (current of ( ) amperes ± ( )% per phase for each heater).

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

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed, each automatic valve actuates to its correct position and the blower starts.
2. Verifying that the blower develops at least the below required vacuum at the rated capacity:
   a) Inboard valves, (50)" H_2O at (100) scfm.
   b) Outboard valves, (50)" H_2O at (240) scfm.

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

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

~~(4.6.1.8.3~~ At least once per 92 days each (S) inch drywell and suppression ~~chamber purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.01) L_a when pressurized to P_a.)~~

SURVEILLANCE REQUIREMENTS (Continued)

If any (1.5) psi or (5) psi leak test results in:

1. A calculated $A/\sqrt{k}$ greater than the specified limit, or

2. A calculated $A/\sqrt{k}$ from a (1.5) psi leak test > 20% of the specified limit,

then the test schedule for subsequent tests shall be reviewed by the Commission.

If two Consecutive (1.5) psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then:

1. A (1.5) psi leak test shall be performed at least once per 9 months until two consecutive (1.5) psi leak tests result in the calculated $A/\sqrt{k}$ within the specified limits, and

2. A (5) psi leak test, performed with the second consecutive successful (1.5) psi leak test, results in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule for only (1.5) psi leak tests may be resumed.

If two consecutive (5) psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then a (5) psi leak test shall be performed at least once per 18 months until two consecutive (5) psi leak tests result in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule for only (1.5) psi leak tests may be resumed.)

Delete NA

CONTAINMENT SYSTEMS

DRYWELL-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.2.4 Differential pressure between the drywell and suppression chamber shall be greater than or equal to (____) psid.*

~~APPLICABILITY: OPERATIONAL CONDITION 1, during the period from:~~

~~a. Within 24 hours after THERMAL POWER is greater than (15%) of RATED THERMAL POWER following STARTUP, to~~

~~b. Within 24 hours prior to reducing THERMAL POWER to less than or equal to (15%) of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.~~

ACTION:

a. With one drywell-suppression chamber differential pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUT-DOWN within the following 24 hours.

b. With both drywell-suppression chamber differential pressure instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUT-DOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the drywell-suppression chamber differential pressure less than (1._) psid, restore the differential pressure to greater than or equal to (1._) psid within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.4.1 The drywell-suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.

4.6.2.4.2 At least two drywell-suppression chamber differential pressure instrumentation channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

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

c. CHANNEL CALIBRATION at least once per ~~18 months~~, *operating cycle*,

with the low differential pressure alarm setpoint ≥ ( ) psid.

*Except for up to 4 hours for required surveillance which reduces the differential pressure.

Delete NA

DRYWELL AND SUPPRESSION CHAMBER ATMOSPHERE DILUTION SYSTEM (If less than two hydrogen recombiners available)

LIMITING CONDITION FOR OPERATION

3.6.6.2 The drywell and suppression chamber atmosphere dilution (CAD) system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the drywell and/or suppression chamber CAD system inoperable, restore the CAD system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The drywell and suppression chamber atmosphere dilution system shall be demonstrated to be OPERABLE:

a. At least once per 31 days by verifying that:

 1. The system contains a minimum of (4350) gallons of liquid nitrogen, and

 2. Each valve, manual, power operated or automatic, in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by:

 1. Cycling each power operated, excluding automatic, valve in the flow path not testable during plant operation through at least one complete cycle of full travel, and

 2. Verifying that each automatic valve in the flow path actuates to its correct position on a ________ isolation test signal.

Delete NA

DRYWELL (AND SUPPRESSION CHAMBER) HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 Two independent drywell (and suppression chamber) hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell (and/or suppression chamber) hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 Each drywell (and suppression chamber) hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Starting the system from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least _______ cfm.

3/4.7 PLANT SYSTEMS

3/4.7.1 PLANT SERVICE WATER SYSTEM - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two independent plant service water system loops shall be OPERABLE with each loop comprised of:

a. Two OPERABLE plant service water pumps and service water actuation instrumentation channels shown in Table 3.7.1-2 shall be OPERABLE with their trip setpoints. A flow path capable of taking suction from Lake Ontario and transferring the water to the associated safety related equipment.

b. Service water supply header discharge water temperature of ≤ 77°F.

c. One OPERABLE Intake Deicing Heater System with Lake Ontario ≤34°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3.

ACTION:

a. With less than 2 plant service water pumps operable in one loop, restore one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With less than 2 plant service water pumps in each loop operable, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one plant service water system loop inoperable, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Also take ACTION required by Specification 3.5.2 and 3.8.1.2.

d. With less than minimum operable service water actuation instrumentation channels operable, take the ACTION required by Table 3.7.1-1.

e. With the service water supply header discharge water temperature over any 24 hour period exceeding 77°F, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. Intake heaters are required to be operable when intake water temperature is < 34°F. A minimum of 21 out of 84 heaters are required to be operable to maintain the required flow for the Service Water System. With less than one Intake Deicing Heater System operable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3/4.7.2 PLANT SERVICE WATER SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent plant service water system loops shall be OPERABLE with each loop comprised of:

a. One OPERABLE plant service water pump and service water actuation instrumentation channels shown in Table 3.7.1-2 shall be OPERABLE with their trip setpoints. A flow path capable of taking suction from Lake Ontario and transferring the water to the associated safety related equipment.

b. Service water supply header discharge water temperature of ≤ 77°F.

c. One OPERABLE Intake Deicing Heater System with Lake Ontario ≤34°F.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5.

ACTION:

a. With less than one service water pump in each loop operable, declare the associated safety related equipment inoperable and take ACTION required by Specifications 3.5.2 and 3.8.1.2.

b. With less than minimum operable service water actuation instrumentation channels operable, take the ACTION required by Table 3.7.1- 1.

c. With the service water supply header discharge temperature over any 24 hour period exceeding 77°F, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel.

d. Intake heaters are required to be operable when intake water temperature is ≤ 34°F. A minimum of 21 out of 84 heaters are required to be operable to maintain the required flow for the Service Water System. With less than one Intake Deicing Heater System operable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS - OPERATING, SHUTDOWN

4.7.1 The plant Service Water System shall be demonstrated OPERABLE:

a. By verifying the water level at the Service Water Pump intake is greater than or equal to 233.1 feet Elevation.

1. At least once per 14 days when the level is greater than 240.0 feet elevation, and

2. At least once per 12 hours when the level is less than or equal to 240.0 feet elevation.

b. Each Service Water actuation instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.7.1-1.

c. At least once per 31 days by verifying that each valve: manual, power operated or automatic, servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

d. At least once per operating cycle during shutdown, by verifying that:

Each automatic valve servicing non-safety related equipment actuates to its isolation position on an isolation test signal.

1. After a simulated test signal, each automatic valve servicing non-safety related equipment actuates to its isolation position. Each associated service water system cross connect and pump discharge valve actuate automatically to their isolation position, and that a single service water pump starts automatically in each division and that the associated pump discharge valve reopens automatically; in order to supply flow to the system safety related components.

2. Each pump runs and maintains service water pump discharge pressure equal to or greater than __* psig with each pump flow equal to or greater than ______* gpm.

e. At least once per 12 hours by recording service water supply header discharge temperature on an operating loop to be within its limit.

f. The current of the heater feeder cables shall be checked weekly at the motor control centers whenever the intake water temperature is $\leq 38^{\circ}F$.

g. The individual heaters shall be monitored once/6 months for rated heater current or as required by large deviations in the feeder checks in 4.7.1. .f. above.

h. Resistance to ground of the heaters shall be checked once/operating cycle.

* To be determined during pre-operational testing.

TABLE 3.7.1-1

SERVICE WATER SYSTEM ACTUATION INSTRUMENTATION CHANNEL OPERABILITY REQUIREMENT

TRIP FUNCTION	ID	MINIMUM INSTRUMENTS OPERABLE (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
Discharge Bay Level	2SWP*LS-30A,B	2	ALL	2
Intake Tunnel 1 & 2 Water Temperature	2SWP*TSL-64A,65A 2SWP*TSL-64B,65B	1/Div 1/Div	ALL ALL	4 4
Service Water Pump Suction Level	2SWP*LS-73A,B	2	ALL	3
Service Water Pump Discharge Flow Train "A"	2SWP*-FSL-96A,C,E	1/pump	ALL	6
Service Water Pump Discharge Flow Train "B"	2SWP*-FSL-96B,D,F	1/pump	ALL	6
Service Water Pumps Discharge Strainer Differential Pressure Train "A"	2SWP*PDT A,C,E	1/strainer	ALL	7,8
Service Water Pumps Discharge Strainer Differential Pressure Train "B"	2SWP*PDT B,F,D	1/strainer	ALL	7,8
Service Water Supply Header Discharge Water Temperature	2SWP*TE-31A,B	2	ALL	8
Service Water Inlet Pressure for EDG-2 (HPCS, Div 3)	2SWP*-PSL-95A,B	2	ALL	5
Control Building Water Flow	2SWP*-FSL-29A,B	2	ALL	1
Control Building Service Water In Temperature	2SWP*-TSL-91A,B	2	ALL	1
Control Building Service Water Out Temperature	2SWP*TE-35A,B	2	ALL	1

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition.

TABLE 3.7.1-1 (Continued)

SERVICE WATER SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 1 Start 2HVK*CHL 1A or B as applicable or establish flow manually.

ACTION 2 Locked close MOV-30A, B.

ACTION 3 Locked open MOV-77A or MOV-77B.

ACTION 4 Place intake heaters in service if lake ≤38°F.

ACTION 5 Lock close MOV-95A or B and declare EDG-2 (HPCS, Div 3) inoperable and follow Specification 3.8.1.

ACTION 6 Declare associated service water pump(s) inoperable and follow Specification 3.7.1.1 or 3.7.1.2 ACTION.

ACTION 7 Manually operate discharge strainers once per 12 hours or declare associated service water pump inoperable.

ACTION 8 Provide an alternate or temporary instrumentation.

TABLE 3.7.1-2

SERVICE WATER SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	ID	TRIP SETPOINT	ALLOWABLE VALUE
Discharge Bay Level	2SWP*LS-30A,B	≤275' Elev.	≤275' 1/4" Elev.
Intake Tunnel 1 & 2 Water Temperature	2SWP*TSL-64A,65A 2SWP*TSL-64B,65B	≥38° F	≥37° F
Service Water Pump Suction Level	2SWP*LS-73A,B	≥234' Elev.	≥233' 12 3/4" Elev.
Service Water Pump Discharge Flow Train "A"	2SWP*FSL-96A,C,E	≥3000 GPM	≥2625 GPM
Service Water Pump Discharge Flow Train "B"	2SWP*FSL-96B,D,F	≥3000 GPM	≥2625 GPM
Service Water Pumps Discharge Strainer Differential Pressure Train "A"	2SWP*PDT A,C,E	6" PSID	≤6 1/4" PSID
Service Water Pumps Discharge Strainer Differential Pressure Train "B"	2SWP*PDT B,D,F	≤6" PSID	≤6 1/4" PSID
Service Water Supply Header Discharge Water Temperature	2SWP*TE-31A,B	**	**
Service Water Inlet Pressure for EDG-2 (HPCS, Div 3)	2SWP*PSL-95A,B	≥25 psig	≥21.5 psig
Control Building Water Flow	2SWP*FSL-29A,B	≥ 225 GPM	≥209 GPM
Control Building Service Water In Temperature	2SWP*TSL-91A,B	≥60° F	≥58° F
Control Building Service Water Out Temperature	2SWP*TC-35A,B	≤72° F .	≤75.5° F

**INDICATION ONLY

TABLE 4.7.1-1

SERVICE WATER SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENT

TRIP FUNCTION	ID	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
Discharge Bay Level	2SWP*LS-30A,B	NA	A	A
Intake Tunnel 1 & 2 Water Temperature	2SWP*TSL-64A,65A 2SWP*TSL-64B,65B	W	A	A
Service Water Pump Suction Level	2SWP*LS-73A,B	NA	A	A
Service Water Pump Discharge Flow Train "A"	2SWP*FSL-96A,C,E	W	A	A
Service Water Pump Discharge Flow Train "B"	2SWP*FSL-96B,D,F	W	A	A
Service Water Pumps Discharge Strainer Differential Pressure Train "A"	2SWP*PDT A,C,E	NA	A	A
Service Water Pumps Discharge Strainer Differential Pressure Train "B"	2SWP*PDT B,D,F	NA	A	A
Service Water Supply Header Discharge Water Temperature	2SWP*TE-31A,B	S	A	A
Service Water Inlet Pressure for EDG-2 (HPCS, Div 3)	2SWP*PSL-95A,B	NA	A	A
Control Building Water Flow	2SWP*FSL-29A,B	NA	A	A
Control Building Service Water In Temperature	2SWP*TSL-91A,B	NA	A	A
Control Building Service Water Out Temperature	2SWP*TC-35A,B	NA	A	A

PLANT SYSTEMS

3/4.7.~~2~~ 3 CONTROL ROOM ~~EMERGENCY FILTRATION SYSTEM~~ *OUTDOOR AIR SPECIAL FILTER TRAIN SYSTEMS*

LIMITING CONDITION FOR OPERATION

3.7.~~2~~ 3 Two independent control room ~~emergency filtration system subsystems~~ *outdoor air special filter trains* shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or ~~3~~ 3, * with one control room ~~emergency filtration subsystem~~ *filter train* inoperable, restore the inoperable ~~subsystem~~ *filter train* to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one control room ~~emergency filtration subsystem~~ *filter train* inoperable, restore the ~~subsystem~~ *inoperable train* to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE ~~subsystem~~ *train* in the ~~(isolation)~~ mode of operation.

2. With both control room ~~emergency filtration subsystems~~ *filter trains* inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the ~~secondary containment~~ *reactor building* and operations with a potential for draining the reactor vessel.

c. The provisions of Specification ~~3.0.3~~ *3.0.4* are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.~~2~~ 3 Each control room ~~emergency filtration subsystem~~ *outdoor air special filter train* shall be demonstrated ~~OPERABLE:~~

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to (120)°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the ~~subsystem~~ *filter train* operates for at least 10 hours with the heaters ~~OPERABLE.~~ *on, in order to reduce the buildup of moisture on the adsorbers and HEPA filters.*

*When irradiated fuel is being handled in the ~~secondary containment.~~ *reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel (with fuel in the vessel.)*

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per operating cycle or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the filter trains by:

1. Verifying that the filters satisfy the in-place Visual (external only), HEPA, and Adsorber testing acceptance criteria by removal of greater than or equal to 99.95% of the DOP or halogenated hydrocarbon gas using the test procedures of ANSI N510-1980, Sections 5, 10, and 12, with the filter train flow rate at 2250 cfm ± 20%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Appendix A, ANSI N509-1980 removes greater than or equal to 99.825% of the methyl iodide when tested in accordance with ANSI N510-1980 at 80°C and 95% RH.

3. Verifying a filter train flow rate of 2250 cfm ± 20% during filter train operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Appendix A, ANSI N509-1980 removes greater than or equal to 99.825% of the methyl iodide when tested in accordance with ANSI N510-1980 at 80°C and 95% RH.

e. At least once per operating cycle by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks in less than 5.5 inches Water Gauge while operating the filter train at a flow rate of 2250 cfm ± 20%.

SURVEILLANCE REQUIREMENTS (Continued)

2. ~~3~~. Verifying that filtration on each of the below ~~(pressurization) mode~~ normal air intake actuation test signals, the ~~subsystem~~ automatically switches to the ~~(pressurization)~~ mode of operation and the control room is maintained at a positive pressure (of (1/8 inch W.G.) relative to the outside atmosphere during ~~subsystem~~ filter train operation at a flow rate less than or equal to ~~(2,000)~~ 2,250 cfm:

~~a) (Smoke detection)~~

a. ~~b)~~ Air intake radiation monitors~~, and~~

~~c) __________~~

3. ~~4~~. Verifying that the heaters dissipate ~~(7.5) ± (0.75)~~ ≥10 Kw when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% ~~(99.95)%*~~ of the DOP when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of ~~(2000)~~ 2250 cfm ± ~~10%~~ 20%.

g. After each complete or partial replacement of a charcoal adsorber from the charcoal bed ~~bank~~ by verifying that the charcoal adsorbers remove ~~(99.95)%*~~ 99.95% of halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of ~~(2000)~~ 2250 cfm ± ~~10%~~ 20%.

~~*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99%, when a filter efficiency of 90% is assumed.~~

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of ~~(automatically)~~ taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than ~~(100)~~ 150 psig.

ACTION:

~~a. With a RCIC discharge line "keep filled" (pressure) (pump failure) alarm instrumentation channel inoperable, perform Surveillance Requirement 4.7.4.a.2 at least once per 24 hours.~~

a. ~~b.~~ With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to ~~(100)~~ 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" (pressure) (pump failure) alarm instrumentation.~~

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

3. ~~4.~~ Verifying that the pump flow controller is in the correct position.

b. *When tested pursuant to Specification 4.0.5 by* ~~At least once per (31) (92) days by~~ verifying that the RCIC pump develops a flow of greater than or equal to (600) gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at (1000 + 20, - 80) psig.*

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per ~~18 months~~ *operating cycle* by:

1. Performing a system functional test which includes simulated automatic actuation (and restart) and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of greater than or equal to (600) gpm in the test flow path when steam is supplied to the turbine at a pressure of (150) ± ~~(15)~~ *1-0* psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal ~~(and on a suppression pool water level - high signal)~~.

~~4. Performing of a CHANNEL CALIBRATION of the discharge line "keep filled" (pressure)(pump failure) alarm instrumentation and verifying the:~~

~~a) High pressure setpoint to be ≤ ( ) psig.~~

~~b) Low pressure setpoint to be ≥ ( ) psig.~~

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

3/4.7.~~8~~ 5 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.~~8~~ 5 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: *OPERATIONAL CONDITIONS 1, 2 and 3 and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.*

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.5.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.~~8~~ 5 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after two ~~four~~ months but within 12 ~~10~~ months of commencing POWER OPERATION and shall include all snubbers. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.
#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of *these* visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per ~~Specifications 4.7.9.d or 4.7.9.e, as applicable.~~ *Surveillance Requirement(s) 4.7.5.d (or 4.7.5.e, as applicable).* ~~However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.~~

c. Functional Tests

During the first refueling shutdown and at least once per operating cycle thereafter during shutdown, a representative sample of at least 10% of the total of (mechanical) snubber(s) either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Surveillance Requirement 4.7.5.d (or 4.7.5.e,) an additional 10% of (that type of) (the) snubber(s) shall be functionally tested).

shall be functionally tested

~~*The value c will be arbitrarily chosen by the applicant and incorporated into the expressions for the representative sample and for the resample prior to the issuance of the Technical Specifications. The expressions are intended for use in plants with larger numbers of safety-related snubbers (>500) and provide a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptable limits. That is, the confidence level will be provided no matter what value is chosen for c. It is advised, however, that discretion be used when initially choosing the value for c because the lower the value of c (the lower the amount of snubbers in the representative sample), the higher the amount of snubbers required in the re-sample will be. To illustrate: If c = 2 and 3 snubbers are found not to meet the functional test acceptance criteria, there will be 70 snubbers in the representative sample and 31 snubbers required for testing in the re-sample; If c = 2 and 4 snubbers fail the functional test, there will be 70 snubbers in the representative sample and 62 snubbers required for testing in the re-sample; If c = 0 and 1 snubber fails the functional test, there will be 35 snubbers in the representative sample and 140 snubbers required for testing in the re-sample; If c = 0 and 2 snubbers fail the functional test, there will be 35 snubbers in the representative sample and 280 snubbers required for testing in the re-sample.~~

~~acceptance criteria selected by the operator, shall be functionally tested either in-place or in a bench test. For each~~ number ~~of snubbers above c which does not meet the functional test acceptance criteria of Specifications 4.7.9.d. or 4.7.9.e, an~~ additional sample ~~selected according to the expression~~ $35\,(1 + \frac{c}{2})\,(\frac{2}{c+1})^2\,(a - c)$ ~~shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample~~.

~~Functional testing shall continue according to the expression~~ $b\,[35(1 + \frac{c}{2})\,(\frac{2}{c+1})^2]$ ~~where b is the number of snubbers found inoperable~~ in the ~~previous re-sample, until no additional inoperable~~ snubbers are ~~found within a sample or until all snubbers have been functionally tested)~~.

The representative sample selected for functional testing* shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)
3. Snubbers within 10 feet of the discharge from a safety relief valve

~~Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*~~

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber ~~(if it is repaired and installed in another position)~~ and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

~~d. Hydraulic Snubbers Functional Test Acceptance Criteria~~

~~The hydraulic snubber functional test shall verify that:~~

~~1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.~~

~~2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.~~

d. ~~e.~~ Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. ~~Drag force shall not have increased more than 50% since the last functional test.~~

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

~~3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.~~

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. ~~f.~~ Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.~~2.1~~.3.

operating cycle
Concurrent with the first inservice visual inspection and at least once per ~~18 months~~ thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.6 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.6.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.6.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

PLANT SYSTEMS

3/4.7.7 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The fire suppression water system shall be operable with:

a. Two operable fire pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header.

b. Operable automatic initiation logic for each fire pump.

c. The provisions of Specification 3.0.4 are not applicable.

APPLICABILITY: At all times.

ACTION:

a. With one of the above required fire pumps inoperable, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a report in accordance with 6.9.2

b. With the fire suppression water system inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. Report to the NRC in accordance with 6.9.2.

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying the minimum contained water supply volume.~~

a. ~~b.~~ At least once per 31 days ~~(on a STAGGERED TEST BASIS)~~ by starting ~~(each)~~ {the} electric motor driven fire ~~suppression~~ pump and operating it for at least ~~15~~ 30 minutes on recirculation flow.}

b. ~~c.~~ At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

c. ~~d.~~ {At least once per 6 months by performance of a system flush.}

d. ~~e.~~ At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. ~~f.~~ At least once per ~~18 months~~ operating cycle by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

~~1. Verifying that each automatic valve in the flow path actuates to its correct position,~~

1. ~~2.~~ Verifying that each fire ~~suppression~~ pump develops at least {2500} gpm at ~~a system head~~ a discharge pressure of ~~(250) feet,~~ 115 psig

2. ~~3.~~ Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. ~~4.~~ Verifying that each fire ~~suppression~~ pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 125 psig.

f. ~~g.~~ At least once per 3 years by performing a flow test of the system in accordance with Chapter ~~5,~~ 8 Section ~~11~~ 16 of the Fire Protection Handbook, 15 ~~14~~th Edition, published by the National Fire Protection Association.

4.7.7.1.2 {The} ~~(Each)~~ diesel driven fire ~~suppression~~ pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

~~1. Verifying the fuel day tank contains at least ( ) gallons of fuel.~~

1. ~~2.~~ Verifying the fuel storage tank contains at least {200} gallons of fuel.

~~3. Starting the fuel transfer pump and transferring fuel from the storage tank to the day tank.~~

2. ~~4.~~ Starting the diesel driven pump from ambient conditions and operating for greater than or equal to 30 minutes on recirculation flow.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per operating cycle ~~18 months~~, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.7.1.3 {The} ~~(Each)~~ diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each ~~(pilot)~~ cell is above the plates,

2. The ~~(pilot)~~ cell specific gravity, corrected to (77)°F and full electrolyte level, is greater than or equal to (1.200),

3. The pilot ~~(pilot)~~ cell voltage is greater than or equal to 2.13 ~~(24)~~ volts, and

4. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that all cell parameters defined in 4.7.7.1.3.a are within specifications. ~~the specific gravity is appropriate for continued service of the battery.~~

c. At least once per operating cycle ~~18 months~~ by verifying that:

1. The cell, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. Cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and sprinkler systems shall be OPERABLE:

a. SPRAY AND SPRINKLER SYSTEMS

	SYSTEM NO.	BUILDING/ELEVATION
1.	W-33	Electrical Tunnel - 35
2.	W-34	Electrical Tunnel - 140
3.	W-35	Electrical Tunnel - 230
4.	W-36	Electrical Tunnel - 315
5.	W-42	Control Bldg. - El. 288'-6"
6.	W-43	Control Bldg. - El. 306'-0"
7.	W-44	Control Bldg. - El. 214'-0" to 306'-0"
8.	W-45	Control Bldg. - El. 214'-0" to 237'-0"
9.	W-45	Control Bldg. - El. 214'-0" to 306'-0"
10.	W-47	Control Bldg. - El. 214'-0" to 237'-0"
11.	W-52	Reactor Bldg. - El. 240'-0"
12.	W-53	Reactor Bldg. - El. 240'-0"
13.	W-55	Reactor Bldg. - El. 175'-0"
14.	W-57	Reactor Bldg. - El. 261'-0"
15.	W-60	Diesel Fire Pump Rm. - El. 261'-0"

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.2 (con't)

b. PRE-ACTION SYSTEMS

	SYSTEM NO.	BUILDING/ELEVATION
1.	W-48	Diesel Generator Bldg. - El. 261'-0"
2.	W-49	Diesel Generator Bldg. - El. 261'-0"
3.	W-50	Diesel Generator Bldg. - El. 261'-0"
4.	W-54	Reactor Bldg. - South El. 175'-0" to 328'-10"
5.	W-56	Reactor Bldg. - North El. 175'-0" to 328'-10"

APPLICABILITY:

Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

A. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a FIRE WATCH PATROL with backup fire suppression equipment.

B. With one or more of the above required pre-action systems inoperable, trip system wet, or within one hour establish a FIRE WATCH PATROL with backup fire suppression equipment.

C. Restore the system to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.

D. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.2 Each of the above required spray and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

Delete NA

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two independent residual heat removal service water (RHRSW) system subsystems shall be OPERABLE with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the (ultimate heat sink) and transferring the water through ~~one~~ RHR heat exchanger.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHRSW pump in each subsystem inoperable, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

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

Delete NA

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

(c. In OPERATIONAL CONDITION 4 or 5 with the RHRSW subsystem(s), which is associated with an (ECCS pump) required OPERABLE by Specification(s) (3.5.2), inoperable, declare the associated (ECCS pump) inoperable and take the ACTION required by Specification(s) (3.5.2).

d. In OPERATIONAL CONDITION 5 with the RHRSW subsystem(s), which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE:

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

b. By verifying the (   ) water level at the intake structure is greater than or equal to (   ) feet MSL;

1. At least once per 14 days when the level is greater than (   ) feet MSL, and

2. At least once per 12 hours when the level is less than or equal to (   ) feet MSL.

c. At least once per (12) months by verifying:

1. The (   ) bottom conditions in the vicinity of the intake structure.

2. The (   ) stage discharge rating curve in the unit vicinity.

d. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a _______ test signal.

Delete NA

PLANT SYSTEMS

ULTIMATE HEAT SINK (Optional)

LIMITING CONDITION FOR OPERATION

3.7.1.3 The (ultimate heat sink) shall be OPERABLE with:

a. A minimum (basin) water level at or above elevation (   ) Mean Sea Level, USGS datum, and

b. An average (basin) water temperature of less than or equal to (.)°F.

(c. (At Least) (two) OPERABLE cooling tower fans.) .

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

ACTION:

With the requirements of the above specification not satisfied:

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

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the plant service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In Operational Condition *, declare the plant service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The (ultimate heat sink) shall be determined OPERABLE at least once per:

a. 24 hours by verifying the average (basin) water temperature and water level to be within their limits.

(b. 31 days by starting each cooling tower fan from the control room and operating the fan for at least 15 minutes.)

(c. 18 months by verifying that each (plant service water) cooling tower fan starts automatically when the associated (plant service water) loop is initiated.

*When handling irradiated fuel in the secondary containment.

Delete NA

PLANT SYSTEMS

3/4.7.3 FLOOD PROTECTION (OPTIONAL*)

LIMITING CONDITION FOR OPERATION

3.7.3 Flood protection shall be provided for all safety related systems, components and structures when the water level of the (   ) exceeds (   ) Mean Sea Level USGS datum at (   ).

APPLICABILITY: At all times.

ACTION:

With the water level at (   ) above elevation (   ) Mean Sea Level USGS datum:

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

b. Initiate and complete within ( ) hours the following flood protection measures:

1. ______________________________,

2. ______________________________, and

3. ______________________________.

SURVEILLANCE REQUIREMENTS

4.7.3 The water level at (   ) shall be determined to be within the limit by:

a. Measurement at least once per 24 hours when the water level is below elevation (   ) Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation (   ) Mean Sea Level ~~USGS datum~~.

(*This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.)

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per operating cycle:

1. By performing a system functional test which includes simulated automatic actuation of the sustem, and:

   a. Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

   b. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle to verify no blockage.

d. At least once per 3 years by performing an air flow test through each open head spray (and) sprinkler header and verifying each open head spray (and) sprinkler nozzle is unobstructed.

PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following CO_2 systems shall be OPERABLE:

	ZONE NO.	BUILDING/ELEVATION
a.	336 XL	Control Bldg. - El. 261'-0"
b.	333 XL	Control Bldg. - El. 261'-0"
c.	342 XL	Control Bldg. - El. 261'-0"
d.	253 XL	Reactor Bldg. - El. 289'-0"

APPLICABILITY:

Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

A. With one or more of the above required CO_2 systems inoperable, within one hour establish a FIRE WATCH PATROL with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

B. Restore the system to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.

C. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

4.7.7.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 4' 0" above bottom of tank and pressure to be greater than 275 psig, and

b. At least once per operating cycle by verifying:

1. The system valves and associated ventilation dampers and fire door release mechanisms actuate, manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each (accessible) nozzle during a "Puff Test".

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight level and 90% of full charge pressure:

	ZONE NO.	BUILDING/ELEVATION
a.	353 SG	Control Bldg. - El. 288'-6"
b.	354 SG	Control Bldg. - El. 288'-6"
c.	362 SG	Control Bldg. - El. 288'-6"
d.	357 XG	Control Bldg. - El. 288'-6"
e.	358 XG	Control Bldg. - El. 288'-6"
f.	374 SG	Control Bldg. - El. 306'-0"
g.	375 SG	Control Bldg. - El. 306'-0"
h.	381 SG	Control Bldg. - El. 306'-0"
i.	376 XG	Control Bldg. - El. 306'-0"

APPLICABILITY:

Whenever equipment protected by the Halon systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a CONTINUOUS FIRE WATCH with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. Restore the system to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight and pressure.

c. At least once per operating cycle by:

   1. Verifying the system and associated ventilation dampers and fire door release mechanisms actuate, manually and automatically, upon receipt of a simulated actuation signal, and

   2. Performance of an air flow test through (accessible) headers and nozzles to assure no blockage.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The fire hose stations shown in Table 3.7.7.5-1 shall be OPERABLE.

APPLICABILITY:

Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.7.5-1 inoperable, route an additional fire hose of equal or greater diameter to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

b. Restore the inoperable fire hose station(s) to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.5 Each of the fire hose stations shown in Table 3.7.7.5-1 shall be demonstrated OPERABLE.

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per operation cycle by:

1. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station:

2. Removing the hose for inspection and re-racking, and

3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure at least 50 psig above the maximum fire main operating pressure.

TABLE 3.7.7.5-1

FIRE HOSE STATIONS

LOCATION~~(*)~~	ELEVATION	HOSE RACK INDENTIFICATION
Control Bldg.	214'-0"	FHR 118
Control Bldg.	214'-0"	FHR 119
Control Bldg.	237'-0"	FHR 113
Control Bldg.	237'-0"	FHR 117
Control Bldg.	250'-0"	FHR 30
Control Bldg.	261'-0"	FHR 116
Control Bldg.	261'-0"	FHR 112
Control Bldg.	288'-6"	FHR 111
Control Bldg.	288'-6"	FHR 115
Control Bldg.	306'-0"	FHR 114
Control Bldg.	306'-0"	FHR 110
Diesel Generator Bldg.	261'-0"	FHR 22
Diesel Generator Bldg.	261'-0"	FHR 33
Reactor Bldg.	175'-0"	FHR 74
Reactor Bldg.	175'-0"	FHR 90
Reactor Bldg.	175'-0"	FHR 100
Reactor Bldg.	198'-0"	FHR 102
Reactor Bldg.	198'-0"	FHR 101
Reactor Bldg.	198'-0"	FHR 103
Reactor Bldg.	215'-0"	FHR 73
Reactor Bldg.	215'-0"	FHR 89
Reactor Bldg.	215'-0"	FHR 99
Reactor Bldg.	240'-0"	FHR 72
Reactor Bldg.	240'-0"	FHR 88
Reactor Bldg.	240'-0"	FHR 98
Reactor Bldg.	261'-0"	FHR 71
Reactor Bldg.	261'-0"	FHR 79
Reactor Bldg.	261'-0"	FHR 87
Reactor Bldg.	261'-0"	FHR 94
Reactor Bldg.	289'-0"	FHR 70
Reactor Bldg.	289'-0"	FHR 78
Reactor Bldg.	289'-0"	FHR 86
Reactor Bldg.	289'-0"	FHR 93

~~(*List all fire hose stations required to ensure the OPERABILITY of safety-related equipment.)~~

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.7.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7.7.6-1 inoperable, route sufficient additional lengths of fire hose of equal or greater diameter located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) within one hour if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, route an additional hose within 24 hours.

b. Restore the inoperable hydrants(s) and/or hose house(s) to operable status within 14 days or prepare and submit a report in accordance with 6.9.2.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months, during March, April or May and during September, October or November, by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure at least 50 psig above the maximum fire main operating pressure.

   2. Replacement of all degraded gaskets in couplings.

   3. Performing a flow check of each hydrant.

PLANT SYSTEMS

TABLE 3.7.7.5-1

FIRE HOSE STATIONS (Continued)

LOCATION	ELEVATION	HOSE RACK IDENTIFICATION
Reactor Bldg.	306'-0"	FHR 69
Reactor Bldg.	306'-0"	FHR 77
Reactor Bldg.	328'-10"	FHR 68
Reactor Bldg.	328'-10"	FHR 76
Reactor Bldg.	328'-10"	FHR 85
Reactor Bldg.	328'-10"	FHR 92
Reactor Bldg.	353'-10"	FHR 67
Reactor Bldg.	353'-10"	FHR 75
Reactor Bldg.	353'-10"	FHR 94
Reactor Bldg.	353'-10"	FHR 91
Aux. Bay North	175'-0"	FHR 97
Aux. Bay North	198'-0"	FHR 104
Aux. Bay North	215'-0"	FHR 96
Aux. Bay North	240'-0"	FHR 95
Aux. Bay South	175'-0"	FHR 83
Aux. Bay South	198'-0"	FHR 82
Aux. Bay South	215'-0"	FHR 81
Aux. Bay South	240'-0"	FHR 80

TABLE 3.7.7.6-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION(*)	HYDRANT NUMBER
Yard	FH 4
Yard	FH 6
Yard	FH 7
Yard	FH 8
Yard	FH 9
Yard	FH 10
Yard	FH 11

(*List all yard fire hydrants and hydrant hose houses required to ensure the OPERABILITY of safety related equipment.)

PLANT SYSTEMS

3/4.7.8 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

listed on fire protection procedures
3.7.8 All fire barrier assemblies, including walls, floor/ceilings, cable tray enclosures and other fire barriers, separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area, and all sealing devices in fire rated assembly penetrations, including fire doors, fire dampers, cable and piping penetration seals and ventilations seals, shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour establish a CONTINUOUS FIRE WATCH on at least one side of the affected assembly(s) and/or sealing device(s) or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly(s) and/or sealing device(s) and establish an hourly fire watch patrol.

b. Restore the non-functional fire barrier penetrations to functional status within 14 days or prepare and submit a report in accordance with 6.9.2.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each of the above required fire rated assemblies and sealing devices shall be verified OPERABLE at least once per operating cycle by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly.

b. Each fire damper and associated hardware.

c. At least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.2 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying the position of each closed fire door at least once per 24 hours.

b. Verifying that doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. Verifying the position of each locked closed fire door at least once per 7 days.

d. Verifying the OPERABILITY of the fire door supervision system by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

e. Inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months.

3/4.7.~~10~~ 9 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~10~~ 9 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With the main turbine bypass system inoperable, ~~within 3 hours~~ restore the system to OPERABLE status ~~or (be in at least STARTUP within the next 8 hours) (determine MCPR to be equal to or greater than the applicable MCPR limit without bypass~~ within ~~the next~~ one hour or ~~take the ACTION required by Specification 3.2.3)~~ ~~(~~reduce THERMAL POWER to less than ~~(~~25~~)~~% of RATED THERMAL POWER within the next 4 hours~~)~~.

SURVEILLANCE REQUIREMENTS

4.7.~~10~~ 9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

~~a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and~~

a. ~~b.~~ ~~18 months~~ Operating cycle by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME ~~to be less than or equal to ( ) seconds.~~ meets the following requirements when measured from the initial movement of the main turbine stop valve or control valve:

   a) 80% of turbine bypass system capacity shall be established within 0.3 seconds.

   b) Bypass valve opening shall start within 0.1 seconds.

3/4.7.9 AREA TEMPERATURE MONITORING (Optional)

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7.9-1 shall be maintained within the limits indicated in Table 3.7.9-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.9-1:

a. For more than eight hours, in lieu of any report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas shown in Table 3.7.9-1 shall be determined to be within its limit at least once per 12 hours.

Discussion on applicability vs. nonapplicability necessary.

TABLE 3.7.9-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
a.	
b.	
c.	
d.	
e.	

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A. C. SOURCES

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A. C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. A separate fuel storuge system containing a minimum of 50,148 gallons of fuel, each for EDG-1 (Div I) and EDG-3 (Div II) and 33,879 gallons of fuel for EDG-2 (HPCS, Div III).

2. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With an offsite circuit inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.1.a.4 and/or 4.8.1.1.2.2.a.4 within 24 hours for the diesel generators associated with the inoperable offsite circuit; restore at least two offsite circuits to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With EDG-1 (Div I) or EDG-3 (Div II) of the above required A. C. electrical power sources inoperable,* demonstrate the OPERABILITY of the A. C. offsite source by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter; and Surveillance Requirement 4.8.1.1.2.1.a.4 within 24 hours; restore the diesel generator to OPERABLE status within 7 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. At the number of failures for the inoperable diesel indicated in Table 4.8-1 perform the Additional Reliability Actions prescribed in Table 4.8-2 and its attachments.

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirement 4.8.1.1.2.1.a.4.
ELECTRIC POWER SYSTEMS

**The maximum total cumulative time that the diesel generators of the onsite emergency AC power system may be in the INOPERABLE status in a given year shall be 17.5 days.

LIMITING CONDITION FOR OPERATION

ACTION: (Cont.)

c. With one offsite circuit and diesel generator EDG-1 (Div I) or EDG-3 (Div II) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter and Surveillance Requirement 4.8.1.1.2.1.a.4 within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With the diesel generator restored to OPERABLE status, follow Action Statement a. With the offsite circuit restored to OPERABLE status, follow Action Statement b.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of EDG-1 (Div I) and EDG-3 (Div II) diesel generators by performing Surveillance Requirement 4.8.1.1.2.1.a.4 within 8 hours unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, follow Action Statement a.

e. With two of the above required diesel generators EDG-1 (Div I) and EDG-3 (Div II) inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With one diesel generator unit restored, follow Action Statement b.

f. With diesel generator EDG-2 (HPCS, Div III) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter and 4.8.1.1.2.1.a.4 within 8 hours; restore the inoperable diesel generator EDG-2 (HPCS, Div III) to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1. The provisions of 3.0.3 do not apply. At the number of failures for the inoperable diesel indicated in Table 4.8-1, perform the Additional Reliability Actions prescribed in Table 4.8-2 and its attachments.

g. If ACTIONS a thru f cannot be met, take the ACTION required by Specification 3.0.4

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

4.8.1.1.2.1 EDG-1 (Div I) and EDG-3 (Div II) diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1 on a STAGGERED TEST BASIS by:

   1. Performance of a channel functional test of the day tank low level alarm.

   2. Verifying the fuel level in the fuel storage tank.

   3. Verifying that the fuel transfer pumps start and transfer fuel from the storage system to the day tank.

   4. Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 $\pm$ 416 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

      a) Manual.
      b) Simulated loss of offsite power by itself.
      c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
      d) An ESF actuation test signal by itself.

   5. Verifying the diesel generator is synchronized, loaded to greater than or equal to 4400 kw and operates with these loads for at least 60 minutes.

   6. Verifying the diesel generator is restored to standby mode.

*The diesel generator start (10 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

7. Verifying the pressure in the air start receivers to be greater than or equal to (240 psia)**

b. At least once per 14 days, rotate the diesel generators on jacking gear.

c. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks.

d. Once per 31 days by checking for and/or removing accumulated water from the fuel oil storage tanks.

e. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1) By verifying it meets the requirements specified in ASTM D975-81 prior to addition to the storage tanks by the following:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40 °C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification,

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

f. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D4057-81, and verifying that total particulate contamination is less than (10 mg/liter)* when checked in accordance with ASTM D2276-78, Method A.

**To be determined during pre-operational testing.

g. At least once per operating cycle, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2) Verifying the diesel generator capability to reject a load of greater than or equal to 1125 kw (EDG-1 Div I) and 750 kw (EDG-3 Div II) while maintaining engine speed <75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3) Verifying the diesel generator capability to reject a load of 4400 kw without tripping. The generator voltage shall not exceed 4576 volts during and following the load rejection.

4) Simulating a loss of offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the autoconnected (shutdown) loads (through the time delay relays) and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6) Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

a) Verifying deenergization of the emergency busses and loads shedding from the emergency busses.

b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected (shutdown) loads through the time delay relays and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

7) Verifying that all automatic diesel trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except engine overspeed and generator differential current.

8) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4840 kw. During the remaining 22 hours of this test, the diesel generator shall be loaded to ≥4400 kw. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.1.g 4)b) *

9) Verifying the diesel generator's capability to:

Manually

a) synchronize with the offsite power while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to its associated day tank.

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* If Surveillance Requirement 4.8.1.1.2.1.g 4)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at ≥ 4400 kw for one hour or until operating temperature has stabilized.

12) Verifying that the time delay relays are OPERABLE with the interval between each load block within ± 10% of its design interval.

13) Verifying that the following diesel generator lockout features prevent diesel generator starting:

a) Turning gear engaged

b) Emergency stop

h. After any modifications

1) which could affect diesel generator interdependence by starting all diesel generators EDG-1 (Div I); EDG-2 (HPCS, Div III); EDG-3 (Div II) simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 600 rpm for EDG-1 (Div I), EDG-3 (Div II); 900 rpm for EDG-2 (HPCS, Div III) in less than or equal to 10 seconds. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4750 kw for EDG-1 (Div I); EDG-3 (Div II) and 2850 kw for EDG-2 (HPCS, Div III)

2) which could affect air starting system by verifying that with all diesel generator air start receivers pressurized to less than or equal to 240 psig and the compressors secured, beginning from ambient conditions the diesel generator starts at least 5 consecutive times and accelerates to 600 rpm ± 3%.

i. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section II Article IWD-5000.

4.8.1.1.2.2 EDG-2 (HPCS, Div III) diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1

1. Verifying the fuel level in the day tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 870 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.
   b) Simulated loss of offsite power by itself.
   c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized and loaded to ≥ 2600 Kw and operate with this load for at least 60 minutes.

6. Verifying the diesel generator is restored to the standby mode

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 215 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. Once per 31 days by checking for and/or removing accumulated water from the fuel oil storage tank

d. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

   1) By verifying it meets the requirements specified in ASTM D975-81 prior to addition to the storage tanks by the following:

SURVEILLANCE REQUIREMENTS

d. 1) a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60 F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40 °C of greater than or equal to 1.9 centistokes,but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification,

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D 1552-79 or ASTM D2622-82.

e) At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D4057-81, and verifying that total particulate contamination is less than (10 mg/liter)* when checked in accordance with ASTM D2276-78, Method A

ASTM D2622-82.

f) At least once operating cycle, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2) Verifying the diesel generator capability to reject a load of greater than or equal to 2433* kw while maintaining engine speed ≤ 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3) Verifying the diesel generator capability to reject a load equal to 2600 kw without tripping. The generator voltage shall not exceed 5842 volts during and following the load rejection.

4) Simulating a loss of offsite power by itself, and:

a) Verifying de-energization of the emergency bus.

b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is loaded. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

*HPCS pump motor rated load of 2433 kw can be achieved by 1673 gpm flow at 1225 psi.

5) Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6) Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) Verifying de-energization of the emergency bus.

   b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads and the auto-connected emergency loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage frequency of the emergency bus shall be maintained at 4160 ± 416 volts 60 ± 1.2 Hz during this test.

7) Verifying that all automatic diesel trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except engine overspeed and generator differential current.

8) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2850 kw. During the remaining 22 hours of this test, the diesel generator shall be loaded ≥ 2600 kw (continuous rating). The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.2. f.4)b).*

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* If Surveillance Requirement 4.8.1.1.2.2. f.4)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated ≥ 2600 kw (continuous rating) for one hour or until operating temperature has stabilized.

9) Verifying the diesel generator's capability to:

   a) Manually synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

11) Verifying that the fuel transfer pump transfers fuel from fuel storage tank to the day tank.

12) Verifying that the following diesel generator lockout features prevent diesel generator starting:

   a) Engine in Maintenance mode
   b) Diesel generator lockout

g. After any modifications

   1) which could affect diesel generator interdependence by starting all diesel generators EDG-1 (Div I); EDG-2 (HPCS, Div III); EDG-3 (Div II) simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 600 rpm for EDG-1 (Div I), EDG-3 (Div II);900 rpm for EDG-2 (HPCS, DivIII) in less than or equal to 10 seconds. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4750 kw for EDG-1 (Div I); EDG-3 (Div II) and 2850 kw for EDG-2 (HPCS, Div III).

   2) which could affect air starting system by verifying that with all diesel generator air start receivers pressurized to less than or equal to 215 psig and the compressors secured, beginning from ambient conditions the diesel generator starts at least 5 consecutive times and accelerates to 870 rpm $\pm$ 3%.

h. At least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution.

Table 4.8.1

DIESEL GENERATOR TEST SCHEDULE

Number of Failure in Last 20 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
≥ 2	At least once per 7 days**

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1:108, Revison 1, August 1977, where the number of tests and failures is determined on a per diesel generator basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 20 valid tests."

**This test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one or less.

TABLE 4.8-2

ADDITIONAL RELIABILITY ACTIONS

No. of failures in last 20 valid test	No of failures in last 100 valid tests	Action
3	6	Within 14 days prepare and maintain a report for NRC audit describing the diesel generator reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	Declare the diesel generator inoperable. Perform a requalification test program for the affected diesel generator. Requalification test program requirements are indicated in Attachment 2 to this table.

ATTACHMENT 1 TO TABLE 4.8-2

REPORTING REQUIREMENT

As a minimum the Reliability Improvement Program report for NRC audit shall include:

a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

b) analysis of failures and determination of root causes of failures

c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

e) the schedule for implementation of each action from d) above

f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

Once a licensee has prepared and maintain an initial report detailing the diesel generator reliability improvement program at his site, as defined above, the licensee need prepare only a supplemental report within 14 days after each failure during a valid demand for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

In addition to the above, submit a yearly data report on the diesel generator reliability.

ATTACHMENT 2 TO TABLE 4.8-2
DIESEL GENERATOR REQUALIFICATION PROGRAM

(1) Perform seven consecutive successful demands without a failure within 30 days of diesel generator being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of diesel generator of being restored to operable status.

(2) If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator of being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of being restored to operable status.

(3) If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive successful demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

(4) Following the second failure during the requalification test program, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

(5) During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

(a) The number of failures in the last 20 valid demands is less than 5.

(b) The number of failures in the last 100 valid demands is less than 11.

(c) In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exceptior will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above).

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above the provisions of those criteria alone will prevail.

ELECTRICAL POWER SYSTEMS

A. C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A. C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator EDG-1 (Div I) or EDG-3 (Div II), and diesel generator EDG-2 (HPCS, Div III) when the HPCS system is required to be OPERABLE, with each diesel generator having:

1. A fuel storage system containing a minimum of 50,148 gallons of fuel for EDG-1 (Div I) or EDG-3 (Div II) and one fuel oil storage system containing 33,879 gallons of fuel for EDG-2 (HPCS, Div III) when HPCS is required to be operable.

2. A fuel transfer pump per required diesel system.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than required offsite circuit and/or diesel generators EDG-1 (Div I) or EDG-3 (Div II) of the above required A. C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, in OPERATIONAL CONDITION 5, with the water level less than 22'-3" above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator EDG-2 (HPCS, Div III) of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator EDG-2 (HPCS, Div III) to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2.1, except for the requirement of 4.8.1.1.2.1a.5 and 4.8.1.1.2.2a.5.

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* When handling irradiated fuel in the secondary containment.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division ~~(1)~~ I, consisting of:
   1. 125 volt battery ~~(1A)~~. 2BYS*BAT 2A
   2. one 125 volt full capacity charger.

b. Division ~~(2)~~ II, consisting of:
   1. 125 volt battery ~~(1B)~~. 2BYS*BAT 2B
   2. one 125 volt full capacity charger.

c. Division ~~(3)~~ III, consisting of:
   1. 125 volt battery ~~(1C)~~. 2BYS*BAT 2C
   2. one 125 volt full capacity charger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division ~~(1)~~ I or Division ~~(2)~~ II battery and/or charger* of the above required D.C. electrical power sources inoperable; restore the inoperable division battery to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division ~~(3)~~ III battery and/or charger* of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to 130 ~~(129)~~ volts on float charge.

* *Surveillance is not required if the inoperable charger is made operable or the installed back-up charger is placed back in service within 15 minutes following the failure.*

b. *Battery 2BYS*BAT 2A, greater than or equal to 492 amperes; battery 2BYS*BAT 2B, greater than or equal to 449 amperes (during the next 118 minutes of the test), and battery 2BYS*BAT 2C, greater than or equal to 8 amperes (during next 8 minutes of the test).*

b. At least once per 92 days and within ~~7~~ *107* days after a battery discharge with battery terminal voltage below ~~(110)~~ volts, or battery overcharge with battery terminal voltage above ~~(150)~~ *142* volts, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,
2. There is no visible corrosion at either terminals or connectors, ~~or the connection resistance of these items is less than (150 x 10-6) ohms, and~~
3. The average electrolyte temperature of ~~(a representative number)~~ ~~of~~ *one out of every five* connected cells is above (60 F).

c. At least once per ~~18 months~~ *operating cycle* by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,
2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,
3. The resistance of each ~~cell-to-cell~~ and terminal connection is ~~less than or equal to (150 x 10-6) ohms, and~~ *within 20% of the resistance readings during initial installation*
4. The battery charger will supply:
   1. For Divisions ~~(1)~~ *I* and ~~(2)~~ *II*, at least (*300*) amperes at a minimum of (*130*) volts for at least (4) hours.
   2. For Division ~~3~~ *III*, at least (*50*) amperes at a minimum of (*125*) volts for at least (4) hours.

d. At least once per ~~18 months~~ *operating cycle*, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for ~~(8)~~ *2* hours for Divisions (1) and (2) and ~~(8)~~ *2* hours for Division (3) when the battery is subjected to a battery service test, or
2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to (*105*) volts.

   a) Battery ~~(1A)~~ *2BYS*BAT 2A*, greater than or equal to (*656*) amperes; battery 2BYS*BAT 2B ~~(1B)~~, greater than or equal to (*576*) amperes; and battery 2BYS*BAT 2C ~~(1C)~~, greater than or equal to (*50*) amperes during the initial 60 seconds of the test.

   b) ~~Battery (1A)~~ ~~2BYS*BAT 2A~~, greater than or equal to (*492*) amperes; battery ~~2BYS*BAT 2B (1B), greater than or equal to (449) amperes; and battery 2BYS*BAT 2C (1C), greater than or equal to (11) amperes during the~~ remainder of the fir~~st hour of the test.~~ ~~*next 118 minutes of the test.*~~

   c) Battery ~~(1A)~~ *2BYS*BAT 2A*, greater than or equal to (*548*) amperes; battery 2BYS*BAT 2B ~~(1B)~~, greater than or equal to (*505*) amperes; and battery 2BYS*BAT 2C ~~(1C)~~, greater than or equal to (*14*) amperes during the remainder of the ~~(8)~~ *2* hour test.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At this once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per ~~18 months~~ *operating cycle* during shutdown performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 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) Average of all connected cells > (1.205)	Not more than .020 below the average of all connected cells Average of all connected cells ≥ (1.195)[b]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than (2) amperes when on float charge.

(c) May be corrected for average electrolyte temperature.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

Numbers in parentheses assume a manufacturer's recommended full charge specific gravity of 1.215.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division ~~(1)~~ *I* or Division ~~(2)~~ *II*, and, when the HPCS system is required to be OPERABLE, Division ~~(3)~~ *III* of the D.C. electrical power sources shall be OPERABLE with:

a. Division ~~(1)~~ *I* consisting of:
   1. 125 volt battery ~~(1A)~~ *2BYS*BAT 2A*
   2. *one* 125 volt full capacity charger.

b. Division ~~(2)~~ *II* consisting of:
   1. 125 volt battery ~~(1B)~~ *2BYS*BAT 2B*
   2. *one* 125 volt full capacity charger.

c. Division ~~(3)~~ *III* consisting of:
   1. 125 volt battery ~~(1C)~~ *2BYS*BAT 2C*
   2. *one* 125 volt full capacity charger.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the Division ~~(1)~~ *I* and/or Division ~~(2)~~ *II* battery and/or charger** of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With Division ~~(3)~~ *III* battery and/or charger** of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.~~3~~ *4* are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

*When handling irradiated fuel in the secondary containment.

***Surveillance is not required if the inoperable charger is made operable or the installed back-up charger is placed in service within 15 minutes following the failure.*

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized with tie breakers open between redundant buses within the unit:

a. A.C. Power Distribution

1. Division I, consisting of:

a. 4160 volt A.C.
b. 600 volt A.C. Load center/MCC's/Dist. panels..
c. 240/120 volt and 208/120 volt A.C. distribution panels.

2. Division II, consisting of:

a. 4160 volt A.C. bus.
b. 600 volt A.C. Load center/MCC's/Dist. panels.
c. 240/120 volt and 208/120 volt A.C. distribution panels.

3. Division III, consisting of:

a. 4160 volt A.C. bus.
b. 600 volt A.C. MCC
c. 240/120 volt and 208/120 volt A.C. distribution panels.

b. D.C. Power Distribution

1. Division I, consisting of 125 volt D.C. Switchgear, MCC and associated Dist. panels. Panels 2BYS*PNL-201A, 2BYS*PNL-202A, 2BY PNL-204A.

2. Division II, consisting of 125 volt D.C. Switchgear, MCC and associated Dist. panels. Panels 2BYS*PNL-201B, 2BYS*PNL-202B, 2BYS*PNL-204B.

3. Division III, consisting of 125 volt D.C. Switchgear, MCC and associated Dist. panels. Panels 2BYS*PNL-201C, 2BYS*PNL-202C, 2BYS*PNL-204C.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. For A.C. power distribution:

1. With either Division I or Division II of the above required A.C distribution system not energized, re-energize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division III of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. For D.C. power distribution:

1. With either Division ~~(1)~~ I or Division ~~(2)~~ II of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division ~~(3)~~ III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct supply breaker alignment. ~~and voltage on the busses/MCCs/panels.~~

ELECTRICAL POWER SYSTEMS

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For A.C. power distribution, Division I or Division II, and when the HPCS system Division III is required to be OPERABLE, with:

1. Division I consisting of:

   a. 4160 volt A.C. bus.
   b. 600 volt A.C. Load Center/MCC's/Dist. Panels.
   c. 240/120 volt and 208/120 volt A.C. distribution panels.

2. Division II consisting of:

   a. 4160 volt A.C. bus.
   b. 600 volt A.C. Load Centers/MCC's/Dist. Panels.
   c. 240/120 volt and 208/120 volt A.C. distribution panels.

3. Division III consisting of:

   a. 4160 volt A.C. bus.
   b. 600 volt A.C. Load Centers/MCC's/Dist. Panels.
   c. 208/120 volt and 208/120 volt A.C. distribution panels.

b. For D.C. power distribution, Division I or Division II, and when the HPCS system Division III is required to be OPERABLE, with:

1. Division I, consisting of 125 volt D.C. Switchgear, MCC and Dist. Panels.
2. Division II, consisting of 125 volt D.C. Switchgear, MCC and Dist. Panels.
3. Division III, consisting of 125 volt D.C. Switchgear, MCC and Dist. Panels.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

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* When handling irradiated fuel in the Reactor Building.

LIMITING CONDITION FOR OPERATION (Continued

ACTION:

a. For A.C. power distribution:

1. With less than Division ~~(1) and/~~ I or Division ~~(2)~~ II of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

2. With Division ~~(3)~~ III of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

b. For D.C. power distribution:

1. With less than Division ~~(1) and/~~ I or Division ~~(2)~~ II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the ~~Auxiliary Building and Enclosure Building~~ Reactor Building and operations with a potential for draining the reactor vessel.

2. With Division ~~(3)~~ III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct supply breaker alignment ~~and voltage on the busses/MCCs/panels~~.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be de-energized*:

a. Circuit numbers (2MHR-CRN3, __, __ and 2MHR-CRN4) in ~~panel~~ MCC 2NHS-C005 ( ).

b. Circuit numbers (__, LATER __ and __) in panel (LATER). *Note: Lighting, Communicate Convenience outlet circuits will be included when they are assigned.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be de-energized ~~at least once per 24 hours** by verifying that the associated circuit breakers are in the tripped condition.~~ *and marked-up prior to power ascension and following final drywell inspection.*

~~*Except~~ during entry into the drywell.

~~**Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.~~

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

1. For (~~6.9~~ 13.8 or 1.25) kV circuit breakers, de-energize the (13.8 or 1.25) kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

2. For ~~480~~ 600 volt MCC circuit breakers, remove the inoperable circuit breaker(s) from service by (racking out the breaker) within 72 hours and verify the inoperable breaker(s) ~~to be (racked out)~~ is in disconnected position at least once per 7 days thereafter.

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

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in (~~6.9~~ 13.8 or 1.25) kV circuits which have their redundant circuit breakers tripped or to ~~480~~ 600 volt circuits which have the inoperable circuit breaker ~~(racked out)~~ disconnected.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be demonstrated OPERABLE:

a. At least once per ~~18 months~~ operating cycle

1. By verifying that the medium voltage (~~4.16~~ 13.8 and 1.25 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers (of each voltage level) and performing:

a) A CHANNEL CALIBRATION of the associated protective relays, and

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed and as specified in Table 3.8.3.2-1.

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. For the lower voltage circuit breakers the nominal trip setpoint and short circuit response times are listed in Table 3.8.4.2-1. Testing of these circuit breakers shall consist of injecting a current in excess of 120% of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value for test current specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.2-1

SERVICE # AND LOCATION	TRIP SET POINT	RESPONSE TIME	SYSTEM POWERED
13.8kv CKT. BRKR.			
2EPS*SWG001	400 amp	5 cycles	2RCS-P1A
2EPS*SWG002	400 amp	5 cycles	2RCS-P1A
2NPS-SWG004	LTR	5 cycles	2RCS-MG1A
2EPS*SWG003	400 amp	5 cycles	2RCS-P1B
2EPS*SWG004	400 amp	5 cycles	2RCS-P1B
2NPS-SWG005	LTR	5 cycles	2RCS-MG1B
4.16kv CKT. BRKR.			
2NNS-SWG011-9	84 amp	5 cycles	2RCS-MG1A
2NNS-SWG013-1	84 amp	5 cycles	2RCS-MG1B
600v A.C. CKT. BRKR.			
1. Type: ITE EF			
2EHS*MCC102	30 amp	1 cycle	HCS, RHS
2EHS*MCC103	30 amp	1 cycle	MSS, CCP
2EHS*MCC302	30 amp	1 cycle	DER, DFR
2EHS*MCC303	30 amp	1 cycle	ICS
2NHS-MCC011	30 amp	1 cycle	WCS, MSS
2NHS-MCC012	30 amp	1 cycle	DFS, RCS
2NHS-MCC014	30 amp	1 cycle	CCF, MSS HCS
2EHS*MCC302	70 amp	1 cycle	RHS, WCS, DRS
2EHS*MCC303	70 amp	1 cycle	RHS, WCS, DRS
2NHS-MCC011	70 amp	1 cycle	RHS, WCS, DRS
2NHS-MCC012	70 amp	1 cycle	RHS, WCS, DRS
2. Type: ITE HE			
2NHS-MCC005-7B	100 amp	1 cycle	2MHR-CRN3
2NHS-MCC005-7C	100 amp	1 cycle	2MHC-CRN4
2NHS-MCC014	30 amp	1 cycle	RCS
FUSES			
1. Type: SHAWMUT			
ATM Midget Amp Trap Fuse	3 amp		Various Control and Instrument circuits
RK-1 Amp Trap Fuse	30 amp		
RK-5 Fault Trap Fuse	30 amp		
2. Type: BUSSMANN			
MIN 1	1 amp		Various Control and Instrument Circuits
MIN 2	2 amp		
MIN 3	3 amp		
MIN 5	5 amp		
MIN 10	10 amp		

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING (SCRAM SOLENOIDS)

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS UPS electric power monitoring channels for each inservice (EPA's) RPS ~~MG set~~ UPS or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power UPS 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~~ UPS from service.

b. With both RPS UPS electric power monitoring channels for an inservice RPS ~~MG set or alternate power supply~~ inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS ~~MG set or alternate power supply~~ UPS from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS power monitoring channels instrumentation shall be determined OPERABLE:

a. At least once per 6 month by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per ~~18 months~~ operating cycle by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage $\leq$ (132) VAC, *VAC (Bus A), *Vac (Bus B), (132v)*

2. Under-voltage $\geq$ (108) VAC, and*VAC (Bus A), *VAC (Bus B), and (108v)*

3. Under-frequency $\geq$ (57) Hz., -0+2%

* *Initial Setpoints, final values to be based on site measurements as defined in RPS Design Specification Sheet 22A3066AB.*

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels (EPA's) for each inservice RPS MG set or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS power monitoring channels instrumentation shall be determined OPERABLE:

a. At least once per 6 month by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per ~~18 months~~ *operating cycle* by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage ≤ ~~(132) VAC~~, **VAC (Bus A), ≤*VAC (Bus B), (132v)**

2. Under-voltage ≥ ~~(108) VAC, and~~ **VAC (Bus A), ≥*VAC (Bus B), and (108v)**

3. Under-frequency ≥ (57) Hz, -0 +2%

* *Initial setpoints, final values to be based on site measurements as defined in RPS Design Specification Sheet 22A3066AB.*

Delete NA

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION (Optional-Not Bypassed)

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves inoperable, bypass the inoperable thermal overload within 8 hours; restore the inoperable thermal overload to OPERABLE status within 30 days or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

Delete NA

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)(No)	SYSTEM(S) AFFECTED

Delete NA

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION (Optional-Bypassed)

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve shown in Table 3.8.4.3-1 shall be bypassed (continuously) (or) (only under accident conditions) (, as applicable,) by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed (continuously) (or) (only under accident conditions) (, as applicable,) by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3.1 The thermal overload protection for the above required valves shall be verified to be bypassed (continuously) (or) (only under accident conditions) (, as applicable,) by an OPERABLE integral bypass device by (verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing) (and) (or)(the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions):

a. At least once per (18 months for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing) (and) (or) (at least once per)(92 days for those thermal overloads which are normally in force during plant operation ~~and are~~ bypassed only under accident conditions).

b. Following maintenance on the motor starter.

(4.8.4.3.2 The thermal overload protection for the above required valves which are continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.)

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Fuel grapple position.
5. Service platform hoist fuel-loaded.

~~(6. Source range monitor countrate.)~~

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

* See Special Test Exceptions 3.10.1 and 3.10.3.##

\# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. *Audible indication in the Control Room* ~~At least one with audible indication in the control room and on the refueling floor,~~

c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

d. The "shorting links" removed from the RPS circuitry prior to ~~and during the time any control rod is withdrawn# and~~ shutdown margin demonstrations. ~~are in progress.~~

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS** and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Except movement of IRM, SRM or special movable detectors.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps:

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

3. At least once per 24 hours.

d. Verifying, ~~that the RPS circuitry "shorting links" have been removed~~ within 8 hours prior to and at least once per 12 hours during~~:~~,

~~1. The time any control rod is withdrawn,##~~ or

1. ~~2.~~ Shutdown margin demonstrations~~,~~, *that the RPS circuitry "shorting links" have been removed.*

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

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours prior to:

  1. The start of CORE ALTERATIONS.
  2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. At least once per 12 hours.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least (24) hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than (24) hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least (24) hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling ~~(platform)~~ ~~(floor)~~ personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling ~~(platform)~~ ~~(floor)~~ personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling ~~(platform)~~ ~~(floor)~~ personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of incore instrumentation and control rods with their normal drive system.

REFUELING OPERATIONS

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds (1200 ± 50) pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail *mounted auxiliary* hoists when the load exceeds ~~(500~~ *1000* ± 50) pounds.

c. Demonstrating operation of the *main and auxiliary hoist* uptravel ~~mechanical~~ stops ~~on the frame mounted and monorail hoists~~ when ~~uptravel brings the top~~ *the grapple is lower than or equal to* ~~of (active) fuel assembly to~~ (8) feet below the ~~(normal fuel storage pool) water level~~. *platform tracks.*

d. Demonstrating operation of the downtravel mechanical cutoff on the main hoist when grapple hook down travel reaches (4) inches below fuel assembly handle.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than (50 ± 10) pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds (485 ± 50) pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds (550 ± 50) pounds.

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1150 ~~(1100)~~ pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 1150 ~~(1100)~~ pounds over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

REFUELING OPERATIONS

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22' 3" ~~(23) feet~~ of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

22'3"
3.9.9 At least ~~(23)~~ feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

~~f. All fuel loading operations shall be suspended.~~

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. *All fuel loading operations shall be suspended.*

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to ~~(22)~~ feet above the top of the reactor pressure vessel flange. 22'3"

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

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

SURVEILLANCE REQUIREMENTS

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

*The shutdown cooling pump be removed from operation for up to 2 hours per 8-hour period.

LIMITING CONDITION FOR OPERATION

3.9.~~11.2~~ Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than ~~(23) feet~~ above the top of the reactor pressure vessel flange.
22' 3"

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than ~~(~~1)% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to (1)% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that the rod worth minimizer is OPERABLE per Specifications 3.1.4.1:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than ~~(~~20~~)~~% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RSCS are bypassed, verify:

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed:

1. That the rod worth minimizer is OPERABLE per Specification 3.1.4.1,

2. That movement of control rods from 75 ~~(50)~~% ROD DENSITY to the RSCS ~~(preset power level)~~ ~~(~~low power setpoint~~)~~ is ~~(blocked or limited to the single notch mode)~~ ~~(~~limited to the approved control rod withdrawal sequence during scram and friction tests~~)~~.

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The ~~("rod-out-notch-override")~~ "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2, *with the RPS circuitry shortening links removed.*

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed (5)% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than (5)% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.4 may be suspended during the performance of the Startup Test Program until either the required 100% of RATED THERMAL POWER trip tests have been completed or the reactor has operated for 120 Effective Full Power Days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The Effective Full Power Days of operation shall be verified to be less than 120, by calculation, at least once per 7 days during the Startup Test Program.

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure ~~5.1-3~~ 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

b. The provisions of Specification ~~6.9.1.9.b~~ 6.6.1.a are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous ~~Releases~~[e] Service Water Effluent A Service Water Effluent B Cooling Tower Blowdown	~~Continuous~~[e ~~f~~] M	~~M~~ ~~Composite~~[~~f~~ e] M	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	~~M~~ ~~Grab Sample~~	~~M~~	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	~~Continuous~~[~~f~~]	~~M~~ ~~Composite~~[~~f~~]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	~~Continuous~~[~~f~~] Q[e]	~~Q~~ ~~Composite~~[~~f~~] Q[e]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

[d]A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

~~[e]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.~~

~~[f]To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.~~

[e]*If the alarm setpoint of the effluent monitor, as determined by the method presented in the ODCM, is exceeded, the frequency of sampling shall be increased to daily until the condition no longer exists. Frequency of analysis shall be increased to daily for principal gamma emitters and an incident composite for H-3, gross alpha, Sr - 89, Sr - 90 and Fe -55.*

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RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.~~2~~ 3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. ~~This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141.*~~

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at ~~least once per 31 days.~~ *prior to each release of a batch of liquid waste.*

~~*Applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river sited plants this is 3 miles downstream only.~~

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RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge ~~when the projected doses due to the liquid effluent, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-3) would exceed 0.06 mrem to the total body~~ or 0.2 mrem to any organ ~~in a 31 day period.~~ *as necessary to meet the requirements of specification 3.11.1.*

APPLICABILITY: At all times.

ACTION:

Refer to Specification 3.11.1.2

~~a.~~ With radioactive ~~liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report,~~ prepare and submit to the ~~Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:~~

~~1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability.~~

2. Action(s) taken ~~to restore the inoperable equipment to OPERABLE status,~~ and

~~3. Summary description of action(s) taken to prevent a recurrence.~~

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

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases from each reactor unit to UNRESTRICTED AREAS shall be projected ~~at least once per 31 days~~ *prior to each discharge of a batch of liquid waste* in accordance with the methodology and parameters in the ODCM.

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS* ~~(Appropriate alternatives to the ACTIONS and SURVEILLANCE REQUIREMENTS below can be accepted if they provide reasonable assurance that in the event of~~ an uncontrolled release of the tanks' contents, the resulting ~~concentrations~~ would be less than the limits of 10 CFR Part 20~~, Appendix B, Table II,~~ Column 2~~,~~ at the nearest potable water supply and the nearest ~~surface~~ water supply in an UNRESTRICTED AREA.)

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. ________
b. ________
c. ________
d. Outside temporary ~~tank~~

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least ~~once per 7 days~~ weekly when radioactive materials are being added to the tank.

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

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, iodine-133 for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

b. The provisions of Specification 6.6.1.a ~~6.9.1.9.b~~ are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133 tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM. ~~by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.~~

NMP-UNIT 2 ~~PWR STS~~

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (µCi/ml)
A. Off Waste Gas Storage ~~Tank~~ System	P ~~Each Tank Grab Sample~~ M	P ~~Each Tank~~ M	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE	P Each PURGE[c] Grab Sample	P Each PURGE[c] Prior to release Each Purge	Principal Gamma Emitters[b]	1×10^{-4}
			H-3, Principal Gamma Emitters[b]	1×10^{-6}, 1×10^{-4}
C. (List other release points where gaseous effluents are discharged from the facility)	M[c,d,~~e~~] Grab Sample	M[c]	Principal Gamma Emitters[b]	1×10^{-4}
	M[d]	M[d]	H-3	1×10^{-6}
D. All Release Types as listed in A, B, C above. Main Stack Reactor/Radwaste Building Vent	Continuous[~~f~~ e]	W[~~g~~ h] Charcoal Sample	I-131	1×10^{-12}
	Continuous[~~f~~ e]	W[~~g~~ h] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[~~f~~ e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[~~f~~ e]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[~~f~~ e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATION

[b]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.~~12.~~

[c]~~Sampling and analysis shall also be performed following shutdown, startup~~, or a THERMAL POWER change ~~exceeding 15 percent of RATED THERMAL POWER~~ within one hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

[d]Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

[e]Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[f]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

[g]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

TABLE 4.11-2 (Continued)

TABLE NOTATION

c. If the Main Stack or Reactor/Radwaste building isotopic monitor is not operable, sampling and analysis shall also be performed following shutdown, start-up or when there is an alert alarm on the offgas pretreatment monitor.

d. Tritium grab samples shall be taken weekly from the Reactor/Radwaste ventilation system when fuel is offloaded until stable tritium release levels can be demonstrated.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time covered by each dose rate calculation made in accordance with specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

f. When the release rate of the Main Stack or Reactor/Radwaste building exceeds its alert alarm setpoint, the iodine and particulate device shall be removed and analyzed to determine the changes in iodine and particulate release rates. The analysis shall be done daily until the release no longer exceeds the alarm setpoint. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

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RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure ~~5.1-3~~ 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.~~2,~~ 3 a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM ~~at least once per 31 days.~~ monthly.

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RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, *IODINE-133,* TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, *iodine-133,* tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at ~~and~~ *or* beyond the SITE BOUNDARY (see Figure ~~5.1-3~~ *5.1-1*) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, *iodine-133,* tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.~~2~~*3*, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.~~3~~*4* and 3.0.~~4~~*5* are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, *iodine-133,* tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM ~~at least once per 31 days.~~ *monthly.*

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RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge ~~when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation in a 31 day period. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed 0.3 mrem to any organ in a 31-day period.~~ *as necessary to meet the requirements of specification 3.11.2.*

APPLICABILITY: At all times.

ACTION:

Refer to Specification 3.11.2.7.

~~a. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:~~

~~1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,~~

~~2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and~~

~~3. Summary description of action(s) taken to prevent a recurrence.~~

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each reactor unit to areas at *or* ~~and~~ beyond the SITE BOUNDARY shall be projected ~~at least once per 31 days~~ *monthly* in accordance with the methodology and parameters in the ODCM.

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RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE ~~(Systems designed to withstand a hydrogen explosion)~~

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen or oxygen in the waste gas holdup system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

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~~RADIOACTIVE EFFLUENTS~~

~~EXPLOSIVE GAS MIXTURE (Systems not designed to withstand a hydrogen explosion)~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.11.2.5A The concentration of hydrogen and/or oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the concentration of hydrogen and/or oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of hydrogen and/or oxygen to within the limit within 48 hours.~~

~~b. With the concentration of hydrogen and/or oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of hydrogen and/or oxygen to less than or equal to 2% by volume without delay.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.11.2.5A The concentrations of hydrogen and/or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and/or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.~~

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~~RADIOACTIVE EFFLUENTS~~

~~EXPLOSIVE GAS~~ MIXTURE (Hydrogen rich systems not designed to withstand a hydrogen explosion)

LIMITING CONDITION FOR OPERATION

3.11.2.5B The concentration of oxygen in the waste gas holdup system shall be ~~limited~~ to less than or equal to 2% by volume whenever the hydrogen ~~concentration exceeds 4% by volume.~~

~~APPL~~ICABILITY: ~~At all times.~~

~~ACTION~~:

~~a.~~ With the concentration of oxygen in the waste gas holdup system ~~greater~~ than 2% by volume but less than or equal to 4% by volume, ~~reduce~~ the oxygen concentration to the above limits within ~~48 hours.~~

~~b.~~ With the concentration of oxygen in the waste gas holdup system ~~greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions~~ of waste gases ~~to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.11.2.5B~~ The concentrations ~~of~~ hydrogen ~~and oxygen in the waste gas~~ holdup ~~system shall~~ be ~~determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors~~ required ~~OPERABLE by Table 3.3-13~~ of Specification 3.3.3.~~11.~~

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~~RADIOACTIVE EFFLUENTS~~

~~GAS STORAGE TANKS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ___ curies noble gases (considered as Xe-133).~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.~~

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RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.6 The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 μCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation.

APPLICABILITY: At all times.

ACTION: With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits; restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.6.1 The radioactivity rate of noble gases at the recombiner discharge shall be continuously monitored in accordance with Table 3.6.14-2.

4.11.2.6.2 The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge:

a. Monthly.

b. Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.

RADIOACTIVE EFFLUENTS

MARK ~~I~~ or II CONTAINMENT ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.11.2.7 VENTING or PURGING of the containment drywell shall be through the Standby Gas Treatment System, unless specification 3.11.2.1.a and b can be met without use of the Standby Gas Treatment System.

APPLICABILITY: Whenever the drywell is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell.

b. The provisions of Specifications 3.0.~~3~~4 and 3.0.~~4~~5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.7 The containment drywell shall be determined to be aligned for VENTING or PURGING through the Standby Gas Treatment System within 4 hours prior to start of and at least once per 12 hours during VENTING or PURGING of the drywell.

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RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.~~3~~4, 3.0.~~4~~5, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

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RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification ~~3.11.1.2.a~~, 3.11.1.2.b, ~~3.11.2.2.a~~, 3.11.2.2.b, ~~3.11.2.3.a~~, or 3.11.2.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with ~~Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance~~ with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4.a.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

(Additional Paragraph to 3.12.1.a)

Deviations are permitted from the required sample schedule if samples are unobtainable due to hazardous conditions, seasonal unavailability, theft, uncooperative residents, or to malfunction of automatic sampling equipment. In the event of the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

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locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification ~~6.9.1.12~~ 6.9.1, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.~~3~~4 and 3.0.~~4~~5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

~~TABLE 3.12-1~~

~~RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[a]~~

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations (DR1-DR40) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY (DR1=DR16); an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR17=DR32); the balance of the stations (DR33-DR40) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.

[a]The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sampling program. The code letters in parentheses, e.g. DR1, A1, provide one way of defining generic sample locations in this specification that can be used to identify the specific locations in the map(s) and table in the ODCM.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average groundlevel D/Q. 1 sample (A4) from the vicinity of a community having the highest calculated annual average groundlevel D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[f]	1 sample upstream (Wa1) 1 sample downstream (Wa2)	Composite sample over 1-month period[g]	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources (Wb1, Wb2), only if likely to be affected[h].	Quarterly	Gamma isotopic[e] and tritium analysis quarterly.
c. Drinking	1 sample of each of 1 to 3 (Wc1 - Wc3) of the nearest water supplies that could be affected by its discharge. 1 sample from a control location (Wc4).	Composite sample over 2-week period[g] when I-131 analysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[i] Composite for gross beta and gamma isotopic analyses[e] monthly. Composite for tritium analysis quarterly.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
d. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wd1).	Semiannually	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations (Ia1 - Ia3) within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas (Ia1 - Ia3) between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[f].	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
	1 sample from milking animals at a control location (Ia4); 16-30 km distant and in the least prevalent wind direction.		
b. Fish and Invertebrates	1 sample of each commercially and recreationally important species in vicinity of plant discharge area. (Ib1 - Ib__).	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[e] on edible portions.
	1 sample of same species in areas not influenced by plant discharge (Ib10 - Ib__).		
c. Food Products	1 sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged (Ic1 - Ic__).	At time of harvest[f]	Gamma isotopic analyses[e] on edible portion.

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~~TABLE 3.12-1 (Continued)~~

~~RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM~~

~~Exposure Pathway and/or Sample~~	~~Number of~~ Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	~~Type and Frequency of Analysis~~
~~c. Food Products (cont'd)~~	Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground ~~level D/Q if milk sampling is not performed~~ (Ic10 - Ic13).	Monthly when available	Gamma isotopic[e] and ~~I-131~~ analysis.
	~~1~~ sample of each of the similar ~~broad~~ leaf vegetation grown ~~15~~-30 km distant in the least prevalent wind direction if milk ~~sampling is not performed~~ ~~(Ic20~~ - Ic23).	Monthly when available	Gamma isotopic[e] and I-131 analysis.

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TABLE 3.12-1 (Continued)

TABLE NOTATION

[a] Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b] One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The freqency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

[c] The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind-direction criteria, other sites that provide valid background data may be substituted.

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TABLE 3.12-1 (Continued)

TABLE NOTATION

[d]Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

[e]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[f]The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

[g]A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

[h]Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[i]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[j]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

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Table 3.12-1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Limiting Condition for Operation

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type of Analysis and Frequency
Radioiodine & Particulates	Samples from 5 locations: 1) 3 samples from off-site locations in different sectors of the highest calculated site average D/Q (based on all site licensed reactors) 2) 1 sample from the vicinity of an established year round community having the highest calculated site average D/Q (based on all site licensed reactors) 3) 1 sample from a control location 10-17 miles distant and in a least prevalent wind direction[d]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent	Radioiodine Canisters analyze once/week for I-131. Particulate Samplers Gross beta radioactivity following filter change,[b] composite (by location) for gamma isotopic analysis[c] once per 3 months, (as a minimum)

Exposure Pathway and/or Sample	Number of Samples[a] and Location	Sampling and Collection Frequency[a]	Type of Analysis and Frequency
Direct Radiation[e]	32 stations with two or more dosimeters to be placed as follows: an inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each land based sector.* The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools and in 2 or 3 areas to serve as control stations.	Once per 3 months	Gamma dose once per 3 months
WATERBORNE			
Surface[f]	1) 1 sample upstream 2) 1 sample from the site's downstream cooling water intake	composite sample over 1 month period[g]	Gamma isotopic analysis[c] once/month. Composite for once per 3 months tritium analysis.
Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis[c]

* At this distance, 8 wind rose sectors are over Lake Ontario.

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type of Analysis and Frequency
INGESTION			
Milk	1) Samples from milk sampling locations in 3 locations within 3.5 miles distance having the highest calculated site average D/Q. If there are none, then 1 sample from milk sample locations in each of 3 areas 3.5-5.0 miles distant having the highest calculated site average D/Q (based on all site licensed reactors)	Twice per month, April-December (samples will be collected in January-March if I-131 is detected in November and December of the preceding year)	Gamma isotopic[c] and I-131 analysis twice per month when animals are on pasture (April-December); once/month at other times (January-March) if required
	2) 1 sample from milk sampling location at a control location (9-20 miles distant and in a least prevalent wind direction)[d]		
Fish	1) 2 samples of commercially or recreationally important species in the vicinity of a site discharge point[h]	Twice per year	Gamma isotopic analysis[c] on edible portions twice per year
	2) 1 sample each of the same species (or of a species with similar feeding habits) from an area at least 5 miles distant from the site.[d]		

Exposure Pathway and/or Sample	Number of Samples[(a)] and Locations	Sampling and Collection Frequency[(a)]	Type of Analysis and Frequency
Food Products	1) 6 samples total (utilizing at least 2 sectors) of fruits and/or vegetables will be collected from available off-site locations of highest calculated site average D/Q (based on all licensed site reactors) 2) 1 sample of each of similar vegetation grown 9-20 miles distant in a less prevalent wind direction	Once per year during harvest season	Gamma isotopic analysis of edible portions (isotopic to include I-131) Once during the harvest season

NOTES FOR TABLE 3.12-1

(a) It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and may be substituted. Actual locations (distance and directions) from the site shall be provided in the Annual Radiological Environmental Operating Report. Highest D/Q locations are based on historical meteorological data for all site licensed reactors.

(b) Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If the gross beta activity in air is greater than 10 times a historical yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(c) Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributed to the effluents from the facility.

(d) The purpose of these samples is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites, such as historical control locations which provide valid background data may be substituted.

(e) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges shall not be used for measuring direct radiation.

(f) The "upstream sample" should be taken at a distance beyond significant influence of the discharge. The "downstream sample" should be taken in an area beyond but near the mixing zone, if possible.

(g) Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g. hourly) relative to the compositing period (e.g. monthly) in order to assure obtaining a representative sample.

(h) In the event commercial or recreational important species are not available as a result of three attempts, then other species may be utilized as available.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	~~20,000*~~ 30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

~~*For drinking water~~ samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, ~~a value of 30,000 pCi/ℓ may be used.~~

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TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS~~a~~ (a)(b)

LOWER LIMIT OF DETECTION (LLD)~~b,c~~ (c)

Analysis	Water (c) (pCi/ℓ)	Airborne Particulate or Gas (pCi/m³)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	~~2000*~~ 3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

~~*If~~ no drinking water pathway exists, a value of 3000 pCi/l may be used.

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TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~11.~~

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ~~Regulatory Guide 4.13.~~ *ANSI N.545 (1975), section 4.3. Allowable exceptions to ANSI N.545 are contained in the Offsite Dose Calculation Manual (ODCM).*

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

DRAFT

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~11~~.

[d]LLD for drinking water samples. ~~If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.~~ *No drinking water pathway exists at the Nine Mile Point site under normal operating conditions due to the direction and distance of the nearest drinking water intake. Therefore, the LLD of the gamma isotopic analysis may be used.*

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of ~~8 km~~ (5 miles) the location in each of the 16 meteorological sectors of ~~the nearest~~ all milk ~~animal~~ animals, and the nearest residence. ~~and the nearest garden* of greater than 50 m² (500 ft²) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m² producing broad leaf vegetation.)~~ *In lieu of a garden census, specifications for vegetation sampling in Table 3.12-1 shall be followed, including analysis of appropriate controls.*

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated ~~dose or dose commitment~~ *D/Q value* greater than the values currently being calculated in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.~~12~~.

b. With a land use census identifying a location(s) that yields a ~~calculated dose or dose commitment~~ *D/Q value* (via the same exposure pathway) *50* ~~20~~ percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated ~~dose or dose commitment(s)~~ *D/Q*, via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.~~12~~, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.~~3~~ *4* and 3.0.~~4~~ *5* are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~11~~.

~~*Broad leaf~~ vegetation sampling of at least three different kinds of vegetation ~~may be performed at the site boundary~~ in each of two different direction sectors ~~with the highest~~ predicted D/Qs in lieu of the garden census. Specifications ~~for~~ broad leaf vegetation sampling in Table 3.12-1.4c shall be followed; ~~including analysis of control samples.~~

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission. *Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available.*

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1~~.11.~~

b. The provisions of Specifications 3.0.~~3~~4 and 3.0.~~4~~5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1~~.11.~~ *Participants in the EPA Cross Check Program may provide the EPA program code designation in lieu of providing results.*

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.4
~~3.0.3~~ This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 requires two control room emergency filtration subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours. As a further example, Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3.0.5
~~3.0.4~~ This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

3.0.3 This specification defines those conditions necessary to declare a system, subsystem, train, component or device operable for its individual specific LCO when an emergency or normal power source becomes inoperable. This presents unnecessary surveillnace and actions required in individual specifications with one power source inoperable.

APPLICABILITY

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL CONDITIONS or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

APPLICABILITY

BASES

4.0.5 This specification ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + {0.38}% delta k/k or R + {0.28}% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by (an insequence) control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

delta k/k

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than (1.06) during the limiting power transient analyzed in Section (15.4) of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than (1.06). The occurrence of scram times longer then those specified should be viewed as an indication of a systematic ~~systemic~~ problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than (3) inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than (20)% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.4) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the 20°C moderator reactor vessel. To allow for potential leakage and imperfect mixing, this concentration is increased by 25%. The required concentration is achieved by having a minimum available quantity of 4700 gallons of sodium-pentaborate solution containing a minumum of 5680 lbs. of sodium-pentaborate. This quantity of solution is a net amount which is above the pump suction. thus allowing for the portion which cannot be injected. The minimum pumping rate of 41.2 gpm per pump provides a negative reactivity insertion rate over the permissable pentaborate solution volume range which adequately compensates for the positive reactivity effects due to temperature and Xenon during shutdown. The temperature versus concentration requirement is necessary to ensure that the sodium-pentaborate remains in solution.

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

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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 peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times (1.02) is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting valuesfor APLHGR *are* ~~is~~ shown in Figures ~~(3.2.1-1, 3.2.1-2 and 3.2.1-3).~~ *3.2.1-1.*

The calculational procedure used to establish the APLHGR shown on Figure~~s~~ ~~(~~3.2.1-1, ~~3.2.1-2 and 3.2.1-3)~~ is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a ~~(TOTAL PEAKING FACTOR of (2.43) for (8 x 8) fuel)~~ {power distribution which would yield the design LHGR at RATED THERMAL POWER}. The ~~flow biased~~ simulated thermal power high ~~upscale~~ scram trip set point ~~setting~~ and flow biased simulated thermal power-upscale control rod block trip set point ~~functions of the APRM instruments~~ must be adjusted to ensure that the MCPR does not become less than 1.06 or that > 1% plastic strain does not occur in the degraded situation. The scram ~~settings~~ and rod block set points ~~settings~~ are adjusted in accordance with the formula in ~~this~~ Specification ~~when the combination of THERMAL POWER and (peak, flux) (MFLPD) indicates a (TOTAL PEAKING FACTOR greater than (2.43)) (higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition). (The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.)~~

3.2.2, whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3463 ~~(2536)~~ Mwt* which corresponds to (105)% of rated steam flow

Vessel Steam Output .... 15 ~~(11)~~ x 10^6 lbm/hr which corresponds to (105)% of rated steam flow

Vessel Steam Dome Pressure.............. (1055) psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 3.1 ~~(4.1)~~ ft^2

b. Small Breaks 0.09 ~~(0.1)~~ ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	(8 x 8)	(13.4)	(1.4)	~~(1.18)~~ 1.20

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection ~~(15._)~~ 6.3.3 of the FSAR.

*This power level meets the Appendix requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at (102)% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of {1.06}, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of {1.06}, the required minimum operating limit MCPR of Specification 3.2.3 is obtained {and presented in Figure 3.2.3-1}.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 ~~{15._}~~ that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-{1}~~{2}~~ is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors were derived using THERMAL POWER and core flow corresponding to {105}% of rated steam flow.

The K_f factors were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the {105}% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the {105}% of rated steam flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MINIMUM CRITICAL POWER RATIO (Continued)

~~The K_f factors shown in Figure 3.2.3-1 are conservative for the General Electric plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20~~ operating ~~limit MCPR used for the generic der~~ivation of K_f.

At THERMAL POWER levels less than or equal to (25)% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at (25)% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. ~~(The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a (95)% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.)~~

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November, 1975.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

3. Qualification of the One Dimensional Core Transient Model For Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program For The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

INSTRUMENTATION

BASES

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. ~~For D.C. operated valves, a (3) second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of (13) seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the (13) second diesel startup.~~ The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the ~~(13)~~ second delay. It follows that checking the valve speeds and the ~~(13)~~ second time for emergency power establishment will establish the response time for the isolation functions. ~~However, to enhance overall system reliability and to~~ monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME~~.~~
for the establishment of emergency power.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is ~~equal to or less than the drift allowance assumed for each trip in the safety analyses.~~
an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is ~~equal to or less than the drift allowance assumed for each trip in the safety analyses.~~
an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

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3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and ~~Appendix (___)~~ *Section 15.8* of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a (position switch) for each of two turbine stop valves provides input to one EOC-RPT system; a (position switch) from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than (30)% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the (time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., (190)ms. ~~less the time allotted for sensor response, i.e., (10)ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test~~ results, i.e., (83)ms, and plant ~~pre-operational test results)~~.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is ~~equal to or less than the drift allowance assumed for each trip in the safety analyses.~~ *an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value ~~also~~ contain additional margin for instrument accuracy and calibration capacity.*

Included in this time are: the time from initial valve movement to reaching the trip setpoint, the response time of the sensor, the response time of the system logic and the time allotted for breaker arc suppression.

The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis ~~that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed~~ *an allowance for instrument drift specifically allocated* for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits *and 3/4.3 Instrumentation*. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis ~~that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed~~ *an allowance for instrument drift specifically allocated* for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with ~~the recommendations of (NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980)~~. *10CFR Part 50, Appendix A. General Design Criteria 19, 41, 60, 61, 63 and 64.*

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.)

The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.)

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost (and is consistent with General Design Criteria 19 of 10 CFR 50.)

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations").

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.7.8 CHLORINE (AND AMMONIA) DETECTION SYSTEM (Optional)

The OPERABILITY of the chlorine (and ammonia) detection system ensures that an accidental chlorine (and/or ammonia) release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine (and/or ammonia), the control room emergency ventilation system will automatically be placed in the (isolation) mode of operation to provide the required protection. (The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", (February 1975) (Revision 1, January, 1977).

MONITORING INSTRUMENTATION (Continued)

~~3/4.3.7.9 CHLORIDE INTRUSION MONITORS (Optional)~~

~~The chloride intrusion monitors provide adequate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available increased sampling frequency provides adequate information for the same purpose.~~

3/4.3.7.10 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.11 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. (The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.)

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

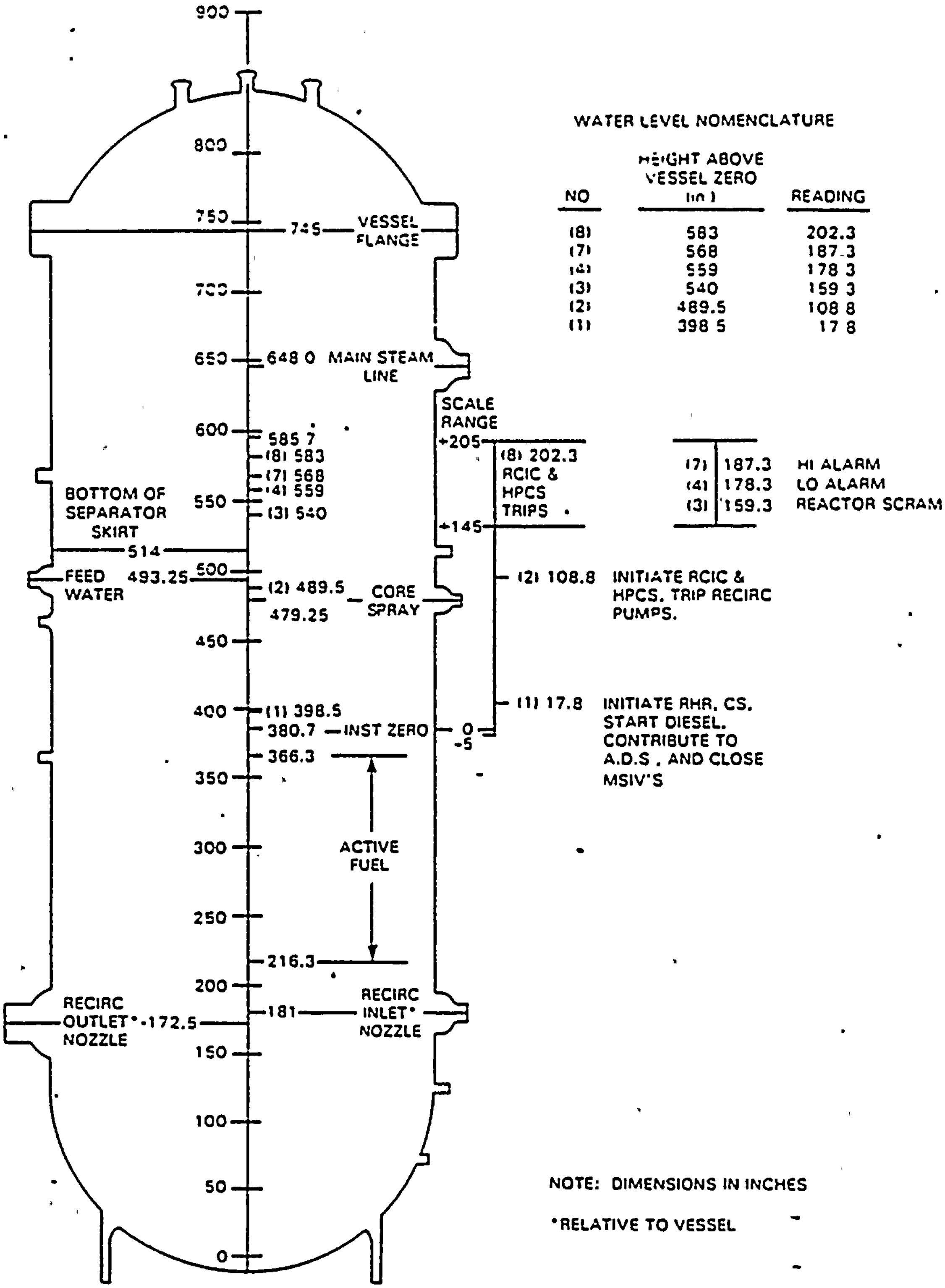
This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action of the suppression pool (and drywell) spray system and the feedwater system/main turbine trip system in the event of feedwater controller failure.

WATER LEVEL NOMENCLATURE

NO	HEIGHT ABOVE VESSEL ZERO (in.)	READING
(8)	583	202.3
(7)	568	187.3
(4)	559	178.3
(3)	540	159.3
(2)	489.5	108.8
(1)	398.5	17.8



Bases Figure B3 4 3-1. Reactor Vessel Water Level

3/4.4.2 SAFETY/RELIEF VALVES

(first paragraph under above)

The safety/relief valves operate during a postulated ATWS event to prevent the reactor coolant system being pressurized above a design allowable value of 1375 psig in accordance with the ASME Code. A total of 17 OPERABLE safety/relief valves is required to limit local pressure at active components to within ASME III allowable design values (Service Level A). All other appropriate ASME III limits are also bounded by this requirement.

3/4.4 · REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. (The limits will ensure an adequate core flow coastdown from either recirculation coop following a LOCA.)

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within (50)°F of each other prior to startup of an idle loop. The loop temperature must also be within (50)°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than ~~(145)~~° F.
100°

3/4.4.2 SAFETY/RELIEF VALVES

~~The safety valve function of the safety relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of (1325) psig in accordance with the ASME Code. A total of (13) OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. (Any combination of (6) SRVs operating in the relief mode and (7) SRVs operating in the safety mode is acceptable.)~~

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

~~The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of (7) valves and lowering the opening setpoint of (2) valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.~~

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. (These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.)

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. ~~(Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that certain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.)~~

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. ~~Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.~~

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action. *Special sampling provisions are provided for unusual outage conditions made necessary by complex maintenance or testing.*

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 SPECIFIC ACTIVITY

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

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking *with off-gas increases* phenomenon which may occur ~~following changes in THERMAL POWER.~~ Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to (20) following a postulated (steam line rupture). The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

~~Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of~~ Isotopic analysis following power changes ~~may be permissible if justified by the data obtained.~~ *which result in off-gas increases will ascertain meeting the reactor water iodine specification, and provide for off-gas monitor re-calibration.*

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

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and start-up and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10CFR50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis of compliance are more fully discussed in FSAR Chapter 5, paragraph 5.3.1.5 "Fracture Toughness".

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and Figure B 3/4.4.6-2.

The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', include an assumed shift in RT_{NDT} for the end-of life fluence.

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, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence inpredicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire data and Bases Figure B 3/4.4.6-2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. *The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.*

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (.) Edition and Addenda through ( ).

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

3/4.4.9 RESIDUAL HEAT REMOVAL

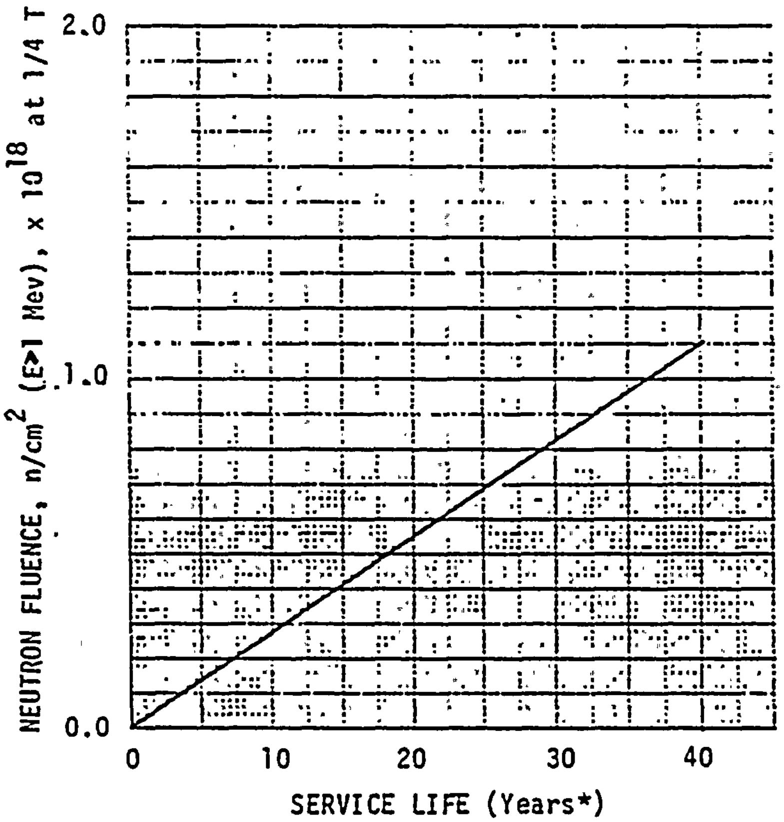
A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in ~~these~~ capsules are provided in Table 4.4.6.1.3 to assure compliance with the requirements of Appendix H to 10CFR50.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

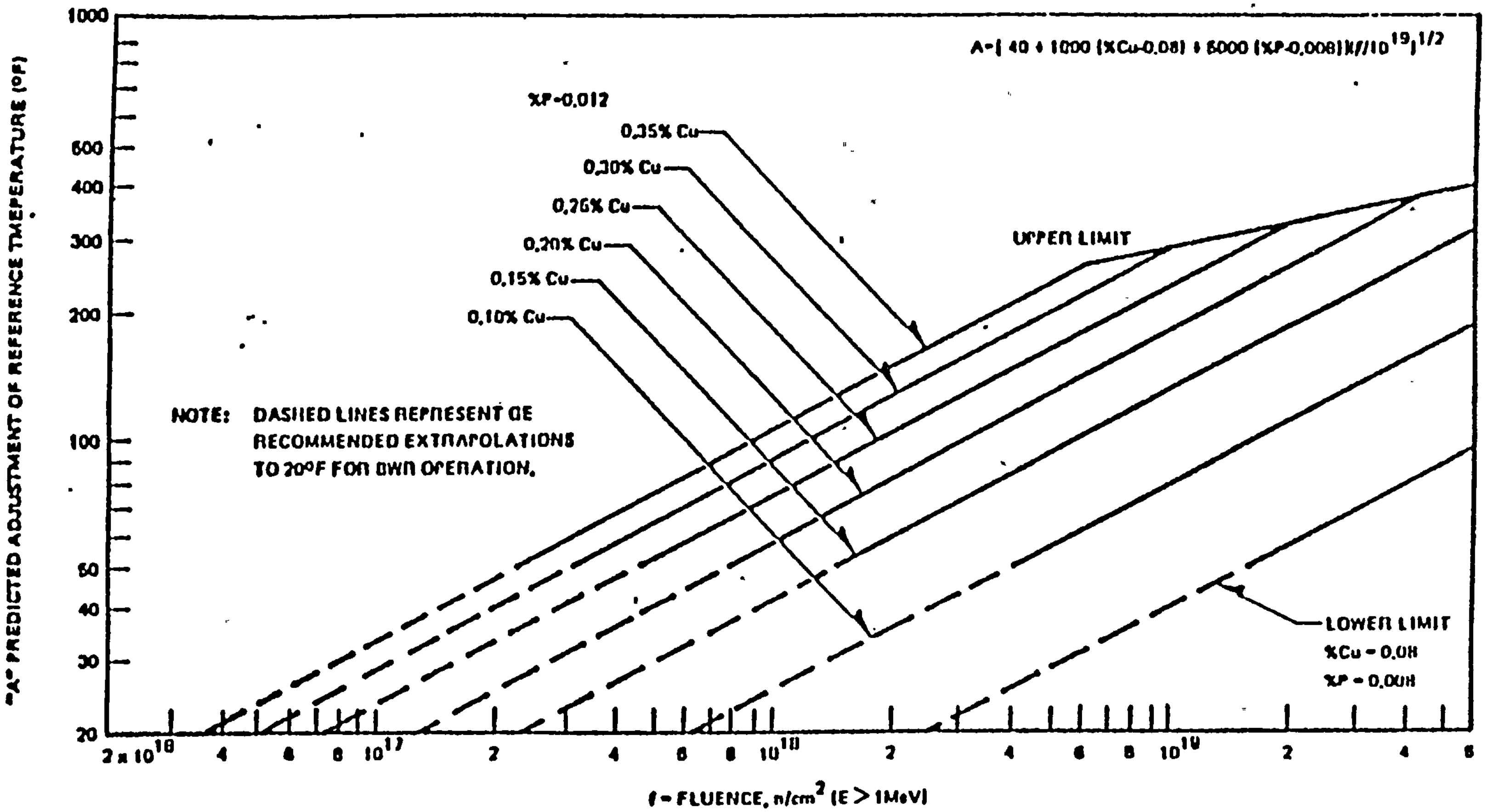
COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	RT_{NDT} °F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS



FAST NEUTRON FLUENCE (E>1 Mev)
AT 1/4T AS A FUNCTION OF SERVICE LIFE*

FIGURE B 3.4.6-1**

*AT of RATED THERMAL POWER AND 90% AVAILABILITY
.ypical figure; to be modified per Specification 4.4.6.1.4



FOR COPPER AND PHOSPHORUS CONTENTS OTHER THAN THOSE PLOTTED, USE THE EXPRESSION FOR "A" GIVEN ON THE FIGURE.

%Cu = WEIGHT PERCENT OF COPPER

%P = WEIGHT PERCENT OF PHOSPHORUS

Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content per NRC Regulatory Guide 1.99 (Adjustments <50°F are Shown in this Figure by Extrapolating the Regulatory Guide 1.99 Curves to Account for Lower BWR Fluences).

Figure B.3/4.4.6-2

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

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

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

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

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

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of (1150) psid, differential pressure between reactor vessel and HPCS suction source, to (0) psid.

ECCS-OPERATING and SHUTDOWN (Continued)

516/1550/6350

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to ~~{445/1550/4825}~~ gpm at differential pressures of {1160/1130/200} psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, {but no credit is taken in the safety analyses for the condensate storage tank water}.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, {a system for which no credit is taken in the safety analysis}, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

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

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds {100} psig. ~~even though low pressure core cooling systems provide adequate core cooling up to (350) psig.~~ *This pressure is sufficiently low that* ~~for which~~ *the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.*

ADS automatically controls (seven) selected safety-relief valves although the safety analysis only takes credit for (six) valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION ~~CHAMBER~~ POOL

The suppression ~~chamber~~ *pool* is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression ~~chamber~~ *pool* minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression ~~chamber~~ *pool* in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

3/4.6.1.2

The leakage rates specified for the main steam line isolation valves, the main steam drain line isolation valves, and the post-accident sampling system gas sample and return line block valves are used to quantify the maximum amount of primary containment atmosphere that could bypass secondary containment and leak directly to the environment after a design basis loss-of-coolant accident. This data is used to determine the radiological consequences of this accident and ensure that the resultant doses are within the limits of the General Design Criteria 19 and 10CFR100.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of General Design Criteria 19 and 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 40.0 ~~(40.4)~~ psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the (main steam line) isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

~~3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM~~

~~Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.~~

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

(Reinforced concrete containment)

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the ~~maximum~~ design pressure of ~~(59)~~ 45.0 psig in the event of a ~~(~~LOCA~~)~~ ~~(steam line break accident)~~. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

~~(Prestressed concrete containment with ungrouted tendons.)~~

~~This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of (48) psig in the event of a LOCA. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. (The tendon wire or strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)~~

~~(The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," January 1976.)~~

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of ~~(40.4)~~ 37.75 psig does not exceed the design pressure of ~~(62)~~ 45.0 psig during ~~(~~LOCA~~)~~ ~~(steam line break)~~ conditions or that the external pressure differential does not exceed the design maximum external pressure differential of ~~(~~4.7~~)~~ psid. The limit of ~~(1.75)~~ 0.75 psig for initial positive containment pressure will limit the total pressure to approximately ~~(40.4)~~ 40 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of ~~(~~340~~)~~°F during ~~(~~LOCA~~)~~ ~~(steam line break)~~ conditions and is consistent with the safety analysis.

3/4.6.1.8 ~~DRYWELL AND SUPPRESSION CHAMBER~~ PRIMARY CONTAINMENT PURGE SYSTEM

~~(The (20) inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a (LOCA or steam line break accident). Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the purge system. To provide assurance that the (20) inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.)~~

3/4.6.1.7

In addition, the maximum drywell average air temperature is also the limiting initial condition used to determine the maximum negative differential pressure acting on the drywell and suppression chamber following inadvertent actuation of the containment sprays.

3/4.6.1.8

The 14" and 12" primary containment purge supply and exhaust automatic isolation valves are required to be closed during plant operation. The two inch supply automatic isolation valves used for inerting are also required to be closed during plant operation. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the purge system. In accordance with the Standard Review plan, 6.2.4, one inboard automatic isolation valve and one outboard automatic isolation valve are provided on each purge line that penetrate the primary containment wall.

The use of the primary containment purge lines during plant operation is restricted to the following:

a. Opening the 2 inch isolation valves to manually add nitrogen to the primary containment to either maintain primary containment pressure within the range of -.5 to .75 psig or to maintain the oxygen concentration limit at 4 volume percent or less.

b. Opening the 14 inch and 12 inch exhaust isolation valves for pressure control of the drywell atmosphere.

Venting of the drywell atmosphere for pressure control will be accomplished by exhausting through piping to the Standby Gas Treatment System (SGTS) where it passes through the SGTS filters and it is released from the main stack to the environment.

The automatic valves will automatically close during one of the following conditions so that offsite radiation doses are maintained below allowable values:

a. LOCA

b. Radiation level of SGTS effluent at the main stack exceeding a predetermined value.

~~DRYWELL AND SUPPRESSION CHAMBER~~ PURGE SYSTEM (Continued)

~~(The use of the drywell and suppression chamber purge lines is restricted to the (6) inch purge supply and exhaust isolation valves since, unlike the (20) inch valves, the (6) inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the (6) inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations.")~~

(Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of those valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.) *The primary containment purge section is described in FSAR Section 9.4.2.*

~~3/4.6.1.9 PRIMARY CONTAINMENT PENETRATION PRESSURIZATION SYSTEM (Optional)~~

~~The OPERABILITY of the primary containment penetration pressurization system is required to meet the restrictions on overall containment leak rate assumed in the accident analyses. (The Surveillance Requirements for determining OPERABILITY are consistent with Appendix "J" of 10 CFR 50.)~~

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of ~~(59)~~ *45* psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~~(1020)~~ *1040* psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed ~~(62)~~ *45* psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given ~~in this specification,~~ containment pressure during the design basis accident is approximately ~~(49)~~ *40* psig which is below the design pressure of ~~(62)~~ *45* psig. Maximum water volume of ~~(148,000)~~ *154,794* ft^3 results in a downcomer submergence of ~~(4'4")~~ *11'0"* and the minimum volume of ~~(142,160)~~ *145,495* ft^3 results in a submergence approximately ~~(four)~~ *18* inches less. ~~The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bogeda Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.~~

CONTAINMENT SYSTEMS

BASES

DEPRESSURIZATION SYSTEMS (Continued)

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

At the initial ~~Under full power~~ operating conditions, blowdown to the ~~from an initial~~ suppression pool ~~chamber~~ water temperature of (90)°F results in a water temperature of approximately 140 ~~(135)~~°F immediately following blowdown which is below the (200)°F used for complete condensation via (T-quencher) ~~(ramshead)~~ devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below (200)°F during any period of relief valve operation with sonic conditions at the discharge exit for (T-quencher) ~~(ramshead)~~ devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. (Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.)

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

(~~In conjunction with the Mark I Containment Short Term Program, a plant~~ unique analysis was performed (see ~~Vermont Yankee letter dated September 13,~~ 1976) 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 of (1.7) psid and a ~~suppression chamber water level corresponding to a downcomer submergence range of (4.29) to (4.54) feet~~ will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces~~.)~~

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment (and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50). Containment isolation within the time limits specified ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are (four pairs) of valves to provide redundancy so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.

3/4.6.5 ~~SECONDARY CONTAINMENT~~ *REACTOR BUILDING*

The Reactor Building ~~Secondary containment~~ is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is in COLD SHUTDOWN or REFUELING, the drywell may be open and the Reactor Building then becomes the only containment.

Establishing and maintaining *sub-atmospheric condition* ~~a vacuum~~ in the reactor building with the standby gas treatment system once per *operating cycle* ~~18 months~~, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the ~~secondary containment.~~ *reactor building.*

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Cumulative operation of the system with the heaters OPERABLE for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of combustible gases ~~hydrogen gas~~ ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. The ~~Either~~ drywell and suppression chamber hydrogen recombiner (~~or the drywell and suppression chamber atmosphere dilution system~~) system is capable of controlling the expected hydrogen and oxygen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. (The drywell (and suppression chamber) hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.) (The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.)

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY ~~FILTRATION SYSTEM~~ *OUTDOOR AIR SPECIAL FILTER TRAIN*

The OPERABILITY of the control room ~~emergency filtration~~ *outdoor air special filter train* system ensures that ~~1) the ambient air temperature does not exceed the allowable temperature for~~ continuous duty rating for the equipment and instrumentation cooled by this ~~system and 2)~~ the control room will remain habitable for operations personnel during and following all design basis accident conditions. Cumulative operation of the system with the heaters OPERABLE for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to {5} rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3. FLOOD PROTECTION (Optional)

The requirement for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation (   ) Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds ~~(200)~~ *150* psig. ~~even though the LPCI mode of the residual heat removal (RHR) system provides adequate core cooling up to (350) psig.~~ *This pressure is sufficiently low that the RCIC system can provide adequate core cooling for events requiring the RCIC system.*

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds ~~(200)~~ *150* psig because RCIC is the primary {non-ECCS} source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

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

3/4.7.1 The Intake Deicing heater specification is to ensure adequate water is available to the service water system. In order to prove that the system is supplying adequate heat to the bar racks, a portable ammeter shall be used to check the full load current of the heaters. The current should be checked on a weekly basis. Current shall be measured for each phase at each of the four motor control center locations. If a major deviation from rated current is detected, further investigation is required to determine if an open circuit exists in the individual heater circuits.

The semiannual check of each heater will verify that the weekly tests have been adequate. The annual check of circuit meggar readings will check against long term degradation of circuit insulations.

3/4.7.~~8~~5 SNUBBERS

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 or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression $35\ (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

3/4.7.6 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.7 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2 systems, Halon systems and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the (weight and pressure) (level) of the tanks. (Level measurements are made by either a U.L. or F.M. approved method.)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.8 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.9 AREA TEMPERATURE MONITORING (Optional)

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of ( )°F.

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the (feedwater controller failure) analysis of FSAR Chapter 15.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. (The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.)

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least Division 1 or 2 of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. Division 3 supplies the high pressure core spray (HPCS) system only.

EDG-1 (Div I) *EDG-1 (Div I)* *EDG-3 (Div II)*

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator ~~(2A)~~ or ~~(2B)~~ is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator ~~(2A)~~ or ~~(2B)~~ as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period diesel generator ~~(2A)~~ or ~~(2B)~~ is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

EDG-3 (Div II) *EDG-1 (Div I)* *EDG-3 (Div II)*

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

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The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of (Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Regulatory Guide 1.137" Fuel-Oil ~~Systems~~ for Standby Diesel Generators", Revision 1, October 1979.)

A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of (Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.") ~~1975~~

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are than tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The (bypassing) ~~(or) (OPERABILITY)~~ of the motor ~~operated~~ valve thermal overload protection (continuously) ~~(or)~~ (~~during accident conditions~~) (by integral bypass devices) ensures that ~~the~~ thermal overload protection (during accident conditions) will not ~~prevent~~ safety related valves from performing their function. The surveillance requirements for demonstrating the (bypassing) (or) (OPERABILITY) ~~of the~~ thermal overload protection ~~(continuously) (and) (or) (during accident conditions)~~ are in accordance with ~~(~~Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.~~)~~

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove (99)% of the assumed (10)% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only ~~one~~ control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as ~~required during~~ REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than ~~(23)~~ 22'-3" feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and ~~(23)~~ 22'-3" feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform-certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

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BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of liquid effluents from all reactors at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." ~~Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations,~~ there is ~~reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141.~~ The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are *generally* consistent with the methodology provided in

There are no drinking water supplies that can be potentially affected by plant operations.

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Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,". Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

~~This specification applies to the release of liquid effluents from each reactor at~~ the site. For units with shared ~~radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.~~

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

~~This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.~~

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this Specification include all those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable

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assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. ~~Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM~~. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

~~This specification applies to the release of gaseous effluents from all reactors at the site.~~

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

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The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

~~This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.~~

3/4.11.2.3 DOSE - IODINE-131, *IODINE 133,* TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, *iodine-1* tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

~~This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.~~

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable".

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RADIOACTIVE EFFLUENTS

BASES

This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

~~This~~ specification applies to the ~~release of gaseous effluents from each reactor at~~ the site. For units with shared radwaste treatment systems, the ~~gaseous effluents from the shared~~ system ~~are proportioned among the units sharing that system.~~

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

~~3/4.11.2.6 GAS STORAGE TANKS~~

~~The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity that provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem in an event of 2 hours duration.~~

~~Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.~~

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.2.6 MAIN CONDENSER

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MARK II CONTAINMENT

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas.

DRAFT

RADIOACTIVE EFFLUENTS

BASES

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles ~~8 km~~ must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

DRAFT

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as ~~an~~ a ~~priori~~ (before the fact) limit representing the capability of a measurement system and not as an ~~a posteriori~~ (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

DRAFT

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information, such as from the door-to-door survey, from an aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. ~~Restricting the census to gardens of greater than 50 m² provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (25 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m².~~

3/4.12/3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

In lieu of a garden census, the significance of the exposure via the garden pathway can be evaluated by the sampling of vegetation as specified in Table 3.12-1.

A milk sampling location, as defined in Section 1, requires that at least 10 milking cows are present at a designated milk sample location. It has been found from past experience, and as a result of conferring with local farmers, that a minimum of 10 milking cows is necessary to guarantee an adequate supply of milk twice per month for analytical purposes. Locations with less than 10 milking cows are usually utilized for breeding purposes which eliminates a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 *Site*

The Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant site comprising approximately 1500 acres, is located on the shores of Lake Ontario, about seven miles northeast of Oswego, New York. An exclusion distance of nearly 4000 feet is provided between the Station and the nearest site boundary to the west, a mile to the boundary on the east, and a mile and a half to the southern site boundary.

Figure 5.1-1 is a Site Boundary Map of Nine Mile Point which allows the identification of gaseous and liquid waste release points. Figure 5.1-1 also defines the unrestricted area within the site boundary that is accessible (except for fenced areas) to member of the public.

CONFIGURATION

5.2.1 The primary containment is a (steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined prestressed concrete vessel in the shape of a truncated cone on top of a water filled suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of *284,620* ~~(221,513)~~ cubic feet. The suppression chamber has a~~n~~ *minimum* air region of ~~(166,400)~~ *192,028* cubic feet and a *minimum* water region of ~~(142,150)~~ *145,495* cubic feet.)

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure ~~(45)~~ psig.

b. Maximum internal temperature: drywell ~~(340)~~°F.
suppression pool ~~(275)°F.~~ *chamber 270°F.*

c. Maximum external pressure ~~(5)~~ *4.7* psig.

d. Maximum floor differential pressure: ~~(25)~~ psid, downward.
~~(5)~~ *10* psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the (Reactor Building, *and north and south auxiliary bays* ~~the Reactor Building recirculation fan room, the equipment access structure and a portion of the main steam tunnel)~~ and has a minimum free volume of ~~(2,550,000)~~ *3,876,630* cubic feet.

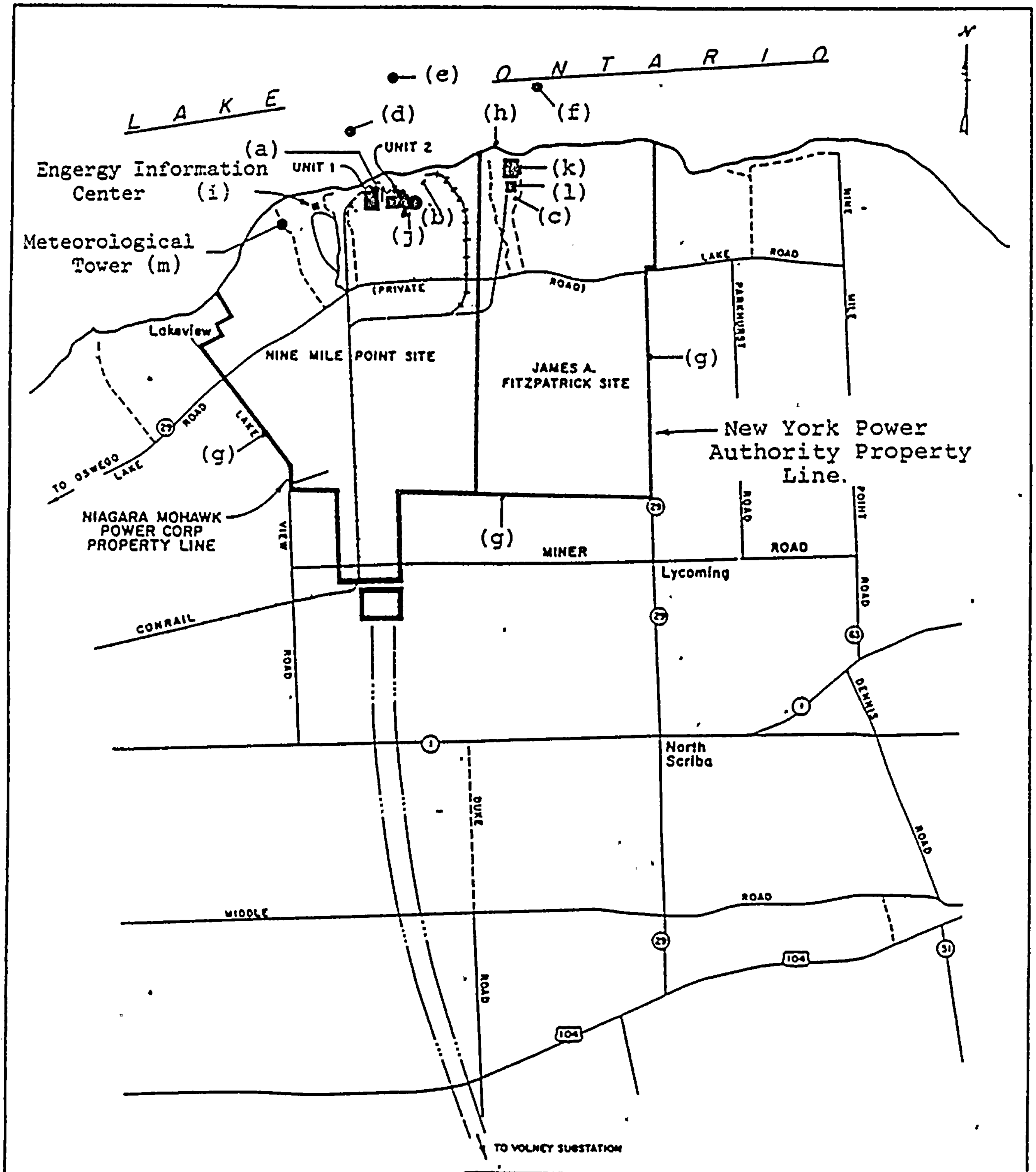

FIGURE 5.1-1

SITE BOUNDARIES

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT SITE

NOTES TO FIGURE 5.1-1

(a) NMP1 Stack (height is 350')

(b) NMP2 Stack (height is 430')

(c) JAFNPP Stack (height is 385')

(d) NMP1 Radioactive Liquid Discharge (Lake Ontario, bottom)

(e) NMP2 Radioactive Liquid Discharge (Lake Ontario, bottom)

(f) JAFNPP Radioactive Liquid Discharge (Lake Ontario, bottom)

(g) Site Boundary

(h) Lake Ontario Shoreline

(i) Energy Information Center (open to the public)

(j) NMP2 Reactor Building Vent

(k) JAFNPP Turbine and Radwaste Building Vents

(l) JAFNPP Reactor Building Vent

(m) Site Meteorological Tower

Additional Information:

- NMP2 Reactor Building Vent is located 187 feet above ground level
- JAFNPP Reactor Building and Turbine Building Vents are located 173 feet above ground level
- JAFNPP Radwaste Building vent is 112 feet above ground level

This figure shall consist of a map of the site area and provide at a minimum, the information described in Section (2.1.2) of the FSAR and meteoralgical tower location.

EXCLUSION AREA

FIGURE 5.1.1-1

This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

LOW POPULATION ZONE

FIGURE 5.1.2-1

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rod(s) clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.88 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is desinged and shall be maintained:

a) In accordance with the code requirements specified in Seciton 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b) For a pressure of:
   1. 1250 psig on the suction side of the recirculation pump.
   2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
   3. 1550 psig from the discharge shutoff valve to the jet pumps.

c) For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal T_{ave} of 533°F.

DESIGN FEATURES

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.~~X~~-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water including all calculational uncertainties and biases as described in Section 4.3 of the FSAR.

b. A nominal 6.18 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for the new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed, or procedural controls are employed to preclude optimum moderation.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertant draining of the pool below elevation 329'7".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more tahn 4050-fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

TABLE 5.7.1-1

~~COMPONENT CYCLIC OR TRANSIENT LIMITS~~

COMPONENT DESIGN CYCLES

COMPONENT	~~CYCLIC OR TRANSIENT LIMIT~~ *DESIGN CYCLES*	DESIGN CYCLE OR TRANSIENT *CONDITIONS*
Reactor	~~{120}~~ heatup and cooldown cycles	~~{~~70~~}~~°F to ~~{560}~~ *565*°F to ~~{~~70~~}~~°F
	~~{~~80~~}~~ step change cycles	Loss of feedwater heaters
	~~{180}~~ *198* reactor trip cycles	~~{~~100~~}~~% to ~~{~~0~~}~~% of RATED THERMAL POWER
	~~{40}~~ *130* hydrostatic pressure and leak tests	Pressurized to ≥ ~~{~~930~~}~~ psig and ≤ ~~{~~1250~~}~~ psig

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The General Superintendent - Nuclear Generation shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Station Shift Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President - Nuclear Generation shall be re-issued to station personnel on an annual basis.

6.2 Organization

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation, this licensed operator shall be present at the controls of the facility.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection* procedures shall be on site when fuel is in the reactor.

* The Radiation Protection qualified individual and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

<u>Facility Staff</u> (Cont'd)

e. A licensed Senior Reactor Operator shall be required in the Control Room during power operations and when the emergency plan is activated. This may be the Station Shift Supervisor - Nuclear or the Assistant Station Shift Supervisor - Nuclear during power operations. When the emergency plan is activated, the Assistant Station Shift Supervisor - Nuclear becomes the Shift Technical Advisor and the Station Shift Supervisor - Nuclear is restricted to the control room until an additional licensed Senior Reactor Operator arrives.

f. A licensed Senior Reactor Operator shall be responsible for all movement of new and irradiated fuel within the site boundary. All core alterations shall be directly supervised by a licensed senior reactor operator who has no other concurrent responsibilities during this operation. A Licensed Operator will be required to manipulate the controls of all fuel handling equipment except movement of new fuel from receipt through dry storage. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

g. A Fire Brigade of five (5) members* shall be maintained on site as defined by 5.1 at all times.

h. Administrative procedures shall be developed and implemented to limit the working hours of facility staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the facility is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications on a temporary basis, the following guidelines shall be followed:

* The Radiation Protection qualified individual and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed four hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

Facility Staff (Cont'd)

1) An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period (all excluding shift turnover time).

3) A break of at least 8-hours should be allowed between work periods (including shift turnover time).

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the General Superintendent - Nuclear Generation or designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Superintendent - Nuclear Generation or designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

Figure 6.2-1

Nine Mile Point Nuclear Station

Management Organization Chart

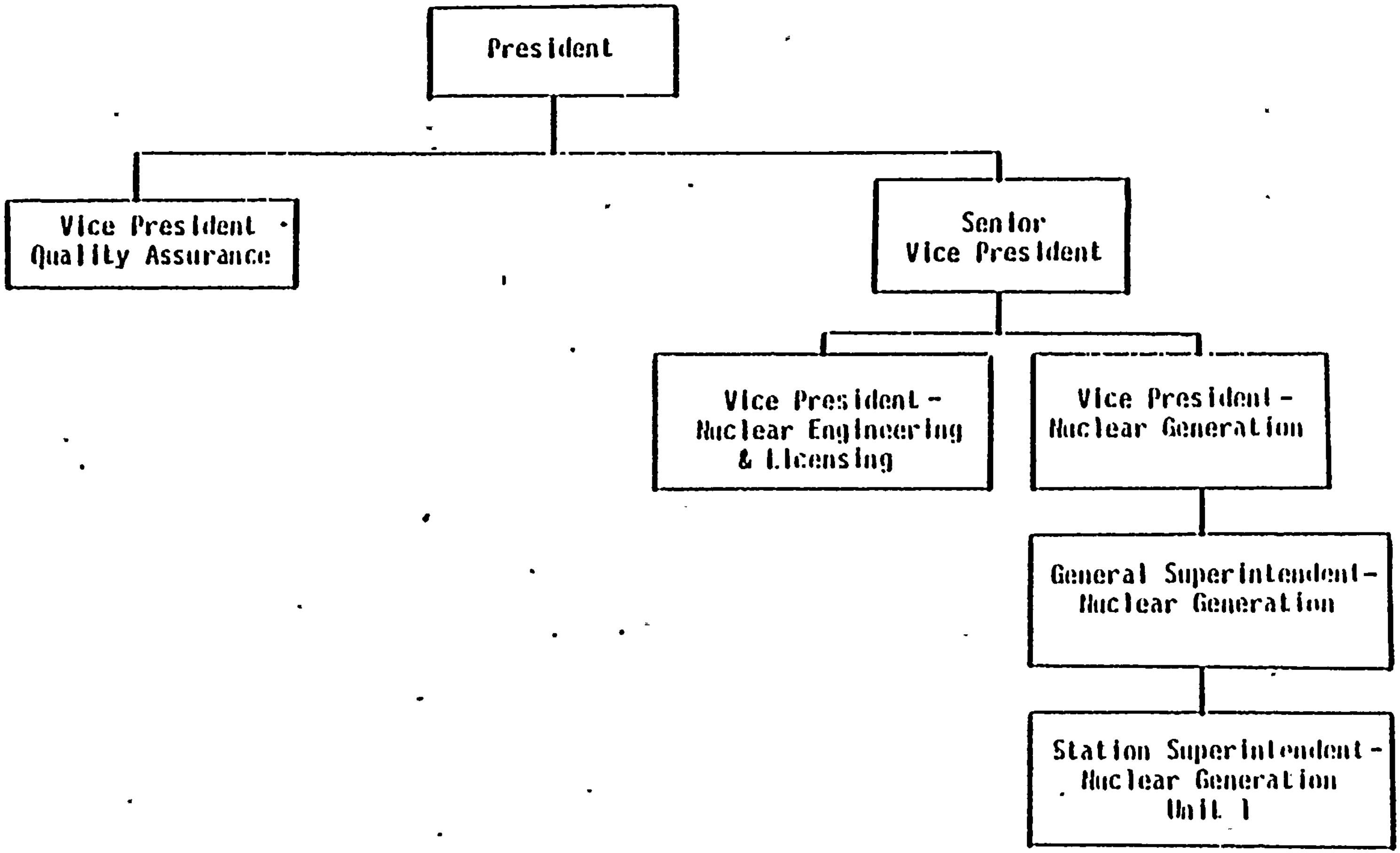

FIGURE 6.2-2
NINE MILE POINT NUCLEAR SITE
OPERATION ORGANIZATION

1,3

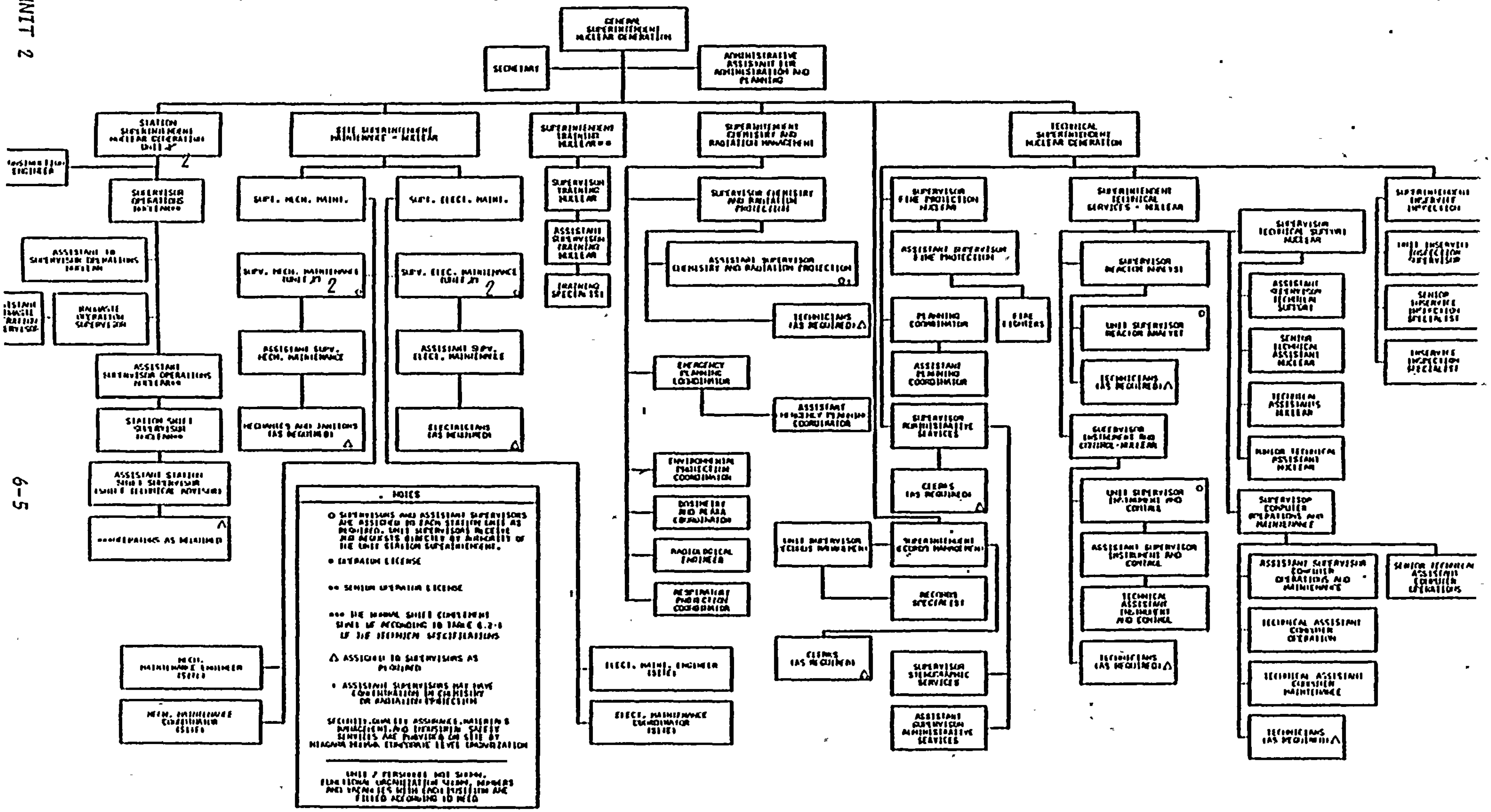

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION(1) (6)

License	Normal Operation	Shutdown Condition	Operation(3) W/O Process Computer	Reactor Startups
Senior Operator	1	1(5)	1	1
Operator	2	1	2	3
Unlicensed(2)	2	1	3	2
Asst. Station Shift Supervisor (Shift Technical Advisor Function) (Senior Operator License)(7)	1	1(4)	1	1

Notes:

(1) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, refuel outages, etc.

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) Hot shutdown condition only.

(5) An additional senior reactor operator who has no other concurrent responsibilities shall supervise all core alterations.

(6) The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 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 of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(7) The Assistant Station Shift Supervisor performs the Shift Technical Advisor function and shall hold a senior reactor operator license.

6.3 Facility Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the 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

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Superintendent-Training Nuclear and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Superintendent-Training Nuclear and Supervisor-Fire Protection, Nuclear and shall meet or exceed the requirements of Appendix R to 10CFR50.

6.5 Review and Audit

6.5.1 Site Operations Review Committee (SORC)

Function

6.5.1.1 The Site Operations Review Committee shall function to advise the General Superintendent-Nuclear Generation on all matters related to nuclear safety.

Composition

6.5.1.2 The Site Operations Review Committee shall be composed of the:

Chairman:	General Superintendent - Nuclear Generation
Member:	Station Superintendent - Nuclear Generation
Member:	Technical Superintendent - Nuclear Generation
Member:	Superintendent Technical Services - Nuclear
Member:	Site Superintendent Maintenance - Nuclear
Member:	Supervisor Instrument and Control -Nuclear
Member:	Superintendent Chemistry and Radiation Management

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SORC activities at any one time.

Meeting Frequency

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman.

Quorum

6.5.1.5 A quorum of the SORC shall consist of the Chairman and four members including alternates.

Responsibilities

6.5.1.6 The Site Operations Review Committee shall be responsible for:

a. Review of all REPORTABLE EVENTS.

b. Review of facility operations to detect potential safety hazards.

c. Performance of special reviews investigations or analyses and reports thereon as requested by the Chairman of the Safety Review and Audit Board.

d. Investigation of violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

Authority

6.5.1.7 The Site Operations Review Committee shall:

a. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6 (a) through (d) above constitutes an unreviewed safety question.

b. Provide immediate written notification to the Vice President - Nuclear Generation and Chairman of the Safety Review and Audit Board of disagreement between the SORC and the General Superintendent - Nuclear Generation; however, the General Superintendent - Nuclear Generation shall have the responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Records

6.5.1.8 The Site Operations Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Vice President - Nuclear Generation and Chairman of the Safety Review and Audit Board..

6.5.2 Technical Review and Control

Activities

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto. Approval of procedures and programs and changes thereto and their safety evaluations, shall be controlled by administrative procedures.

6.5.2.2 Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the General Superintendent-Nuclear Generation.

Activities (Cont'd)

6.5.2.3 Proposed modifications to unit structures, systems and components that affect nuclear safety shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to structures, systems and components and the safety evaluations shall be approved prior to implementation by the General Superintendent-Nuclear Generation; or the Station Superintendent-Nuclear Generation; or the Technical Superintendent-Nuclear Generation as previously designated by the General Superintendent-Nuclear Generation.

6.5.2.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the General Superintendent-Nuclear Generation to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary such review shall be performed by the appropriate designated station review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications and their safety evaluations shall be reviewed by the General Superintendent-Nuclear Generation; or by the Station Superintendent-Nuclear Generation, or the Technical Superintendent-Nuclear Generation as previously designated by the General Superintendent-Nuclear Generation.

6.5.2.6 The General Superintendent-Nuclear Generation shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President-Nuclear Generation.

6.5.2.7 The facility security program, and implementing procedures, shall be reviewed at least every 12 months. Recommended changes shall be approved by the General Superintendent-Nuclear Generation and transmitted to the Vice President-Nuclear Generation, and to the Chairman of the Safety Review and Audit Board.

6.5.2.8 The facility emergency plan, and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the General Superintendent-Nuclear Generation and transmitted to the Vice President-Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

Activities (Cont'd)

6.5.2.9 The General Superintendent-Nuclear Generation shall assure the performance of a review by a qualified individual/organization of changes to the Radiological Waste Treatment systems.

6.5.2.10 Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

6.5.2.11 Review of changes to the Process Control Program and the Offsite Dose Calculation Manual. Approval of any changes shall be made by the General Superintendent - Nuclear Generation or his designee before implementation of such changes.

6.5.2.12 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Vice President-Nuclear Generation and the Safety Review and Audit Board.

6.5.3 Safety Review and Audit Board (SRAB)

Function

6.5.3.1 The Safety Review and Audit Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices
i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

Composition

6.5.3.2 The Safety Review and Audit Board shall be composed of the:

Chairman:	Staff Engineer or Manager or Vice President
Member:	General Superintendent-Nuclear Generation
Member:	Staff Engineer - Nuclear
Member:	Staff Engineer - Mechanical or Electrical
Member:	Staff Engineer - Environmental
Member:	Consultant (See 6.5.3.4)

Alternates

6.5.3.3 Alternate members shall be appointed in writing by the SRAB Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SRAB activities at any one time.

Consultants

6.5.3.4 Consultants shall be utilized as determined by the SRAB Chairman to provide expert advice to the SRAB.

Meeting Frequency

6.5.3.5 The SRAB shall meet at least once per six months.

Quorum

6.5.3.6 A quorum of SRAB shall consist of the Chairman or his designated alternate and a majority of SRAB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.3.7 The SRAB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Site Operations Review Committee.

Audits

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the SRAB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occuring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Facility Emergency Plan and implementing procedures at least once every 12 months.

f. The Facility Security Plan and implementing procedures at least once every 12 months.

g. The Facility Fire Protection Program and implementing procedures at least once per two years.

h. Any other area of facility operation considered appropriate by the SRAB, the Vice President - Nuclear Generation or the Vice President - Nuclear Engineering and Licensing.

i. The radiological environmental monitoring program and the results thereof at least once per 12 months.

j. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

k. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

Authority

6.5.3.9 The SRAB shall report to and advise the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing on those areas of responsibility specified in Section 6.5.3.7 and 6.5.3.8.

Records

6.5.3.10 Records of SRAB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRAB meeting shall be prepared, approved and forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 30 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 e,f,g and h above, shall be prepared, approved and forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 90 days following completion of the review.

6.6 Reportable Event Action

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.72 and 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Vice President - Nuclear Generation and to the SRAB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRAB and the Vice President - Nuclear Generation within 10 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved by the General Superintendent-Nuclear Generation or designee prior to implementation and periodically as set forth in each document.

6.8 Procedures (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the General Superintendent-Nuclear Generation or designee within 14 days of implementation.

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 Director of Inspection and Enforcement Regional Office I, King of Prussia, Pennsylvania 19406, unless otherwise noted.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1 Routine Reports (Cont'd)

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Occupational Exposure Report. A tabulation shall be submitted on an annual basis which includes the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. 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 shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience including documentation of challenges to the safety relief valves or safety valves, shall be submitted on a monthly basis, which will include a narrative of operating experience, to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of I&E, no later than the 15th of each month following the calendar month covered by the report.

1/ This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

6.9.1 Routine Reports (Continued)

d. Annual Radiological Environmental Operating Report*.

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1, 1985.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls as appropriate, and with environmental surveillance reports from the previous 5 years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.6.22.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.6.21; discussion of all deviations from the sampling schedule of Table 3.6.20-1; and discussion of all analyses in which the LLD required in Table 4.6.20-1 was not achievable.

* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.9.1 Routine Reports (cont'd)

e. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin on January 1, 1985.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the Offsite Dose Calculation Manual.

6.9.1 Routine Reports (cont'd)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and,

f. Solidifcation agent or absorbent (e.g., cement)

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.6.20.

Changes to the Process Control Program (PCP) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

6.9.1 Routine Reports (cont'd)

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

6.9.2 Fire Protection Program Reports

a. Submit a special report to the appropriate Regional Office as follows:

- Notify the Director of the appropriate Regional Office by telephone within 24 hours.
- Confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and
- Follow-up in writing within 14 days after the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to an operable status.

b. Submit a special report to the Director of the appropriate Regional Office within 30 days following the event outlining the plans and procedures to be used to restore the inoperable equipment to an operable status.

6.9.3 Special Reports

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.4.6.14 (12 months)

b. Safety Class 1 Inservice Inspection, Specification (3.4.8.a) (Three months)

c. Safety Class 2 Inservice Inspections, Specification (3.4.8.b) (Three months)

d. Safety Class 3 Inservice Inspections; Specification (3.4.8.c) (Three months)

e. Primary Containment Leakage Testing, Specification 4.6.1.2 a,b,c (Three months)

f. Secondary Containment Leakage Testing, Specification 4.6.5.1.c (Three months)

g. Sealed Source Leakage In Excess Of Limits, Specification 4.7.6.3 (Three months)

h. Calculate Dose from Liquid Effluent in Excess of Limits, Specification 3.11.1.2.a (30 days from the end of the affected calendar quarter).

i. Calculate Air Dose from Noble Gases Effluent in Excess of Limits, Specification 3.11.2.2.a (30 days from the end of the affected calendar quarter).

j. Calculate Dose from I-131, I-133, H-3 and Radioactive Particulates with half lives greater than eight days in Excess of Limits, Specification 3.11.2.3.a (30 days from the end of the affected calendar quarter).

k. Calculated Doses from Uranium Fuel Cycle Source in Excess of Limits Specification 3.11.4.a (30 days from the end of the affected calendar year).

l. Inoperable Gaseous Radwaste Treatment System, Specification 3.11.2.4 (30 days from the event).

m. Environmental Radiological Reports. With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting level of Table 6.9.3-1, when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within thirty (30) days from the end of the calendar quarter a special report identifying the cause(s) for exceeding the limits, and define the corrective action to be taken.

Table 6.9.3-1

REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr/Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10.0	1,000	60	1,000
Cs-137	50	20.0	2,000	70	2,000
Ba/La-140	200			300	

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6.10 Record Retention

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. REPORTABLE EVENT REPORTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

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6.10 Record Retention (Continued)

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

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

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by oneor more of the following:

6.12 High Radiation Area (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor or designate in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1 areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem** shall be provided with locked doors to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Station Shift Supervisor or designate on duty and/or the Radiation Protection Supervisor or designate. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify in accordance with site approved procedures ~~accordingly~~, the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the drywell, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* by accessible passage and permanently fixed ladders
** measurement made at 18" from source of radioactivity

6.13 Fire Protection Inspection

6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

6.14 Systems Integrity

Procedure shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.

6.15 Iodine Monitoring

Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.8.c of NUREG 0578.