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ILLINOIS POWER COMPANY

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727

September 28, 1984

Docket No. 50-461

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Clinton Power Station Unit 1
DRAFT Technical Specifications

Dear Mr. Schwencer:

Illinois Power is enclosing a revised mark-up of the Clinton Power Station "DRAFT" Technical Specifications for your review. The changes indicated in the enclosure are provided as a result of our review of plant specific design information, information contained in the Clinton Power Station Final Safety Analysis Report, requirements/commitments delineated in the Clinton Power Station Safety Evaluation Report (NUREG-0853) through Supplement No. 3, and the Technical Specifications of recent licensees.

We request your review of the enclosure to support our schedule for "cold license" operator training beginning in November 1984. Should you have any questions or require additional information, please contact us.

Sincerely yours,

F. A. Spangenberg

F. A. Spangenberg
Director - Nuclear Licensing and Configuration
Nuclear Station Engineering

FAS:RFP/lm

Enclosure

cc: B. L. Siegel, NRC Clinton Licensing Project Manager
NRC Resident Office
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety/w/o enclosure

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SECTION 1.0

DEFINITIONS

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

ACTION

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

AVERAGE PLANAR EXPOSURE

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

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

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

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

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

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

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

b. All drywell equipment hatches are closed and sealed.

c. The drywell airlock door(s) are closed and sealed ~~is OPERABLE~~ pursuant to Specification 3.6.2.~~3~~1. CPS

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

e. The suppression pool is OPERABLE pursuant to Specification 3.6.3.1.

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

E-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump 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

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to (complete suppression of the electric arc between the fully open contacts) ~~(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: CPS

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

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

(FRACTION OF LIMITING POWER DENSITY

1.14 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). CPS

(FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

> Insert Attached CPS

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.~~15~~ 8 IDENTIFIED LEAKAGE shall be: CPS

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

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

ISOLATION SYSTEM RESPONSE TIME

1.1~~6~~ 9 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. CPS

LIMITING CONTROL ROD PATTERN

1.~~17~~ 20 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. CPS

LINEAR HEAT GENERATION RATE

1.~~18~~ 21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. CPS

LOGIC SYSTEM FUNCTIONAL TEST

1.~~19~~ 22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc., of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested. CPS

(MAXIMUM FRACTION OF LIMITING POWER DENSITY CPS

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

~~MAXIMUM TOTAL PEAKING FACTOR~~

~~1.20 The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.~~ CPS

> Insert Attached

Insert

MEMBER(S) OF THE PUBLIC

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

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MINIMUM CRITICAL POWER RATIO

1.2~~1~~5 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel). CPS

> Insert Attached 1.26 CPS

OPERABLE - OPERABILITY

1.2~~2~~7 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). CPS

OPERATIONAL CONDITION - CONDITION

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

1.~~25~~30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall. CPS

PRIMARY CONTAINMENT INTEGRITY

1.~~26~~31 PRIMARY CONTAINMENT INTEGRITY shall exist when: CPS

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

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

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

b. All containment equipment hatches are closed and sealed.

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

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

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

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

> Insert Attached 1.32, 1.33 CPS

Insert

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.26 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, and in the conduct of the environmental radiological monitoring program. | CPS

PROCESS CONTROL PROGRAM (PCP)

1.32 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low-level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low-level radioactive waste disposal sites. | CPS

PURGE - PURGING

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

RATED THERMAL POWER

1.~~27~~ 34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2894 MWT. | CPS

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.~~28~~ 35 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. | CPS

REPORTABLE ~~OCCURRENCE~~ EVENT

1.~~29~~ 36 A REPORTABLE OCCURRENCE shall be any of those conditions specified in ~~Specifications 6.9.1.8 and 6.9.1.9.~~ Section 50.73 to 10CFR Part 50. | CPS.

ROD DENSITY

1.~~30~~ 37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY. | CPS

SECONDARY CONTAINMENT INTEGRITY

1.~~31~~ 38 SECONDARY CONTAINMENT INTEGRITY shall exist when: | CPS

a. All Secondary Containment ~~(Auxiliary Building)~~ 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) (, as applicable) secured in its closed position, except as provided in Table 3.6.6.2-1 of Specification 3.6.6.2. | CPS

b. All Secondary Containment ~~(Auxiliary Building)~~ equipment hatches and blowout panels are closed and sealed. | CPS

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.6.3. |

d. (At least one) ~~(The)~~ door in each access to the Secondary Containment (Auxiliary Building) is closed (, except for normal entry and exit). | CPS

e. The sealing mechanism associated with each Secondary Containment ~~(Auxiliary Building)~~ penetration, e.g., welds, bellows or O-rings, is OPERABLE. | CPS

(f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.6.1.a.) | CPS

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SHUTDOWN MARGIN

1.~~32~~ 9 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. |CPS

> Insert 1.40, 1.41, 1.42 |CPS

STAGGERED TEST BASIS

1.~~33~~ 4 A STAGGERED TEST BASIS shall consist of: |CPS

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

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

THERMAL POWER

1.~~34~~ 4 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. |CPS

~~TOTAL PEAKING FACTOR~~

~~1.35 The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.~~ |CPS

TURBINE BYPASS SYSTEM RESPONSE TIME

1.~~36~~ 45 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the (monitored parameter exceeds its actuation setpoint at the channel sensor) (turbine bypass control unit generates a turbine bypass valve flow signal) until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured. |CPS

UNIDENTIFIED LEAKAGE

1.~~37~~ 46 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE. |CPS

> Insert Attached 1.47 - 1.50

SITE BOUNDARY

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

~~SOLIDIFICATION~~

1.41 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP). | CPS

SOURCE CHECK

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

1.0 DEFINITIONS (Continued)

UNRESTRICTED AREA*

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

VENTILATION EXHAUST TREATMENT SYSTEM*

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

VENTING*

1.49 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, 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.

WASTE GAS HOLDUP SYSTEM*

1.50 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

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NO CHANGE

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

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

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NO CHANGE

TABLE 1.2

OPERATIONAL CONDITIONS

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

#The reactor mode switch may be placed in the 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.

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

NO CHANGE

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

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- No CHANGE

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, 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.

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TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	$\leq$ 0.66 W+48%, with a maximum of	$\leq$ 0.66 W+51%, with a maximum of
2) High Flow Clamped	$\leq$ 111.0% of RATED THERMAL POWER	$\leq$ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ ~~1074~~ 1064.7 psig	$\leq$ ~~1089~~ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ ~~10.0~~ 8.91 inches above instrument zero*	$\geq$ ~~9.4~~ 8.31 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	$\leq$ 52.0 inches above instrument zero*	$\leq$ 52.6 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	$\leq$ ~~6~~ 6.2% closed	$\leq$ ~~7~~ 7.2% closed
7. Main Steam Line Radiation - High	$\leq$ ~~(2.5)~~ 3.0 x full power background	$\leq$ ~~(3.0)~~ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ ~~1.73~~ 1.68 psig	$\leq$ ~~1.93~~ 1.88 psig
9. Scram Discharge Volume Water Level - High	$\leq$ (~~30~~ LATER)% of full scale	$\leq$ (~~39~~ LATER)% of full scale
10. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ ~~44.3~~ 530 psig	$\geq$ ~~41~~ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

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

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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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.

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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.

CPS

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. CPS

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. CPS

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

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

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

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

DRAFT

NO CHANGE

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.

DRAFT

NO CHANGE.

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

BASES

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 1974 Edition, including Addenda through ~~Winter 1975~~ Summer 1973, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure, 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the ASME Boiler and Pressure Vessel Code, 1975 Edition, including Addenda through ~~Winter~~ Summer 1975 for the reactor recirculation piping, which permits a maximum pressure transient of (120)%, ~~(1380)~~ psig, of design pressure, 1250 psig for suction piping and 1650 psig for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

equaling 1500 (Suction) psig and 1980 (discharge

2.1.4 REACTOR VESSEL WATER LEVEL

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

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.~~ Insert Attached

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems, and backup to the APRM prior to entering the "RUN" Mode. ~~The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section (15.1.12) of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately (1)% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to (21)% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold criterion of (170) cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.~~

Insert Attached

2. Average Power Range Monitor

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

Insert 2.2.1

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.

Insert 1. IRM Neutron Flux - High

For BWR/6 plants, the role of the IRM in responding to potential Rod Withdrawl Error (RWE) transients is greatly diminished due to use of a dual channel Rod Pattern Control System. Consequently specific analysis of IRM response to the RWE is no longer required.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

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

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

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 > to FRTP). CPS

3. Reactor Vessel Steam Dome Pressure-High

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

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. ~~The results reported in Section 15 show that scram and isolation of all process lines, except main steam, at this level adequately protects the fuel and the pressure barrier, because MCPR 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. CB

5. Reactor Vessel Water Level-High

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

6. Main Steam Line Isolation Valve-Closure

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

7. Main Steam Line Radiation-High

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

DRAFT

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Drywell Pressure-High

and the primary containment

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 the primary containment. The trip setting was selected as low as possible to minimize heat loads of equipment located within the primary containment without causing spurious trips. CPS

9. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water 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 (~~25~~) gallons of water. CPS

LATER

10. Turbine Stop Valve-Closure

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

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

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincident ~~with~~ failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 20 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the two-out-of-four 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 15.2.2 of the Final Safety Analysis Report. CPS

DRAFT

No Change

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

12. Reactor Mode Switch Shutdown Position

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

13. Manual Scram

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

DRAFT

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND
SURVEILLANCE REQUIREMENTS

DRAFT

NO CHANGE

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

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

DRAFT

NO CHANGE

SURVEILLANCE REQUIREMENTS

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

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

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

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

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifi-ations. 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

DRAFT

NO CHANGE

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.

DRAFT

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. Within 12 ~~one~~ 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. CPS

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

DRAFT

3/4.1.2 REACTIVITY ANOMALIES

No CHANGE

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.

DRAFT

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

   1. Within one hour:

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

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

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

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

      c) Comply with Surveillance Requirement 4.1.1.c.

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

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

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

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

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable 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 by bypassing ~~on the RGDS analyzer card~~, or CP

      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 (by bypassing) ~~on the RGDS analyzer card~~, or | CPS

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

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

3. The provisions of specification 3.0.4 are not applicable. | CPS

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

a. At least once per 7 days, and

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

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

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*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 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 CPS

   2. Open when the scram signal is reset.

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

DRAFT

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43 ~~44~~	29 ~~28~~	13 ~~12~~
950	0.31	0.81	1.44
~~1085~~ 1050	0.32 ~~0.28~~	~~0.91~~ 0.86	~~1.62~~ 1.57

CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43 ~~44~~	29 ~~28~~	13 ~~12~~
950	0.38	1.09	2.09
~~1085~~ 1050	0.39 ~~0.34~~	~~1.20~~ 1.14	~~2.30~~ 2.22

CPS

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

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds)		
	43 ~~44~~	29 ~~28~~	13 ~~12~~
950	0.30	0.78	1.40
~~1085~~ 1050	0.31 ~~0.27~~	~~0.88~~ 0.84	~~1.58~~ 1.53

CPS

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 5.
4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupy adjacent locations in any direction, including the diagonal, to another such control rod.

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

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

DRAFT

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

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

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

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

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

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

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

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

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

      b) OPERABLE.

   4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed ~~( )(**)~~ 5. | CPS

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

d. The provisions of Specification 3.0.4 are not applicable. | CPS

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

DRAFT

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

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

a) Restore the inoperable accumulator to OPERABLE status, or

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

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

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

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

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

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

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 (by bypassing) ~~on the RGDS analyzer card~~, or | CPS

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

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

c. The provisions of Specification 3.0.4 are not applicable. | CPS

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is ~~(greater than or equal to)~~ 1520 +30, -0 psig unless the control rod is inserted and disarmed or scrammed. CPS

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of 1520 +30, -0 psig on decreasing pressure.

2. ~~(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)~~ starting at normal operating pressure with no control rod drive pump operating. CPS

DRAFT

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

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

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

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

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

      a) Electrically by bypassing ~~on the RGDS analyzer card~~, or |CPS

      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 by bypassing ~~on the RGDS analyzer card~~, or |CPS

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

c. The provisions of specification 3.0.4 are not applicable. |CPS

*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.

DRAFT

NO CHANGE

SURVEILLANCE REQUIREMENTS

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

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

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

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

DRAFT

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

   1. Determine the position of the control rod by ~~an~~ the alternate control rod position indicator, or CPS

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

   3. When THERMAL POWER is:

      a) Within the low power setpoint of the RPCS:

         1) Declare the control rod inoperable, and

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

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

         1) Electrically (by bypassing) ~~on the RODS analyzer card,~~ or CPS

         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 both position indicators of a withdrawn control rod inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of specification 3.0.4 are not applicable. CPS

*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.

DRAFT

SURVEILLANCE REQUIREMENTS

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

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

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

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.4.b, and

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

CPS

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CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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3/4.1.4 CONTROL ROD PROGRAM CONTROLS

CONTROL ROD WITHDRAWAL

LIMITING CONDITION FOR OPERATION

3.1.4.1 Control rods shall not be withdrawn.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod pattern control system (RPCS) shall be OPERABLE. with its function compatible with any existing inoperable control rods. | CP

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#.

ACTION:

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

1. THERMAL POWER less than or equal to the low power setpoint, ~~(20)% of RATED THERMAL POWER~~, control rod movement shall not be permitted, except by a scram. | CPS
2. THERMAL POWER greater than the low power setpoint ~~(20)% of RATED THERMAL POWER~~, control rod withdrawal shall not be permitted. | CPS

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

1. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable, this inoperable control rod may be bypassed ~~in the rod gang drive system (RGDS)~~ provided that the SHUTDOWN MARGIN has been determined to be equal to or greater than required by Specification 3.1.1. | CPS
2. With up to eight control rods inoperable for causes other than addressed in ACTION b.1, above, ~~one of~~ these inoperable control rods may be bypassed ~~in the RGDS~~ provided that: | CPS
   a) The control rod to be bypassed is inserted and the directional control valves are disarmed either:
      1) Electrically by bypassing ~~on the RGDS analyzer card~~, or | CPS
      2) Hydraulically by closing the drive water and exhaust water isolation valves.
   b) All inoperable control rods are separated from all other inoperable control rods by at least two control cells in all directions.
   c) There are not more than 3 inoperable control rods in any RPCS group.
3. The position and bypassing of an inoperable control rod(s) is verified by a second licensed operator or other technically qualified member of the unit technical staff.

*See Special Test Exception 3.10.2

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

INSERT TO replace ACTION 3.1.4.2.b (p 3/4 1-16)

b. With an inoperable control rod(s) and THERMAL POWER less than or equal to the low power setpoint, OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RPCS provided that:

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

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

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

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

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

b. In OPERATIONAL CONDITION 5*:

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

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

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

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

1. The temperature of the sodium pentaborate solution in the storage tank is ~~within the limits of Figure 3.1.5-1.~~ greater than or equal to 70°F. CPS

2. The available volume of sodium pentaborate solution is within ~~greater than or equal to 3542 gallons.~~ the limits of Figure 3.1.5-1. CPS

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Verifying the continuity of the explosive charge.

2. Determining that 4246 the available net weight of sodium pentaborate is greater than or equal to ~~4600~~ lbs and the concentration of ~~boron~~ sodium pentaborate in solution is within the limits of Figure 3.1.5-1 by chemical analysis.* CPS

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. CPS

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

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

2. Demonstrating 1400 that the pump relief valve setpoint is less than or equal to ~~(1600)~~ psig and verifying that the relief valve does not actuate during recirculation to the test tank. CPS

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 ~~a~~ temperature rise of the sodium pentaborate solution in the storage tank ~~by at least __ °F within __ minutes~~ after the heaters are energized. CPS

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

**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.

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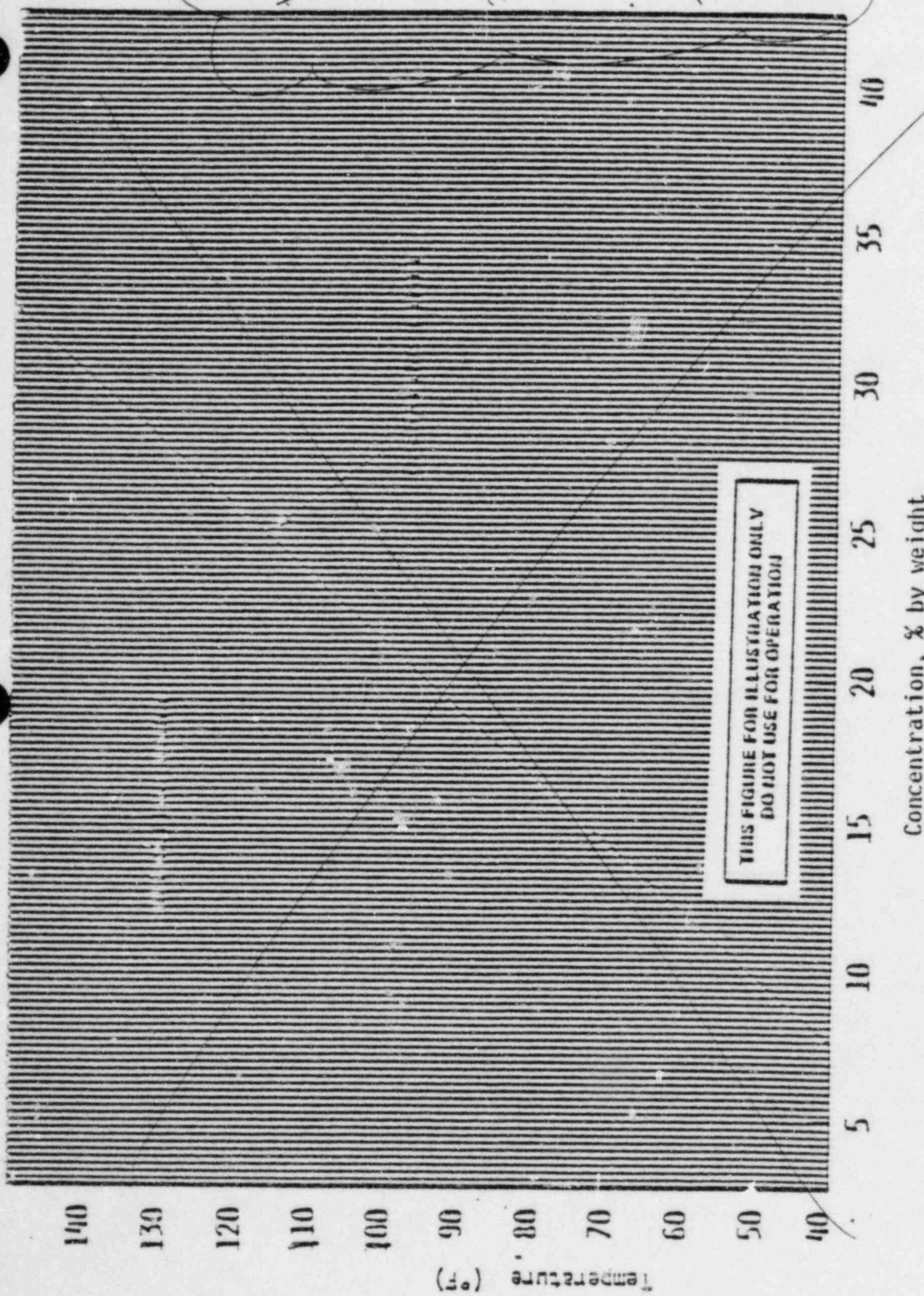

SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

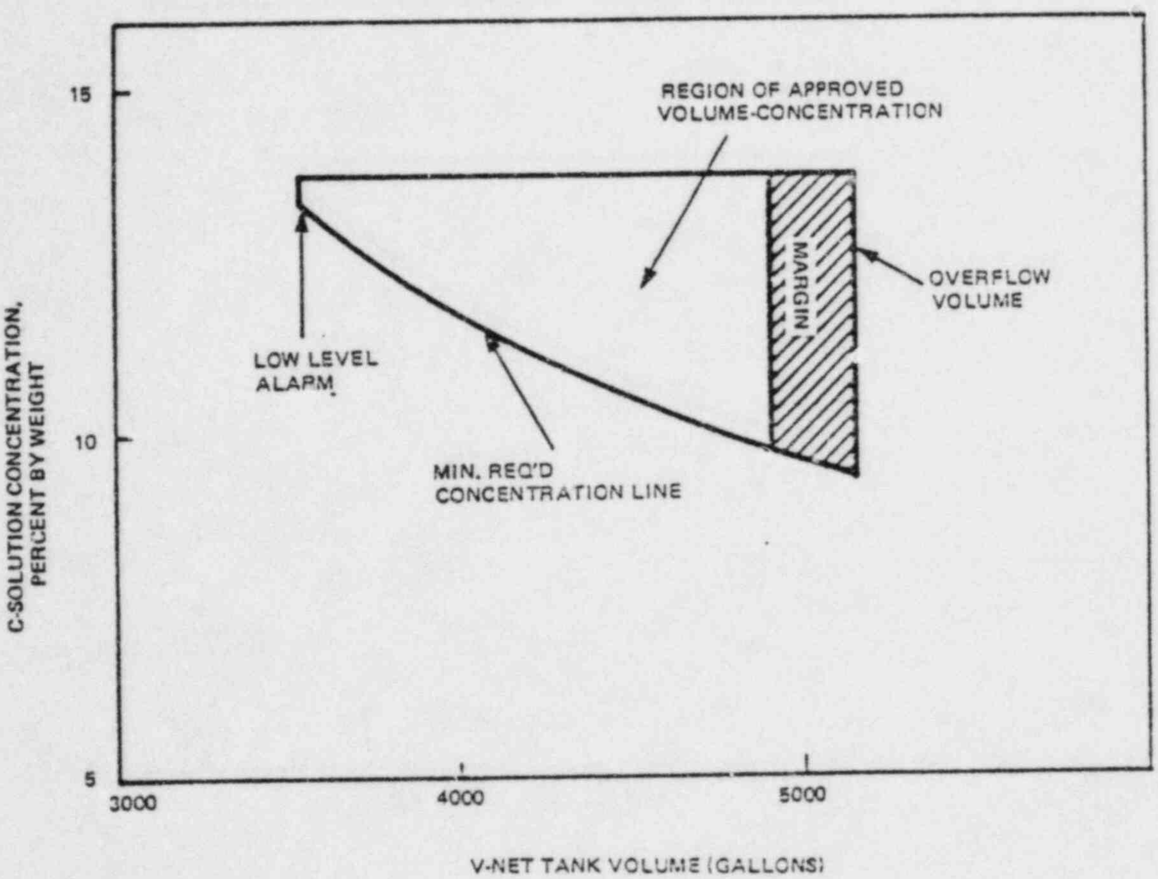

Figure 3.1.5-1 WEIGHT PERCENT SODIUM PENTABORATE SOLUTION AS A FUNTION OF NET TANK VOLUME

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

NO CHANGE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

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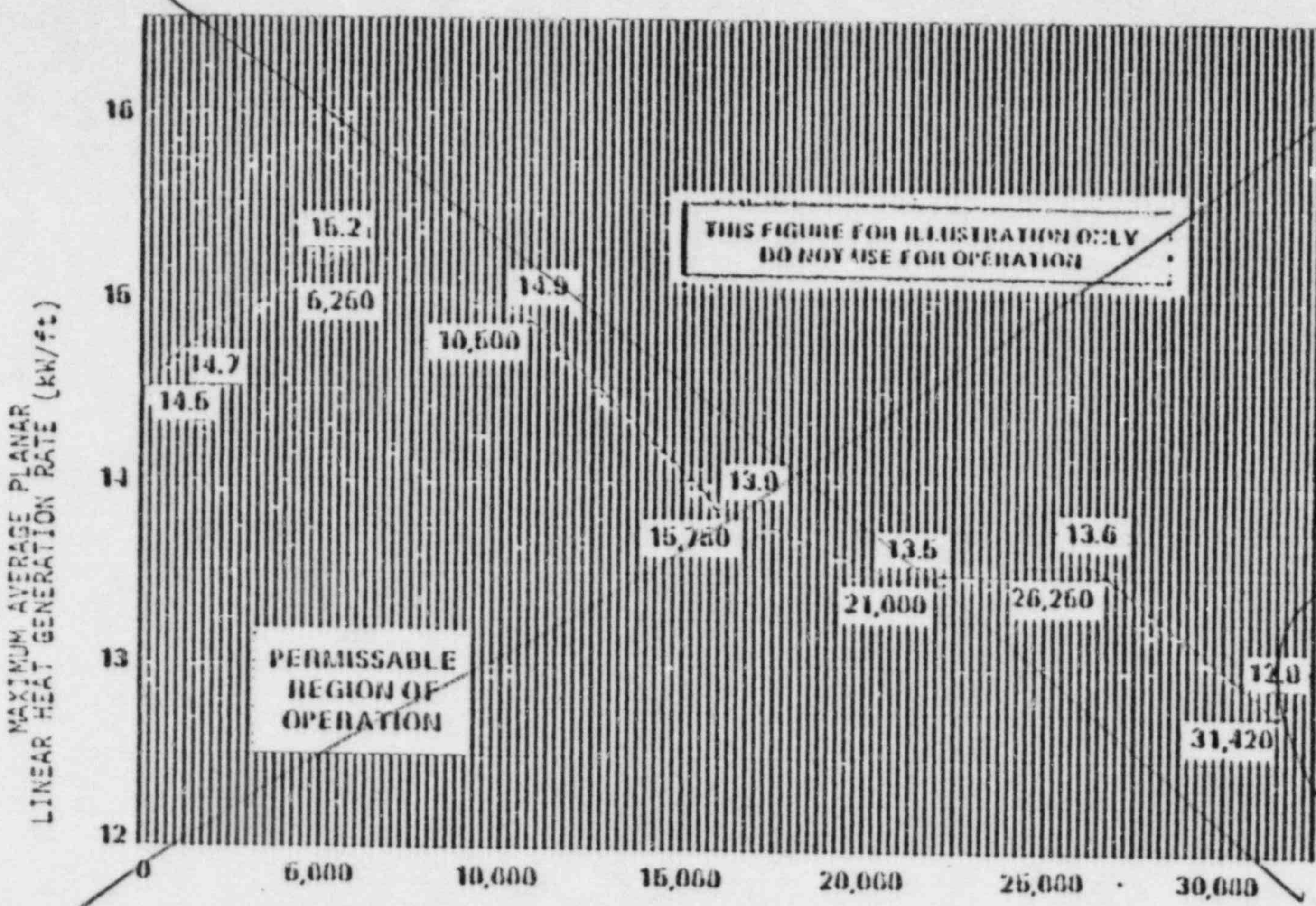

AVERAGE PLANAR EXPOSURE (MWd/t)

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES (   )
Figure 3.2.1-1

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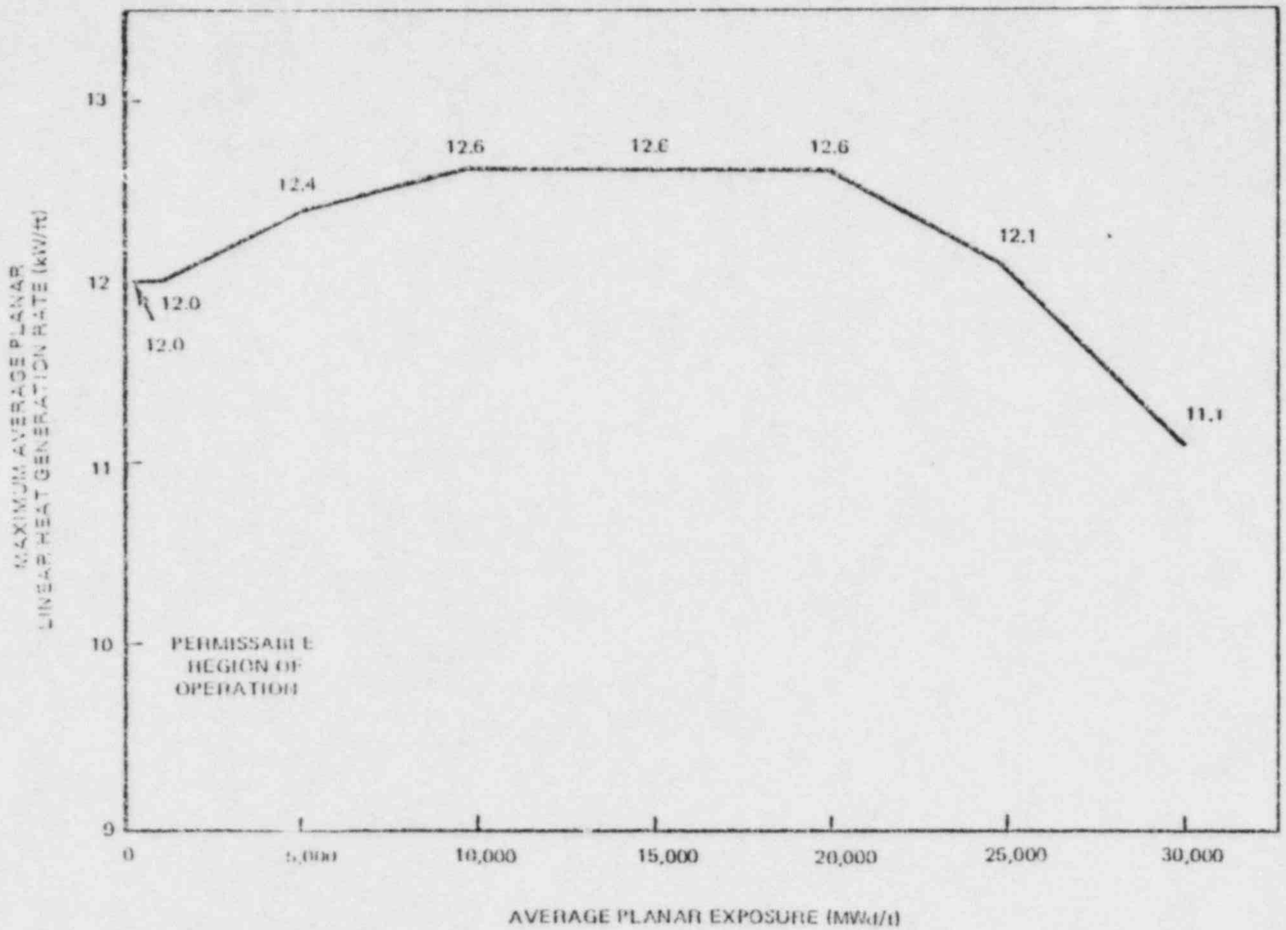

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Average Planar Exposure
Initial Core Fuel Types - High Enrichment
Figure 3.2.1-1

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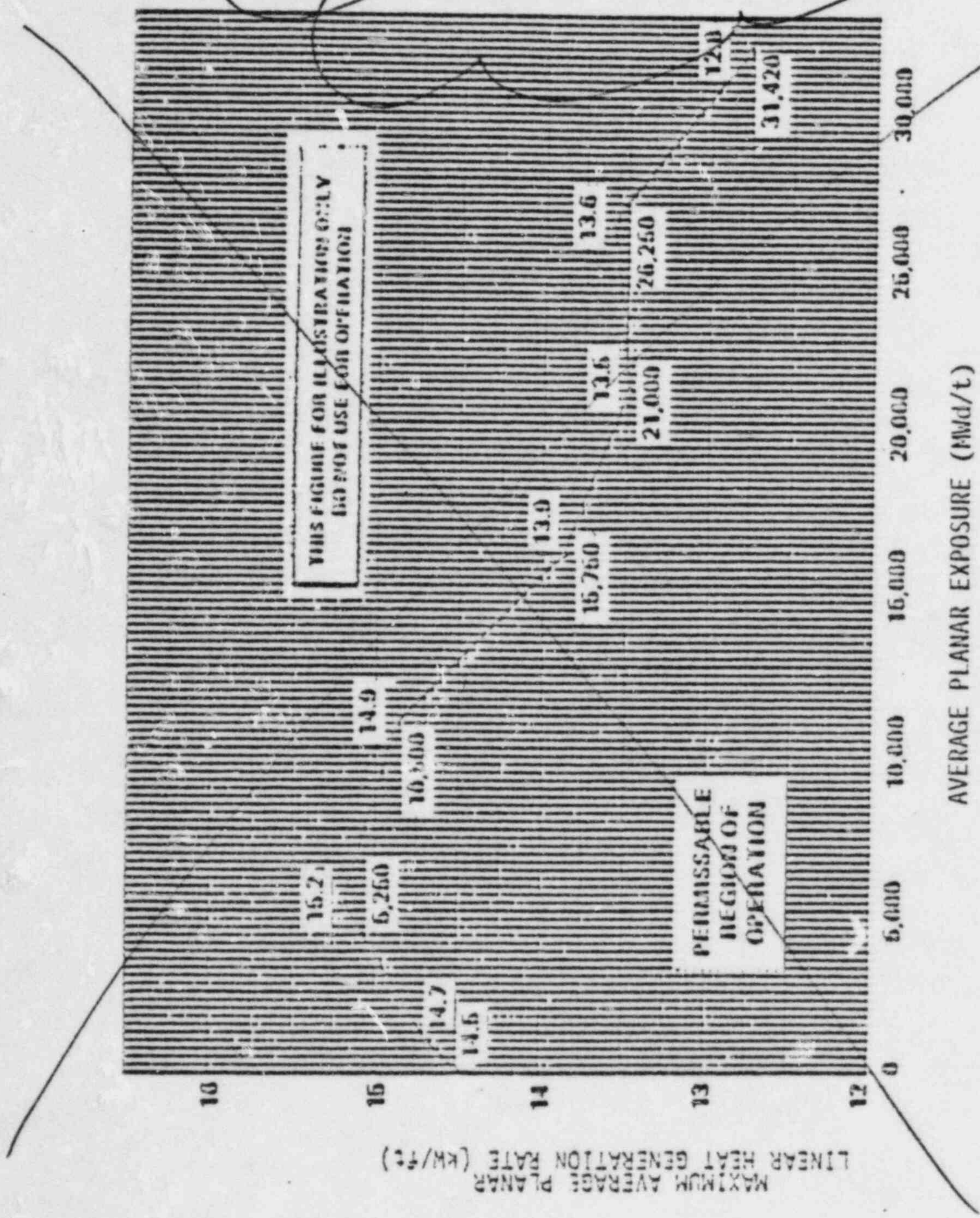
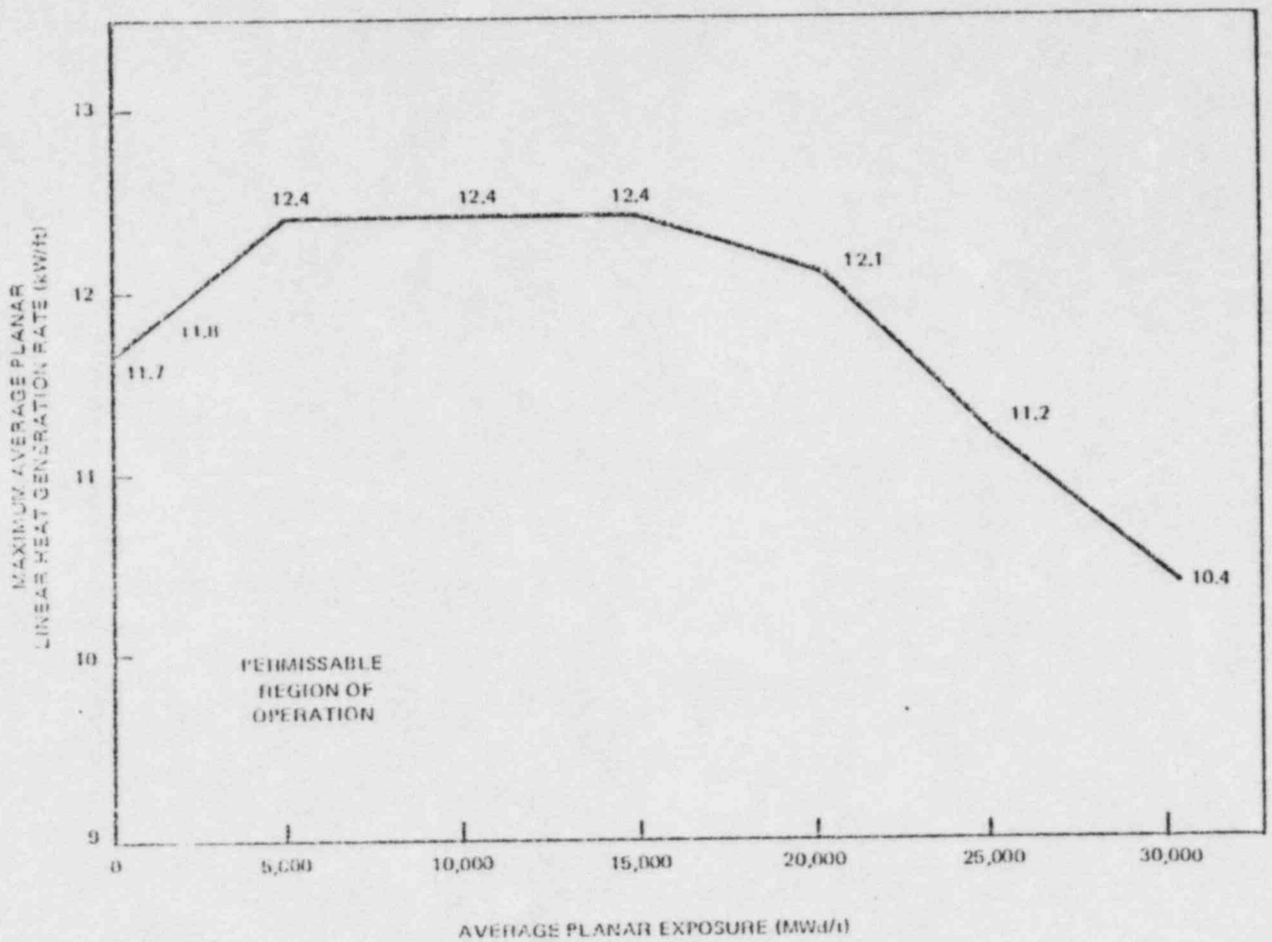


MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURES
INITIAL CORE FUEL TYPES ( )
Figure 3.2.1-2



Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Average Planar Exposures
Initial Core Fuel Types Medium Enrichment
Figure 3.2.1–2

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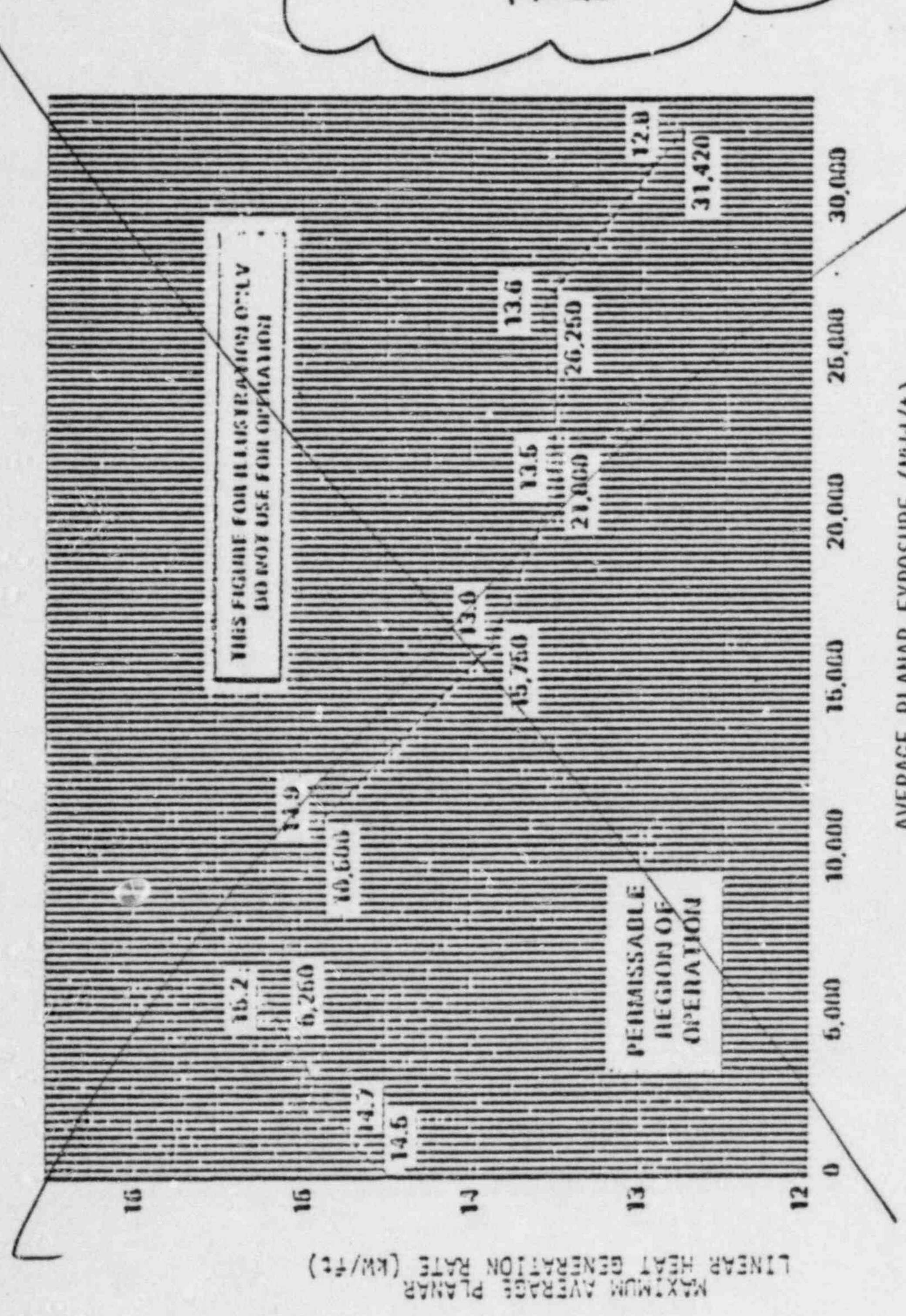

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE INITIAL CORE FUEL TYPES ( )

Figure 3.2.1-3

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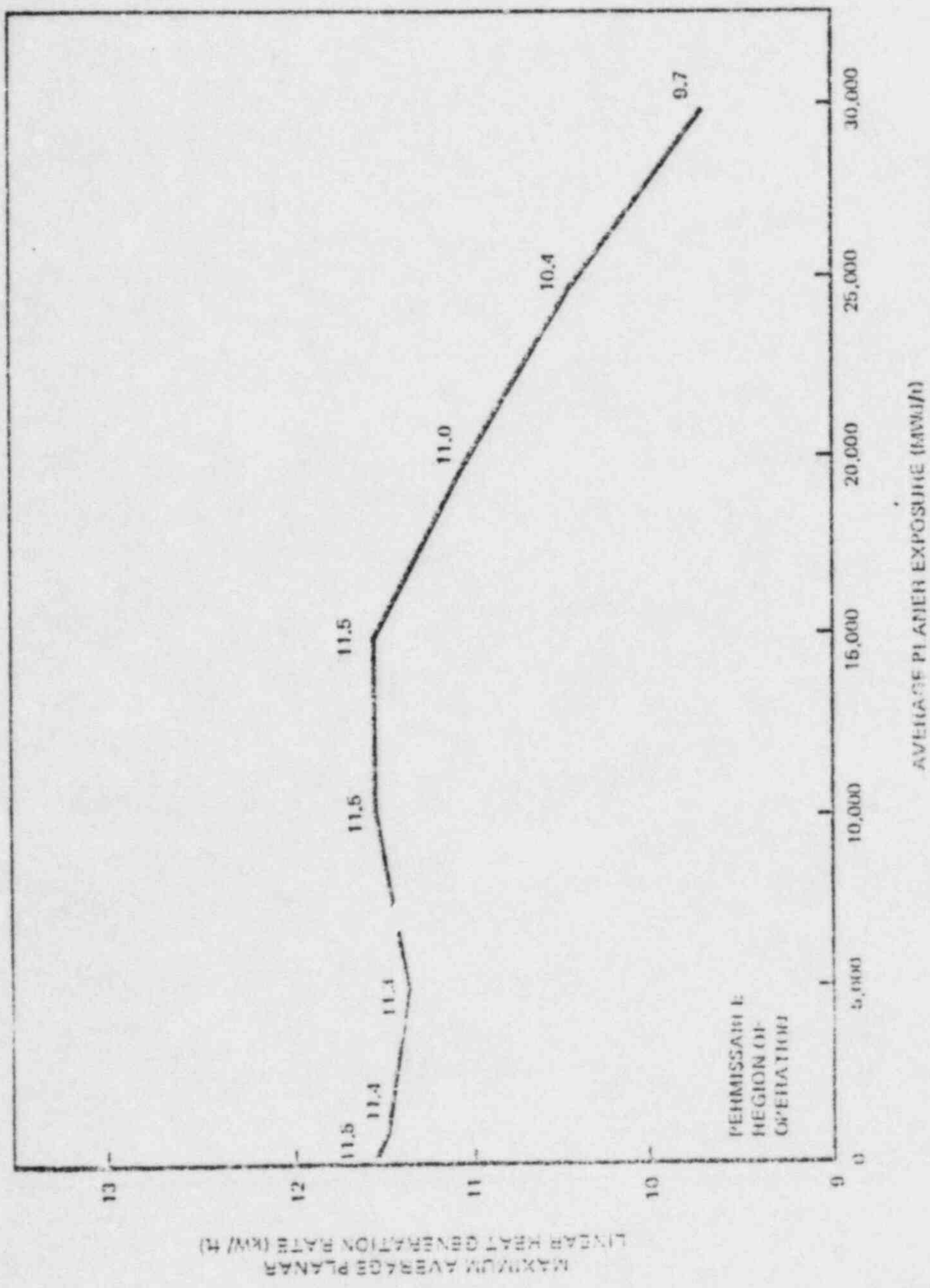

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure
Initial Core Fuel Types Natural Enrichment
Figure 3.2.1-3

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

CPS

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 ~~(112.5)~~ 84.5 million lbs/hr.
T = Lowest value of the ratio of ~~(design TPF divided by the MTPF obtained for any class of fuel in the core)~~ (FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)). T is ~~always less than or equal to 1.0~~ applied only if less than or equal to 1.0.
~~(Design TPF for 8 x 8 fuel = 2.43.)~~

CPS

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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) greater than or equal to ~~(2.43)~~ (FRTP). CPS

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

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

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both $MCPR_f$ and $MCRP_p$ limits at indicated core flow and THERMAL POWER as shown in ~~Curve A of~~ Figures 3.2.3-1 and 3.2.3-2, ~~provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2.~~ | CPS

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 equal to or greater than both $MCPR_f$ and $MCPR_p$ as shown in Curves A and B of Figures 3.2.3-1 and 3.2.3-2 as follows:~~

~~1. From beginning-of-cycle (BOC) to end-of-cycle (EOC) minus 2000 MWd/t, Curve A.~~

~~2. From EOC minus (2000) MWd/t to EOC, Curve B.~~

CPS

~~(b. With the main turbine bypass system inoperable per Specification 3.7.7, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than both $MCPR_f$ and $MCPR_p$, as shown in Figures 3.2.3-1 and 3.2.3-2 by the main turbine bypass inoperable curve.)~~

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

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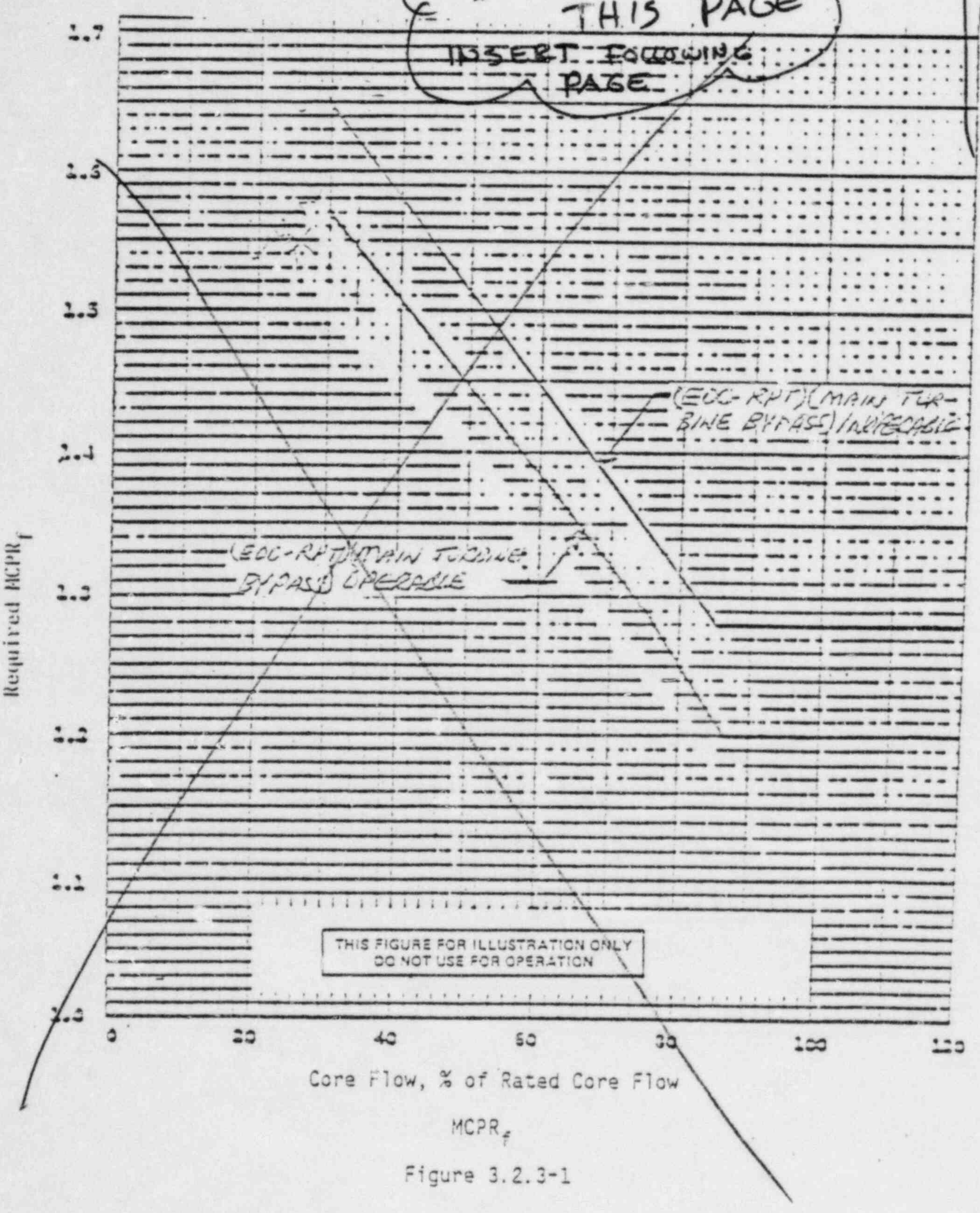
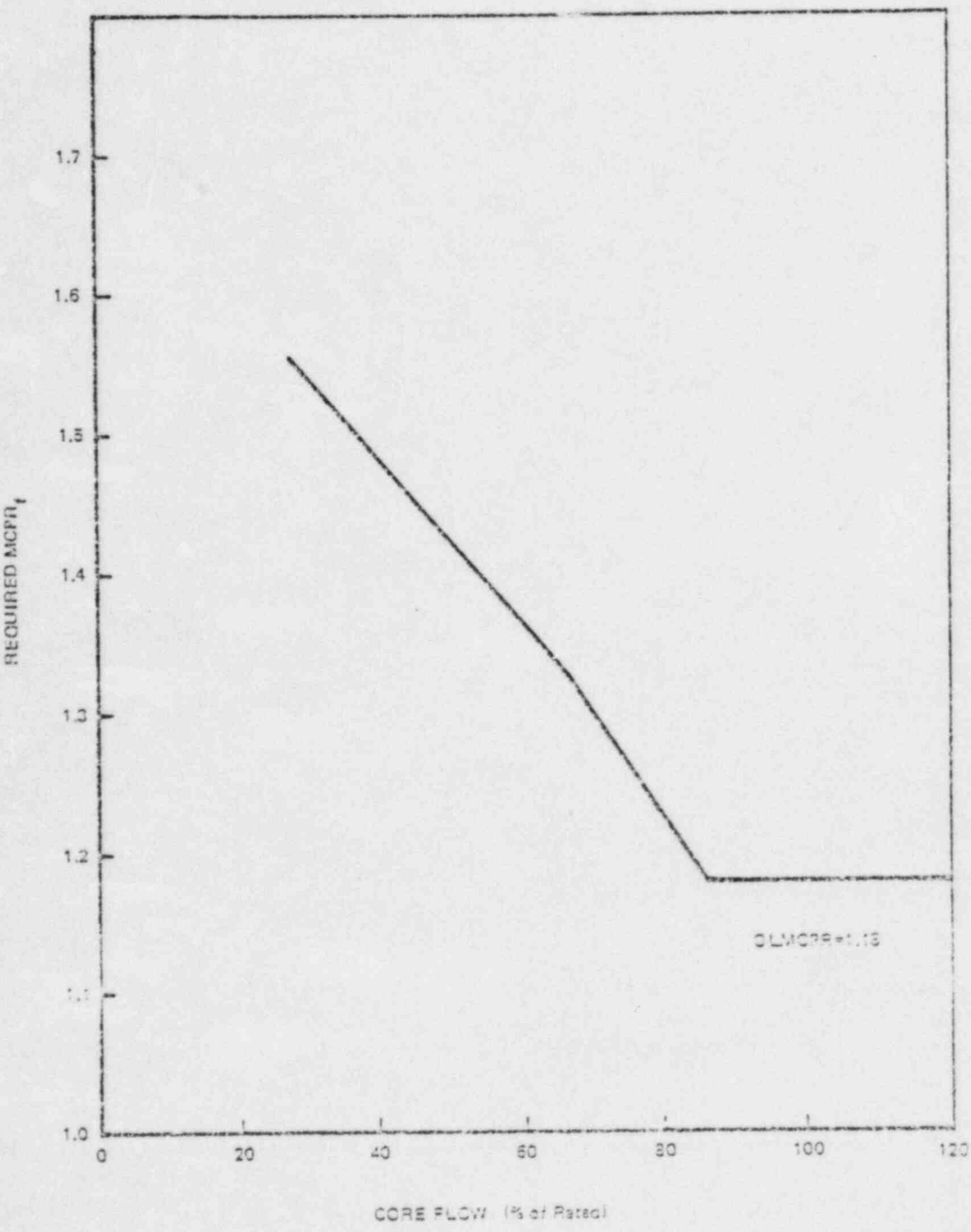


Core Flow, % of Rated Core Flow

MCPR$_f$

Figure 3.2.3-1



Clinton $MCPR_f$ (K_f) Versus Core Flow
Figure 3.2.3-1

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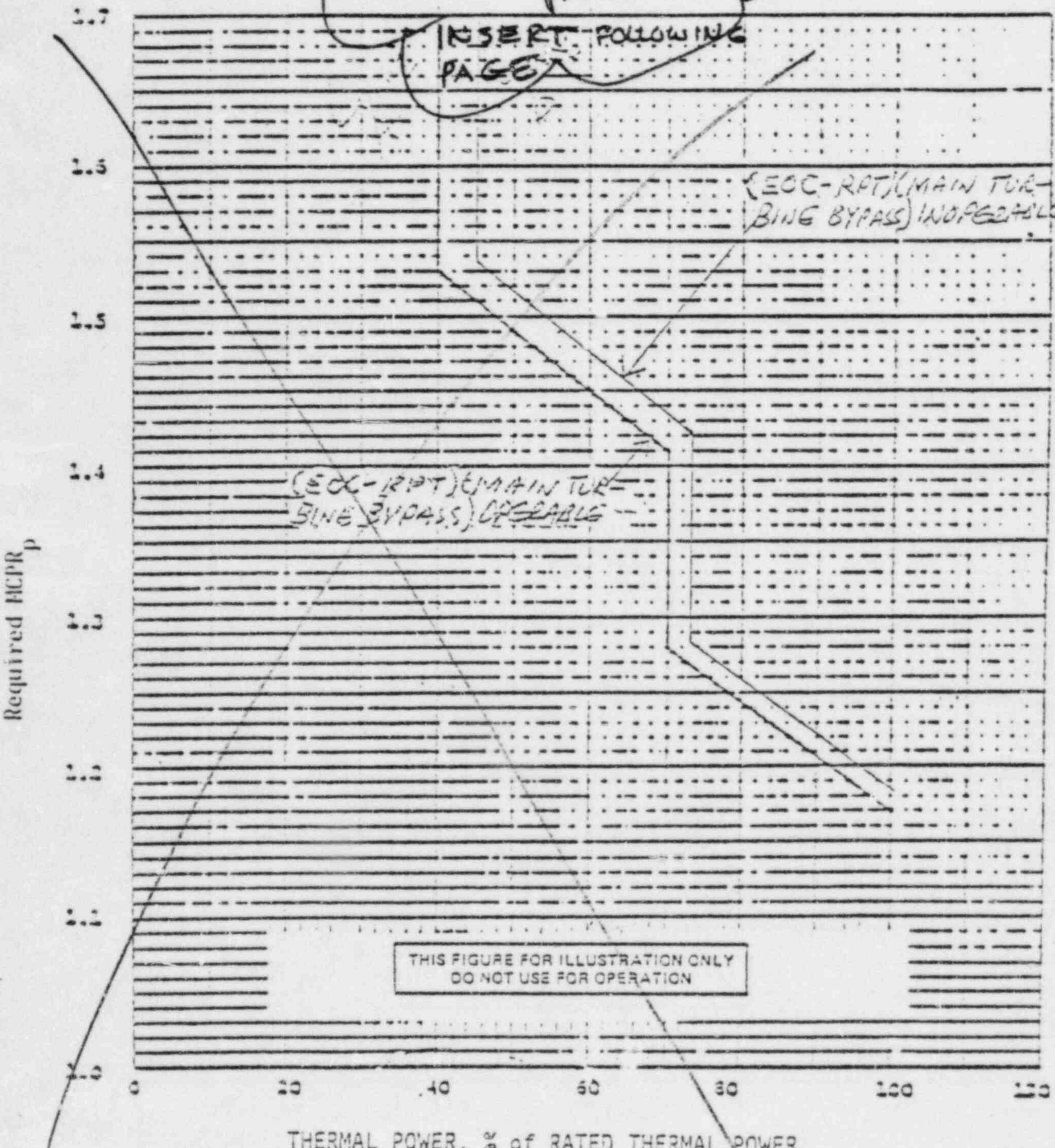
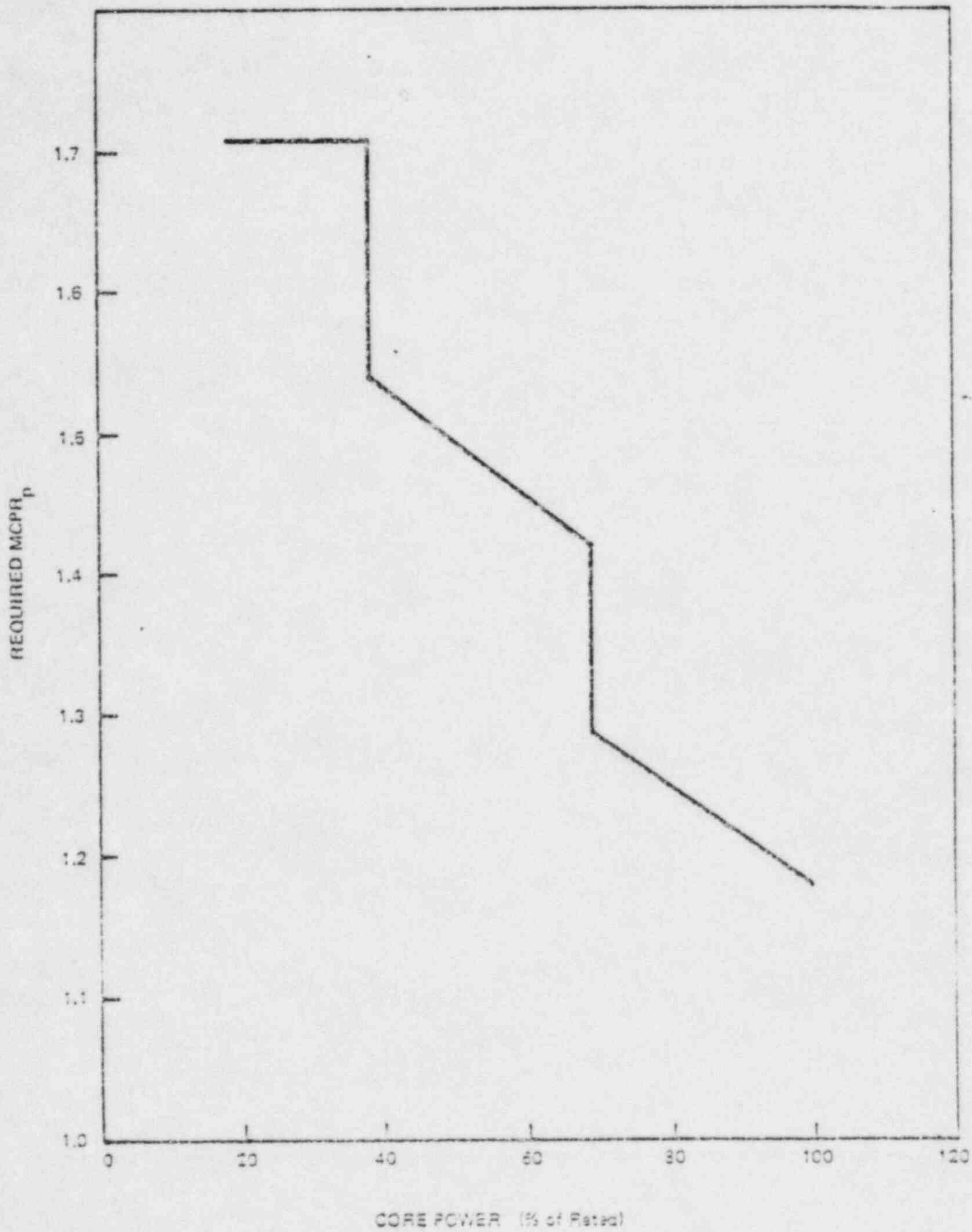


$MCPR_p$

Figure 3.2.3-2



Clinton MCPR$_p$ (K_p) Versus Power
Figure 3.2.3-2

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3/4.2.3 MINIMUM CRITICAL POWER RATIO (Optional - ODYN Option B)

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit shown in Figure 3.2.3-1 times the K_f shown in Figure 3.2.3-2, (provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2) with:

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

where:

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

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

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

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

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

τ_i = average scram time to motch (39) of all rods measured in the i^{th} surveillance test, and

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

APPLICABILITY:

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

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

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

(b. With the main turbine bypass system inoperable per Specification 3.7.4, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than both $MCPR_f$ and $MCPR_p$, as shown in Figures 3.2.3-1 and 3.2.3-2 by the main turbine bypass inoperable curve.)

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

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR with

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

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

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

a. At least once per 24 hours,

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

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

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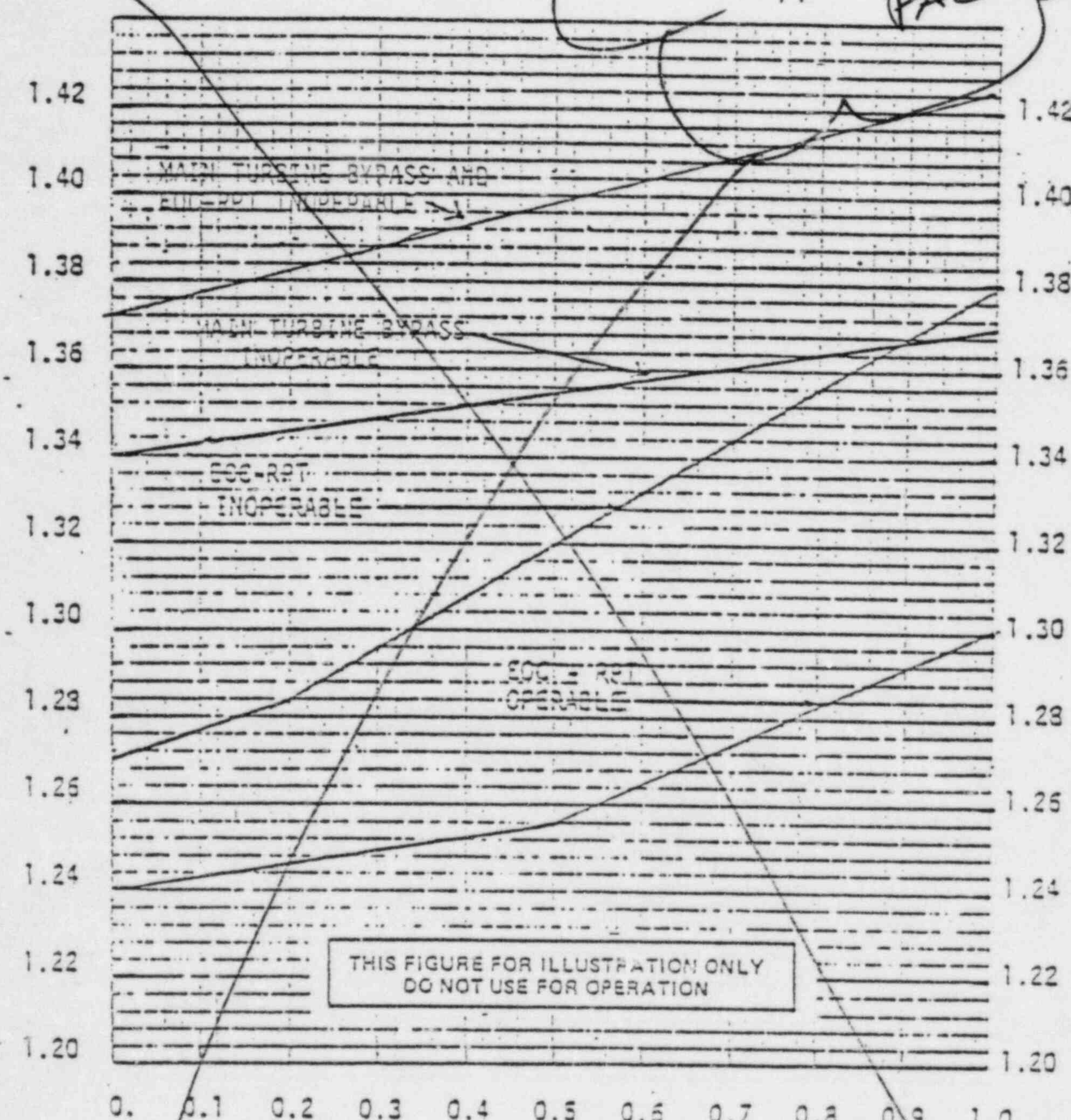
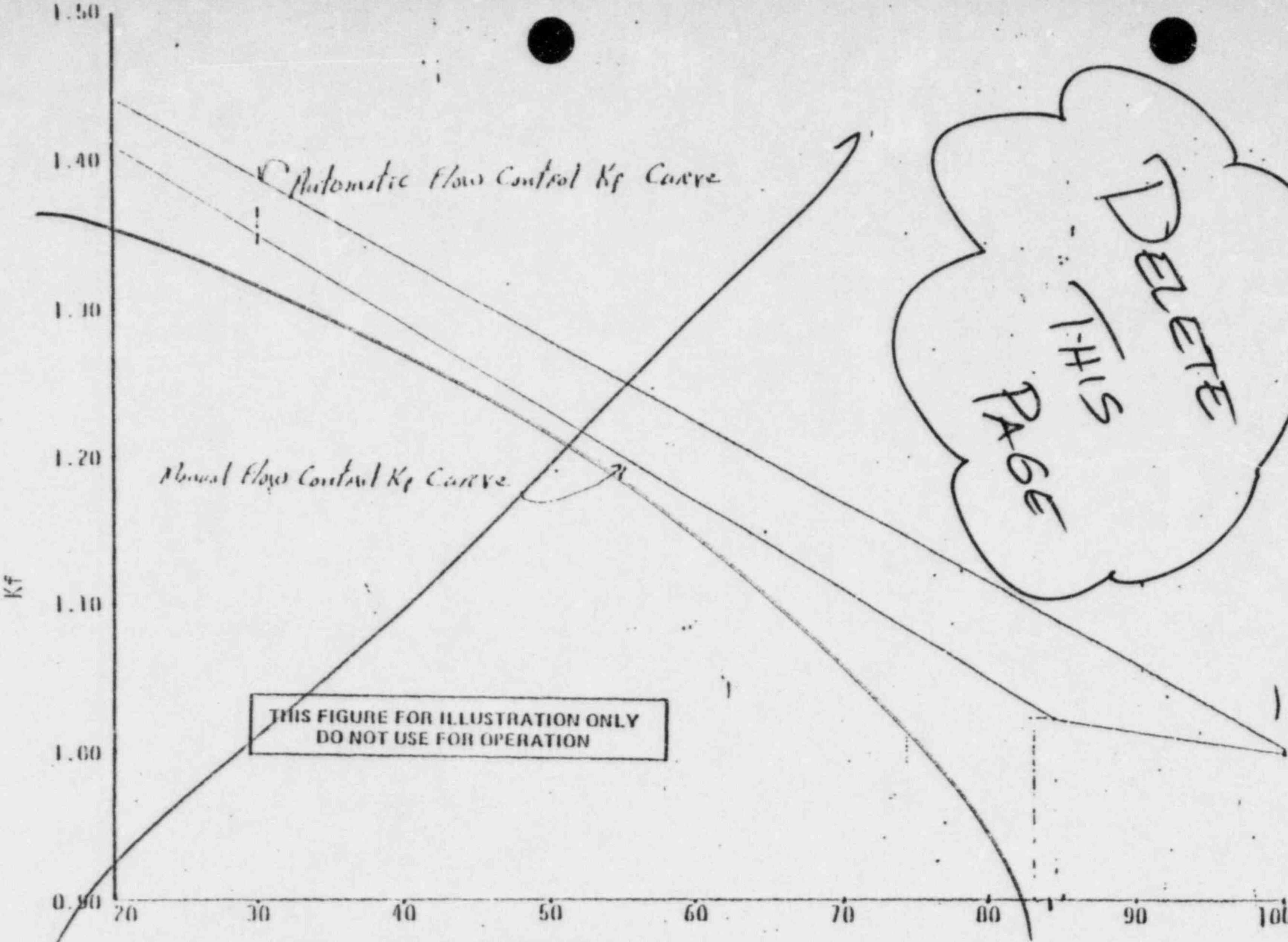


MINIMUM CRITICAL POWER RATIO (MCPR)

VERSUS τ AT RATED FLOW

Figure 3.2.3-1



Core Flow, % Rated Core Fow

K_f. FACTOR

Figure 3.2.3-2

CPS

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3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

~~As shown in Table 3.3.1-1.~~ See insert (next page)

CPS

SURVEILLANCE REQUIREMENTS

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

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

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include {at least two} ~~(both)~~ {logic trains such that all logic trains are tested at least once per 36 months and} one channel per trip ~~(system)~~ {function} 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 {function} ~~(system)~~.

CPS

INSERT

TO 3.3.1

ACTION:

1. All Functional Units of Table 3.3.1-1 other than Reactor Mode switch Shutdown Position.
   a. With one of the four channels required for any Trip Function inoperable, operation may continue for 48 hours, after which time the inoperable channel shall be placed in the tripped condition.
   b. With two of the four channels required for any Trip Function inoperable, place one channel in the tripped condition within one hour provided no tripped channel for that Trip Function already exists.
   c. With three of the four channels required for any Trip Function inoperable, take the ACTION required by Table 3.3.1-1.
   d. The provisions of Specification 3.0.4 are not applicable.

2. Reactor Mode Switch shutdown Position as shown in Table 3.3.1-1.

CLINTON - UNIT 1

3/4 3-2

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TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT *	APPLICABLE OPERATIONAL CONDITIONS	~~TOTAL NO. OF CHANNELS~~	SPECIFIC CHANNEL OR FUNCTIONAL REQUIREMENTS	~~CHANNELS TO TRIP~~	~~MINIMUM OPERABLE CHANNELS~~	ACTION	CPS
1. Intermediate Range Monitors:							
a. Neutron Flux - High	2	~~4~~	Note a	~~2~~	~~3(a)~~	1#	
	3, ~~4~~	~~4~~	Note a	~~2~~	~~3(a)~~	2	
	5~~(b)~~	~~4~~	Notes a, b	~~2~~	~~3(a)~~	3	
b. Inoperative	2	~~4~~		~~2~~	~~3~~	1#	
	3, 4	~~4~~		~~2~~	~~3~~	2	
	5	~~4~~		~~2~~	~~3~~	3	
2. Average Power Range Monitor:							
a. Neutron Flux - High, Setdown	2	~~4~~	Note c	~~2~~	~~3(c)~~	1#	
	3, ~~4~~	~~4~~	Note c	~~2~~	~~3(c)~~	2	
	5~~(b)~~	~~4~~	Notes b, c	~~2~~	~~3(c)~~	3	
b. Flow Biased Simulated Thermal Power - High	1	~~4~~	Note c	~~2~~	~~3(c)~~	4 ~~1~~#	
c. Neutron Flux - High	1	~~4~~	Note c	~~3~~	~~3(c)~~	4 ~~1~~#	
d. Inoperative	1, 2	~~4~~		~~2~~	~~3(c)~~	1#	
	3, ~~4~~	~~4~~		~~2~~	~~3(c)~~	2	
	5	~~4~~		~~2~~	~~3(c)~~	3	
3. Reactor Vessel Steam Dome Pressure - High	1, 2~~(d)~~	~~4~~	Note d	~~2~~	~~3~~	1#	
4. Reactor Vessel Water Level - Low, Level 3	1, 2	~~4~~		~~2~~	~~3~~	1#	
5. Reactor Vessel Water Level - High, Level 8	1~~(e)~~	~~4~~	Note e	~~2~~	~~3~~	4 ~~1~~#	
6. Main Steam Line Isolation Valve - Closure	1~~(e)~~	~~4~~	Note e	~~2~~	~~3(a)~~	4 ~~1~~#	

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

CPS

FUNCTIONAL UNIT *	APPLICABLE OPERATIONAL CONDITIONS	SPECIFIC CHANNEL OR FUNCTIONAL REQUIREMENTS ~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM OPERABLE CHANNELS~~	ACTION
7. Main Steam Line Radiation - High	1, 2~~(d)~~	~~4~~ Note d	~~2~~	~~3~~	1#
8. Drywell Pressure - High	1, 2~~(f)~~	~~4~~ Note f	~~2~~	~~3~~	1#
9. Scram Discharge Volume Water Level - High	1, 2 5~~(g)~~	~~4~~ ~~4~~ Note g	~~2~~ ~~2~~	~~3~~ ~~3~~	1# 3
10. Turbine Stop Valve - Closure	1~~(h)~~	~~4~~ Note h	~~2~~	~~3~~	5 ~~1~~#
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1~~(h)~~	~~4~~ Note h	~~2~~	~~3~~	5 ~~1~~#
12. Reactor Mode Switch Shutdown Position	1, 2, 3, 4, 5 ~~3, 4~~ ~~5~~	~~4~~ ~~4~~ ~~4~~	~~2~~ ~~2~~ ~~2~~	~~3~~ ~~3~~ ~~3~~	6 ~~4~~ ~~5~~ ~~6~~
13. Manual Scram	1, 2 3, 4 5	~~4~~ ~~4~~ ~~4~~	~~2~~ ~~2~~ ~~2~~	~~3~~ ~~3~~ ~~3~~	1 ~~7~~ 2 ~~8~~ 3 ~~9~~

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* All Functional Units have "any-two-from-four" scram logic.

SEE ATTACHED REWRITE

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TABLE 3.3.1-1 (Continued)

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REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, POWER OPERATION and/or STARTUP, as applicable, may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum OPERABLE Channels requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 2 - With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels requirement, verify all insertable control rods to be fully inserted and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - With the number of OPERABLE channels:

a. One less than the Total Number of Channels, CORE ALTERATIONS may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within one hour.

2. The Minimum OPERABLE Channels requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing for Specifications 4.3.1.1 and 4.3.1.2.

b. Less than required by the Minimum OPERABLE Channels requirement, suspend all operations involving CORE ALTERATIONS* and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 4 - With the number of OPERABLE channels less than the Total Number of Channels, be in at least HOT SHUTDOWN within 12 hours.

ACTION 5 - With the number of OPERABLE channels less than the Total Number of Channels, verify all insertable control rods to be fully inserted within one hour and at least once per hour thereafter.

ACTION 6 - With the number of OPERABLE Channels less than the Total Number of Channels, suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the Refuel position within one hour.

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

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in HOT SHUTDOWN within 12 hours.

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

ACTION 3 - Suspend all operations involving CORE ALTERATIONS,* insert all insertable control rods and lock the reactor mode switch in the Shutdown position within one hour.

CPS

ACTION 4 - Be in at least the STARTUP mode within 8 hours

ACTION 5 - Initiate a reduction in THERMAL POWER within 15 minutes, and be at less than the turbine first stage pressure bypass setpoint within 8 hours.

ACTION 6 - With less than four channels OPERABLE, verify that all insertable control rods are fully inserted and lock the reactor mode switch in the Shutdown position within one hour.** Verify all insertable control rods to be fully inserted at least once per hour thereafter.

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

** The mode switch position may be changed only for the purpose of maintaining, repairing and testing the mode switch.

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~~TABLE 3.3.1-1 (Continued)~~

~~REACTOR PROTECTION SYSTEM INSTRUMENTATION~~

~~ACTION (Continued)~~

ACTION 7 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 8 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels requirement, lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or 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.

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

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

~~#The provisions of Specification 3.0.4 are not applicable.~~ CPS

(a) ~~A~~ An IRM channel is inoperable if there are less than two inputs per channel. CPS

(b) The "shorting links" shall be removed from the RPS circuitry ~~(or the rod pattern control system shall be OPERABLE)~~ prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations are being performed per Specificiation 3.10.3. CPS

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than ~~11~~ 17 LPRM inputs to an APRM channel. CPS

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

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

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

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

(h) This function shall be automatically bypassed when turbine first stage pressure is ~~< (250) psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.~~ SEE INSERT NEXT PAGE CPS

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

DEC 8 1982

INSERT TO NOTE (h) FROM
TABLE 3.3.1-1 (p. 3/4 3-6)

(h) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 30% of turbine first stage pressure in psig, at valves wide open turbine throttle steam flow, equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER. To allow for instrumentation accuracy, calibration and drift a set point of 25.4% of scale of turbine first stage pressure in psig is used.

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TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)	
1. Intermediate Range Monitors:		
a. Neutron Flux - High	NA	
b. Inoperative	NA	
2. Average Power Range Monitor*:		
a. Neutron Flux - High, Setdown	NA	
b. Flow Biased Simulated Thermal Power - High	≤ (0.09)(**)	CPS
c. Neutron Flux - High	≤ (0.09)	
d. Inoperative	NA	
3. Reactor Vessel Steam Dome Pressure - High	≤ ~~(0.35)~~ 0.33	CPS
4. Reactor Vessel Water Level - Low, Level 3	≤ ~~(0.30)~~ 1.03	
5. Reactor Vessel Water Level - High, Level 8	≤ ~~(0.30)~~ 1.03	
6. Main Steam Line Isolation Valve - Closure	≤ ~~(0.06)~~ 0.04	
7. Main Steam Line Radiation - High	NA	
8. Drywell Pressure - High	NA	
9. Scram Discharge Volume Water Level - High	NA	
10. Turbine Stop Valve - Closure	≤ (0.06)	CPS
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤ ~~(0.07)#~~ 0.05 #	CPS
12. Reactor Mode Switch Shutdown Position	NA	
13. Manual Scram	NA	

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. ~~(This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)~~ CPS

**(Not) including simulated thermal power time constant, 6 ± ~~1~~ 0.6 seconds.

#Measured from start of turbine control valve fast closure.

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TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

(g) Calibrate trip unit at least once per 31 days. ~~(BWR/6 relay only)~~ CPS

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

(i) This calibration shall consist of (verifying) ~~(adjustment, as required of)~~ the 6±~~1~~ 0.6 second simulated thermal power time constant.

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

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

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3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION: INSERT (NEXT PAGE)

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. CP

b. With the number of OPERABLE channels:

1. For a main steam line isolation trip function, one less than the Total Number of Channels, operation may proceed and the provisions of Specification 3.0.4 are not applicable provided the following conditions are satisfied:

   a) The inoperable channel is placed in the tripped condition within one hour.

   b) The Minimum OPERABLE Channels requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

2. For all other trip functions, less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

*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.

INSERT TO ACTION 3.3.2

p. 3/4 3-10

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. For the main steam line isolation Trip Functions:

1. With one of the four channels required for any Trip Function inoperable, operation may continue for 48 hours, after which time the inoperable channel shall be placed in the tripped condition.

2. With two of the four channels required for any Trip Function inoperable, place one channel in the tripped condition within one hour provided no tripped channel for that Trip Function already exists.

3. With three of the four channels required for any Trip Function inoperable take the ACTION required by Table 3.3.2-1.

c. For all other Trip Functions:

1. With less than the minimum number of OPERABLE channels required for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour.

2. With less than the minimum number of OPERABLE channels required for both trip systems, place one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

d. The provisions of Specification 3.0.4 are not applicable.

---

*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, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued) CPS

c. With the number of OPERABLE Channels:

1. For a main steam line isolation trip function, less than the Minimum OPERABLE Channels requirement, place at least one of the inoperable channels in the tripped condition and take the ACTION required by Table 3.3.2-1.

2. For all other trip functions, 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.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

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

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one ~~(both)~~ logic train such that all logic trains are tested at least once per 36 months and one channel per trip ~~(system)~~ function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip ~~(system)~~ function. CPS CPS

~~**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.~~ (SEE PREVIOUS/ABOVE INSERT) CPS

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TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	~~4, 5, 8,~~ ~~(b)(e)~~	2	1, 2, 3 and #	20
b. Drywell Pressure - High	~~2, 7, 8, 10, 11~~ ~~(b)(e)~~	2	1, 2, 3	20
~~c. Plant Exhaust Plenum Radiation - High~~	~~(6)~~ ~~(b)(e)~~	~~2~~	~~1, 2, 3 and *#~~	~~21~~ e
f ~~d.~~ Manual Initiation	~~2, 4, 5, (6), 7, 8, 10, 11~~	1/system	1, 2, 3 and *#	26

SEE INSERT (NEXT PAGE)

TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~ †	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM OPERABLE CHANNELS~~	APPLICABLE OPERATIONAL CONDITION	ACTION
2. MAIN STEAM LINE ISOLATION	DELETE THESE COLUMNS					
a. Reactor Vessel Water Level-Low Low Low, Level 1	~~1, 11~~	~~4~~	~~2~~	~~3~~	1, 2, 3	20
b. Main Steam Line Radiation - High (d)	~~1, 5~~	~~4~~	~~1~~	~~3~~	1, 2, 3	23
c. Main Steam Line Pressure - Low	~~1~~	~~4~~	~~2~~	~~3~~	1	23
d. Main Steam Line Flow - High	~~1~~	~~4~~	~~2~~ ~~(c)~~	~~3~~	1	23
e. Condenser Vacuum - Low	~~1~~	~~4~~	~~2~~	~~3~~	1, 2,** 3**	23
f. Main Steam Line Tunnel Temperature - High	~~1~~	~~4~~	~~2~~	~~3~~	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp. - High	~~1~~	~~4~~	~~2~~	~~3~~	1, 2, 3	23
h. Main Steam Line Turbine Bldg. Temperature - High	~~1~~	~~4~~	~~2~~	~~3~~	1, 2, 3	23
i. Manual Initiation	~~1~~	~~2/system~~	~~1/system~~	~~2/system~~	1, 2, 3	22

† All Functional Units have "any-two-from-four" isolation logic.

CPS CPS CPS CPS CPS CPS CPS

DEC. 8 1982

INSERT TO TABLE 3.3.2-1

1. PRIMARY CONTAINMENT ISOLATION

Parts c, d and e

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITIONS	ACT
1. PRIMARY CONTAINMENT ISOLATION			
c. Containment Building Fuel Transfer Ventilation Plenum Radiation - High	2	1, 2, 3 and *	21
d. Containment Building Exhaust Radiation - High	2	1, 2, 3 and *	21
e. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation - High	2	1, 2, 3 and *	21

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	~~6(b)(e)(f)~~	2	1, 2, 3, and #	25
b. Drywell Pressure - High	~~6(b)(e)(f)~~	2	1, 2, 3	25
~~c. Fuel Handling Area Ventilation Exhaust Radiation - High High~~	~~6(b)(f)~~	~~2~~	~~1, 2, 3, and *~~	~~25~~
~~d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High~~	~~6(b)(f)~~	~~2~~	~~1, 2, 3, and *~~	~~25~~
g. ~~e.~~ Manual Initiation	~~6(d)~~ ~~6(d)~~	~~(1/group)~~ ~~(1/group)~~	1, 2, 3 *	~~(26)~~ ~~(25)~~
~~f.~~				
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	~~4~~	~~(1)~~ 1/valve	1, 2, 3	27
b. Δ Flow Timer	~~4~~	~~1~~ 1/valve	1, 2, 3	27
c. Equipment Area Temperature - High	~~4~~	~~(1)~~ 8/valve(j)	1, 2, 3	27
d. Equipment Area Δ Temp. - High	~~4~~	~~(1)~~ 8/valve(j)	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	~~4~~	~~2~~ 2/valve	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	~~4~~	~~(1)~~ 1/valve	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	~~4~~	~~(1)~~ 1/valve	1, 2, 3	27
h. SLCS Initiation (g)	~~4(g)~~	~~NA~~ 1	1, 2, 3	27
i. Manual Initiation	~~4~~	~~1/system~~ 1/valve	1, 2, 3	26

SEE INSERT (NEXT PAGE)

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INSERT TO TABLE 3.3.2-1

3. SECONDARY CONTAINMENT ISOLATION

Parts c, d, e and f

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTI
3. SECONDARY CONTAINMENT ISOLATION			
c. Containment Building Fuel Transfer Ventilation Plenum Radiation - High	2	1, 2, 3 and *	25
d. Containment Building Exhaust Radiation - High	2	1, 2, 3 and *	25
e. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation - High	2	1, 2, 3 and *	25
f. Fuel Building Ventilation Exhaust Radiation - High	2	***	25

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				CPS
a. RCIC Steam Line Flow - High	6	2	1, 2, 3	27
b. RCIC Steam Line Flow - High Timer		1	1, 2, 3	27 CPS
c. ~~b.~~ RCIC Steam Supply Pressure - Low	6 ((h))	1	1, 2, 3	27
d. ~~c.~~ RCIC Turbine Exhaust Diaphragm Pressure - High	6	2	1, 2, 3	27 CPS
e. ~~d.~~ RCIC Equipment Room Ambient Temperature - High	6	1	1, 2, 3	27 CPS
f. ~~e.~~ RCIC Equipment Room Δ Temp. - High	6	1	1, 2, 3	27 CPS
g. ~~f.~~ Main Steam Line Tunnel Ambient Temperature - High	6	1	1, 2, 3	27 CPS
h. ~~g.~~ Main Steam Line Tunnel Δ Temp. - High	6	1	1, 2, 3	27 CPS
i. ~~h.~~ Main Steam Line Tunnel Temperature Timer	6	1	1, 2, 3	27 CPS
~~i. RHR Equipment Room Ambient Temperature - High~~	6	~~1~~	~~1, 2, 3~~	~~27~~ CPS
~~j. RHR Equipment Room Δ Temp. - High~~	6	~~1~~	~~1, 2, 3~~	~~27~~ CPS
j. ~~k.~~ Drywell Pressure - High[(h)]	6	1	1, 2, 3	27 CPS
k. ~~l.~~ Manual Initiation	6 ((i))	1~~/system~~	1, 2, 3	26 CPS

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL (a)~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	~~2, 3~~	~~(1)~~ 2	1, 2, 3	28
b. RHR Equipment Area Δ Temp. - High	~~2, 3~~	~~(1)~~ 2	1, 2, 3	28
c. RHR/RCIC Steam Line Flow - High	~~(5)~~	~~2~~ 1	1, 2, 3	28
d. Reactor Vessel Water Level - Low, Level 3	~~2, 3~~	2	1, 2, 3	28
e. Reactor Vessel Water Level - Low Low Low, Level 1[k]	~~(5)~~	~~(1)~~ 2	1, 2, 3	28
f. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	~~3~~	2	1, 2, 3	28
g. Drywell Pressure - High[k]	~~(5)~~	2	1, 2, 3	28
h. Manual Initiation	~~2, 3~~	1~~/system~~	1, 2, 3	26

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CPS

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

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

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

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

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

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

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

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

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

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

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

NOTES

primary and/or secondary

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

** May be bypassed ~~with reactor steam pressure ≤(1043) psig and all turbine stop valves closed.~~ by use of keylock bypass switches (S24B). CPS

SEE INSERT #1

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

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

(b) Also actuates the standby gas treatment system.

(c) Also actuates the control room emergency filtration system in the isolation mode of operation.

(d) Also trips and isolates the mechanical vacuum pumps.

~~(e) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.~~ Delete CPS

(f) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.

SEE INSERT #2 CPS

~~(g) Closes only RWCU system isolation valve(s)~~

~~((h) Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.)~~

~~((i) Manual initiation isolates ___ only and only with a coincident reactor vessel water level-low, level 3.)~~

INSERTS TO NOTES FOR TABLE 3.3.2-1

(p. 3/4 3-16)

INSERT # 2

(g) Manual Switch closes RWCU system inboard isolation valves F001, F028, F053, F040 and outboard isolation valves F004, F039, F034 and F054.

(h) Requires RCIC system steam supply pressure low coincident with drywell pressure high for isolation of vacuum breaker isolation valves.

(i) Manual initiation isolates outboard steam supply line isolation valve (F064) and the RCIC pump suction from suppression pool valve (F031) only, and only following a manual or automatic (Reactor Vessel Water Level 2) initiation of the RCIC system.

(j) There are eight temperature sensor locations, each capable of closing inboard valves F001, F028, F053 and F040. Similarly, there are eight temperature sensor locations closing outboard valves F004, F039, F034, and F054. For each location both ambient and delta temperature instrumentation is provided.

(k) Isolates containment spray valve F028 only for ten minutes following LOCA.

INSERT # 1

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

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TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
1. PRIMARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ ~~(51)~~ 45.5 inches*	$\geq$ ~~(53)~~ 47.7 inches	CPS
b. Drywell Pressure - High	$\leq$ ~~1.73~~ 1.68 psig	$\leq$ ~~1.93~~ 1.88 psig	CPS
SEE INSERT (NEXT PAGE) ~~(c. Plant Exhaust Plenum Radiation - High~~	~~$\leq$ ( ) mR/hr**~~	~~$\leq$ ( ) mR/hr**)~~	CPS
f ~~d.~~ Manual Initiation	NA	NA	CPS
2. MAIN STEAM LINE ISOLATION			
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ ~~(150)~~ 145.5 inches*	$\geq$ ~~(152)~~ 147.7 inches	CPS
b. Main Steam Line Radiation - High	$\leq$ ~~(2.5)~~ 3.0 x full power background	$\leq$ ~~(3.0)~~ 3.6 x full power background	CPS
c. Main Steam Line Pressure - Low	$\geq$ ~~(850)~~ 849 psig	$\geq$ ~~(838)~~ 837 psig	CPS
d. Main Steam Line Flow - High	$\leq$ ~~(203)~~ 170 psid	$\leq$ ~~(216)~~ 178 psid	CPS
e. Condenser Vacuum - Low	$\geq$ ~~(9.0)~~ 8.5 inches Hg. ~~(absolute pressure)~~ (vacuum)	$\geq$ ~~(8.7)~~ 7.6 inches Hg. ~~(absolute pressure)~~ (vacuum)	CPS
f. Main Steam Line Tunnel Temperature - High	$\leq$ ( )°F LATER	$\leq$ ( )°F.	CPS
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ ~~( )~~ 54.5°F	$\leq$ ~~( )~~ 60°F	CPS
h. Main Steam Line Turbine Bldg. Temperature - High	$\leq$ ~~( )~~ 131.2°F	$\leq$ ~~( )~~ 138°F	CPS
i. Manual Initiation	NA	NA	

DEC. 8 1982

(p. 3/4 3-17)

INSERT FOR TABLE 3.3.2-2

Trip Function	Trip Setpoint	Allowable Value
c. Containment Bldg. Fuel Transfer Ventilation Plenum Radiation - High	$\leq 100 \frac{mR}{hr}$	$\leq 500 \frac{mR}{hr}$
d. Containment Bldg. Exhaust Radiation - High	$\leq 100 \frac{mR}{hr}$	$\leq 500 \frac{mR}{hr}$
e. Containment Bldg. Continuous Containment Purge (CCP) Exhaust Radiation - High	(LATER)	

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
3. SECONDARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low Low, Level 2	≥ - ~~(51)~~ 45.5 inches*	≥ - ~~(53)~~ 47.7 inches	CPS
b. Drywell Pressure - High	≤ ~~1.73~~ 1.68 psig	≤ ~~1.93~~ 1.88 psig	CPS
~~c. Fuel Handling Area Ventilation Exhaust Radiation - High High~~	~~≤ (4.5) mR/hr**~~	~~≤ (5.5) mR/hr**~~	CPS
~~d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High~~	~~≤ (35) mR/hr**~~	~~≤ (35) mR/hr**~~	
~~e.~~ g. Manual Initiation	NA	NA	CPS
~~f.~~	AA	AA	CPS
4. REACTOR WATER CLEANUP SYSTEM ISOLATION			
a. Δ Flow - High	≤ ~~(68)~~ 55 gpm	≤ ~~(77)~~ 62.1 gpm	CPS
b. Δ Flow Timer	≥ 45 ~~( ) ± ( )~~ seconds	≤ 47 ~~( ) ± ( )~~ seconds	CPS
c. Equipment Area Temperature - High	≤ ( )°F LATER	≤ ( )°F	CPS
d. Equipment Area Δ Temp. - High	≤ ( )°F	≤ ( )°F	CPS
e. Reactor Vessel Water Level - Low Low, Level 2	≥ - ~~(51)~~ 45.5 inches*	≥ - ~~(53)~~ 47.7 inches	CPS
f. Main Steam Line Tunnel Ambient Temperature - High	≤ ( )°F LATER	≤ ( )°F	CPS
g. Main Steam Line Tunnel Δ Temp. - High	≤ ~~( )~~ 54.5°F	≤ ~~( )~~ 60°F	CPS
h. SLCS Initiation	NA	NA	
i. Manual Initiation	NA.	NA	

SEE INSERT (NEXT PAGE)

INSERT FOR TABLE 3.3.2-2, p. 3/4 3-18

	Trip Setpoint	Allowable Value
c. Containment Bldg. Fuel Transfer Ventilation Plenum Radiation - High	100 $\frac{mR}{hr}$	500 $\frac{mR}{hr}$
d. Containment Bldg. Exhaust Radiation - High	100 $\frac{mR}{hr}$	500 $\frac{mR}{hr}$
e. Containment Bldg. Continuous Containment Purge (CCP) Exhaust Radiation - High	(LATER)	
f. Fuel Bldg. Ventilation Exhaust Radiation - High	10 $\frac{mR}{hr}$	17 $\frac{mR}{hr}$

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION			
a. RCIC Steam Line Flow - High	$\leq$ ~~(310)~~ 257.5" H_2O	$\leq$ ~~(324)~~ 266" H_2O	CPS
b. RCIC Steam Line Flow - High Timer	$\geq$ 3 seconds	$\leq$ 13 seconds	
c. ~~b.~~ RCIC Steam Supply Pressure - Low	$\geq$ ~~(60)~~ psig	$\geq$ ~~(53)~~ 52 psig	
d. ~~c.~~ RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ ~~(10)~~ psig	$\leq$ ~~(20)~~ psig	CPS
e. ~~d.~~ RCIC Equipment Room Ambient Temperature - High	$\geq$ ( )°F $\leq$ ( )°F LATER	$\geq$ ( )°F $\leq$ ( )°F	CPS
f. ~~e.~~ RCIC Equipment Room Δ Temp. - High	$\leq$ ~~( )~~ 21°F	$\leq$ ~~( )~~ 26.5°F	CPS
g. ~~f.~~ Main Steam Line Tunnel Ambient Temperature - High	$\leq$ ( )°F LATER	$\leq$ ( )°F	CPS
h. ~~g.~~ Main Steam Line Tunnel Δ Temp. - High	$\leq$ ~~( )~~ 54.5°F	$\leq$ ~~( )~~ 60°F	CPS
i ~~h.~~ Main Steam Line Tunnel Temperature Timer	30 minutes ~~( ) ± ( ) seconds~~	~~( ) ± ( ) seconds~~	CPS
j ~~i.~~ RHR Equipment Room Ambient Temperature - High	$\leq$ ( )°F LATER	$\leq$ ( )°F	CPS
~~j.~~ RHR Equipment Room Δ Temperature - High	$\leq$ ~~( )~~ 74.2°F	$\leq$ ~~( )~~ 79.6°F	CPS
~~k.~~ Drywell Pressure - High	$\leq$ ~~1.73~~ 1.68 psig	$\leq$ ~~1.93~~ 1.88 psig	CPS
~~l.~~ Manual Initiation	NA	NA	CPS

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
6. RHR SYSTEM ISOLATION			
a. RHR Equipment Area Ambient Temperature - High	≤ ( )°F LATER	≤ ( )°F	CPS
b. RHR Equipment Area Δ Temperature - High	≤ 74.2 ~~( )~~°F	≤ 79.6 ~~( )~~°F	CPS
c. RHR/RCIC Steam Line Flow - High	≤ 179.5 ~~(140)~~" H_2O	≤ 188 ~~(151)~~" H_2O	CPS
d. Reactor Vessel Water Level - Low, Level 3	≥ 8.9 ~~10~~ inches*	≥ 8.3 ~~9.4~~ inches*	CPS
e. Reactor Vessel Water Level - Low Low Low, Level 1	≥ - 145.5 ~~(150)~~ inches*	≥ - 147.7 ~~(152)~~ inches*	CPS
f. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 135 psig**	≤ 150 psig**	CPS
g. Drywell Pressure - High	≤ 1.68 ~~1.73~~ psig	≤ 1.88 ~~1.93~~ psig	CPS
h. Manual Initiation	NA	NA	

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

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

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)~~#~~
~~1. PRIMARY CONTAINMENT ISOLATION~~	
~~a. Reactor Vessel Water Level - Low Low, Level 2~~	~~≤ (13)[(a)]~~
~~b. Drywell Pressure - High~~	~~≤ (13)[(a)]~~
~~(c. Plant Exhaust Plenum Radiation - High[(b)]~~	~~≤(13)[(a)]~~
~~d. Manual Initiation~~	~~NA~~
} DELETE	
~~2.~~ MAIN STEAM LINE ISOLATION	
1 ~~a.~~ Reactor Vessel Water Level - Low Low Low, Level 1	~~(~~1.0~~)~~ ~~*/≤ (13)[(a)]**~~
~~b. Main Steam Line Radiation - High[(b)]~~	~~(1.0) */≤ (13)[(a)]**~~
2 ~~c.~~ Main Steam Line Pressure - Low	~~(~~1.0~~)~~ ~~*/≤ (13)[(a)]**~~
3 ~~d.~~ Main Steam Line Flow - High	~~(~~0.5~~)~~ ~~*/≤ (13)[(a)]**~~
~~e. Condenser Vacuum - Low~~	~~(NA)~~
~~f. Main Steam Line Tunnel Temperature - High~~	~~(NA)~~
~~g. Main Steam Line Tunnel Δ Temp. - High~~	~~(NA)~~
~~h. Main Steam Line Turbine Bldg.~~	~~NA~~
~~i. Manual Initiation~~	~~NA~~
~~3. SECONDARY CONTAINMENT ISOLATION~~	
~~a. Reactor Vessel Water Level - Low Low, Level 2~~	~~≤(13)[(a)]~~
~~b. Drywell Pressure - High~~	~~≤ (13)[(a)]~~
~~c. Fuel Handling Area Ventilation Exhaust Radiation - High High(b)~~	~~≤ (13)[(a)]~~
~~d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High(b)~~	~~≤ (13)[(a)]~~
~~e. Manual Initiation~~	~~NA~~
~~f. ______~~	~~______~~
~~4. REACTOR WATER CLEANUP SYSTEM ISOLATION~~	
~~a. Δ Flow - High~~	~~≤ (13)[(a)(##)]~~
~~b. Δ Flow Timer~~	~~(NA)~~
~~c. Equipment Area Temperature - High~~	~~(NA)~~
~~d. Equipment Area Δ Temp. - High~~	~~(NA)~~
~~e. Reactor Vessel Water Level - Low Low, Level 2~~	~~≤ (13)[(a)]~~
~~f. Main Steam Line Tunnel Ambient Temperature - High~~	~~(NA)~~
~~g. Main Steam Line Tunnel Δ Temp. - High~~	~~(NA)~~
~~h. SLCS Initiation~~	~~NA~~
~~i. Manual Initiation~~	~~NA~~

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TABLE 3.3 2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	≤ (13)(a)(###)
b. RCIC Steam Supply Pressure - Low	≤ (13)(a)
c. RCIC Turbine Exhaust Diaphragm Pressure - High	(NA)
d. RCIC Equipment Room Ambient Temperature - High	(NA)
e. RCIC Equipment Room Δ Temp. - High	(NA)
f. Main Steam Line Tunnel Ambient Temp. - High	(NA)
g. Main Steam Line Tunnel Δ Temp. - High	(NA)
h. Main Steam Line Tunnel Temperature Timer	(NA)
i. RHR Equipment Room Ambient Temperature - High	(NA)
j. RHR Equipment Room Δ Temp. - High	(NA)
k. Drywell Pressure - High	NA
l. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	(NA)
b. RHR Equipment Area Δ Temp. - High	(NA)
(c. RHR/RCIC Steam Line Flow - High	(NA))
d. Reactor Vessel Water Level - Low, Level 3	≤ (13)(a)
(e. Reactor Vessel Water Level - Low Low Low, Level 1	≤ (13)(a))
f. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	(NA)
(g. Drywell Pressure - High	NA)
h. Manual Initiation	NA

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.

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

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

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

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

(##Time delay of ( ) seconds.)

(###Time delay of ( ) seconds.)

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TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	~~OPERATIONAL~~ OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and #
b. Drywell Pressure - High	S	M	R	1, 2, 3
SEE INSERT (NEXT PAGE) ~~(c. Plant Exhaust Plenum Radiation - High~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and *)~~
f. ~~d.~~ Manual Initiation	NA	~~(M)[(a)](R)~~	NA	1, 2, 3 and ~~*#~~
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h. Main Steam Line Turbine Bldg. Temperature - High	NA	M	Q	1, 2, 3
i. Manual Initiation	NA	~~(M)[(a)](R)~~	NA	1, 2, 3

INSERT TO TABLE 4.3.2.1-1, p. 3/4 3-23

UNDER 1. PRIMARY CONTAINMENT ISOLATION

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERAT NAL CONDITIONS IN WHIC SURVEILLANCE REQUIRE
c. Containment Bldg. Fuel Transfer Ventilation Plenum Radiation - High	S	M	R	1, 2, 3 and *
d. Containment Bldg. Exhaust Radiation - High	S	M	R	1, 2, 3 and *
e. Containment Bldg. Continuous Containment Purge (CCP) Exhaust Radiation - High	S	M	R	1, 2, 3 and *

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TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and #
b. Drywell Pressure - High	S	M	R	1, 2, 3
SEE INSERT (NEXT PAGE) ~~c. Fuel Handling Area Ventilation Exhaust Radiation - High High~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and *~~
~~d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and *~~
g ~~e.~~ Manual Initiation	NA	~~(M[a])~~ (R)[e]	NA	1, 2, 3 and *
~~f.~~				
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Δ Flow Timer	NA	M	Q	1, 2, 3
c. Equipment Area Temperature - High	NA	M	Q	1, 2, 3
d. Equipment Area Ventilation Δ Temp. - High	NA	M	Q	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	NA	M	Q	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	NA	M	Q	1, 2, 3
h. SLCS Initiation	NA	M(b)	NA	1, 2, 3
i. Manual Initiation	NA	~~(M[a])~~ (R)[e]	NA	1, 2, 3

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INSERT TO TABLE 4.3.2.1-1, p. 3/4 3-24

UNDER 3. SECONDARY CONTAINMENT ISOLATION

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN W[illegible] SURVEILLANCE REQUI[illegible]
c. Containment Bldg. Fuel Transfer Ventilation Plenum Radiation - High	S	M	R	1,2,3 and *
d. Containment Bldg. Exhaust Radiation - High	S	M	R	1,2,3 and *
e. Containment Bldg. Continuous Containment Purge (CCP) Exhaust Radiation - High	S	M	R	1,2,3 and *
f. Fuel Bldg. Ventilation Exhaust Radiation - High	S	M	R	1,2,3 and *

INSERT TO NOTES TO TABLE 4.3.2.1-1

** May be bypassed by use of keylock switch S21B

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3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

c. With either ADS trip system "~~A~~ 1" or "~~B~~ 2" inoperable, restore the inoperable trip system to OPERABLE status within: CPS

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

2. 72 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP (SYSTEM) (FUNCTION)[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION	
A. DIVISION 1 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 (Minimum Flow)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~31~~	CPS
c ~~d.~~ Reactor Vessel Pressure-Low (LPCS Permissive)	~~(1)~~ 4	1, 2, 3 4*, 5*	~~32~~ 33 33	CPS
d ~~e.~~ Reactor Vessel Pressure-Low (LPCI Permissive)	~~(1)~~ 4	1, 2, 3 4*, 5*	~~32~~ 33 33	CPS
e ~~f.~~ LPCI Pump A Start Time Delay Relay	~~(~~1~~)~~	1, 2, 3, 4*, 5*	32	CPS
~~g. LPCI Pump A Discharge Flow-Low (Minimum Flow)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~31~~	
~~h.~~ Division 1 Bus Power Monitor##	(2)	1, 2, 3, 4*, 5*	34	
~~i.~~ Manual Initiation	2 ~~(1)/(system)~~	1, 2, 3, 4*, 5*	35	
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "~~A~~ 1"# ADS LOGIC 'A' AND 'E'				CPS
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~~	1, 2, 3	32	CPS
f. LPCI Pump A Discharge Pressure-High (Permissive)	2 ~~1~~	1, 2, 3	32	CPS
g. Manual Initiation	2 ~~1/valve~~	1, 2, 3	35	

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

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

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP ~~(SYSTEM)~~ (FUNCTION)(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	~~(2)~~ 4(b)	1, 2, 3, 4*, 5*	~~(30)~~ 36
b. Drywell Pressure - High	~~(2)~~ 4(b)	1, 2, 3	~~(30)~~ 36
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	32
d. RCIC 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 (Minimum Flow)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~[illegible]~~
~~g. HPCS System Flow Rate Low (Minimum Flow)~~	~~(1)~~	~~1, 2, 3, 4*, 5*~~	~~[illegible]~~
h. HPCS Bus Power Monitor##	(1)	1, 2, 3, 4*, 5*	34
~~i.~~ Manual Initiation	~~1~~ 2	1, 2, 3, 4*, 5*	35

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	1/bus	1/bus	1/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)

(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 (and the suppression pool makeup system).
(c) Provides signal to close HPCS pump ~~discharge~~ valve only.
(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.

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

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

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

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

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

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

ACTION 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: CPS

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.* CPS

ACTION ~~(36)~~ (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. CPS

ACTION ~~(37)~~ (38) - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate. CPS

ACTION ~~(38)~~ (39) - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour*; operation may than continue until performance of the next required CHANNEL FUNCTIONAL TEST. CPS

*The provisions of Specification 3.0.4 are not appliable.

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TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
A. DIVISION 1 TRIP SYSTEM			
1. RHR-A (LPCI MODE) AND LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ ~~(150)~~ 145.5 inches*	≥ ~~(162)~~ 147.7 inches	CPS
b. Drywell Pressure - High	≤ ~~(1.89)~~ 1.68 psig	≤ ~~(1.94)~~ 1.88 psig	CPS
~~c. LPCS Pump Discharge Flow-Low~~	~~≥ ( ) gpm~~	~~≥ ( ) gpm~~	
d. Reactor Vessel Pressure-Low (LPCS Permissive)	≥ (484) psig, decreasing	≥ (469) psig, decreasing	
e. Reactor Vessel Pressure-Low (LPCI Permissive)	≥ (484) psig, decreasing	≥ (469) psig, decreasing	
f. LPCI Pump A Start Time Delay Relay	≤ (5) seconds	≤ ( ) seconds	
~~g. LPCI Pump A Discharge Flow-Low~~	~~≥ ( ) gpm~~	~~≥ ( ) gpm~~	CPS
~~h.~~ Division 1 Bus Power Monitor	≥ ( ) (volts)	≥ ( ) (volts)	
~~i.~~ Manual Initiation	NA	NA	
	LATER		
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM ~~"A"~~ 1 ADS LOGIC 'A' AND 'E'			CPS
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ ~~(160)~~ 145.5 inches*	≥ ~~(152)~~ 147.7 inches	
b. Drywell Pressure - High	≤ ~~(1.89)~~ 1.68 psig	≤ ~~(1.94)~~ 1.88 psig	CPS
c. ADS Timer	~~(≥ 90,)~~ ≤ (105) seconds	~~(≥ 90,)~~ ≤ (117) seconds	
d. Reactor Vessel Water Level-Low, Level 3	≥ ~~(11.4)~~ 8.9 inches*	≥ ~~(10.8)~~ 8.3 inches	
e. LPCS Pump Discharge Pressure-High	≥ (145) psig, increasing	≥ ~~(140)~~ psig, increasing	
f. LPCI Pump A Discharge Pressure-High	≥ (125) psig, increasing	≥ ~~(122)~~ 125 psig, increasing	
g. Manual Initiation	NA	NA 115	

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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	≥ ~~(150)~~ 145.5 inches*	≥ ~~(152)~~ 147.7 inches	CPS
b. Drywell Pressure - High	≤ ~~(1.89)~~ 1.68 psig	≤ ~~(1.94)~~ 1.88 psig	CPS
c. Reactor Vessel Pressure-Low (LPCI Permissive)	≥ ~~( )~~ 484 psig, decreasing	≥ ~~( )~~ 469 psig, decreasing	
d. LPCI Pump ~~(B)~~ Start Time Delay Relay	≤ (5) seconds	≤ ( ) seconds	
~~e. LPCI Pump Discharge Flow-Low~~	~~≥ ( ) gpm~~	~~≥ ( ) gpm~~	CPS
~~f.~~ Division 2 Bus Power Monitor	≥ ( ) (volts)	≥ ( ) (volts)	
~~g.~~ Manual Initiation	NA	NA	
	LATER	LATER	
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM ~~"B"~~ 2 ADS LOGIC 'B' AND 'F'			CPS
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ ~~(150)~~ 145.5 inches*	≥ ~~(152)~~ 147.7 inches	
b. Drywell Pressure - High	≤ ~~(1.89)~~ 1.68 psig	≤ ~~(1.94)~~ 1.88 psig	CPS
c. ADS Timer	~~(≥ 90,)~~ ≤ 105 seconds	~~(≥ 90,)~~ ≤ 117 seconds	
d. Reactor Vessel Water Level-Low, Level 3	≥ ~~(11.4)~~ 8.9 inches*	≥ ~~(10.8)~~ 8.3 inches	
e. LPCI Pump (B and C) Discharge Pressure-High	≥ 125 psig, increasing	≥ ~~(125)~~ 115 psig, increasing	
f. Manual Initiation	NA	NA	

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ ~~(51)~~ 45.5 inches*	≥ ~~(53)~~ 47.7 inches
b. Drywell Pressure - High	≤ ~~(1.89)~~ 1.68 psig	≤ ~~(1.94)~~ 1.88 psig
c. Reactor Vessel Water Level - High, Level 8	≤ ~~(52)~~ 52.0 inches*	≤ ~~(52.6)~~ 54.2 inches
d. RCIC Storage Tank Level - Low	≥ ~~(X+3)~~ inches**	≥ ~~(X)~~ inches**
e. Suppression Pool Water Level - High	≤ ~~(Y-3)~~ Y-1.1 inches#	≤ ~~(Y)~~ inches#
~~f. Pump Discharge Pressure - High~~	~~≥ ( ) psig~~	~~≥ ( ) psig~~
~~g. HPCS System Flow Rate - Low~~	~~≥ ( ) gpm~~	~~≥ ( ) gpm~~
~~h.~~ HPCS Bus Power Monitor	≥ ( ) (volts)	≥ ( ) (volts)
~~i.~~ Manual Initiation	NA	NA
	LATER	
D. LOSS OF POWER		
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage) (###)	a. 4.16 kv Basis - (2940)±(161) volts	(2940)±(315) volts
	b. 120 v Basis - (84)±(4.6) volts	(84)±(9) volts
	c. ≤ (10) sec. time delay	≤ (10) sec. time delay
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - (3727)±(9) volts	(3727)±(21) volts
	b. 120 v Basis - (106.5)±(0.25) volts	(106.5)±(0.60) volts
	c. (10)±(0.5) sec. time delay	(10)±(1) sec. time delay

CPS

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

**X is the ~~value that ensures adequate NPSH and precludes air entry due to vortexing.)~~ SEE INSERT (NEXT PAGE).

#Y is ~~(5) inches above normal water level.)~~ SEE INSERT (NEXT PAGE).

###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.

CPS

INSERT FOR NOTES TO TABLE 3.3.3-2
BOTTOM OF p. 3/4 3-34

**X is the minimum level to allow time to switch to suppression pool suction.

Y is suppression pool high water level.

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TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

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

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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 1 TRIP SYSTEM					
1. RHR-A (LPCI MODE) AND LPCS SYSTEM					
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*	
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3	
~~c. LPCS Pump Discharge Flow-Low~~	~~NA~~	~~M~~	~~Q~~	~~1, 2, 3, 4*, 5*~~	CPS
c ~~d.~~ Reactor Vessel Pressure-Low (LPCS)	S	M	R(a)	1, 2, 3, 4*, 5*	
d ~~e.~~ Reactor Vessel Pressure-Low (LPCI)	S	M	R(a)	1, 2, 3, 4*, 5*	
e ~~f.~~ LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*	
~~g. LPCI Pump A Flow-Low~~	~~NA~~	~~M~~	~~Q~~	~~1, 2, 3, 4*, 5*~~	CPS
~~h.~~ Division I Bus Power Monitor	NA	M	(NA)	1, 2, 3, 4*, 5*	
~~i.~~ Manual Initiation	NA	~~M(b)~~ ~~(R)~~	NA	1, 2, 3, 4*, 5*	
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM ~~"A"~~ # 1					CPS
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3	
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3	
c. ADS Timer	NA	M	Q	1, 2, 3	
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3	
e. LPCS Pump Discharge Pressure-High	~~NA~~ S	M	~~Q~~ R(a)	1, 2, 3	CPS
f. LPCI Pump A Discharge Pressure-High	~~NA~~ S	M	R ~~Q~~(a)	1, 2, 3	CPS
g. Manual Initiation	NA	~~M(b)~~ ~~(R)~~	NA	1, 2, 3	

ADS LOGIC 'A' AND 'E'

CPS

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TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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INFORMATION FROM [illegible] TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION ~~(Optional)~~ CPS

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

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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

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

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ 45.5 ~~(51)~~ inches*	~~≥ (53) inches~~ N/A	CPS
2. Reactor Vessel Pressure - High	≤ 1127 ~~(1135)~~ psig	~~≤ (1150) psig~~ N/A	CPS

*See Bases Figure B3/4 3-1.

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TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

*Calibrate trip unit at least once per 31 days

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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 ~~25%~~ 40% of RATED THERMAL POWER. CPS

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.

REPLACE WITH INSERT (NEXT PAGE)

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 (or both) trip system(s) 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).

INSERT FOR ACTION
STATEMENTS TO 3.3.4.2
p. 3/4 3-43

b. With one of the four channels required for any Trip Function inoperable, operation may continue for 48 hours after which time the inoperable channel shall be placed in the tripped condition.

c. With two of the four channels required for any Trip Function inoperable, place one channel in the tripped condition within one hour provided no tripped channel for that Trip Function exists.

d. With three of the four channels required for any Trip Function inoperable, reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

e. The provisions of specification 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

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

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

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

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TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	SPECIFIC CHANNEL OR FUNCTIONAL REQUIREMENTS (a)	~~MINIMUM OPERABLE CHANNELS (PER TRIP SYSTEM)(a)~~	
			CPS
1. Turbine Stop Valve - Closure		~~2~~(b)	CPS
2. Turbine Control Valve-Fast Closure		~~2~~(b)	CPS

(a) All Trip Functions have "any-two-from-four" logic. CPS

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

(b)This function shall be automatically bypassed when turbine first stage pressure is less than or equal to ~~(250) psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.~~ 130% of turbine first stage pressure in psia at valves wide open turbine throttle steam flow equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER. To allow for instrumentation accuracy, calibration and drift a setpoint of 25.4% of turbine first stage pressure is used. CPS

DRAFT

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
1. Turbine Stop Valve-Closure	≤ ~~(~~5~~)~~% closed	≤ ~~(~~7~~)~~% closed	CPS
2. Turbine Control Valve-Fast Closure	≥ ~~(500)~~ 530 psig	≥ ~~(414)~~ 465 psig	CPS

DRAFT

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milliseconds)	
1. Turbine Stop Valve-Closure	≤ ~~(104)~~ 140	CPS
2. Turbine Control Valve-Fast Closure	≤ ~~(~~140~~)~~	CPS

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#

CPS

#Calibrate trip unit at least once per 31 days.

DRAFT

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 ~~130~~ 150 psig. CPS

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS		TRIP SETPOINT	ALLOWABLE VALUE	
a.	Reactor Vessel Water Level - ~~(~~Low Low, Level 2~~)~~	≥ -~~(51)~~ 45.5 inches*	≥ -~~(53)~~ 47.7 inches	CPS
b.	Reactor Vessel Water Level - High, Level ~~(~~8~~)~~	≤ ~~(52)~~ 52.0 inches*	≤ ~~(52.6)~~ 54.2 inches	CPS
c.	RCIC Storage Tank Level - Low	≥ ~~(14)~~ X+3 inches**	≥ ~~(9)~~ X inches	CPS
d.	Suppression Pool Water Level - High	≤ ~~(5)~~ Y-1.1 inches***	≤ ~~(21)~~ Y inches	CPS
e.	Manual Initiation	NA	NA	

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

** X = Value that ensures NPSH and no vortexing. CPS

*** Y = High water level based on pool load analysis. CPS

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TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	
a. Reactor Vessel Water Level - (Low Low, Level 2)	S	M	R(a)	CPS
b. Reactor Vessel Water Level - High, Level (8)	S	M	R(a)	CPS
c. RCIC Storage Tank Level - Low	~~NA~~ S	M	~~Q~~ R	CPS
d. Suppression Pool Water Level - High	~~NA~~ S	M	~~Q~~ R	CPS
e. Manual Initiation	NA	~~M(b)~~ (R)	NA	CPS

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

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

DRAFT

NO CHANGE

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.

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TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION	
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	2	1, 2	60	
b. ~~Intermediate Rod Withdrawal Limiter Setpoint~~ RWL - High Power Setpoint	2	1, ~~2~~	60	CPS
2. APRM				
a. Flow Biased Neutron Flux - Upscale	~~6~~ 3	1	61	CPS
b. Inoperative	~~6~~ 3	1, 2, 5	61	
c. Downscale	~~6~~ 3	1	61	
d. Neutron Flux - Upscale, Startup	~~6~~ 3	2, 5	61	
3. SOURCE RANGE MONITORS				
a. Detector not full in[(a)]	~~4~~	~~2, 5~~	~~61~~	SEE INSERT (NEXT PAGE) CPS
b. Upscale[(b)]	~~4~~	~~2, 5~~	~~61~~	
c. Inoperative[(b)]	~~4~~	~~2, 5~~	~~61~~	
d. Downscale[(c)]	~~4~~	~~2, 5~~	~~61~~	
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[(d)]	6	2, 5	61	
b. Upscale	6	2, 5	61	
c. Inoperative	6	2, 5	61	
d. Downscale[(d)]	6	2, 5	61	
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	~~(~~2~~)~~	1, 2, 5*	62	CPS
b. Scram Trip Bypass	~~(~~2~~)~~	~~(1, 2,)~~ 3, 4, 5*	62	
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	2	1	62	
~~b. Inoperative~~	~~2~~	~~1~~	~~62~~	CPS
~~c. (Comparator) (Downscale)~~	~~2~~	~~1~~	~~62~~	
7. REACTOR MODE SWITCH				CPS
a. Shutdown Mode	2	3,4	62	
b. Refuel Mode	2	5	62	

CLINTON-1

INSERT TO ITEM 3, "SOURCE RANGE MONITORS", UNDER "CONTROL ROD BLOCK INSTRUMENTATION" TABLE 3.3.6-1
p. 3/4 3-55

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
3. SOURCE RANGE MONITORS			
a. Detector not full in [(a)]	3	2	61
	2	5	61
b. Upscale [(b)]	3	2	61
	2	5	61
c. Inoperative [(b)]	3	2	61
	2	5	61
d. Downscale [(c)]	3	2	61
	2	5	61

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3/4 3-55

DRAFT

NO CHANGE

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

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

ACTION 61 - With the number of OPERABLE Channels:

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

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

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

NOTES

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

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

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

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

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

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TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	(**) psig turbine first stage pressure ~~(20)% of RATED THERMAL POWER~~	(**) psig turbine first stage pressure ~~(20 + 15, - 0)% of RATED THERMAL POWER~~
b. ~~Intermediate Rod Withdrawal Limiter Setpoint~~ RWL - High Power Setpoint	~~≤ 70% of RATED THERMAL POWER~~ (**) psig turbine first stage pressure	~~≤ 70% of RATED THERMAL POWER~~ (**) psig turbine first stage pressure
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1 x 10^5 cps	≤ ~~1.5~~ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ ~~2~~ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 division of full scale	≤ (110/125) division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division of full scale	≥ (3/125) division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ (32.5) inches	≤ (34) inches
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ ~~(108)~~% of rated flow	≤ ~~(111)~~% of rated flow
~~b. Inoperative~~	~~NA~~	~~NA~~
~~c. (Comparator) (Downscale)~~	~~≤ (10)% flow deviation~~	~~≤ (11)% flow deviation~~
7. REACTOR MODE SWITCH		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

~~*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.~~

SEE INSERT/ADDITION (NEXT PAGE)

CPS

*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.

** Corresponding to analytical values of 20% +15%, -0% (Low Power Setpoint) and 70% +0%, -15% (Rod-High Power Setpoint). Percentage values are of RATED THERMAL POWER. First stage turbine pressure values to be determined during start-up test program.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

DRAFT

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION$^{(a)}$	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED	
1. ROD PATTERN CONTROL SYSTEM					
a. Low Power Setpoint	NA	S/U$^{(b)(e)}$ D$^{(c)}$,M$^{(d)(e)}$	Q	1, 2	
b. ~~Intermediate Rod Withdrawal Limiter Setpoint~~ RWL - High Power Setpoint	NA	S/U$^{(b)(e)}$ D$^{(c)}$,M$^{(d)(e)}$	Q	1~~, 2~~	CPS
2. APRM					
a. Flow Biased Neutron Flux - Upscale	~~(NA)~~	S/U$^{(b)}$,M	~~(Q)~~	1	CPS
b. Inoperative	NA	S/U$^{(b)}$,M	NA	1, 2, 5	
c. Downscale	~~(NA)~~	S/U$^{(b)}$,M	~~(Q)~~	1	CPS
d. Neutron Flux - Upscale, Startup	~~(NA)~~	S/U$^{(b)}$,M	~~(Q)~~	2, 5	
3. SOURCE RANGE MONITORS					
a. Detector not full in	NA	S/U$^{(b)}$,W	NA	2, 5	
b. Upscale	NA	S/U$^{(b)}$,W	Q	2, 5	
c. Inoperative	NA	S/U$^{(b)}$,W	NA	2, 5	
d. Downscale	NA	S/U$^{(b)}$,W	Q	2, 5	
4. INTERMEDIATE RANGE MONITORS					
a. Detector not full in	NA	S/U$^{(b)}$,W	NA	2, 5	
b. Upscale	NA	S/U$^{(b)}$,W	Q	2, 5	
c. Inoperative	NA	S/U$^{(b)}$,W	NA	2, 5	
d. Downscale	NA	S/U$^{(b)}$,W	Q	2, 5	
5. SCRAM DISCHARGE VOLUME					
a. Water Level-High	NA	~~(M)~~ ~~(Q)~~	R	1, 2, 5*	CPS
b. Scram Trip Bypass	NA	M	NA	~~(1, 2, 5)~~* 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~~	CPS
~~c. (Comparator) (Downscale)~~	~~NA~~	~~S/U$^{(b)}$,M~~	~~Q~~	~~1~~	
7. REACTOR MODE SWITCH					
a. Shutdown Mode	NA	R	NA	3,4	CPS
b. Refuel Mode	NA	R	NA	5	

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NO CHANGE

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

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

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

e. Includes reactor manual control multiplexing system input.

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

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3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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Table 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION	
~~1. Auxiliary Bldg/Fuel Handling Area Vent Exhaust Radiation Monitor~~	~~3~~	~~1,2,3 and *~~	~~≤ (4) mR/hr$^{(a)(b)}$~~	~~70~~	CPS
~~2. Off-Gas Post-treatment Radiation Monitor~~	~~2~~ ~~1~~	~~1, 2~~ ~~3 and **~~	~~≤ ( ) cpm$^{(c)}$~~ ~~≤ ( ) cpm~~	~~71~~ ~~72~~	CPS
1 ~~3.~~ Main Control Room Air Intake ~~Ventilation~~ Radiation Monitor	2/(intake)	1,2,3,5 and *	≤ 10 ~~(5)~~ mR/hr	~~73~~ 70	CPS
~~4. Auxiliary Bldg/Fuel Handling Area Pool Sweep Exhaust Radiation Monitors~~	~~3~~	~~***~~	~~≤ (35) mR/hr$^{(a)(b)}$~~	~~74~~	CPS
~~5. Standby Gas Treatment System Exhaust Radiation Monitor~~	~~1~~	~~1, 2, 3 and ****~~	~~≤ (5) mR/hr$^{(d)}$~~	~~75~~	CPS
2 ~~6.~~ Area Monitors					CPS
a. Criticality Monitors					
1) New Fuel Storage Vault	1	#	≤ 50 ~~(15)~~ mR/hr ~~(d)~~ **	~~76~~ 71	CPS
2) Spent Fuel Storage Pool	1	##	≤ 2.5 ~~(15)~~ mR/hr ~~(d)~~ **	~~76~~ 71	CPS
b. Control Room Direct Radiation Monitor	1	At all times	≤ 2.5 mR/hr **	~~76~~ 71	CPS

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Table 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

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

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

~~***With irradiated fuel in the spent fuel storage pool or building.~~

~~****When the standby gas treatment system is operating.~~

~~(a)Also isolates the primary and secondary containment purge and vent penetrations, valve group(s) ( ).~~

~~(b)Also starts the standby gas treatment system.~~

~~(c)Time delay before off-gas system discharge valve closure ____ ± ____ seconds.~~

** ~~(d)~~Alarm only.

~~(e)Also isolates the secondary containment purge and vent penetrations, valve group(s) ( ).~~

#With fuel in the new fuel storage vault.

##With fuel in the spent fuel storage pool.

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

~~ACTION 70 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours. In addition, with the plant vent stack (noble gas) monitor inoperable, restore the inoperable (noble gas) monitor to OPERABLE status within 24 hours or place the inoperable (noble gas) monitor in the tripped condition.~~

~~ACTION 71 -~~

~~a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within one hour.~~

~~b. With both of the required monitors inoperable, be in at least HOT SHUTDOWN within 12 hours.~~

~~ACTION 72 - With no monitors OPERABLE, determine that the off gas treatment system is not bypassed or be in at least COLD SHUTDOWN within 24 hours. Restore the inoperable monitors to OPERABLE status prior to entering OPERATING CONDITIONS 1, 2 and 3.~~

ACTION ~~73~~ 70 -

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

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

~~ACTION 74 -~~

~~a. With one of the required monitors inoperable, place the inoperable channel in the (downscale) tripped condition within one hour.~~

~~b. With two of the required monitors inoperable, initiate and maintain operation of at least one standby gas treatment subsystem within 12 hours.~~

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION (Continued)

~~ACTION 75 - With the required monitor inoperable, verify that the accident monitor on the release line shown in Table (3.3.7.5-1) is OPERABLE and restore the inoperable monitor to OPERABLE status within 7 days. Otherwise, declare the system inoperable.~~ CPS

ACTION ~~76~~ 71 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours. CPS

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Table 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	CONDITIONS IN WHICH SURVEILLANCE REQUIRED	
~~1. Auxiliary Bldg/Fuel Handling Area Vent Exhaust Radiation Monitor~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and *~~	CPS
~~2. Off-Gas Post-treatment Radiation Monitor~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and **~~	
1~~3~~. Main Control Room Air Intake ~~Ventilation~~ Radiation Monitor	S	M	R	1, 2, 3, 5 and *	CPS
~~4. Auxiliary Bldg/Fuel Handling Area Pool Sweep Exhaust Radiation Monitors~~	~~S~~	~~M~~	~~R~~	~~***~~	CPS
~~5. Standby Gas Treatment System Exhaust Radiation Monitor~~	~~S~~	~~M~~	~~R~~	~~1, 2, 3 and ****~~	
2~~6~~. Area Monitors					CPS
a. Criticality Monitors					
1) New Fuel Storage Vault	S	M	R	#	
2) Spent Fuel Storage Pool	S	M	R	##	
b. Control Room Direct Radiation Monitor	S	M	R	At all times	

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

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

~~***With irradiated fuel in the spent fuel storage pool or building.~~

~~****When the standby gas treatment system is operating.~~

#With fuel in the new fuel storage vault.

##With fuel in the spent fuel storage pool.

CPS

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SEISMIC MONITORING INSTRUMENTATION

NO CHANGE

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. CPS

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.

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TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSERT

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE	
1. Triaxial Time-History Accelerographs		SEE INSERT (NEXT PAGE)	CPS
a. Concrete pad (outside) El. 712'	~~0.02 to 20g~~	1	
b. Fuel Bldg., El. 712'	~~0.02 to 20g~~	1	
c. Containment Bldg., El. 851'	~~0.02 to 20g~~	1	
2. Triaxial Peak Accelerographs			
a. Fuel Bldg., El. 707'6"	~~0 to 5g~~	1	
b. DG & HVAC Bldg., El. 712'	~~0 to 5g~~	1	
c. Containment Bldg., El. 778'	~~0 to 5g~~	1	
3. Triaxial Seismic Switches			
a. Fuel Bldg., El. 712'	0.1 to 30 Hz/0.005 to 0.2g	1(a)(b)	CPS
4. Triaxial Response-Spectrum Recorders			
a. Circulating Water House	1 to 30 Hz ~~2 to 25 Hz~~	1~~(a)~~	CPS
b. Digital Cassette with Playback Feature (Active)	1 to 30 Hz	1	CPS

(a) With main ~~reactor~~ control room ~~indication and~~ annunciation. CPS

(b) Adjustable setpoint. CPS

INSERT TO TABLE 3.3.7.2-1

p. 3/4 3-68

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. Concrete pad (outside) El. 712'	1 to 30 Hz /	1[a]
b. Fuel Bldg., El. 712'	1 to 30 Hz /	1[a]
c. Containment Bldg., El. 851'	1 to 30 Hz /	1[a]
d. Containment Bldg., El. 778'	1 to 30 Hz /	1[a]
e. Control Bldg., El. 737'	1 to 30 Hz /	1[a]
2. Triaxial Peak Accelerographs (Passive)		
a. Fuel Bldg., El. 707'6"	0 to 20 Hz/0.02 to 3g	1
b. DG Bldg., El. 712'	0 to 20 Hz/0.02 to 3g	1
c. Containment Bldg., El. 778'	0 to 20 Hz/0.02 to 3g	1

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TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION—SURVEILLANCE REQUIREMENTS

SEE INSERT (NEXT PAGE)

INSTRUMENTS AND SENSOR LOCATIONS	~~CHANNEL CHECK~~	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	CP
1. Triaxial Time-History Accelerographs				CP
a. Concrete pad (outside), El. 712'	~~M~~	SA	R	
b. Fuel Bldg., El. 712'	~~M~~	SA	R	
c. Containment Bldg., El. 851'	~~M~~	SA	R	
2. Triaxial Peak Accelerographs				
a. Fuel Bldg., El. 707'6"	~~NA~~	NA	R	
b. DG & HVAC Bldg., El. 712'	~~NA~~	NA	R	
c. Containment Bldg., El. 778'	~~NA~~	NA	R	
3. Triaxial Seismic Switches				
a. Fuel Bldg., El. 712'	~~M(a)~~	SA	R	CP
4. Triaxial Response-Spectrum Recorders				
a. Circulating Water House	~~NA~~	NA	R(a)	CP
b. Digital Cassette with Playback Feature (Active)		SA	R	CP

(a) ~~Except seismic trigger.~~ CP

INSERT TO TABLE 4.3.7.2-1
p. 3/4 3-69

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs		
a. Concrete pad (outside) El. 712'	SA	R
b. Fuel Bldg., El. 712'	SA	R
c. Containment Bldg., El. 851'	SA	R
d. Containment Bldg., El. 778'	SA	R
e. Control Bldg., El. 737'	SA	R
2. Triaxial Peak Accelerographs (Passive)		
a. Fuel Bldg., El. 707'6"	NA	
b. DG Bldg., El. 712'	NA	
c. Containment Bldg., El. 778'	NA	

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METEOROLOGICAL MONITORING INSTRUMENTATION

NO CHANGE

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.

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TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE	
a. Wind Speed		
1. Elev. 768 ft.	1	CPS
2. Elev. ~~933~~ 932 ft. MSL	1	
b. Wind Direction		
1. Elev. 768 ft.	1	CPS
2. Elev. ~~933~~ 932 ft. MSL	1	
c. Air Temperature Difference		
1. Elev. 768/~~933~~ 932 ft. MSL	1	CPS

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TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	
a. Wind Speed			
1. Elev. 768 ft.	D	SA	CPS
2. Elev. ~~933~~ 932 ft. MSL	D	SA	
b. Wind Direction			
1. Elev. 768 ft.	D	SA	CPS
2. Elev. ~~933~~ 932 ft. MSL	D	SA	
c. Air Temperature Difference			
1. Elev. 768/~~933~~ 932 ft. MSL	D	SA	CPS

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REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown monitoring instrumentation channels shown in Table 3.3.7.4-1 shall be OPERABLE with readouts on the ~~Reactor~~ Remote Shutdown Panel. ||CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE	
1. Reactor Vessel Pressure	1	
2. Reactor Vessel Water Level	1	
3. Safety/Relief Valve Position, 3 valves	1/valve	
4. Suppression Pool Water Level	1	
5. Suppression Pool Water Temperature	~~1~~ 3	CPS
6. Drywell Temperature	1	
7. RHR System Flow	1	
8. Shutdown Service Water ~~System Flow~~ Pump Discharge Pressure	1	CPS
~~9. Shutdown Service Water Temperature~~	~~1~~	CPS
9 ~~10.~~ RCIC System Flow	1	CPS
10 ~~11.~~ RCIC Turbine Speed	1	CPS
11 ~~12.~~ RCIC Storage Tank Level	1	CPS

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TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	
1. Reactor Vessel Pressure	M	R	
2. Reactor Vessel Water Level	M	R	
3. Safety/Relief Valve Position	M	NA	
4. Suppression Pool Water Level	M	R	
5. Suppression Pool Water Temperature	M	R	
6. Drywell Temperature	M	R	
7. RHR System Flow	M	R	
8. Shutdown Service Water Pump Discharge Pressure ~~System Flow~~	M	R	CPS
~~9. Shutdown Service Water Temperature~~	~~M~~	~~R~~	CPS
9 ~~10.~~ RCIC System Flow	M	R	CPS
10 ~~11.~~ RCIC Turbine Speed	M	R	CPS
11 ~~12.~~ RCIC Storage Tank Level	M	R	CPS

ACCIDENT MONITORING INSTRUMENTATION

NO CHANGE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION	
1. Reactor Vessel Pressure	2	1	1	80	
2. Reactor Vessel Water Level	2	1	1, 2	80	
3. Suppression Pool Water Level	~~2~~ 4	~~1~~ 2	1, 2	80	CPS
4. Suppression Pool Water Temperature	2/sector	1/sector	1, 2	80	
~~5. Drywell Radiation~~	~~2~~	~~1~~	~~1, 2~~	~~80~~	CPS
5 ~~6.~~ Drywell Pressure	2	1	1, 2	80	
6 ~~7.~~ Drywell Air Temperature	2	1	1, 2	80	
7 ~~8.~~ Drywell Hydrogen Concentration ~~Analyzer and Monitor~~	12	1	1, 2	80	
8 ~~9.~~ Containment Pressure	2	1	1, 2	80	
9 ~~10.~~ Containment Temperature	2	1	1, 2	80	
10 ~~11.~~ Containment Hydrogen Concentration ~~Analyzer and Monitor~~	2	1	1, 2	80	
11 ~~12.~~ Safety/Relief Valve Acoustic Monitor	1 ~~2~~/valve**	1/valve**	1, 2	80a	
~~13. In-Core Thermocouples~~	~~(4)/(1 per core quadrant)~~	~~(2)/(1 each of 2 core quadrants)~~	~~1, 2~~	~~80~~	
12 ~~14.~~ Containment/Drywell High Range ~~Area Gross Gamma~~ Radiation Monitors	1 ~~2~~ 4***	2*	1, 2, 3	81	
13 ~~15.~~ HVAC Stack High Radio-activity Monitor #	1	1	1, 2, 3	81	
14 ~~16.~~ SGTS Exhaust High Radio-activity Monitor #	1	1	1, 2, 3	81	
~~17.~~ ~~#~~	~~(1)~~	~~1~~	~~1, 2, 3~~	~~81~~	

*One each for ~~secondary~~ containment and drywell. CPS

#High range noble gas monitors.

**Thermocouples in the SRV discharge line can serve as backup to the SRV position indication should the position indication become inoperable. CPS

***Two each for containment and drywell.

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Table 3.3.7.5-1 (Continued)

NO CHANGE

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

Action 80 -

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

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

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

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

b. 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 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.

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TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2
2. Reactor Vessel Water Level	M	R	1, 2
3. Suppression Pool Water Level	M	R	1, 2
4. Suppression Pool Water Temperature	M	R	1, 2
~~5. Drywell Radiation~~	~~M~~	~~R~~	~~1, 2~~
5 ~~6.~~ Drywell Pressure	M	R	1, 2
6 ~~7.~~ Drywell Air Temperature	M	R	1, 2
7 ~~8.~~ Drywell Hydrogen Concentration Analyzer and Monitor	D* ~~(H)(NA)~~	~~Q#~~ N/A	1, 2
8 ~~9.~~ Containment Pressure	M	R	1, 2
9 ~~10.~~ Containment Temperature	M ~~(H)(NA)~~	R	1, 2
10 ~~11.~~ Containment Hydrogen Concentration Analyzer and Monitor	D* ~~H~~	~~Q#~~ N/A	1, 2
11 ~~12.~~ Safety/Relief Valve Acoustic Monitor	NA	R	1, 2
~~13. In Core Thermocouples~~	~~M~~	~~R~~	~~1, 2~~
12 ~~14.~~ Containment/Drywell High Range ~~Area Gross Gamma~~ Radiation Monitors	M	R	1, 2, 3
13 ~~15.~~ HVAC Stack High Radioactivity Monitor	M	R	1, 2, 3
14 ~~16.~~ SGTS Exhaust High Radioactivity Monitor	M	~~M~~ R	1, 2, 3
~~15 17.~~ ~~#~~	~~M~~	~~M~~	~~1, 2, 3~~

~~^Using sample gas containing:~~
~~a. One volume percent hydrogen, balance nitrogen.~~
~~b. Four volume percent hydrogen, balance nitrogen.~~
#High range noble gas monitors.
*Accomplished automatically using sample gas supply integral to analyzer equipment.

CPS

CPS

CPS

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NO CHANGE

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

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

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. Performance of a:

1. CHANNEL CHECK at least once per:

   a) 12 hours in CONDITION 2*, and

   b) 24 hours in CONDITION 3 or 4.

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

b. Performance of a CHANNEL FUNCTIONAL TEST:

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

2. At least once per 31 days.

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

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

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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. | CPS

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 Two independent chlorine detection system subsystems shall be OPERABLE with their (alarm) (trip) setpoints adjusted to actuate at a chlorine concentration of less than or equal to ~~1~~ 5 ppm. CPS

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one chlorine detection system 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 chlorine mode of operation.

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine 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~~ 6 months. CPS

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No CHANGE

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

   1. Less than, but more than one-half of, the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, restore the inoperable Function A instrument(s) to OPERABLE status within 14 days or, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specifications 4.6.1.7 and 4.6.2.6.

   2. One less than the Total Number of Instruments shown in Table 3.3.7.9-1 for Function B, or one-half or less of the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, or with any two or more adjacent instruments inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specifications 4.6.1.7 and 4.6.2.6.

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

SURVEILLANCE REQUIREMENTS

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

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

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

Control Building 781'

Control Building 800'

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TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	SMOKE (x/y)
a. Control Room	210/0	326/0
b. ~~Control Room~~ Auxiliary Electrical Equipment Room	0/0	4/0
c. Cable Spreading Rooms		
Div. 1 Control Building 781'	0/5	5/0
Div. 2 Control Building 781'	0/5	5/0
d. Switchgear Rooms		
Div. 1 Aux. Bldg. 781'	0/0	8/0
Div. 2 Aux. Bldg. ~~781'~~ 781'	0/0	10/0
Div. 3 Cont. Bldg. ~~781'~~ 781'	0/0	6/0
Cont. Bldg. 825'	0/0	20/0
e. Station Batteries		
Div. 1 781' Aux. Bldg.	0/0	1/0
Div. 2 781' Aux. Bldg.	0/0	1/0
Div. 3 781' Cont. Bldg.	0/0	1/0
Div. 4 781' Cont. Bldg.	0/0	1/0
f. Inverter Rooms		
Div. 1 781' Cont. Bldg.	0/0	1/0
Div. 2 781' Cont. Bldg.	0/0	1/0
Div. 4 781' Cont. Bldg.	0/0	1/0
g. Diesel Generator Rooms		
Div. 1 737' Diesel Gen. Bldg.	0/5	0/0
Div. 2 737' Diesel Gen. Bldg.	0/4	0/0
Div. 3 737' Diesel Gen. Bldg.	0/4	0/0
h. Diesel Gen. Day Tank Rooms		
Div. 1 737' Diesel Gen. Bldg.	0/0	0/0
Div. 2 737' Diesel Gen. Bldg.	0/0	0/0
Div. 3 737' Diesel Gen. Bldg.	0/0	0/0
i. Diesel Fuel Oil Storage Tank Rms.		
Div. 1 712' Diesel Gen. Bldg.	0/0	0/0
Div. 2 712' Diesel Gen. Bldg.	0/0	0/0
Div. 3 712' Diesel Gen. Bldg.	0/0	0/0

~~(List all detectors in areas required to ensure the OPERABILITY of safety-related equipment.)~~

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

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

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TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)	TOTAL NUMBER OF INSTRUMENTS SMOKE (x/y)	
707'6"			
j. Safety Related Pumps			
RHR "A" ~~712'~~ Aux. Bldg.	0/0	5/0	CPS
RHR "B" ~~712'~~ Aux. Bldg.	0/0	5/0	
RHR "C" ~~712'~~ Aux. Bldg.	0/0	3/0	
LPCS ~~712'~~ Aux. Bldg.	0/0	5/0	
HPCS 712' ~~Aux.~~ Fuel Bldg.	0/0	5/0	
RCIC ~~712'~~ 778' Aux. Bldg	0/0	4/0	
S.L.C. ~~701'~~ 723'2" Containment Bldg.	0/0	2/0	
R Recirc ~~712'~~ Containment Bldg.	4/0	0/0	
F.P.C. 712' Fuel Bldg.	0/0	2/0	
SSW Div. 1 699' Screen House	0/0	3/0	
SSW Div. 2 699' Screen House	0/0	3/0	
SSW Div. 3 699' Screen House	0/0	2/0	
k. Fuel Storage 755' Fuel Bldg.	0/0	6/0	CPS

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LOOSE-PART DETECTION SYSTEM

NO CHANGE

LIMITING CONDITION FOR OPERATION

3.3.7.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, in lieu of any other report required by Specifition 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 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. CHANNEL CHECK at least once per 24 hours,

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

c. CHANNEL CALIBRATION at least once per 18 months.

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3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM (Optional)

DELETE THIS SECTION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

CPS

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

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

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

a. At least once per 7 days by:

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

      a) For the overspeed protection control system;

         1) Four high pressure turbine control valves, and

         2) Four low pressure turbine interceptor valves.

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

         1) Four high pressure turbine throttle stop valves,

         2) Four high pressure turbine reheat stop valves,

         3) Four high pressure turbine control valves, and

         4) Four low pressure turbine interceptor valves.

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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DELETE

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INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

7 1 3.3.7.11-1

3.3.~~3~~.1~~0~~ 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). CP

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, 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 3.3.7.11-1 ~~3.3-12~~. 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. CP

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

SURVEILLANCE REQUIREMENTS

4.3.3.11 ~~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~~ 4.3.7.11-1 CP

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TABLE 3.3.7.11-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste ~~Effluent Line~~ Discharge Process Radiation Monitor	~~(~~1~~)~~	110
2. GROSS ~~BETA OR GAMMA~~ RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Plant Service Water ~~System~~ Effluent ~~Line~~ Process Radiation Monitor	~~(~~1~~)~~	~~112~~ 111
b. ~~Component Cooling Water System Effluent Line~~ Shutdown Service Water Effluent Process Radiation Monitor	~~(~~1~~)~~	~~112~~ 111
c. Fuel Pool Heat Exchanger Service Water Radiation Monitor	1	111
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	~~(~~1~~)~~	~~113~~ 112
b. ~~Discharge Canal~~ Plant Service Water Effluent Line	~~(~~1~~)~~	~~113~~ 112
c. Plant Circulating Water Effluent Line	1	112
4. ~~RADIOACTIVITY RECORDERS*~~		
~~a. Liquid Radwaste Effluent Line~~	~~(1)~~	~~115~~

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

TABLE 3.3.7.11-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5. TANK LEVEL INDICATING DEVICES**		
a. Cycled Condensate Storage	(1)	~~114~~ 113
b. Reactor Core Isolation Cooling Storage	(1)	~~114~~ 113
~~c.~~	~~(1)~~	~~114~~

CPS

~~**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 overflows and surrounding area drains connected to the liquid radwaste treatment system.~~

CPS

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TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release: CPS

a. At least two independent samples are analyzed in accordance with Specification ~~4~~ 3.11.1.1.~~2~~, and CPS

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 111 - Not Applicable.~~ CPS

ACTION ~~112~~ 111 - 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 limit of detection of at least 10-' microcuries/ml. CPS

ACTION ~~113~~ 112 - 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 curves generated in situ may be used to estimate flow. CPS

ACTION ~~114~~ 113 - 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. CPS

~~ACTION 115 - 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.~~ CPS

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TABLE 4.3.7.11-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Discharge ~~Effluent Line~~ Process Radiation Monitor	~~P~~ D	P	R(3)	Q(1)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Plant Service Water ~~System~~ Effluent Process Radiation Monitor ~~Line~~	D	M	R(3)	Q(2)
b. Shutdown Service Water Effluent Process Radiation Monitor ~~Component Cooling Water System Effluent Line~~	D	M	R(3)	Q(2)
c. Fuel Pool Heat Exchanger Service Water Radiation Monitor	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Plant Service Water Effluent Line ~~Discharge Canal~~	D(4)	N.A.	R	Q
c. Plant Circulating Water Effluent Line	D(4)	N.A.	R	Q

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TABLE 4.3.7.11-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. ~~RADIOACTIVITY RECORDERS~~				
~~a. Liquid Radwaste Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
5. TANK LEVEL INDICATING DEVICES				
a. Cycled Condensate Storage	D*	N.A.	R	Q
b. Reactor Core Isolation Cooling	D*	N.A.	R	Q
~~c. ______~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~

*During liquid additions to the tank.

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TABLE 4.3.7.11-1 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall ~~also~~ demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists: 8

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

(2) The CHANNEL FUNCTIONAL TEST shall ~~also~~ demonstrate that control room alarm annunciation occurs if any of the following conditions exists: 8

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

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. ~~For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)~~ CP5

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

Subsequent CHANNEL CALIBRATION shall be performed using the initial radioactive standards or other standards of equivalent quality or radioactive sources that have been related to the initial calibration. CP5

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INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.~~3.11~~ 7.12 The radioactive gaseous effluent monitoring instrumentation channels shown in Table ~~3.3-13~~ 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM. CPS

APPLICABILITY: As shown in Table ~~3.3-13~~ 3.3.7.12-1.

ACTION: Incert Attached

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table ~~3.3-13.~~ 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.~~ CPS

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

SURVEILLANCE REQUIREMENTS

4.3.~~3.11~~ 7.12 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table ~~4.3-12.~~ 4.3.7.12-1

Restore the inoperable instrumentation to OPERABLE status within 30 days or, in lieu of a Licensee Event Report, explain why the inoperability was not corrected within the time specified in the next Semiannual Radioactive Effluent Release Report. CPS

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Insert Attached

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EFFLUENT MONITORING SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	(1)	*	123
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127
d. Effluent System Flow Rate Measuring Device	(1)	*	122
e. Sampler Flow Rate Measuring Device	(1)	*	122
2A. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor	(1)	**	125
b. Hydrogen or Oxygen Monitor	(1)	**	125
2B. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor	(2)	**	126
b. Hydrogen or Oxygen Monitor	(2)	**	126

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TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. POST-TREATMENT AIR EJECTOR OFFGAS PRM			
a. High Range Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release	1	**	121
b. Effluent System Flow Rate Monitor	1	**	123
c. PRM Flow Rate Monitor	1	**	123
2. STATION HVAC EXHAUST PRM			
a. High Range Noble Gas Activity Monitor	1	*	121
b. Low Range Noble Gas Activity Monitor	1	*	121
c. Iodine Activity Monitor	1	*	122
d. Particulate Activity Monitor	1	*	122

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TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Insert Attached

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. REACTOR BUILDING VENTILATION/PURGE MONITORING SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	124
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127
d. Effluent System Flow Rate Monitor	(1)	*	122
e. Sampler Flow Rate Monitor	(1)	*	122
4. MAIN STACK MONITORING SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	123
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127
d. Effluent System Flow Rate Monitor	(1)	*	122
e. Sampler Flow Rate Monitor	(1)	*	122
5. TURBINE BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	123
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127

CPS

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
2. STATION HVAC EXHAUST PRM (Con't)			
e. PRM Flow Rate Monitor	1	*	123
f. Effluent System Flow Rate Monitor	1	*	123
3. STANDBY GAS TREATMENT SYSTEM EXHAUST PRM			
a. High-High Range Noble Gas Activity Monitor	1	***	121
b. High Range Noble Gas Activity Monitor	1	***	121
c. Low Range Noble Gas Activity Monitor	1	***	121
d. High Range Iodine Activity Monitor	1	***	122
e. Low Range Iodine Activity Monitor	1	***	122
f. Particulate Activity Monitor	1	***	122
g. PRM Flow Rate Monitor	1	***	123
h. Effluent System Flow Rate Monitor	1	***	123

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TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Incert Attached

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. TURBINE BUILDING VENTILATION MONITORING SYSTEM (Continued)			
d. Effluent System Flow Rate Monitor	(1)	*	122
e. Sampler Flow Rate Monitor	(1)	*	122
6. AUXILIARY BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	123
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127
d. Flow Rate Monitor	(1)	*	122
e. Sampler Flow Rate Monitor	(1)	*	122
7. FUEL STORAGE AREA VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	(1)		123
b. Iodine Sampler	(1)	*	127
c. Particulate Sampler	(1)	*	127
d. Flow Rate Monitor	(1)	*	122
e. Sampler Flow Rate Monitor	(1)	*	122

CPS

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	124
5. PRE-TREATMENT AIR EJECTOR OFFGAS PRM			
a. Noble Gas Activity Monitor	1	**	125

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

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

Insert Attached Actions

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

a. The offgas system is not bypassed, and

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

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

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

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

ACTION 124 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend release of radioactive effluents via this pathway.

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

ACTION 126 - 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.

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

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

*** During standby gas treatment system operation.

ACTION 121 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and analyzed for gross noble gas activity within 24 hours.

ACTION 122 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, within 4 hours after the channel has been declared inoperable, samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 123 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATION

ACTION 124 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the main condenser offgas treatment system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours. If the recombiner temperature remains constant and THERMAL POWER has not changed, the sampling frequency may be changed to 8 hours.

ACTION 125 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, gases from the main condenser offgas treatment system may be released to the environment for up to 72 hours provided:

a. The offgas treatment system is not bypassed, and

b. The post-treatment air ejector offgas PRM high range noble gas activity monitor is OPERABLE.

ACTION 126 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway shall be terminated.

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TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Insert Attached

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EFFLUENT MONITORING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R(3)	Q(1)	*
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	*
2A. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems 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	**
2B. MAIN CONDENSER OFFGAS TREATMENT 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	**

Form 115-8
Dev. 12-66

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
1. POST-TREATMENT AIR EJECTOR OFFGAS PRM					
a. High Range Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release	D	D	R(2)	Q(1)	*
b. Effluent System Flow Rate Monitor	D	N.A.	N.A.	N.A.	*
c. PRM Flow Rate Monitor	D	N.A.	N.A.	N.A.	*
2. STATION HVAC EXHAUST PRM					
a. High Range Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Low Range Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
c. Iodine Activity Monitor	W	N.A.	N.A.	N.A.	*
d. Particulate Activity Monitor	W	N.A.	N.A.	N.A.	*

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TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Insert Attached

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BUILDING VENTILATION/PURGE MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. MAIN STACK MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

CPS

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
e. PRM Flow Rate Monitor	D	N.A.	R	Q	*
f. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
3. STANDBY GAS TREATMENT SYSTEM EXHAUST PRM					
a. High-High Range Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. High Range Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
c. Low Range Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
d. High Range Iodine Activity Monitor	W	N.A.	N.A.	N.A.	*
e. Low Range Iodine Activity Monitor	W	N.A.	N.A.	N.A.	*
f. Particulate Activity Monitor	W	N.A.	N.A.	N.A.	*
g. PRM Flow Rate Monitor	D	N.A.	R	Q	*
h. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Insert Attached

CPS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. TURBINE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
6. AUXILIARY BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Actvity 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	*

Form 115-3
Rev. 12-66

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
4. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	NA	Q(3)	M	**
5. PRE-TREATMENT AIR EJECTOR OFFGAS PRM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	**
6. STATION HVAC EXHAUST (ACCIDENT RANGE) PRM			(LATER)		
7. STANDBY GAS TREATMENT SYSTEM EXHAUST (ACCIDENT RANGE) PRM			(LATER)		

TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

This page not needed for CPS

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

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TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

This page not needed for CPS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
9. TURBINE GLAND SEAL CONDENSER VENT AND MECHANICAL VACUUM PUMP EXHAUST MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
10. CONDENSER AIR EJECTOR RADIOACTIVITY MONITOR					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***

DRAFT

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

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

~~(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation 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.~~

(~~2~~ 1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

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

(~~3~~ 2) 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.)~~ Subsequent CHANNEL CALIBRATION shall be performed using the initial radioactive standards or other standards of equivalent quality or radioactive sources that have been related to the initial calibration

(~~4~~ 3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. ~~One~~ 0.0 volume percent hydrogen, balance nitrogen, and
2. ~~Four~~ 2.0 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.~~

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3/4.3.~~9~~ 8 PLANT SYSTEMS ACTUATION INSTRUMENTATION

CP

LIMITING CONDITION FOR OPERATION

3.3.~~9~~ 8 The plant systems actuation instrumentation channels shown in Table 3.3.~~9~~ 8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.~~9~~ 8-2. CP

APPLICABILITY: As shown in Table 3.3.~~X~~ 8-1. CP

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.~~X~~ 8-2, 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.~~ system/loop inoperable. CP

b. For the containment spray system:

1. With the number of OPERABLE channels containment spray loop less than required by the Minimum OPERABLE Channels per ~~Trip System~~ requirement for one ~~trip~~ containment spray loop ~~system~~, place at least one inoperable channel in the tripped condition within one hour or declare the associated ~~system~~ loop inoperable. CP

2. With the number of OPERABLE channels containment spray loop less than required by the Minimum OPERABLE Channels per ~~Trip System~~ requirement for both ~~trip~~ loops ~~systems~~, declare the associated ~~system~~ loop inoperable. CP

c. 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 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.

CP

SURVEILLANCE REQUIREMENTS

4.3.~~X~~8.1 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.~~X~~8.1-1. |CPS

4.3.~~X~~8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. |CPS

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TABLE 3.3.~~X~~8-1 |CPS

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER ~~TRIP SYSTEM~~ CONTAINMENT SPRAY LOOP	APPLICABLE OPERATIONAL CONDITIONS	
1. CONTAINMENT SPRAY SYSTEM			|CPS
a. Drywell Pressure-High	~~1~~ 2	1, 2, 3	|CPS
b. Containment Pressure-High	~~1~~ 2	1, 2, 3	|CPS
c. Reactor Vessel Water Level-Low Low Low, Level 1	~~1~~ 2	1, 2, 3	|CPS
d. Timers			
(1). ~~System A~~ Loop A	1	1, 2, 3	|CPS
(2). ~~System B~~ Loop B	1	1, 2, 3	|
e. Manual Initiation	1	1, 2, 3	|

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS	
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High, Level (8)	3	1	|CPS

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TABLE 3.3.~~X~~ 8-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	
1. CONTAINMENT SPRAY SYSTEM			CPS
a. Drywell Pressure-High	≤ 1.68 ~~(1.83)~~ psig	≤ 1.88 ~~(1.94)~~ psig	CPS
b. Containment Pressure-High	≤ 8.0 ~~(9)~~ psig	≤ 8.5 ~~(9)~~ psig	CPS
c. Reactor Vessel Water Level-Low Low Low, Level 1	≥ ~~(150)~~ 145.5 inches*	≥ ~~(149)~~ 147.7 inches	CPS
d. Timers			
1). ~~System A~~ Loop A	≤ ~~(10)~~ minutes	≤ ~~(10)~~ minutes	CPS
2). ~~System B~~ Loop B	≤ ~~(11)~~ 10 minutes	≤ ~~(11)~~ 10 minutes	CPS
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High, Level ~~(8)~~ 8	≤ 52.0 ~~(54.5)~~ inches*	≤ 53.5 ~~(56.0)~~ inches	CPS

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

TABLE 4.3.~~X~~1-1 (Continued)

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	OPERATIONAL CHANNEL CALIBRATION	CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CONTAINMENT 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
e. Manual Initiation	NA	M	NA	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level ~~(8)~~	~~(NA)~~	~~(M)~~	~~(R)~~	1

CPS
CPS
CPS

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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 CPS

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.

DRAFT

NO CHANGE

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

DRAFT
NO CHANGE

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

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IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F, ~~and:~~ when reactor vessel pressure in the steam space is greater than 25 psig*, and : | C

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 ~~70°F,~~ 50°F or | CPS

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

* Below 25 psig the temperature differential is not applicable. | CPS

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3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 Of the following safety/relief valves, the safety valve function of at least 6 valves and the relief valve function of at least 5 valves other than those satisfying the safety valve function requirement shall be OPERABLE with the specified lift settings:

±15 psi Relief
±1% Safety

Number of Valves	Function	Setpoint* (psig) ± 1%
7	Safety	1165
5	Safety	1180
4	Safety	1190
1	Relief	1103
8	Relief	1113
7	Relief	1123

CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than ~~(105)°F~~ 110°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is ~~(105)°F~~ 110°F or greater, place the reactor mode switch in the Shutdown position. CPS CPS

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

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The acoustic monitor~~(s)~~ for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be ((20) ± (5) psig) (   ) by performance of a:

a. CHANNEL ~~(~~FUNCTIONAL TEST~~)/(~~CHECK~~)~~ initially, then at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months. ~~(**)~~ CPS

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

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

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

*The 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.)~~ CPS

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SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ± ~~1%~~ 2% Open	Close	Relief Function Setpoint* (psig) ± 15 psi Open	Close
F051D	1033	926	1103	1003
F051C	1073	936	1113	1013
F047F	1113	946	1113	1013
F051B	1113	946	1113	1013
F051G ~~F051A~~	1113	946	1113	1013
~~(F051G)~~	~~-~~	~~-~~	~~-~~	~~-~~

CPS

CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The drywell atmosphere particulate radioactivity monitoring system,

b. The drywell sump flow monitoring system, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. Drywell air cooler condensate flow 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. CPS

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

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

d. 1 gpm leakage at a reactor coolant system pressure of ~~(~~1000~~)~~ ± ~~(~~10~~)~~ psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1. CPS

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. CPS

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

(*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.)

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SURVEILLANCE REQUIREMENTS

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

a. Monitoring the drywell atmospheric ~~(~~particulate~~)~~ ~~(~~and~~)~~ ~~(~~gaseous~~)~~ radioactivity at least once per 12 hours, CPS

b. Monitoring the drywell sump flow rate ~~or the (gaseous) (particulate) radioactivity~~ at least once per 12 hours, and. CPS

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, and

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

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

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

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

b. CHANNEL CALIBRATION at least once per 18 months.

DRAFT

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
1E12F041A	LPCI from RHRA Testable Chk
1E12F041B	LPCI from RHRB Testable Chk
1E12F041C	LPCI from RHRC Testable Chk
1E12F042A	LPCI from RHRA Shutoff
1E12F042B	LPCI from RHRB Shutoff
1E12F042C	LPCI from RHRC Shutoff
1E12F023	RHR B Supp to Rx Head Spray
1E21F005	LPCS Inj Isol
1E21F006	LPCS Inj Testable Chk
1E22F004	HPCS Inj Isol
1E22F005	HPCS Testable Chk Disc
1E51F066	RCIC Pmp Disch to Rx Testable Check
1E51F013	RCIC Pmp Disch to Rx Otbd Isol
1E51F064	RHR & RCIC St Supp Otbd Isol
1C41F004A	SBLC Pump 1A Disch Explosive V/O
1C41F004B	SBLC Pump 1B Disch Explosive V/O

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)

LATER

CPS

DRAFT

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

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

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

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

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

c. At all other times:

1. With the:

   a. Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours or ~~, restore the chloride concentration to within the limit within 24 hours, or~~

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

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

CPS

2. The provisions of Specification 3.0.3 are not applicable.

DRAFT

SURVEILLANCE REQUIREMENTS

NO CHANGE

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

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

b. Analyzing a sample of the reactor coolant for:

 1. Chlorides at least once per:

  a) 72 hours, and

  b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

 2. Conductivity at least once per 72 hours.

 3. pH at least once per:

  a) 72 hours, and

  b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

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

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

 2. 24 hours at all other times.

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

 1. 7 days, and

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

DRAFT

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

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

NO CHANGE

DRAFT

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

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

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

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

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 ~~I-13~~ or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 ~~I-1131~~ together with the following additional information. | CPS

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LIMITING CONDITION FOR OPERATION (Continued)

NO CHANGE

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
2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or
3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.

Additional Information

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

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.

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TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

NO CHANGE

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
#Until the specific activity of the primary coolant system is restored to within its limits.

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

~~a. A maximum heatup of 100°F in any one hour period,~~
~~b. A maximum cooldown of 100°F in any one hour period,~~
See Insert CPS

b. ~~c.~~ A maximum temperature change of less than or equal to 20°F ~~(10)°F~~ in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and CPS

c. ~~d.~~ The reactor vessel flange and head flange metal temperatures shall be maintained greater than or equal to 70°F when reactor vessel head bolting studs are under full tension. CPS

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

~~4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C' as applicable, at least once per 30 minutes.~~ CPS

See Insert

INSERT replacing 3.4.6.1. a & b (pg 3/4 4-17)

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.

INSERT replacing 4.4.6.1.1 (pg 3/4 4-17)

4.4.6.1.1 During system heatup, cooldown and in-service leak and hydrostatic testing operations, the reactor vessel pressure and metal temperature of the reactor vessel flange surfaces, bottom head inside surface, as measured by the bottom head drain temperature, shall be determined to be within the operating limits defined by Figure 3.4.6.1-1 at least once per 30 minutes.

bottom head outside surface and

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SURVEILLANCE REQUIREMENTS (Continued)

SEE INSERT

~~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.~~ CPS

~~4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.~~ CPS

SEE INSERT

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F~~:~~ when reactor vessel head bolting studs are under full tension: CPS

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

1. ≤ ~~100°F~~ 90°F, at least once per 12 hours.

2. ≤ (80)°F, at least once per 30 minutes. CPS

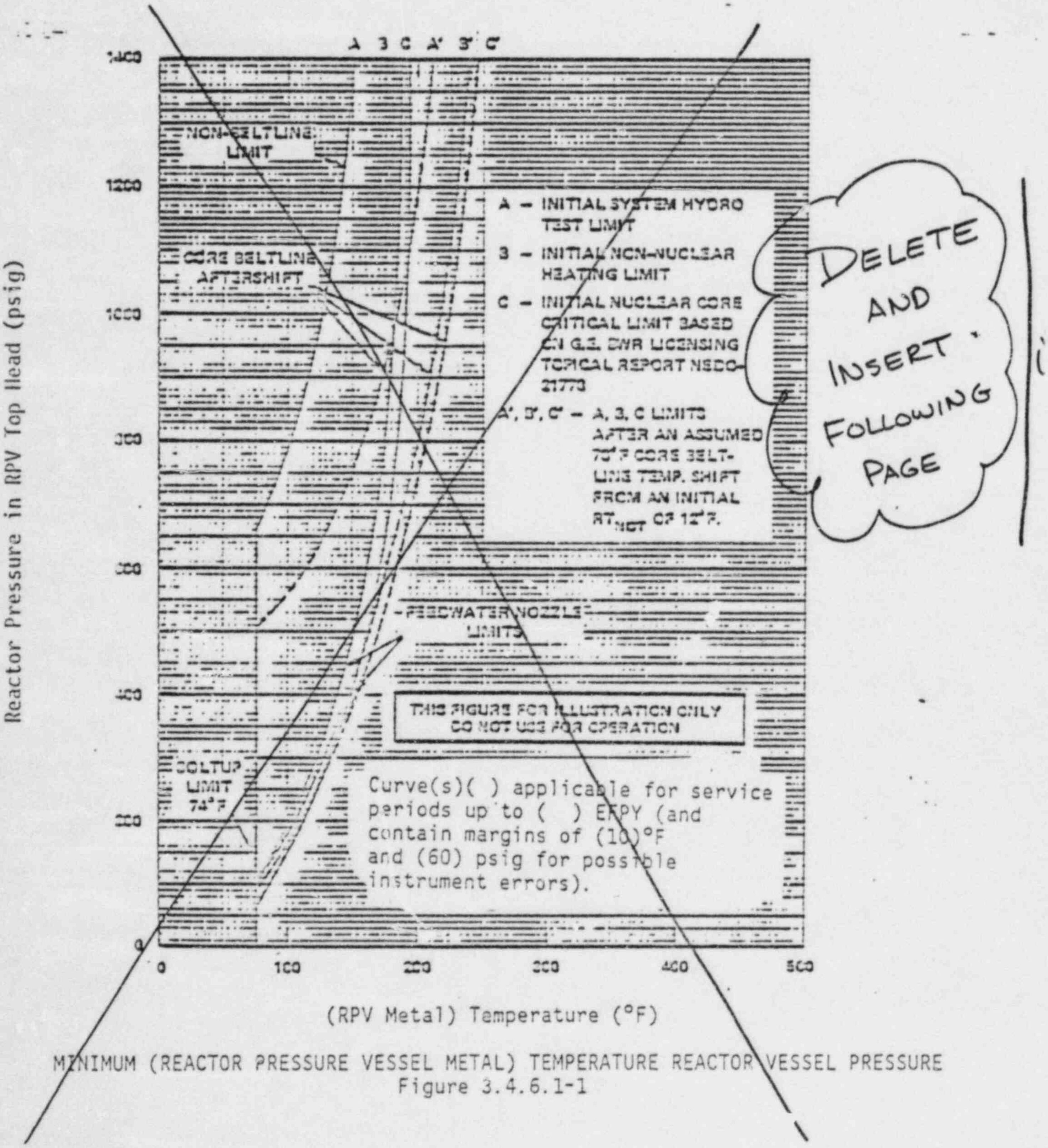
b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs, except that 10% of the bolting studs may be fully tensioned to below 70°F. CPS

INSERT to replace 4.4.6.1.2 (pg. 3/4 4-18)

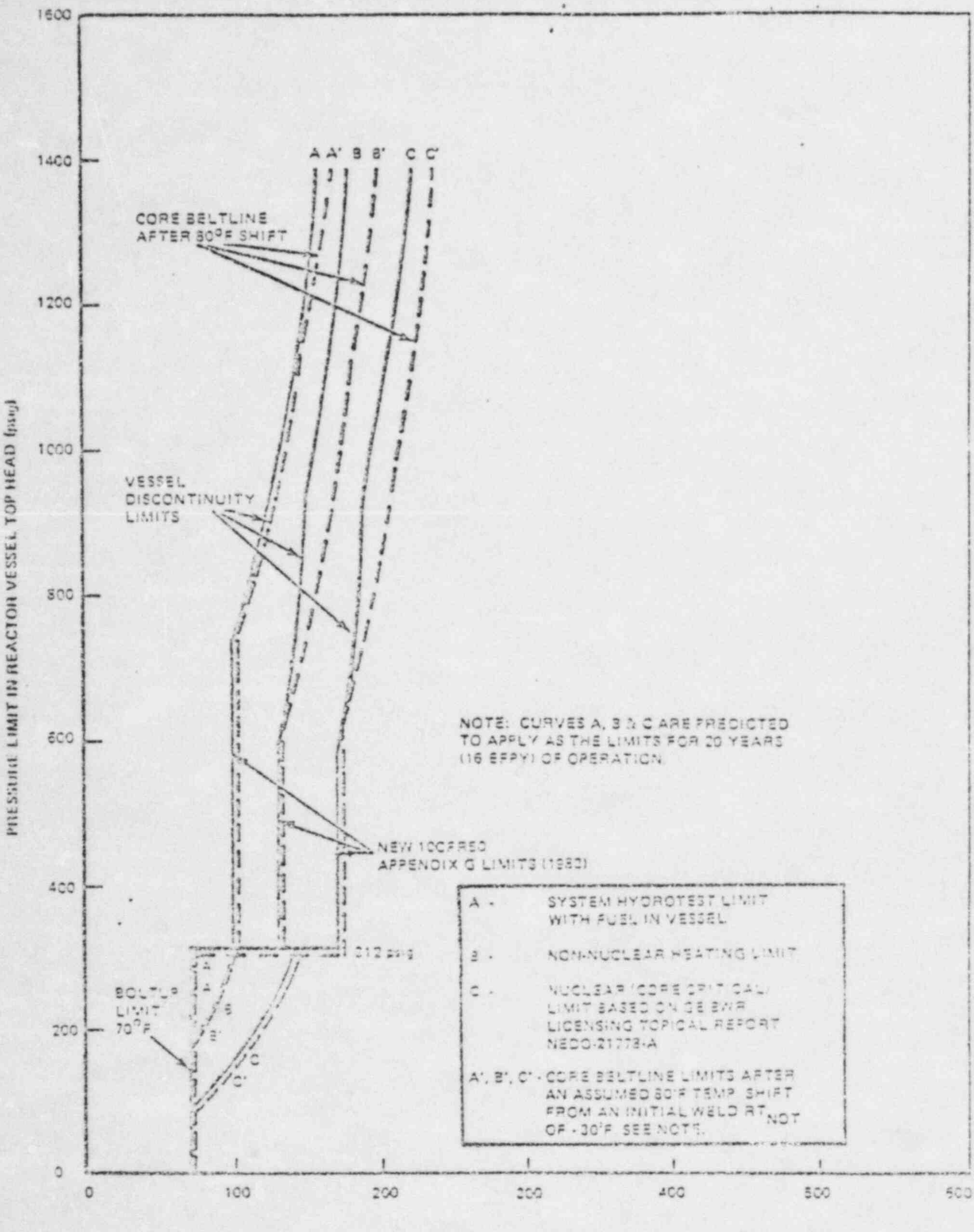
4.4.6.1.2 The reactor steam space coolant temperature shall be determined to be within the heatup and cooldown limits of 100°F in any 1 hour at least once per 30 minutes.

INSERT to replace 4.4.6.1.3 (pg 3/4 4-18)

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined as a function of time and THERMAL POWER as required by 10CFR50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1.



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



PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
1600
1400
1200
1000
800
600
400
200
0
A A' B B' C C'
CORE BELTLINE
AFTER 80°F SHIFT
VESSEL
DISCONTINUITY
LIMITS
NOTE: CURVES A, B & C ARE PREDICTED
TO APPLY AS THE LIMITS FOR 20 YEARS
(16 EFPY) OF OPERATION.
NEW 10CFR50
APPENDIX G LIMITS (1983)
BOLTUP
LIMIT
A - SYSTEM HYDROTEST LIMIT
WITH FUEL IN VESSEL
B - NON-NUCLEAR HEATING LIMIT
C - NUCLEAR (CORE CRITICAL)
LIMIT BASED ON GE BWR
LICENSING TOPICAL REPORT
NEDO-21778-A
A', B', C' - CORE BELTLINE LIMITS AFTER
AN ASSUMED 80°F TEMP. SHIFT
FROM AN INITIAL WELD RT NDT
OF -30°F. SEE NOTE.
0
100
200
400
500
MINIMUM REACTOR VESSEL METAL TEMPERATURE - (°F)

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TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
~~1~~ Item 2 (131C8981G1)	~~3°~~ 3°	0.86	10
~~2~~ Item 2 (131C8981G1)	~~183°~~ 177°	0.86	~~30~~ 20
~~3~~ Item 2 (131C8981G1)	~~177°~~ 183°	0.86	~~Standby~~ Spare

CPS

DRAFT

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

No CHANGE

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

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*Not applicable during anticipated transients.

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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~~ stroke times greater than or equal to ~~3~~ 2.5 and less than or equal to 5 seconds. The stroke time average of the fastest valves in each of the four steam lines shall be greater than or equal to 3 seconds. CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

~~a.~~ With one or more MSIVs inoperable:

a. ~~1.~~ Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either: CPS

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

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

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

~~b. The provisions of Specification 3.0.4 are not applicable.~~ CPS

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying ~~full closure~~ stroke time between ~~3~~ 2.5 and 5 seconds when tested pursuant to Specification 4.0.5. The stroke time average of the fastest valves in each of the four steam lines shall be verified to be greater than or equal to 3 seconds. CPS

DRAFT

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

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3/4.4.9 RESIDUAL HEAT REMOVAL

NO CHANGE

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 alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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

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

##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

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

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COLD SHUTDOWN

NO CHANGE

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 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

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

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

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

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

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. ECCS division 1 consisting of:

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

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

3. ~~At least~~ 7 OPERABLE ADS valves. | CPS

b. ECCS division 2 consisting of:

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

2. ~~At least~~ 7 OPERABLE ADS valves. | CPS

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

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

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

#See Special Test Exception 3.10.5.

MAR 16 1983

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LIMITING CONDITION FOR OPERATION

NO CHANGE

ACTION:

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

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

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

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

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

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

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

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

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

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

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

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

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

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LIMITING CONDITION FOR OPERATION (Continued)

NO CHANGE

ACTION: (Continued)

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

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

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

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

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

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

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

f. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the useage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

1. LPCS pump develops a flow of at least 5010 gpm against a test line pressure greater than or equal to ~~300~~ (**) psig. CPS

2. LPCI pump develops a flow of at least 5050 gpm against a test line pressure greater than or equal to ~~110~~ (**) psig. CPS

3. HPCS pump develops a flow of at least 5010 gpm against a test line pressure greater than or equal to ~~305~~ (**) psig. CPS

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

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

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.

*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. CPS

SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per 18 months:

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

b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig ~~(*)~~ and observing ~~the expected change in the indicated valve position and that the acoustic tail-pipe monitor alarms~~. CPS

c) Performing a CHANNEL CALIBRATION of the accumulator ~~backup compressed gas system~~ low pressure alarm system and verifying an alarm setpoint of ~~( ) + ( ), - ( )~~ 140 psig on decreasing pressure. CPS

that either:

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

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

~~(*)~~ 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. CPS

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3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

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

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

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

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

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

   1. From the suppression pool, or

   2. When the suppression pool level is less than the limit or is drained, from the RCIC storage tank containing at least 125,000 available gallons of water, equivalent to a level of ~~( )~~% 95%.

CPS

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 reactor vessel to steam dryer pool gates are removed, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

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SURVEILLANCE REQUIREMENTS

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

4.5.2.2 The HPCS system shall be determined OPERABLE at least once per 12 hours by verifying the RCIC storage tank required volume when the ~~con-densate~~ RCIC storage tank is required to be OPERABLE per Specification 3.5.2.e. CPS

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3/4.5.3 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least 146,400 ft^3, equivalent to a level of 18'11".

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least ~~(~~93,600~~)~~ ft^3, equivalent to a level of ~~(~~12'8"~~)~~, except that the suppression pool level may be less than the limit or may be drained provided that: CPS

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 RCIC storage tank contains at least 125,000 available gallons of water, equivalent to a level of ~~( )%~~ 95%, and CPS

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

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

ACTION:

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

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

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

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SURVEILLANCE REQUIREMENTS

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

a. 18'11" at least once per 24 hours. for operational conditions 1, 2 and 3 CPS

b. 12'8" at least once per (12) hours. for operational conditions 4, 5 CPS

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

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

b. Verify footnote conditions * to be satisfied.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, ~~(15.0)~~ 9.0 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. |CPS

b. At least once per 31 days by verifying that all containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

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

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

*See Special Test Exception 3.10.1

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

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CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to: SEE INSERT (NEXT PAGE.

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

1. L_a, 0.65 percent by weight of the containment air per 24 hours at P_a, 9.0 psig, or

2. L_t, (___) percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (___) psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.4-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.4-1,) subject to Type B and C tests when pressurized (in accordance with Table 3.6.4-1 of Specification 3.6.4) (or) (to P_a, (15.0) psig) (,as applicable).

c.(*)Less than or equal to (11.5) (46.0) scf per hour for (any one) (all four) main steam line((s) through the) isolation valve(s) when tested at P_a, 9.0 psig.

d. A combined leakage rate of less than or equal to 0.12 L_a for all penetrations shown in Table 3.6.4-1 of Specification 3.6.4 as secondary containment bypass leakage paths when pressurized in accordance with Table 3.6.4-1.

e. A combined leakage rate of less than or equal to (1 gpm times the total number of) (3 gpm for all) (ECCS and RCIC) containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at (1.10) P_a (16.5) psig.

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

ACTION:

With:

a. The measured overall integrated containment leakage rate exceeding 0.75 L_a ~~or 0.75 L_t, as applicable,~~ or CPS

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

c. The measured leakage rate exceeding ~~(11.5) (46.0)~~ 28 scf per hour for ~~(~~any one~~)~~ ~~(all four)~~ main steam line~~(s)~~ through the~~)~~ isolation valve(s), or CPS

d. The combined leakage rate for all penetrations shown in Table 3.6.4-1 as secondary containment bypass leakage paths exceeding ~~0.12~~ 0.08 L_a, or CPS

e. The measured combined leakage rate ~~for all (ECCS and RCIC) containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding (~~ of less than or equal to 1 gpm times the total number of ~~such valves)~~ ~~(3 gpm),~~ of containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm when tested at 1.1 Pa (9.9 psig). CPS

~~(*Exemption to Appendix J of 10 CFR 50.)~~ CPS

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INSERT (pg 3/4 6-2)

a. An overall leakage rate of less than La, the maximum allowable leakage rate, which is equal to 0.65 % by weight of the containment volume per 24 hours at Pa, the maximum calculated design basis containment pressure, equal to 9.0 psig

b. A combined leakage rate of less than 0.60 La for all primary containment penetrations and isolation valves listed in Table 3.6.4-1, except that the main steam isolation valves and valves that are hydrostatically tested are excluded from this criteria. The penetrations and valves are tested at 9 psig.

c. Equal to or less than 28 scf per hour for any one main steam line through the isolation valve(s) when tested at 9.0 psig.

d. A combined leakage rate of less than 0.08 La for all penetrations shown in Table 3.6.4-1 as secondary containment bypass leakage paths when tested at 9.0 psig.

e. The measured combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment when tested at 1.1 Pa (9.9 psig).

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

a. The overall integrated leakage rate(s) to less than ~~or equal to~~ 0.75 L_a ~~or 0.75 L_t, as applicable, and~~ , and CPS

b. The combined leakage rate for all penetrations and all valves listed in FSAR Table 6.2-47 ~~Table 3.6.4-1,~~ except for main steam line isolation valves* ~~(~~and valves which are hydrostatically leak tested per FSAR Table 6.2-47 ~~3.6.4-1,) subject to Type B and C tests~~ to less than or equal to 0.60 L_a, and CPS

c. The leakage rate to less than 28 ~~(11.5) (46.0)~~ scf per hour for ~~(~~any one~~)~~ ~~(all four)~~ main steam line ~~(s)~~ through the~~)~~ isolation valve(s), and CPS

d. The combined leakage rate for all penetrations shown in FSAR Table 6.2-47 ~~3.6.4-1~~ as through line ~~secondary containment bypass~~ leakage class 2 ~~paths~~ to less than ~~or equal to~~ 0.08 ~~(0.12)~~ L_a, and CPS

e. The combined leakage rate for ~~all (ECCS and RCIC)~~ containment isolation valves ~~in~~ hydrostatically tested ~~lines which penetrate the primary containment~~ to less than ~~or equal to (~~1 gpm ~~times the total number of such valves) (3 gpm),~~ prior to increasing reactor coolant system temperature above 200°F. CPS

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N56.8 - 1981: CPS

SEE INSERT (NEXT PAGE)

a. ~~Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 9.0 psig, or at P_t, ( ) psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.~~

b. If any periodic Type A test fails to meet 0.75 L_a ~~or 0.75 L_t, as applicable,~~ the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a ~~or 0.75 L_t, as applicable,~~ a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a ~~or 0.75 L_t, as~~ applicable, at which time the ~~above~~ test schedule specified in 4.6.1.2.a may be resumed. CPS

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

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a ~~or 0.25 L_t, as applicable.~~ CPS

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

INSERT p. 3/4 6-3 (To 4.6.1.2)

a. An overall containment integrated leakage rate test shall be conducted at the first refueling shutdown (not more than 3 years subsequent to the preoperational test) and at intervals not to exceed 5 years thereafter.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at ~~P_a~~ 9.0 psig. ~~or P_t, ( ) psig, as applicable.~~ CP

d. Penetration and valve ~~Type B and C~~ tests shall be conducted with gas at ~~P_a~~ 9.0 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Penetrations using continuous leakage monitoring systems,

~~(4. Valves pressurized with fluid from a seal system,)~~ CP

4 ~~5.~~ ~~(ECCS and RCIC)~~ Containment isolation valves ~~in~~ hydrostatically tested, ~~lines which penetrate the primary containment,~~ and

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

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

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

g. Penetration ~~Type B~~ tests ~~for penetrations~~ employing a continuous leakage monitoring system shall be conducted at ~~P_a~~ 9.0 psig, at intervals no greater than once per 3 years. CP

~~(h. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 9.9 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.)~~ CP

h ~~i.~~ ~~(ECCS and RCIC)~~ Containment isolation valves that are ~~in~~ hydrostatically tested ~~lines which penetrate the primary containment~~ shall be leak tested at least once per ~~18 months.~~ 2 years. CP

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

j ~~k.~~ The provisions of Specification 4.0.2 are not applicable to ~~24 month or 40 ± 10 month~~ the surveillance intervals. CP

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

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate at P_a, 9.0 psig:

1. For the personnel air lock, elevation ~~823'~~ 823'-3", of less than or equal to $0.02\ L_a$. CPS

2. For the personnel air lock, elevation ~~757'~~ 741'-0", of less than or equal to $0.05\ L_a$. CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

*See Special Test Exception 3.10.1.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to ~~(~~5~~)~~ scf per hour when the gap between the door seals is pressurized to P_a, 9.0 psig. CPS

b. By conducting an overall air lock leakage test at P_a, 9.0 psig, and verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months#,

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

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

#The provisions of Specification 4.0.2 are not applicable.

~~*Exemption to Appendix J of 10 CFR 50.~~ CPS

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

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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:

1. 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.

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SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying:

1. Blower OPERABILITY by starting the blower~~(s)~~ from the control room and operating the blower~~(s)~~ for at least 15 minutes.

2. Heater OPERABILITY by demonstrating electrical continuity of the heating element circuitry.

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

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position, the blower~~(s)~~ start~~(s)~~.

2. Verifying that the blower~~(s)~~ develop~~(s)~~ at least the ~~below~~ required vacuum and ~~vacuum at the~~ rated capacity indicated below and the heater ~~(temperatures rise to ≥ ( )°F within ( ) minutes)~~ ~~(~~draws dual unit ~~(~~3.7~~)~~ to ~~(~~4.6~~)~~ amperes per phase~~)~~ for each of the two 9-element units.

   a) Inboard ~~valves,~~ system ~~60~~ 15" H_2O at 100 scfm.

   b) Outboard ~~valves,~~ system ≥~~50~~ 15" H_2O at ~~240~~ 100 scfm for each of the two blowers.

d. By verifying the ~~(~~flow, pressure, temperature and pressure differential~~)~~ ~~(operating)~~ instrumentation to be OPERABLE per Table 4.6.1.4-1.

INSERT TO 3.6.1.4 ACTIONS

2. With both MSIV leakage control systems inoperable, restore at least one of the leakage control systems 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.

3. The provisions of Specification 3.0.4 are not applicable.

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TABLE 4.6.1.4-1

MSIV LEAKAGE CONTROL SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	SETPOINTS	
1. FLOW				
a. MSIV-LC Inbd. Flow	M	R	≤ 28 scfh	CPS
b. MSIV-LC Inbd. Flow	M	R	≤ 28 scfh	
c. MSIV-LC Inbd. Flow	M	R	≤ 28 scfh	
d. MSIV-LC Inbd. Flow	M	R	≤ 28 scfh	
2. PRESSURE — SEE INSERT (NEXT PAGE)				CPS
a. MSIV-LC Inbd. Pressure	M	R	--	
b. MSIV-LC Inbd. Pressure	M	R	--	
c. MSIV-LC Inbd. Pressure	M	R	--	
d. MSIV-LC Inbd. Pressure	M	R	--	
e. Reactor Pressure	M	R	--	
f. MSIV-LC Inbd. Pressure	M	R	--	
3. TEMPERATURE				
a. MSIV-LC Inbd. Temperature	M	R	~~NA~~ N/A*	CPS
b. MSIV-LC Inbd. Temperature	M	R	~~NA~~ N/A*	
c. MSIV-LC Inbd. Temperature	M	R	~~NA~~ NA*	
d. MSIV-LC Inbd. Temperature	M	R	~~NA~~ NA*	
4. PRESSURE DIFFERENTIAL				
a. Dilution Flow ΔP-Inbd.	M	R	≤ ~~7"~~ 5" H_2O	CPS
b. Dilution Flow ΔP-Outbd.	M	R	≤ ~~20"~~ 15" H_2O	

* Heater OPERABILITY verified by demonstrating electrical continuity of the heating element circuitry covered in 4.6.1.4.a.2. | CPS

INSERT TO TABLE 4.6.1.4-1

p. 3/4 6-8

2. PRESSURE	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	SETPOINTS
a. MSIV-LC Inbd. Pressure Reactor Pressure Interlock	M	R	≤ 20 psig
b. MSIV-LC Inbd. Pressure Isolation	M	R	≤ 5 psig
c. MSIV-LC Inbd. Pressure Steamline Pressure Interlock	M	R	≤ 20 psig
d. MSIV-LC Outbd. Pressure Reactor Pressure Interlock	M	R	≤ 20 psig
e. MSIV-LC Outbd. Pressure Steamline Pressure Interlock	M	R	≤ 20 psig
f. MSIV-LC Outbd Pressure Depressurization Check	M	R	≤ 0.5 psig

DRAFT

CONTAINMENT STRUCTURAL INTEGRITY

NO CHANGE

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.6.1.5.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

DRAFT

CONTAINMENT INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Containment to secondary containment differential pressure shall be maintained between ~~-0.1~~ -0.25 and ~~+1.5~~ +0.25 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the containment to secondary containment differential pressure outside of the specified limits, restore the differential pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The containment to secondary containment differential pressure shall be determined to be within the limits at least once per 12 hours.

DRAFT

CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Containment average air temperature shall not exceed 120°F.

APPLICABILITY: OPERATIONAL CONDITION: 1, 2 and 3.

ACTION:

With the containment average air temperature greater than 120°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 containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	________	________
b.	________	________
c.	________	________
d.	________	________
e.	________	________
f.	________	________

DRAFT

(DRYWELL AND) CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The (drywell and) containment purge (6) inch supply and exhaust isolation valves shall be OPERABLE and:

a. Each (20) inch purge valve shall be sealed closed.

b. Each (6) inch purge valve may be open for purge system operation with such operation limited to (90) hours per 365 days for reducing airborne activity pressure control.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a (20) inch (drywell and) containment 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.

b. With a (6) inch (drywell and) containment purge supply and/or exhaust isolation valve(s) inoperable or open (for more than (90) hours per 365 days) for other than inverting, deinverting on 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.

(c. With a (drywell and) containment purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.3 and/or 4.6.1.8.4, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.)

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Each (20) inch (drywell and) containment purge supply and exhaust isolation valve shall be verified to be sealed closed at least once per 31 days.

(4.6.1.8.2 The cumulative time that the (6) inch (drywell and) containment purge supply and exhaust isolation valves have been open during the past 365 days shall be determined at least once per 7 days.)

DRAFT

SURVEILLANCE REQUIREMENTS (Continued)

(4.6.1.8.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed (20) inch (drywell and) containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.05) L_a when pressurized to P_a.)

(4.6.1.8.4 At least once per 92 days each (6) inch (drywell and) containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.01) L_a when pressurized to P_a.)

DRAFT

WATER POSITIVE SEAL ISOLATION VALVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Two independent water positive seal isolation valve leakage control system (WPS-IVLCS) divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one WPS-IVLCS division inoperable, restore the inoperable division to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9 Each WPS-IVLCS division shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the:

   1. Division 1:

      a) Sealing water supply tank water level greater than or equal to ___ inches.

      b) Sealing water supply tank pressure between ___ and ___ psig.

   2. Division 2 fuel pool water level greater than or equal to ___ feet.

NA-CPS

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

c. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the system throughout its operating sequence, and verifying that each automatic valve actuates to its correct position.

   2. Demonstrating that the Division 1 APS-IVLCS maintains the Division 1 WPS-IVLCS seal water supply tank pressure between ___ and ___ psig when ________________.

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

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

   2. CHANNEL CALIBRATION at least once per 18 months.

DRAFT

AIR POSITIVE SEAL ISOLATION VALVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.10 Two independent air positive seal isolation valve leakage control system APS-IVLCS divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one APS-IVLCS division inoperable, restore the inoperable division to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.10 Each APS-IVLCS shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying division pressure greater than or equal to ___ psig.

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

c. At least once per 18 months by performance of a functional test which includes simulated actuation of the system throughout its operating sequence, and verifying that each automatic valve actuates to its correct position.

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

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

   2. CHANNEL CALIBRATION at least once per 18 months.

NA-CPS

DRAFT

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. After each opening ~~closing~~ of the drywell equipment hatch by visual inspections of the seals during closing. CPS

b. At least once per 31 days by verifying that all drywell penetrations** not capable of being closed by OPERABLE design bases drywell automatic ~~isolation~~ valves and required to be closed during ~~accident~~ conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4. CPS

c. By verifying at least one of the drywell airlock doors is closed and sealed. ~~each drywell air lock OPERABLE per Specification 3.6.2.3.~~ (A visual inspection will be performed at least once per 31 days.) CPS

d. By verifying the suppression pool OPERABLE per Specification 3.6.3.1.

e. By verifying at least once per 6 months that only one door in the air lock can be opened at a time. CPS

*See Special Test Exception 3.10.1.

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

DRAFT

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell ~~bypass~~ capability ~~leakage~~ shall be less than or equal to ~~10% of the minimum acceptable A/√k design value of 1.00 ft².~~ the maximum allowable leakage considering an A/√k value of 1.0 ft² and a differential pressure of 3 psi.

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the maximum allowable ~~minimum acceptable A/√k design value of 1.00 ft²~~, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.2 ~~The~~ A drywell bypass leakage rate test shall be conducted on the same schedule as the containment integrated leakage rate test specified in surveillance requirement 4.6.1.2.a ~~at least once per 18 months~~ at a ~~n~~ ~~initial~~ differential pressure of 3.0 psi. ~~and the A/√k shall be calculated from the measured leakage~~. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test.

Xa. If any drywell bypass leakage test fails to meet the specified limit, the schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least every 18 9 months until two consecutive tests meet the limit, at which time the ~~18 month~~ test schedule in 4.6.1.2.a may be resumed. X

b. The provisions of Specification 4.0.2 are not applicable.

The acceptable leak rate at 3 psi differential pressure is 3600 scfm which corresponds to 10% of the maximum allowable leakage rate using a value of $A/\sqrt{k} = 1.0$ ft².

DRAFT

DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 Each drywell air lock shall be OPERABLE with:

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

(b. An overall air lock leakage rate of less than or equal to (2) scf per hour at P_a, 9.0 psig.)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one drywell air lock door inoperable:

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

2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

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

4. The provisions of Specification 3.0.4 are not applicable.

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

NA-GPS

*See Special Test Exception 3.10.1.

DRAFT

SURVEILLANCE REQUIREMENTS

~~4.6.2.3 Each drywell air lock shall be demonstrated OPERABLE:~~

~~(a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to (2) scf per hour when the gap between the door seals is pressurized to P_a, 9.0 psig.)~~

~~(b. By conducting an overall air lock leakage test at (P_a), 9.0 psig and verifying that the overall air lock leakage rate is within its limit:~~

~~1. At least once per 6 months#.~~

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

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

~~(#The provisions of Specification 4.0.2 are not applicable.)~~

~~(*Exemption to Appendix J of 10 CFR 50.)~~

~~(**Except that the inner door need not be opened to verify interlock OPERABILITY when the drywell is inerted, provided that the inner door interlock is tested within 8 hours after the drywell is de-inerted.)~~

DRAFT

DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.2.4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CF

4.6.2.4.2 Reports Any abnormal degradation of the drywell structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

DRAFT

DRYWELL INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.2.5 Drywell to containment differential pressure shall be maintained between -0.1 and ~~+1.5~~ +1.0 psid. CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.5 The drywell to containment differential pressure shall be determined to be within the limits at least once per 12 hours.

DRAFT

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.2.6 Drywell average air temperature shall not exceed ~~150~~ 135°F. CP

APPLICABILITY: OPERATONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than ~~150~~ 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. CP

SURVEILLANCE REQUIREMENTS

4.6.2.6 The drywell average air temperature shall be the ~~arithmetical~~ average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours: CPS

	Elevation	Azimuth
~~a.~~		
~~b.~~		
~~c.~~		
~~d.~~		
~~e.~~		
~~f.~~		

SEE INSERT (NEXT PAGE) CPS

DRAFT

3/4.6.3 DEPRESSURIZATION SYSTEMS

NO CHANGE

SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Volume between 146,400 ft^3 and 150,300 ft^3, equivalent to a level between 18'11" and 19'5", and a

b. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

1. 105°F during testing which adds heat to the suppression pool.
2. 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.
3. 120°F with the main steam line isolation valves closed following a scram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

b. In OPERATIONAL CONDITION 1 or 2 with the suppression pool average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression pool average water temperature greater than 105°F during testing which adds heat to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the suppression pool average water temperature greater than:
   a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
   b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.
3. With the suppression pool average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water temperature instrumentation channel in any redundant ~~any~~ pair~~(s)~~ of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel~~(s)~~ to OPERABLE status within 7 days ~~or verify suppression pool water temperature to be within the limits at least once per 12 hours.~~ (a sector being any three sequentially located SRV downcomers). CPS

d. With both suppression pool water temperature instrumentation channels in any redundant ~~any~~ pair~~(s)~~ of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~.~~ (a sector being any three sequentially located SRV downcomers). CPS

SURVEILLANCE REQUIREMENTS

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

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

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression pool average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

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

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

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying (at least) sixteen suppression pool water temperature instrumentation channels, at least two channels in each suppression pool sector, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months,

with the water high temperature alarm setpoint for ≤ ~~( )~~ 93°F

CPS

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(CONTAINMENT~~) (AND) (DRYWELL)~~ SPRAY

CPS

LIMITING CONDITION FOR OPERATION

3.6.3.2 The (containment~~) (and) (drywell)~~ spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of: CPS

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression pool through a ~~shutdown service water~~ RHR heat exchanger, and the (containment) (and) (drywell) spray sparger. CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one (containment~~) (and/or) (drywell)~~ spray 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. CPS

b. With both (containment~~) (and/or) (drywell)~~ spray loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours. CPS

SURVEILLANCE REQUIREMENTS

4.6.3.2 The (containment~~) (and) (drywell)~~ spray mode of the RHR system shall be demonstrated OPERABLE: CPS

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 300 gpm on recirculation flow through the RHR heat exchanger~~(s) and the suppression pool spray sparger~~ when tested pursuant to Specification 4.0.5. CPS

c. At least once per 18 months by performance of a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual spraying of coolant into the (containment~~) (drywell)~~ may be excluded from this test. CPS

(d. By performance of an air or smoke flow test of the (containment~~) (and) (drywell)~~ spray nozzles at least once per 5 years and verifying that each spray nozzle is unobstructed.) CPS

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

CONTAINMENT SYSTEMS

DRAFT

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump; and

b. An OPERABLE flow path capable of recirculating water from the suppression pool through a ~~shutdown service water~~ RHR heat exchanger. CP

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least ~~7,700~~ 5050 gpm on recirculation flow through the RHR heat exchangers to the suppression pool when tested pursuant to Specification 4.0.5. CP

*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.

DRAFT

SUPPRESSION POOL MAKEUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.4 The suppression pool makeup system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool makeup line inoperable, restore the inoperable makeup line 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 the upper containment pool water level less than the limit, restore the water level to within the limit 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 upper containment pool water temperature greater than the limit, restore the upper containment pool water temperature to within the limit 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

4.6.3.4 The suppression pool makeup system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the upper containment pool water:

   1. Level to be greater than or equal to ~~17'0"~~ 827'-3", and CPS

   2. Temperature to be less than or equal to 120°F.

b. At least once per 31 days by verifying that:

   1. The upper containment steam dryer storage pool/reactor pool gate is in position (closed). CPS

   ~~(1. The drywell storage/fuel transfer pool gate is removed.)~~

   2. Each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secure in position, is in its correct position.

c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual makeup of water to the suppression pool may be excluded from this test.

DRAFT

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The containment and drywell isolation valves ~~and the reactor instrumentation line excess flow check valves~~ shown in Table 3.6.4-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.4-1. CPS

APPLICABILITY: (As shown in Table 3.6.4-1.) (OPERATIONAL CONDITIONS 1, 2 and 3 and **)

ACTION:

a. With one or more of the containment or drywell isolation valves shown in Table 3.6.4-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

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

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

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

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

   Otherwise, in Operational Condition **, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

~~b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.4-1 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:~~ CPS

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.4.2 Each automatic isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on an isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic valve shown in Table 3.6.4-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

~~4.6.4.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a (10) psid differential pressure.~~

DELETE

~~(4.6.4.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:~~

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

~~[b. At least once per 18 months by removing (at least one) (the) explosive squib(s) from (at least one) (the) explosive valve (, such that each explosive squib in each explosive valve will be tested at least once per 36 months) and initiating the explosive squib(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.])~~

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TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
1. Automatic Isolation Valves						
a. Primary Containment						
1) Main Steam Line C	5	1	1, 2, 3		No	9.0
1B21-F022C				3-5		
1B21-F028C				3-5		
1B21-F067C				STD		|CPS
2) Main Steam Line A	6	1	1, 2, 3		No	9.0
1B21-F022A				3-5		
1B21-F028A				3-5		
1B21-F067A				STD		|CPS
3) Main Steam Line D	7	1	1, 2, 3		No	9.0
1B21-F022D				3-5		
1B21-F028D				3-5		
1B21-F067D				STD		|CPS
4) Main Steam Line B	8	1	1, 2, 3		No	9.0
1B21-F022B				3-5		
1B21-F028B				3-5		
1B21-F067B				STD		|CPS
5) Feedwater/RHR Line A	9		1, 2, 3			9.0
1B21-F032A				0.5	Yes	|CPS
1E12-F053A		3		39		

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	
Automatic Isolation Valves (Continued)							
Primary Containment (Continued)							
6) Feedwater/RHR Line B	10		1, 2, 3				
1B21-F032B				0.5	Yes	9.0	CPS
1E12-F053B		3		39	NO		
7) RHR Shutdown Cooling	14	3	1, 2, 3		~~Yes~~ g No	9.0	CPS
1E12-F008				39			
1E12-F009				39			
8) RHR A To Fuel Pool Cooling	15	3	1, 2, 3			9.0	
1E12-F037A ~~(b)~~ e				STD	No		CPS
9) RHR B To Fuel Pool Cooling	16	3	1, 2, 3			9.0	
1E12-F037B ~~(b)~~ e				STD	No		CPS
10) RHR A/LPCS Test Line	18	11	1, 2, 3 & ##		No	9.9	
1E12-F024A				90			
1E12-F011A, 1E12-F011				STD			
1E21-F012				STD			CPS
1E12-F064A				8			
11) RHR C Test Line	19	11	1, 2, 3 & ##		No	9.9	
1E12-F021				STD			
1E12-F064C				8			CPS
12) RHR B Test Line	20	11	1, 2, 3 & ##		No	9.9	
1E12-F024B				90			
1E12-F011B				STD			
13) 1E12-F064B				8			CPS
14 ~~13~~) HPCS Test Line	33	8	1, 2, 3 & ##		No	9.9	
1E22-F023				STD			
(13) RCIC Suction	28		1, 2, 3	STD	NO	9.9	
1E51-F031							

Renumber from here

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Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINEMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
Automatic Isolation Valves (Continued)						
Primary Containment (Continued)						
14) Supp. Pool Cleanup Suction	34	8	1, 2, 3 & ##		Yes	9.9
1S-F004				STD		
> INSERT IN PENETRATION ORDER ATTACHED						
15) RCIC/RHR Head Spray	42		1,2,3		NO	9.0
1E51-F013				15		
1E~~51~~ 12-F023				STD		
16) RCIC Steam Supply	43	6	1, 2, 3		~~Yes~~ No	9.0
1E51-F063				33		
1E51-F064				33		
1E51-F076				STD		
17) RCIC Turb Vac BKR Line	44	7	1, 2, 3		No	9.0
~~1E51-F077~~				~~9~~		
1E51-F078				STD		
18) Main Steam Drain Line	45	1	1, 2, 3		Yes	9.0
1B21-F016				STD		
1B21-F019				STD		
19) Comp. Cooling Water Supply	46	8	1, 2, 3		Yes	9.0
1CC049				STD		
1CC050				STD		
1CC12~~6~~ 7				STD		
20) Comp. Cooling Water Return	47	8	1, 2, 3		Yes	9.0
1CC053				STD		
1CC054				STD		
1CC060				STD		

CPS CPS CPS CPS CPS CPS

INSERT INTO TABLE 3.6.4-1 IN Penetration Order

HPCS Injection Line 1E22-F004	35	1, 2, 3	27	NO	9.0
RCIC 1E51-F019	40	1, 2, 3	STD	NO	9.9
RCIC 1E51-F077	41	1, 2, 3	STD	NO	9.0

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
Automatic Isolation Valves (Continued)						
Primary Containment (Continued)						
~~21) SX To Contm. Cooler~~	~~48~~	~~8~~	~~____~~		~~No~~	~~9.9~~
~~1SX088A~~						
~~1SX089A~~						
INSERT Attached						
22) Make-up Condensate	50	8	1, 2, 3		Yes	9.0
0MC009				STD		
0MC010				STD		
23) Fuel Pool Cool/ Cleanup Supply	52	8	1, 2, 3		No	9.0
1FC036				STD		
1FC037				STD		
24) Fuel Pool Cool/ Cleanup Return	53	8	1, 2, 3		No	9.0
1FC007				STD		
1FC008				STD		
25) Fire Protection	56	8	1, 2, 3		Yes	9.0
1FP052				STD		
1FP051				STD		
26) Instrument Air Supply	57	8	1, 2, 3		Yes	9.0
1IA005				≤5		
1IA006				≤5		
27) Instrument Air Bottles	58	8	1, 2, 3		Yes	9.0
1A012B				STD		
IA012A				STD		

Breathing Air	49	1, 2, 3	STD	YES	9.0
ORA 026					
ORA 027			STD		

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Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	
Automatic Isolation Valves (Continued)							
Primary Containment (Continued)							
28) Service Air Supply	59	8	1, 2, 3		Yes	9.0	
1SA030				≤ 5			CPS
1SA029				≤ 5			
29) RWCU Suction Line	60	4	1, 2, 3		No	9.0	
1G33-F001				15			
1G33-F004				15			
30) RWCU Return To Filter	61	4	1, 2, 3		No	9.0	
1G33-F053				15			CPS
1G33-F054				15			
31) Hydrogen Recombiner Supply	62	8	1, 2, 3		Yes	9.0	CPS
1HG008				STD			
32) RWCU To RHR/FW	64	4	1, 2, 3		No	9.0	
1G33-F040				15			
1G33-F039				15			
33) RWCU Transfer To Radwaste	65	8	1, 2, 3 & ##		Yes	9.0	
1WX019				≤1			CPS
1WX020				≤1			
34) DW/Cont. Equip. Drain	69	8	1, 2, 3		No	9.0	
1RE021				STD			CPS
1RE022				STD			

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Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
Automatic Isolation Valves (Continued)						
Primary Containment (Continued)						
35) DW/Cont. Floor Drain	70	8	1, 2, 3		No	9.0
1RF021				STD		
1RF022				STD		
36) Hydrogen Recombiner Supply	71	8	1, 2, 3		Yes	9.0
1HG001				STD		
37) Cycle Condensate	85	8	1, 2, 3		Yes	9.0
1CY017				STD		
1CY016				STD		
38) RWCU Letdown	86	4	1, 2, 3 & ##			9.0
1G33-F028				15	Yes	
1G33-F034				15		
39) SX From Recir. Pump	88	11	1, 2, 3		No	9.~~9~~0
1CC071				STD		
1CC072				STD		
40) Containment HVAC Supply	101		1, 2, 3		Yes	9.0
1VR001A		10		6 ~~10~~		
~~1VR002A~~		8				
1VR001B		10		6 ~~10~~		
~~1VR002B~~		8				
41) Hydrogen Recombiner Return	72	8	1, 2, 3		Yes	9.0
1HG004				STD		

Place in Penetration Number Order

Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINEMNT AND DRYWELL ISOLATION VALVES

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VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	
Automatic Isolation Valves (Continued)							
Primary Containment (Continued)							
42) SX to Recir. Pump	78	11	1, 2, 3		No	9.~~9~~0	CPS
1CC074				STD			
1CC073				STD			
43) Supp Pool Cleanup Return	79	8	1, 2, 3		Yes	9.~~9~~0	
1SF001				STD			
1SF002				STD			
44) Fire Protection	81	8	1, 2, 3		Yes	9.0	CPS
1FP050				STD			
1FP092				STD			
45) Fire Protection	82	8	1, 2, 3		Yes	9.0	CPS
1FP053				STD			
1FP054				STD			
46) Containment HVAC Exhaust	102		1, 2, 3		Yes	9.0	CPS
1VQ004A		10		~~10~~ 6			
~~1VQ006A~~		8					
1VQ004B		10		~~10~~ 6			
~~1VQ006B~~		8					
47) Plant Chilled Water Supply	103	8	1, 2, 3		Yes	9.0	CPS
1WO001A				STD			
1WO001B				STD			
48) Plant Chilled Water Return	104	8	1, 2, 3		Yes	9.0	CPS
1WO002A				STD			
1WO002B				STD			
Control Bldg HVAC	106		1, 2, 3		Yes	9.0	
1VR007B				6			
1VR007A				6			

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Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
Automatic Isolation Valves (Continued)						
Primary Containment (Continued)						
49) DW Chilled Water Supply	107	8	1, 2, 3		No	9.0
1VP004B				STD		
1VP005B				STD		
50) DW Chilled Water Return	108	8	1, 2, 3		No	9.0
1VP014B				STD		
1VP015B				STD		
51) DW Chilled Water Supply	109	8	1, 2, 3		No	9.0
1VP004A				STD		
1VP005A				STD		
52) DW Chilled Water Return	110	8	1, 2, 3		No	9.0
1VP014A				STD		
1VP015A				STD		
53) Hydrogen Recombiner Supply	166	8	1, 2, 3		Yes	9.0
1HG005				STD		
54) SX Supply	205	8	1, 2, 3		No	9.9
1SX088B				STD		
1SX089B				STD		
55) Instrument Air Bottles	206	8	1, 2, 3		No	9.0
1IA013B				STD		
56) SX From Contm. Cooler	208	8	1, 2, 3		No	9.9
1SX096A				STD		
1SX097A				STD		

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)
Automatic Isolation Valves (Continued)						
b. Drywell						
1) Drywell HVAC Supply	101	10	1, 2, 3	N/A	No	N/A
1VQ001A						
1VQ001B						
2) Drywell HVAC Exhaust	102		1, 2, 3	N/A	No	N/A
1VQ002		10				
1VQ005		10				
1VQ003		3				

CPS

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS	
2. Manual Isolation Valves					
a. Primary Containment					
~~None~~ 1) RHR/LPCI A Injection 1E12-F044A	15	NO	9.0	At All Times[a]	CPS
2) RHR/LPCI B Injection 1E12-F044B	16	NO	9.0	At All Times[a]	CPS
b. Drywell					
None					
3. Test Connections, Vents and Drains[b]					
a. Primary Containment					
1) Main Steam Line C 1B21-F025C ~~1B21-F026C~~	5	No	N/A	At all times[a]	CPS
2) Main Steam Line B 1B21-F025B ~~1B21-F026B~~	6	No	N/A	At all times[a]	CPS
3) Main Steam Line D 1B21-F025B ~~1B21-F026D~~	7	No	N/A	At all times[a]	CPS
4) Main Stream Line A 1B21-F025A ~~1B21-F026A~~	8	No	N/A	At all times[a]	CPS

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
5) Feedwater/RHR Line A 1B21-F063A ~~1B21-F064A~~ e INSERT Attached	9	~~No~~ e Yes	~~N/A~~ e 9.0	At all times (a)
6) Feedwater/RHR Line B 1B21-F063B ~~1B21-F064B~~ e Insert Attached	10	~~No~~ e Yes	~~N/A~~ e 9.0	At all times (a)
7) RHR A Suction 1E12-F334A 1E12-F335A	11	No	~~N/A~~ e 9.9	At all times (a)
8) RHR B Suction 1E12-F334B 1E12-F335B	12	No	~~N/A~~ e 9.9	At all times (a)
9) RHR C Suction 1E12-F334C 1E12-F335C	13	No	~~N/A~~ e 9.9	At all times (a)
10) RHR Shutdown Cooling 1E12-F001 ~~1E12-F002~~ e	14	No	~~N/A~~ e 9.0	At all times (a)

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INSERT to Table 3.6.4-1

5)		PEN. NUM	
	1B21-F030A	9	NO
	1E12-F058A		NO
	1E12-F349A		NO
6)	1B21-F030B	10	NO
	1E12-F058B		NO
	1E12-F349B		NO

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
11) RHR/LPCI A Injection 1E12-F107A 1E12-F108A 1E12-F331A 1E12-F332A 1E12-F329A 1E12-F330A	15	No	N/A	At all times (a)
12) RHR/LPCI B Injection 1E12-F107B 1E12-F108B 1E12-F331B 1E12-F332B 1E12-F329B 1E12-F330B	16	No	N/A	At all times (a)
13) RHR/LPCI C Injection 1E12-F351 1E12-F352	17	No	N/A	At all times (a)
14) RHR A Test Line 1E12-F365A 1E12-F366A 1E12-F346 1E12-F347	18	No	N/A	At all times (a)

No CHANGE

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
15) RHR C Test Line 1E12-F353 1E12-F354	19	No	N/A	At all times (a)
16) RHR B Test Line 1E12-F365A 1E12-F366A	20	No	N/A	At all times (a)
17) RHR A HX PRV	24	No	N/A	At all times (a)
18) RHR B HX PRV	26	No	N/A	At all times (a)
19) RCIC Pump Suction 1E51-F336 1E51-F337	28	No	N/A	At all times (a)
20) RHR HX PRV	31	No	N/A	At all times (a)
21) LPCS Pump Suction 1E21-F331 1E21-F334	32	No	N/A	At all times (a)
22) HPCS Test To Supp Pool	33	No	N/A	At all times (a)
23) Supp Pool Cleanup Pump Suction 1SF034	34	No	N/A	At all times (a)

NO CHANGE

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
24) HPCS Pump Discharge 1E22-F021 1E22-F022	35	No	N/A	At all times (a)
25) LPCS Pump Discharge 1E21-F013 1E21-F014	36	No	N/A	At all times (a)
26) HPCS Pump Suction	37	No	N/A	At all times (a)
27) RCIC Min Flow	40	No	N/A	At all times (a)
28) RCIC Turb Steam Exhaust 1E51-F041 1E51-F342 1E51-F083 1E51-F343	41	No	N/A	At all times (a)
29) Head Spray 1E51-F367 1E51-F368	42	No	N/A	At all times (a)
30) RCIC Turb Steam Supply 1E51-F399 1E51-F400	43	No	N/A	At all times (a)

NO CHANGE

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
31) RCIC Turb Vacuum Breaker 1E51-F080 1E51-F344 1E51-F082 1E51-F345 1E51-F375 1E51-F376	44	No	N/A	At all times (a)
32) Main Stream Drain Line 1B21F017 1B21F018	45	No	N/A	At all times (a)
33) CCW Supply 1CC164	46	No	N/A	At all times (a)
34) CCW Return 1CC165	47	No	N/A	At all times (a)
35) SX Supply	48	No	N/A	At all times (a)
36) Makeup Condensate 1MC011	50	No	N/A	At all times (a)
37) Fuel Pool Cool/Cleanup Supply 1FC092	52	No	N/A	At all times (a)

NO CHANGE

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
38) Fuel Pool Cool/ Cleanup Return 1FC093	53	No	N/A	At all times [(a)]
39) Fire Protection	56	No	N/A	At all times [(a)]
40) Instrument Air 1IA039	57	No	N/A	At all times [(a)]
41) Instrument Air Bottles	58	No	N/A	At all times [(a)]
42) Service Air Line 1SA047	59	No	N/A	At all times [(a)]
43) RWCU Pump Suction 1G33-F002 1G33-F003	60	No	N/A	At all times [(a)]
44) RWCU Return 1G33-F061 1G33-F062	61	No	N/A	At all times [(a)]
45) Hydrogen Recombiner 1HG019	62	No	N/A	At all times [(a)]
46) CRD Pump Discharge 1C11-F128 1C11-F129	63	No	N/A	At all times [(a)]

NO CHANGE

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
47) RWCU Return	64	No	N/A	At all times (a)
48) Hydrogen Recombiner 1HG016	71	No	N/A	At all times (a)
49) Hydrogen Recombiner 1HG017	72	No	N/A	At all times (a)
50) SX To Recir Pump 1CC170	78	No	N/A	At all times (a)
51) Supp Pool Cleanup Return 1SF023	79	No	N/A	At all times (a)
52) Fire Protection	81	No	N/A	At all times (a)
53) Fire Protection	82	No	N/A	At all times (a)
54) Cycle Condensate 1CY019	85	No	N/A	At all times (a)
55) RWCU Letdown 1G33-F069 1G33-F070	86	No	N/A	At all times (a)
56) SX From Recir Pump 1CC171	88	No	N/A	At all times (a)

NO CHANGE

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)					
Primary Containment (Continued)					
~~57)~~	~~RHR HX A Vent~~ ~~1E12-F404A~~ ~~1E12-F405A~~	~~89~~	~~No~~	~~N/A~~	~~At all times (a)~~
58)	Containment HVAC Supply 1VR003	101	No	9.0 ~~N/A~~	At all times (a)
59)	Containment HVAC Return 1VQ007	102	No	9.0 ~~N/A~~	At all times (a)
	Containment HVAC 1VR011	106	NO	9.0	At all times (a)
60)	Drywell Chilled Water 1VP044B 1VP077D	107	No	9.0 ~~N/A~~	At all times (a)
61)	Drywell Chilled Water 1VP047B 1VP077B	108	No	9.0 ~~N/A~~	At all times (a)
62)	Drywell Chilled Water 1VP044A 1VP077C	109	No	9.0 ~~N/A~~	At all times (a)
63)	Drywell Chilled Water 1VP047A 1VP077A	110	No	N/A	At all times (a)
64)	Hydrogen Recombiner 1HG018	166	No	N/A	At all times (a)
~~65)~~	~~RHR HX B Vent~~ ~~1E12-F404B~~ ~~1E12-F405B~~	~~172~~	~~No~~	~~N/A~~	~~At all times (a)~~

INSERT ATTACHED

CPS

CPS

CPS

INSERT to Table 3.6.4-1

Containment HVAC				
1VR012	113	No	9.0	At all times(a)
Standby Liquid Control				
1C41-F322	116	No	9.0	At all times(a)

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Test Connections, Vents and Drains (Continued)				
Primary Containment (Continued)				
66) SX Return	204	No	N/A	At all times (a)
67) SX Supply 1SX132	205	No	N/A	At all times (a)
68) Instrument Air Bottles	206	No	N/A	At all times (a)
69) SX Return	208	No	N/A	At all times (a)
b. Drywell				
1) Drywell HVAC Supply 1VQ011	101	No	N/A	At all times (a)
2) Drywell HVAC Exhaust 1VQ012	102	No	N/A	At all times (a)
4. Other Isolation Valves				
a. Primary Containment				
1) Main Steam Line C 1E31-F001J 1B21-F098C(c)	5	No	9.0	1, 2, 3
2) Main Steam Line A 1E32-F001A 1B21-F098A(c)	6	No	9.0	1, 2, 3

No CHANGE

DRAFT

Renumber as necessary

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Other Isolation Valves (Continued)				
Primary Containment (Continued)				
3) Main Steam Line D 1E32-F001N 1B21-F098D(c)	7	No	9.0	1, 2, 3
4) Main Steam Line B 1E32F001E 1B21F098D(c)	8	No	9.0	1, 2, 3
5) Feedwater/RHR Line A 1B21-F010A 1B21-F065A	9	Yes	9.0	1, 2, 3
6) Feedwater/RHR Line B 1B21-F010B 1B21-F065B	10	Yes	9.0	1, 2, 3
7) RHR A Suction Line 1E12-F004A	11	No	9.9	1, 2, 3
8) RHR B Suction Line 1E12-F004B	12	No	9.9	1, 2, 3
9) RHR C Suction Line 1E12-F105	13	No	9.9	1, 2, 3
RHR A Shutdown Cooling 1E12-F467	14	No	9.0	At all times(a)

CPS

Renumber as necessary

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS	
Other Isolation Valves (Continued)						
Primary Containment (Continued)						
10)	RHR/LCPI A Injection 1E12-F027A 1E12-F042A~~(c)~~ 1E12-F028A~~(c)~~	15	No	9.0	1, 2, 3	CPS
11)	RHR/LPCI B Injection 1E12-F027B 1E12-F042B~~(c)~~ 1E12-F028B~~(c)~~	16	No	9.0	1, 2, 3	CPS
12)	RHR/LPCI C Injection 1E12F042C ~~1E12F041C(c)~~	17	No	9.0	1, 2, 3	CPS
13)	RHR A Suction Relief ~~1E12-F005(c)~~ 1E12-F017A	21	No	9.9	1, 2, 3	CPS
14)	RHR Shutdown Cool Relief 1E12-F005~~(c)~~	23	No	9.0	1, 2, 3	CPS
15)	RHR A HX Relief Line 1E12-F055A~~(c)~~ ~~1E12-F103A(c)~~ ~~1E12-F104A(c)~~	24	No	9.0	1, 2, 3	CPS

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

Renumber as necessary

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Other Isolation Valves (Continued)				
Primary Containment (Continued)				
16) RHR B Suction Relief 1E12-F017B(c)	25	No	9.0	1, 2, 3
17) RHR B HX Relief Line 1E12-F055B(c) 1E12-F103B(c) 1E12-F104B(c)	26	No	9.0	1, 2, 3
18) RHR/LPCI B Inj. Relief 1E12-F025B(c)	27	No	9.0	1, 2, 3
19) RCIC Suction 1E51-F031	28	No	9.9	1, 2, 3
20) RHR B Suction Relief 1E12-F101(c)	29	No	9.0	1, 2, 3
21) RHR/LPCI C Inj. Relief 1E12-F025C(c)	30	No	9.0	1, 2, 3
22) RHR to RCIC Suction Relief 1E12-F036(c)	31	No	9.0	1, 2, 3
23) LPCS Suction Line 1E21-F001	32	No	9.9	1, 2, 3

No CHANGE

Renumber as necessary

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS	
Other Isolation Valves (Continued)					
Primary Containment (Continued)					
24) HPCS Test Line Relief 1E22-F014 1E22-F035 1E22-F039	33	No	9.9	1, 2, 3	
~~25) HPCS Injection Line~~ ~~1E22-F004~~ ~~1E22-F005(c)~~	~~35~~	~~No~~	~~9.0~~	~~1, 2, 3~~	CPS
26) LPCS Injection Line 1E21-F005 ~~1E21-F005(c)~~	36	No	9.0	1, 2, 3	CPS
27) HPCS Injection Line 1E21-F015	37	No	9.9 ~~9.0~~	1, 2, 3	CPS
28) LPCS Pump Relief Line 1E21-F018 1E21-F031	38	No	9.9 ~~9.0~~	1, 2, 3	CPS
29) RCIC Min Flow Relief 1E51-F090	40	No	9.9 ~~9.0~~	1, 2, 3	CPS
30) RCIC Turbine Exhaust 1E51-F068	41	No	9.9	1, 2, 3	

Renumber as necessary

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS	
Other Isolation Valves (Continued)					
Primary Containment (Continued)					
~~31) RCIC/RHR Head Spray~~	~~42~~	~~No~~	~~9.0~~	~~1, 2, 3~~	CPS
~~1E12-F019(c)~~					
~~1E51-F065(c)~~					
Insert ~~1E51-F066(c)~~					
32) Instrument Air Bottles	58	Yes	9.0	1, 2, 3	CPS
1IA042B					
> INSERT					
33) RHR Flush Line	76	No	9.9	1, 2, 3	
1E12-F030					
34) RHR/LPCI A Injec. Relief	87	No	9.0	1, 2, 3	
1E12-F025(c)					
35) RHR HX A Vent	89	No	9.9	1, 2, 3	
1E12-F074A					
1E12-F073A(c)					
1E12-F110A(c)					
1E12-F111A(c)					
36) DW Chilled Water Relief	107	No	9.0	1, 2, 3	CPS
1VP023B(c)					

INSERT

CLINTON-1

INSERT 3/4 6-54A

Insert to Table 3.6.4-1

Penetration Number Order

SX to Containment Cooler				
1SX088A				
1SX089A	48	NO	9.0	
CRD	63	S	9.0	1,2,3
1C11-F122				
1C11-F083				
Containment HVAC Supply	101	YES	9.0	1,2,3
1VR002A				
1VR002B				
Containment HVAC Exhaust	102	YES	9.0	1,2,3
1VQ006A				
1VQ006B				

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

Renumber as necessary

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS	
Other Isolation Valves (Continued)					
Primary Containment (Continued)					
37) DW Chilled Water Relief 1VP027B	108	No	9.0	1, 2, 3	
38) DW Chilled Water Relief 1VP023A	109	No	9.0	1, 2, 3	
Insert					CPS
39) DW Chilled Water 1VP027A	110	No	9.0	1, 2, 3	
40) Containment Press 1CM003A(c)(d)	150	No	N/A	1, 2, 3	CPS
Insert					
41) Suppression Pool Level 1CM002A(c)(d) 1CM003~~A~~ B(c)(d)	157	No	N/A	1, 2, 3	
42) Suppression Pool Level ~~1CM001A(c)(d)~~ 1SM010(c)(d) ~~1E51-F377A(c)(d)~~ ~~1E51-F377C(c)(d)~~	158	No		1, 2, 3	

[illegible]sert to Table 3.6.4-1

Penetration o[illegible] number

Standby Liquid Control	116	No	9.0	1,2,3
1C41-F317				
1C41-F318				
Drywell Pressure				
1CM051(d)	151	No	N/A	123

Renumber as necessary

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Other Isolation Valves (Continued)				
Primary Containment (Continued)				
43) Suppression Pool Level 1E22-F330(c)(d) 1E22-F334(c)(d)	159	No		1, 2, 3
44) Suppression Pool 1SM009(c)(d)	160	No		1, 2, 3
45) RHR HX Vent B 1E12-F074B 1E12-F073B(c) 1E12-F110B(c) 1E12-F111B(c)	172	No	9.9	1, 2, 3
46) Suppression Pool Level 1E51-F377B(c)(d) 1E51-F377D(c)(d) 1SM011(c)(d) 1CM004B(c)	177	No		1, 2, 3
47) Suppression Pool Level 1E11-F328 1E22-F332	179	No		1, 2, 3

No CHANGE

Renumber as necessary

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Other Isolation Valves (Continued)				
Primary Containment (Continued)				
48) Suppression Pool Level 1SM008(c)(d)	181	No	N/A	1, 2, 3
49) Supression Pool Level 1CM002B(c)(d)	183	No	N/A	1, 2, 3
INSERT Attached				
50) Instrument Air Bottles 1IA042A	206	No	9.0	1, 2, 3
51) RHR A/LPCS Test Line 1E12-F064A 1E21-F011	18	No	9.9	1, 2, 3 & ##
52) RHRC Test Line 1E12-F064C	19	No	9.9	1, 2, 3 & ##
53) RHR B Test Line 1E12-F064B	20	No	9.9	1, 2, 3 & ##
54) HPCS Test Line 1E22-F012	33	No	9.9	1, 2, 3 & ##
55) RCIC Min. FLow Line 1E51-F019	40	No	9.9	1, 2, 3

CPS

CLINTON-1

Insert to Table 3.6.4-1

RCIC 1E51-F377A (d)(e)	200	No	N/A	1,2,3
Drywell Pressure 1CM053 (d)	203	No	N/A	1,2,3

Insert 3/4 6-57A

DRAFT

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	APPLICABLE OPERATIONAL CONDITIONS
Other Isolation Valves (Continued)				
Primary Containment (Continued)				
56) Instrument Air Bottles				
1IA012A	58	Yes	9.0	1, 2, 3
1IA013A	206	No	9.0	1, 2, 3
57) SX Return	204	No	9.9	1, 2, 3
1SX096B				
1SX097B				

b. Drywell

None

No Change

(a)May be opened on an intermittent basis under administrative control.

(b)Not subject to Type C leakage tests - sealed with fluid from a seal system.

(c)Not subject to Type C leakage tests.

(d)Excess flow check valve.

((e)Opens on an isolation signal. Valve(s) will be open during Type A test. Type C test not required.)

(*But greater than or equal to (3) seconds.)

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

(#The provisions of Specification 3.0.4 are not applicable.)

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

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TO CONTAINMENT

3/4.6.5 DRYWELL POST-LOCA VACUUM BREAKERS |CPS

LIMITING CONDITION FOR OPERATION

to containment

3.6.5 All drywell post-LOCA vacuum breakers shall be OPERABLE and closed. |CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

to containment

a. With one drywell post-LOCA 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. |CPS

to containment

b. With one drywell post-LOCA vacuum breaker open, 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. |CPS

to containment |CPS

c. With the position indicator of an OPERABLE drywell post-LOCA vacuum breaker inoperable, verify the vacuum breaker to be closed at least once per 24 hours by visual inspection. Otherwise, declare the vacuum breaker inoperable.

SURVEILLANCE REQUIREMENTS

to containment

4.6.5 Each drywell post-LOCA vacuum breaker shall be: |CPS

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days by

   a) Cycling the vacuum breaker through at least one complete 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) Verifying the pressure differential required to open the vacuum breaker, from the closed position, to be less than or equal to 0.2 psid, and

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

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3/4.6.6 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inches of vacuum water gauge. CP

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.
2. At least one door in each access to the secondary containment is closed.
3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to ~~194~~ 188 seconds, and CP
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 secondary containment at a flow rate not exceeding 4000 CFM.

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

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SECONDARY CONTAINMENT AUTOMATIC ISOLATION (DAMPERS)~~(VALVES)~~ |CPS

LIMITING CONDITION FOR OPERATION

3.6.6.2 The secondary containment ventilation system automatic isolation (dampers)~~(valves)~~ shown in Table 3.6.6.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.6.2-1. |CPS

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation (dampers)~~(valves)~~ shown in Table 3.6.6.2-1 inoperable, maintain at least one isolation (damper)~~(valve)~~ OPERABLE in each affected penetration that is open, and within 8 hours either: |CPS

a. Restore the inoperable (damper)~~(valve)~~(s) to OPERABLE status, or |CPS

b. Isolate each affected penetration by use of at least one deactivated automatic (damper)~~(valve)~~ secured in the isolation position, or |CPS

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

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

Otherwise, in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each secondary containment ventilation system automatic isolation (damper)~~(valve)~~ shown in Table 3.6.6.2-1 shall be demonstrated OPERABLE: |CPS

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 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 secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

THIS PAGE OPEN PENDING RECEIPT OF CONTAINMENT SYSTEMS INFORMATION FROM THE APPLICANT

DRAFT

TABLE 3.6.6.2-1

SECONDARY CONTAINMENT VENTILATION ~~SYSTEM~~ AUTOMATIC ISOLATION (DAMPERS)(VALVES)

(DAMPER)(VALVE) FUNCTION	MAXIMUM ISOLATION TIME* (Seconds)
1. Fuel Building Supply Damper ~~(Valve)~~, Outboard, 1VF004Y	~~(5)~~ 4
2. Fuel Building Supply Damper ~~(Valve)~~, Inboard, 1VF006Y	~~(5)~~ 4
3. Fuel Building Exhaust Damper ~~(Valve)~~, Outboard, 1VF09Y	~~(5)~~ 4
4. Fuel Building Exhaust Damper ~~(Valve)~~, Inboard, 1VF07Y	~~(5)~~ 4

* A margin of 2 seconds is included in the specified value

(#The provisions of Specification 3.0.4 are not applicable.)

DRAFT

No CHANGE

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

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

   2. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

THIS PAGE OPEN PEND...

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~(*)~~ 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%. |CPS

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~(**)~~ 0.175%; and |CPS

3. Verifying a subsystem flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~(**)~~ 0.175%. |CPS

d. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

   a) LOCA, and

   b) Fuel handling accident.

2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~(8)~~ 7.5 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%. |CPS

3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   INSERT Attached

   ~~(b. Simulated automatic imitation signal.)~~ |CPS

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5. Verifying that the heaters dissipate 20.0 ± 2.0 kw when tested in accordance with ANSI N510-~~1975~~ 1980.

Insert to 4.6.6.3.d.3.b

b. Simulated automatic start in response to any one of the following signals:

1) High Drywell Pressure.

2) Low Reactor Water Level-2.

3) High Radiation in exhaust air from the fuel transfer floor of the containment.

4) High Radiation in the containment building ventilation exhaust.

5) High Radiation in the fuel building ventilation exhaust.

6) High Radiation in the continuous containment purge exhaust duct.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~(*)~~ 0.05% in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ~~±~~ + 10%, -0%. CPS

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~(**)~~ 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ~~±~~ + 10%, -0%. CPS

~~(*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).)~~

~~(**0.175% value applicable when a charcoal adsorber efficiency of 99% is assumed, or 1% value applicable when a charcoal adsorber efficiency of 95% is assumed, or 10% value applicable when a charcoal adsorber efficiency of 90% is assumed in the NRC staff's safety evaluation.)~~ CPS

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3/4.6.7 ATMOSPHERE CONTROL

CONTAINMENT ~~AND DRYWELL~~ HYDROGEN RECOMBINER SYSTEMS |CPS

LIMITING CONDITION FOR OPERATION

3.6.7.1 Two independent containment ~~and drywell~~ hydrogen recombiner systems shall be OPERABLE. |CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment ~~and/or drywell~~ 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. |CPS

SURVEILLANCE REQUIREMENTS

4.6.7.1 Each containment ~~and drywell~~ hydrogen recombiner system shall be demonstrated OPERABLE: |CPS

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum reaction chamber temperature increases to greater than or equal to ~~(700)~~ 600 °F within ~~(90)~~ LATER minutes.

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)~~ LATER ohms. |CPS

3. Verifying during a recombiner system functional test that the reaction chamber temperature increases to greater than or equal to 1150°F within 2 hours and is maintained between 1177°F and 1223°F for at least 2 hours.

4. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials, etc.

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SURVEILLANCE REQUIREMENTS (Continued)

c. By measuring the leakage rate:

1. As a part of the average 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 P_a, (15.0) 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.

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CONTAINMENT AND DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.2 Two independent containment and drywell hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one containment and/~~or~~ drywell hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours. CPS

SURVEILLANCE REQUIREMENTS

4.6.7.2 Each containment and drywell hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   ~~1.~~ Starting the system from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least 200 scfm.

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CONTAINMENT AND DRYWELL HYDROGEN IGNITION SYSTEM
~~DRYWELL PURGE SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.6.7.3 Two independent and drywell ~~purge system~~ hydrogen ignition system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: (LATER)

~~With one drywell purge subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.7.3 Each drywell purge system subsystem shall be demonstrated OPERABLE:~~

~~a. At least once per 92 days by:~~

~~1. Starting the subsystem from the control room, and~~

~~2. Verifying that the system operates for at least 15 minutes.~~

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

~~1. Verifying a subsystem flow rate of at least (500) cfm during subsystem operation for at least 15 minutes.~~

~~2. Verifying the pressure differential required to open the vacuum breakers on the drywell purge compressor discharge lines, from the closed position, to be less than or equal to (1.0) psid.~~

~~c. Verifying the OPERABILITY of the drywell purge compressor discharge line vacuum breaker isolation valve differential pressure actuation instrumentation with the opening setpoint of (1.0) psid 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.~~

CPS

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3/4.7 PLANT SYSTEMS

3/4.7.1 SHUTDOWN SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1. Three independent shutdown service water (SX) system subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE SX pump, and

b. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water through~~:~~
~~1. For the Division I and II subsystems, the associated RHR heat exchanger.~~ CF
~~2. For the Division III subsystem, the HPCS pump heat exchanger.~~ See Insert

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

ACTION:

a. With SX Division 1 and/or 2 subsystems inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification(s) ~~3.4.9.2~~ 3.4.9.1, 3.4.9.2, 3.5.1, 3.5.2, 3.6.3.2, 3.6.3.3, 3.9.11.1 and/or 3.9.11.2. CP CP

b. With SX Division III subsystems inoperable, declare the HPCS inoperable and take the ACTION required by Specification 3.5.1 or 3.5.2.

SURVEILLANCE REQUIREMENTS

4.7.1 Each shutdown 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. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to the correct position on a LOCA test signal. CPS

INSERT replacing 3.7.1.b.1 & 2 (p. 3/4 7-1)

the associated RHR heat exchanger, associated ECCS pump room coolers, associated switchgear heat removal condensing units, associated ECCS pump heat exchangers, and associated diesel generator cooling heat exchangers.

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3/4.7.2 CONTROL ROOM VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room ventilation system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one control room ventilation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one control room ventilation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the high radiation mode of operation.

2. With both control room ventilation subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room ventilation 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~~. 104°F CPS

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than (*)% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ~~4000~~ 3000 cfm ± 10%. CPS

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than (**)%; and

3. Verifying a subsystem flow rate of ~~4000~~ 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1975. CPS

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than (**)%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~(6)~~ 8 inches Water Gauge while operating the subsystem at a flow rate of ~~4000~~ 3000 cfm ± 10%. CPS

2. Verifying that on a high chlorine actuation test signal, the subsystem automatically switches to the chlorine mode of operation and the ~~valves~~ dampers close within ~~( )~~ ≤2 seconds.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that on a smoke mode actuation test signal, the subsystem automatically switches to the smoke 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 operation at a flow rate less than or equal to 4000 cfm.

4. Verifying that on a high radiation actuation test signal, the subsystem automatically switches to the high radiation 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 operation at a flow rate less than or equal to ~~4000~~ 3000 cfm. CPS

5. Verifying that the heaters dissipate ~~(7.5) ± (0.75) Kw~~ 16 kW for the make-up filter package when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than (*)% in accordance with ANSI N510-1975 while operating the system at a flow rate of ~~4000~~ 3000 cfm ± 10%. CPS

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than (*)% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of ~~4000~~ 3000 cfm ± 10%. CPS

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).~~ CPS

~~**0.175% value applicable when a charcoal adsorber efficiency of 99% is assumed, or 1% value applicable when a charcoal adsorber efficiency of 95% is assumed, or 10% value applicable when a charcoal adsorber efficiency of 90% is assumed in the NRC staff's safety evaluation.~~ CPS

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3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than ~~130~~ 150 psig. CPS

ACTION: With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to ~~130~~ 150 psig within the following 24 hours. CPS

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

3. Verifying that the pump flow controller is in the correct position.

b. (At least once per 92 days) ~~(when tested pursuant to Specification 4.0.5)~~ by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path (with a system head corresponding to reactor vessel operating pressure) when steam is being supplied to the turbine at 1000 + 20, - 80 psig.* CPS CPS

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

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation ~~(~~and restart~~)~~ and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded. | CPS

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of ~~150 + 15, - 0~~ 135 psig.* | CPS

3. Verifying that the suction for the RCIC system is automatically transferred from the RCIC storage tank to the suppression pool on a RCIC storage tank water level-low signal and on a suppression pool water level - high signal.

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

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3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers listed in Tables 3.7.4-1 and 3.7.4-2 shall be OPERABLE.

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 4.7.4.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7.4-1 and 3.7.4-2. If less than two 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.

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PLANT SYSTEMS

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

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b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) that attachments to the foundation or supporting structure are 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 Surveillance Requirements 4.7.4.d or 4.7.4.e, as applicable. However, when a fluid part of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE by 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 18 months thereafter during shutdown, a representative sample of:

1. At least 10% of the total of hydraulic snubbers listed in Table 3.7.4-1 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Surveillance Requirement 4.7.4.d an additional 10% of that type of snubber shall be functionally tested.
2. That number of mechanical snubbers which follows the expression $35(1 + \frac{c}{2})$, where C=(*), the allowable number of snubbers not meeting the

(*)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.

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SURVEILLANCE REQUIREMENTS (Continued)

No. Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; ~~and~~ or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.4f. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems. |CPS

d. Refueling Outage ~~Transient Event~~ Inspection

>Insert Attached

~~An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.~~ CPS

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

Insert 4.7.4.d

d. At each refueling, the systems which have the potential for a severe dynamic event shall be inspected to determine if there has been a severe dynamic event. In the case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom-of-motion using one of the following: 1) manually induced snubber movement; or 2) evaluation of in-place snubber piston setting; or 3) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced or repaired before returning to power. The requirements of Specification 4.7.8.b are independent of the requirements of this Specification

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SURVEILLANCE REQUIREMENTS (Continued)

Functional Tests (Continued)

acceptance criteria (selected by the licensee), 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 Specification 4.7.4.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 in Table 3.7.4-2 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. Each snubber within 5 feet of heavy equipment ,i.e., valve, pump, turbine, motor, etc.
3. Each snubber within 10 feet of the discharge from a safety relief valve

Tables 3.7.4-1 and 3.7.4-2 may be used jointly or separately as the basis for the sampling plan.(*)

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.)

CPS

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept' line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

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SURVEILLANCE REQUIREMENTS (Continued)

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Functional Tests (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 any 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 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.

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 surveillance 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.

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SURVEILLANCE REQUIREMENTS (Continued)

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

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SURVEILLANCE REQUIREMENTS (Continued)

Replace with Attached Page

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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 6.10.3.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintanence records for each snubber listed in Tables 3.7.4-1 and 3.7.4-2 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.

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SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.4e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Replacement Program

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

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TABLE 3.7.4-1

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON AND LOCATION	SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON AND LOCATION	SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON AND LOCATION
1B33-S301A	Recirculation System,	1B33-S359A	Recirculation System,	B21-S101A	Main Steam System,
-S301B	Drywell	-S359B	Drywell	-S101B	Drywell
-S302A		-S360A		-S101C	
-S302B		-S360B		-S101D	
-S303A		-S361A		-S102A	
-S303B		-S361B		-S102B	
-S304A		-S362A		-S102C	
-S304B		-S362B		-S102D	
-S305A		-S363A		-S103A	
-S305B		-S363B		-S103B	
-S306A		-S369A		-S103C	
-S306B		-S369B		-S103D	
-S351A		-S370A		-S104A	
-S351B		-S370B		-S104B	
-S352A		-S371A		-S104C	
-S352B		-S371B		-S104D	
-S353A		-S372A		-S105A	
-S353B		-S372B		-S105B	
-S354A		-S373A		-S105C	
-S354B		-S373B		-S105D	
-S356A		-S374A		-S106A	
-S356B		-S374B		-S106B	
-S357A		-S375A		-S106C	
-S357B		-S375B		-S106D	
-S358A				-S107A	
-S358B				-S107B	
				-S107C	
				-S107D	
				-S108B	
				-S108C	

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Insert Figure 4.7.4-1 attached

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*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7.4-1 provided that a revision to Table 3.7.4-1 is included with the next License Amendment request.

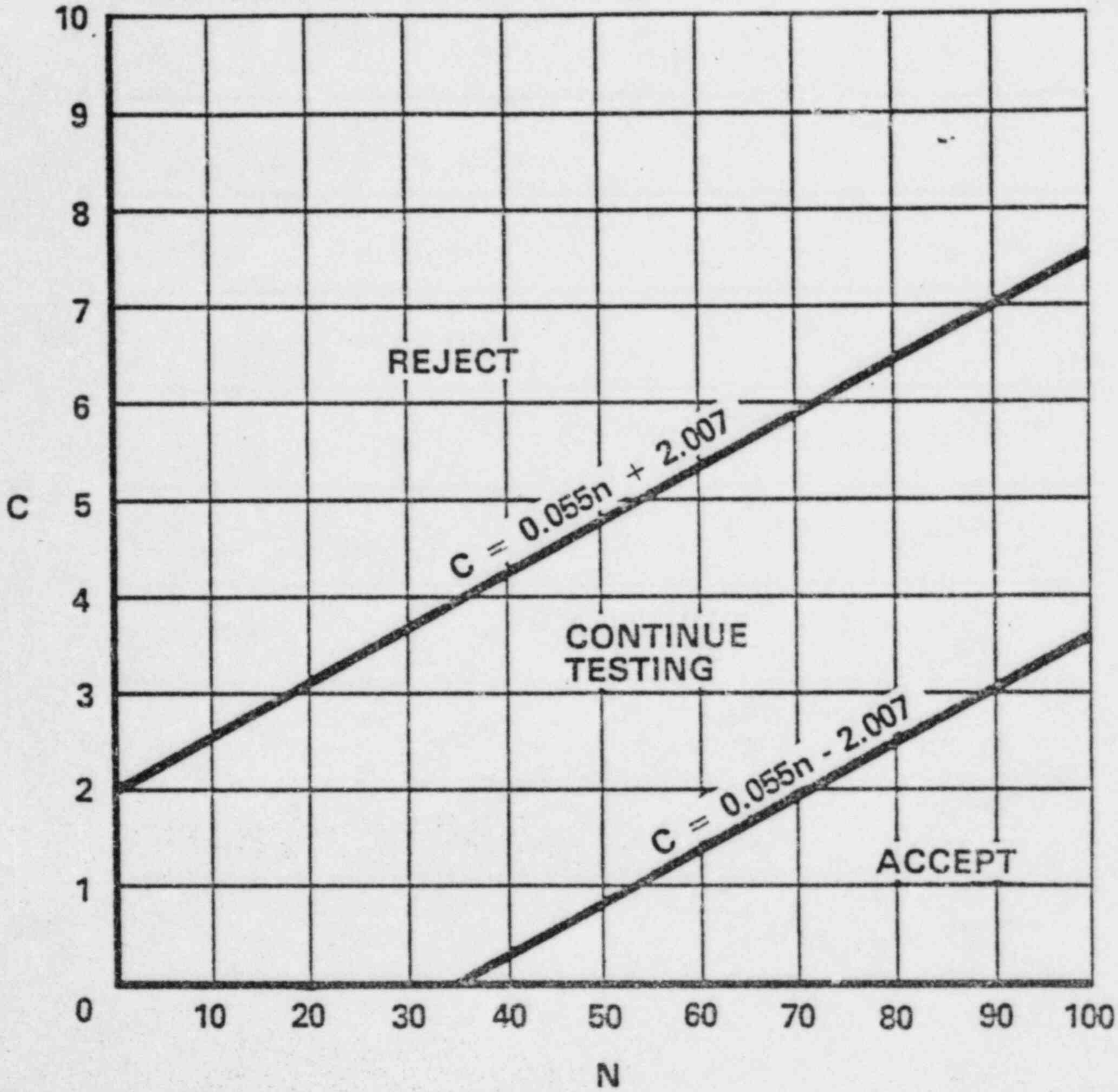

FIGURE 4.7.4-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

TABLE 3.7.4-2

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION

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*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7.4-2 provided that a revision to Table 3.7.4-2 is included with the next License Amendment request.

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3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or ~~5~~ 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination. |CF

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

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No Change

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

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3/4.7.6 FIRE SUPPRESSION SYSTEMS

FIRE PROTECTION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The fire protection water system shall be OPERABLE with:

a. Two fire protection pumps, each with a capacity of 2500 gpm, or one fire protection pump and one plant service water pump, with their discharge aligned to the fire suppression header,

b. An OPERABLE flow path capable of taking suction from the cooling lake and transferring the water through distribution piping with OPERABLE sectionalizing control valves to the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.6.2 and 3.7.6.5. CPS

APPLICABILITY: At all times.

ACTION:

a. With one of the above required pumps inoperable, restore the inoperable pump to OPERABLE status within 7 days or provide an alternate backup pump. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire protection water system otherwise inoperable, establish a backup fire protection water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 The fire protection water system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 6 months by performance of a system flush.

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

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SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each fire protection pump develops at least 2500 gpm at a system head of ~~110~~ 130 psi, CPS

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that fire protection pumps 1A and 1B start sequentially to maintain the fire protection water system pressure greater than or equal to 75 psig and 70 psig, respectively.

e. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.6.1.2 Each diesel driven fire protection pump shall be demonstrated OPERABLE:

a. At least once per 31 days by;

1. Verifying the fuel day tank contains at least 250 gallons of fuel.

2. Starting the pump from ambient conditions and operating for greater than or equal to 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.6.1.3 Each diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each ~~(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~~ 1.260,

3. The ~~(pilot) cell~~ individual battery voltage is greater than or equal to ~~(24)~~ 12 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 the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. Battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

4.7.6.1.4 Each plant service water pump required OPERABLE by Specification 3.7.6.1 shall be demonstrated OPERABLE ~~(~~per Specification 4.0.5~~)~~.

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SPRAY AND SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following spray and sprinkler systems shall be OPERABLE:

a.	1FP01SA	Div I	Diesel Gen. Fuel Storage Tank Room
b.	1FP01SB	Div II	Diesel Gen. Fuel Storage Tank Room
c.	1FP01SC	Div III	Diesel Gen. Fuel Storage Tank Room
d.	1FP02SA	Div I	Diesel Gen. Day Tank Room
e.	1FP02SB	Div II	Diesel Gen. Day Tank Room
f.	1FP02SC	Div III	Diesel Gen. Day Tank Room
g.	1FP15SA	Div I	Cable Spread Area
h.	1FP15SB	Div II	Cable Spread Area
i.			Control Room HVAC Charcoal Deluge
j.			Standby Gas Treatment System Charcoal Deluge

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APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each of the above required spray and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on an ____ test signal, and |CPS

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

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NO CHANGE

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity.

3. By a visual inspection of each nozzle's spray area to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray and sprinkler header and verifying each open head spray and sprinkler nozzle is unobstructed.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

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CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The following low pressure and high pressure CO_2 systems shall be OPERABLE:

a. Division I diesel generator room

b. Division II diesel generator room

c. Division III diesel generator room

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 continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.6.3.1 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.

4.7.6.3.2 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 ___ LATER and pressure to be greater than 275 psig, and | CPS

b. At least once per 18 months by verifying:

1. The system, including associated ventilation system fire dampers, actuates, manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each ~~(accessible*)~~ nozzle during a "Puff Test." | CPS

~~(*Accessible nozzles.)~~

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HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The ~~following~~ PGCC Halon systems listed below shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure: |CPS

~~a. (Plant dependent - to be listed by name and location.)(*)~~
~~b.~~ control cabinets (U701 through U744), PGCC under floor area 800'0". |CPS
~~c.~~

APPLICABILITY: Whenever equipment protected by the Halon systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight and pressure.

c. At least once per 18 months by:

   1. Verifying the system, including associated ventilation system fire dampers and fire door release mechanisms, actuates, manually and automatically, upon receipt of a simulated actuation signal, and

   2. Performance of a flow test through ~~(accessible*)~~ headers and nozzles to assure no blockage. |CPS

~~(*Accessible headers and nozzles.)~~ |CPS

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FIRE HOSE STATIONS

No Change

LIMITING CONDITION FOR OPERATION

3.7.6.5 The fire hose stations shown in Table 3.7.6.5-1 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 3.7-5 inoperable, provide gated wye (s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the addition hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the fire hose stations shown in Table 3.7.6.5-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

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TABLE 3.7.6.5-1

NO CHANGE

FIRE HOSE STATIONS

LOCATION AND ELEVATION	HOSE RACK INDENTIFICATION
a. Containment, Elevation 775'	
1. Northeast quadrant outside Main Steam Tunnel	AZ - 30°
2. Norhtwest quadrant near stairs	AZ - 313°
3. South under Fuel Transfer Tube	AZ - 180°
b. Containment, Elevation 778'	
1. Northeast quadrant outside Main Steam Tunnel	AZ - 30°
2. Southeast quadrant near stairs	AZ - 140°
3. Southwest quadrant outside RWCU Backwash Rec. Tank Room	AZ - 246°
4. Northwest quadrant near stairs	AZ - 330°
c. Containment, Elevation 803'	
1. Northeast quadrant near Equipment Hatch	AZ - 35°
2. Southeast quadrant near stairs	AZ - 146°
3. Southwest quadrant near RWCU Precoat Pumps	AZ - 246°
4. Northwest quadrant near stairs	AZ - 318°
d. Containment, Elevation 828'	
1. Northwest quadrant	AZ - 276°
2. Northeast quadrant	AZ - 32°
3. Southeast quadrant	AZ - 140°
e. Fuel Building, Elevation 712'	
1. Outside HPCS Pump Room	AH - 105°
f. Auxilary Building, Elevation 707'	
1. LPCS Pump Room	AB - 124°
2. Outside of RHR Pump 1A Room	U - 121°
3. RHR Pump 1A Room	U - 117°
4. RCIC Pump Room	X - 110°
5. Outside of RHR Pump 1B Room	S - 107°
6. RHR Pump 1B Room	V - 105°
7. RHR Pump 1C Room	AB - 102°

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No Change

TABLE 3.7.6.5-1 (Continued)

FIRE HOSE STATIONS

LOCATION AND ELEVATION		HOSE RACK INDENTIFICATION
g.	Auxilary Building, Elevation 737'	
1.	Outside RHR HX 1A Room	U - 117°
2.	Outside RHR HX 1B Room	U - 110°
3.	Outside of MSIV Rooms	X - 112°
h.	Control/Diesel Gen. Building, Elevation 702'	
1.	Between HZ Recombiners	T - 130°
i.	Control/Diesel Gen. Building, Elevation 719'	
1.	North between 480v Substations O and P	T - 130°
2.	Outside of Diesel Gen. 1A Fuel Oil Storage Tank Room	AE - 128°
3.	Outside of Diesel Gen. 1B Fuel Oil Storage Tank Room	AE - 126°
4.	Diesel Gen. 1A Fuel Oil Storage Tank Room South Wall	AJ - 128°
5.	Diesel Gen. 1C Fuel Oil Storage Tank Room South Wall	AJ - 126°
j.	Control/Diesel Gen. Bldg., Elevation 737'	
1.	Diesel Gen. 1C Room near Entrance	AD - 126°
2.	Diesel Gen. 1C Room East Wall	AF - 126
3.	Diesel Gen. 1A Room near entrance	AC - 129°
4.	Diesel Gen. 1A Room East Wall	AF - 129°
5.	Across from Diesel Gen. 1A Room entrance	AA - 128°
6.	Across from Diesel Gen. 1B Room entrance	AA - 130°
7.	Diesel Gen. 1B Room near entrance	AC - 129°
8.	Diesel Gen. 1B Room West Wall	AF - 129°
k.	Control Building, Elevation 781'	
1.	Stairwell - southwest corner	AA - 124°
2.	Outside Div. 3 Battery Room	AA - 130°
3.	Div. 1 Cable Spreading Area	Y - 128°
4.	Passsage outside Div. 1 Inverter Room	Y - 130°
5.	Outside Div. 4 Inverter Room	V - 124°
6.	Passage outside Div. 2 Inverter Room	V - 130°
7.	Div. 2 Cable Spreading Area	T - 128°

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TABLE 3.7.6.5-1 (Continued)

FIRE HOSE STATIONS

LOCATION AND ELEVATION	HOSE RACK INDENTIFICATION
l. Control Building, Elevation 825'	
1. Next to Auxilary Building HTG MCC	AA - 130°
2. Next to 480v Substation A	AA - 133°
m. Screen House	
1. North of Plant Service Water Strainers	LATER

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NO CHANGE

3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire barrier assemblies , including walls, floor/ceilings, cable tray enclosures and other fire barriers, separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area, and all sealing devices in fire rated assembly penetrations, including fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals and ventilation seals, shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour establish a continuous fire watch on at least one side of the affected assembly(s) and/or sealing device(s) or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly(s) or sealing device(s) and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each of the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE at least once per 18 months by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly.

b. Each fire window, fire damper and associated hardware.

c. At least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

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NO CHANGE

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. That each locked-closed fire door is closed at least once per 7 days.

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test of these mechanisms at least once per 18 months.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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NO CHANGE

3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.8-1 shall be maintained within the limits indicated in Table 3.7.8-1.

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 3.7.8-1:

a. For more than eight hours, in lieu of any report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 The temperature in each of the areas shown in Table 3.7.8-1 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
a. RHR "A" Pump Room	
b. RHR "A" HX Room	
c. RHR "B" Pump Room	
d. RHR "B" HX Room	
e. RHR "C" Pump Room	
f. HPCS Pump Room	
g. LPCS Pump Room	
h. RCIC Room	
i. SGTS "A" Room	
j. SGTS "B" Room	
k. Div I DG Room	
l. Div II DG Room	
m. Div III DG Room	
n. SX "A" Pump Room	
o. SX "B" Pump Room	
p. SX "C" Pump Room	
q. SBLC Area	
r. Div I Battery Room	
s. Div II Battery Room	
t. Div III Battery Room	

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TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	SENSOR	DISPLAY	TEMPERATURE LIMIT (°F)
a. RHR "A" Pump Room	ITE-CM292	ITR-CM326	200
b. RHR "A" HX Room	ITE-CM293		
c. RHR "B" Pump Room	ITE-CM287		
d. RHR "B" HX Room	ITE-CM288		
e. RHR "C" Pump Room	ITE-CM289		
f. HPCS Pump Room	ITE-CM295		
g. LPCS Pump Room	ITE-CM294		
h. RCIC Room	ITE-CM290		
~~i. SGTS "A" Room~~			
~~j. SGTS "B" Room~~			
~~k. Div I DG Room~~			
~~l. Div II DG Room~~			
~~m. Div III DG Room~~			
~~n. SX "A" Pump Room~~			
~~o. SX "B" Pump Room~~			
~~p. SX "C" Pump Room~~			
~~q. SBLC Area~~			
~~r. Div I Battery Room~~			
~~s. Div II Battery Room~~			
~~t. Div III Battery Room~~			

INSERT

INSERT (Table 3.7.8-1)

	AREA	SENSOR	DISPLAY	TEMP. LIMIT (°F)
i.	RCIC Inst. Panel Rm.	ITE-CM291	ITR-CM326	200
j.	Radwaste Pipe Tunnel	ITE-CM311		
k.	Aux. Bldg. Below MST	ITE-CM313		
l.	RWCU "A" Pump Rm.	ITE-CM314		
m.	RWCU "B" Pump Rm.	ITE-CM315		
n.	RWCU "C" Pump Rm.	ITE-CM316		
o.	Aux. Bldg. Aisle El. 707'6"	ITE-CM286		
p.	Aux. Bldg. Aisle El. 737'0"	ITE-CM310		
q.	Aux. Bldg. Aisle El. 737'0"	ITE-CM312		
r.	Aux. Bldg. Steam Tunnel	ITE-CM319		
s.	Fuel Pool Clg. Ht. Ex. Rm.	ITE-CM308	ITR-CM327	135
t.	Fuel Pool Clg. Ht. Ex. Rm.	ITE-CM309		135
u.	Fuel Bldg. Gen. Area El. 712'0"	ITE-CM295		115
v.	Fuel Bldg. Gen. Area El. 712'0"	ITE-CM296		
w.	Fuel Bldg. Gen. Area El. 712'0"	ITE-CM297		
x.	Fuel Bldg. Gen. Area El. 712'0"	ITE-CM298		
y.	Fuel Bldg. Pipe Vlv. Rm.	ITE-CM299		
z.	Fuel Bldg. Pipe Vlv. Rm.	ITE-CM300		
aa.	Fuel Pool Clg. Pump Rm.	ITE-CM301		
bb.	Fuel Pool Clg. Pump Rm.	ITE-CM302		
cc.	Fuel Bldg. Gen. Area El. 737'0"	ITE-CM304		
dd.	Fuel Bldg. Gen. Area El. 737'0"	ITE-CM305		
ee.	Fuel Bldg. Gen. Area El. 737'0"	ITE-CM306		
ff.	Fuel Bldg. Gen. Area El. 737'0"	ITE-CM307		
gg.	Aux. Bldg. Gas Cont. Boundary	ITE-CM324		200
hh.	Aux. Bldg. Gas Cont. Boundary	ITE-CM325		200
ii.	MSIV "A" Room	ITE-CM317		200

INSERT (Table 3.7.8-1) (Cont'd.)

	AREA	SENSOR	DISPLAY	TEMP. LIMIT (°F)
jj.	MSIV "B" Room	ITE-CM318	ITR-CM327	200
kk.	Fuel Bldg. Gen. Area El. 744'0"	ITE-CM320	↓	115
ll.	Fuel Bldg. Gen. Area El. 744'0"	ITE-CM321		↓
mm.	Fuel Bldg. Gen. Area El. 744'0"	ITE-CM322		
nn.	Fuel Bldg. Gen. Area El. 744'0"	ITE-CM323	↓	↓

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 The main turbine bypass system shall be OPERABLE.

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

ACTION:

With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or ~~(be in at least STARTUP within the next 6 hours)~~ ~~(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. CPS

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME ~~to be less than or equal to ( ) seconds.~~ meets the following requirements when measured from the initial movement of main turbine stop valve or control valve: CPS

   a. 80% of turbine bypass system capacity shall be established within 0.3 seconds.

   b. Bypass valve opening shall start within 0.1 seconds.

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. A separate day fuel tank containing a minimum of ~~560~~ (LATER) gallons of fuel for diesel generators 1A and 1B and ~~340~~ (LATER) gallons of fuel for diesel generator 1C.

2. A separate fuel storage system containing a minimum of ~~47,000~~ (LATER) gallons of fuel for diesel generators 1A and 1B and ~~28,000~~ (LATER) gallons of fuel for diesel generator 1C.

3. A separate fuel transfer pump.

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APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator 1A or 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and diesel generators 1A and 1B to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and diesel generator 1A or 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators 1A and 1B to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LIMITING CONDITION FOR OPERATION (Continued) NO CHANGE

ACTION (Continued)

c. With diesel generator 1C of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

d. With diesel generator 1A, 1B or 1C of the above required A.C. electrical power sources inoperable, in addition to ACTION a, b or c, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. With diesel generators 1A and 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 1A and 1B to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators 1A and 1B to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day fuel tank.
   2. Verifying the fuel level in the fuel storage tank.
   3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel 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 1A and 1B voltage and frequency shall be Insert Attached ~~4160 ± 420 volts and 60 ± (1.2) Hz~~ within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals: | CPS
      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 ECCS actuation test signal by itself.
   5. Verifying the diesel generator is synchronized, loaded to greater than or equal to 3869 kw for diesel generator 1A, 3875 kw for diesel generator 1B and 2200 kw for diesel generator 1C in less than or equal to ~~(60)~~ 90 seconds, and operates with this load for at least 60 minutes. | CPS
   6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.
   ~~(7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to (250) psig.)~~ | CPS

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 ~~and engine-mounted~~ fuel tanks. | CPS

* Insert Attached

INSERT TO 4.8.1.1.2.a.4

+420, -0 volts and 60 + 6, -0 Hz and the generator
1c voltage and frequency shall be 4160 ± 420 and
60 ± 1.2 Hz

*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall 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.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per (31 [if ground water table is equal to or higher than the bottom of the tank]) (92) days by removing accumulated water from the fuel storage tank(s).

d. At least once per 92 days and from new fuel oil prior to addition to the storage tanks, by obtaining a sample in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken from new fuel or prior to addition to the storage tank, as applicable, verify in accordance with the tests specified in ASTM-D975-77 that the sample has:

a) A water and sediment content of less than or equal to 0.05 volume percent.

b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to ____ but less than or equal to ____ or an API gravity @ 60°F of greater than or equal to ____ degrees but less than or equal to ____ degrees.

2. Within one week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

3. Within two weeks after obtaining the sample, verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137, Position 2.a, are met when tested in accordance with ASTM-D975-77.

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e. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1120 kw for diesel generators 1A and 1B, and greater than or equal to ~~1865~~ 1995 kw for diesel generator 1C while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz. INSERT Attached 8-4B C

3. Verifying the diesel generator capability to reject a load of 3869 kw for diesel generators 1A, 3875 kw for diesel generator 1B and 2200 kw for diesel generator 1C without tripping. The generator voltage shall not exceed ~~(4784)~~ volts for diesel generator 1A and 1B and 5824 volts for diesel generator 1C during and following the load rejection. CPS

c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic vicosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

diesel generator 1A an 1B voltage at 4160 ± 420 volts and frequency at 60 +6, -2 Hz and, diesel generator 1C

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SURVEILLANCE REQUIREMENTS (Continued)

4. Simulating a loss of offsite power by itself, and:

   a) For divisions 1 and 2:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads ~~through the load sequencer~~ and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test. |CPS

   b) For division 3:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with ~~(the permanently connected)~~ ~~(~~its~~)~~ loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test. |CPS

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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 1A and 1B voltage and frequency shall be ~~(~~4160~~)~~ ± ~~(~~420~~)~~ volts and ~~(~~60~~)~~ ± ~~(~~1.2~~)~~ Hz within (1~~2~~ 0) seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. |CPS

~~6. Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.~~ |CPS

6 ~~7~~. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and: |CPS

   a) For divisions 1 and 2:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

shall be 4160 + 420, -0 volts and frequency 60 + 6, -2 Hz and diesel generator 1C voltage and frequency

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2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

b) For division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads and the auto-connected emergency loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

7 ~~8.~~ Verifying that all automatic diesel generator trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except:

a) For divisions 1 and 2, engine overspeed, generator differential current, and engine overcrank.

b) For division 3, engine overspeed and generator differential current, and engine overcrank | CPS

8 ~~9.~~ Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 425~~2~~6 kw for diesel generator 1A, 426~~2~~3 kw for diesel generator 1B and 2420 kw for diesel generator 1C. During the remaining 22 hours of this test, the diesel generator shall be loaded to 3869 kw for diesel generator 1A, 3875 kw for diesel generator 1B and 2200 kw for diesel generator 1C. The generator INSERT ATTACHED voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained ~~within these limits~~ during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.e.4.a)2) and b)2)*. at 4160±420 volts and 60±1.2 Hz | CPS | CPS

*If Surveillance Requirements 4.8.1.1.2.e(4).a)2 and b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at (continuous rating) for one hour or until operating temperatures have stabilized.

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1A and 1B voltage and frequency shall be 4160 +420, -0 volts and 60 +6, -2 Hz and the generator 1C

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SURVEILLANCE REQUIREMENTS (Continued)

9 ~~10~~. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4078 kw for diesel generator 1A, 4082 kw for diesel generator 1B and 2350 kw for diesel generator 1C. ICPS

10 ~~11~~. Verifying the diesel generator's capability to: ICPS

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

11 ~~12~~. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

~~(13. Verifying that with all diesel generator air start receivers pressurized to less than or equal to (250) psig and the compressors secured, the diesel generator starts at least 5 times from ambient conditions and accelerates to (900) rpm ± 3% in less than or equal to (13) seconds.)~~

~~(14. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.)~~

~~15. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 1A and 1B.~~

12 ~~16~~. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Maintenance mode.

b) Diesel generator lockout.

f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all three diesel generators accelerate to at least 870 rpm in less than or equal to 10 seconds.

g. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a (sodium hypochlorite) or equivalent solution, and ICPS

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SURVEILLANCE REQUIREMENTS (Continued)

No CHANGE

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section 11 Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Numer of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency. With the exception of the semiannual fast start, no starting time requirements are required to meet the valid test requirements of Regulatory Guide 1.108.

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A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator 1A or 1B, and diesel generator 1C when the HPCS system is required to be OPERABLE, with each diesel generator having:

1. A day fuel tank containing a minimum of ~~560~~ (LATER) gallons of fuel for diesel generators 1A and 1B and ~~340~~ (later) gallons of fuel for diesel generator 1C.
2. A fuel storage system containing a minimum of ~~47,000~~ (LATER) gallons of fuel for diesel generators 1A and 1B and ~~28,000~~ (LATER) gallons of fuel for diesel generator 1C.

(3. A fuel transfer pump.)

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APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the offsite circuits and/or diesel generators 1A or 1B of the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 23 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator 1C of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

*When handling irradiated fuel in the secondary containment.

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3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division I, consisting of:

   1. 125 volt battery 1A.
   2. 125 volt full capacity charger.

b. Division II, consisting of:

   1. 125 volt battery 1B.
   2. 125 volt full capacity charger.

c. Division III, consisting of:

   1. 125 volt battery 1C.
   2. 125 volt full capacity charger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division I or Division II battery and/or charger of the above required D.C. electrical power sources inoperable, restore the inoperable division to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to (129)-volts on float charge.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and/or within 7 days after a battery discharge with battery terminal voltage below (110) volts, or battery overcharge with battery terminal voltage above (150) volts, by verifying that: CPS

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items ~~is less than (150×10^{-6}) ohms, and~~ meet the requirements of IEEE 484-1975. CPS

3. The average electrolyte temperature of ~~(a representative number)~~ the pilot cells of connected cells is above ~~60~~ 65°F. CPS

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The resistance of each cell(-to-cell) and terminal connection ~~is less than or equal to (150×10^{-6}) ohms, and~~ meets the requirements of IEEE 484-1975 CPS

4. The battery charger will supply at least ~~(400) amperes~~ 300 amperes for Div. I, II and 100 amperes for Div. III. at a minimum of ~~(125)~~ 105 volts for at least (4) hours. CP

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design duty cycle when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to ~~125~~ 105 volts. CPS

   a) Battery 1A, greater than or equal to ~~543~~ (Later) amperes; battery 1B, greater than or equal to ~~129~~ (Later) amperes; and battery 1C, greater than or equal to ~~126~~ (Later) amperes during the initial 60 seconds of the test. CPS

   b) Battery 1A, greater than or equal to ~~245~~ (Later) amperes and battery 1B greater than or equal to ~~309~~ (Later) amperes during the remainder of the first half hour of the test.

   c) Battery 1C, greater than or equal to ~~75~~ (Later) amperes during the remainder of the first hour of the test. CPS

SURVEILLANCE REQUIREMENTS (Continued)

d) Battery 1A, greater than or equal to ~~230~~ (Later) amperes and battery 1B, greater than or equal to ~~294~~ (Later) amperes during the remainder of the first hour and half of the test.

e) Battery 1A, greater than or equal to ~~135~~ (Later) amperes; battery 1B, greater than or equal to ~~79~~ (Later) amperes; and battery 1C, greater than or equal to ~~78~~ (Later) amperes during the remainder of the 4 hour test.

CPS

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At this once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months during shutdown performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 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)~~ 1.200 - 1.220

(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.

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D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

NO CHANGE

3.8.2.2 As a minimum, Division I or Division II, and, when the HPCS system is required to be OPERABLE, Division III, of the D.C. electrical power sources shall be OPERABLE with:

a. Division I consisting of:
   1. 125 volt battery 1A.
   2. 125 volt full capacity charger.

b. Division II consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity charger.

c. Division III consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity charger.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *

ACTION:

a. With less than the Division I and/or Division II battery and/or charger of the above required 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 III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

*When handling irradiated fuel in the secondary containment.

DEC 8 1982

ELECTRICAL POWER SYSTEMS

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3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized: ~~with tie breakers open (both) between redundant buses within the unit (and between units at the same station):~~ CPS

a. A.C. power distribution:

1. Division I, consisting of: CPS

a) 4160 volt A.C. bus 1A1.

b) 480 volt ~~A.C. MCCs A and 1A.~~ Unit Subs A and 1A

c → d ~~c~~) 120 volt A.C. distribution panels in 480 volt ~~MCCs A and 1A.~~ Auxiliary Building MCC 1A1 and Control Building MCC E2.

2. Division II, consisting of: CPS

a) 4160 volt A.C. bus 1B1.

b) 480 volt ~~A.C. MCCs B and 1B.~~ Unit Subs B and 1B

c) → d ~~c~~) 120 volt A.C. distribution panels in 480 volt ~~MCCs B and 1B.~~ Auxiliary Building MCC 1B1 and Control Building MCC F2.

3. Division III, consisting of:

a) 4160 volt A.C. bus 1C1.

b) 480 volt A.C. AB MCC 1C and AB MCC 1C1 and SSW MCC 1C

c) 120 volt A.C. distribution panels in 480 volt ABMCC 1C and ABMCC 1C1.

> 4. Insert Attached CPS

b. D.C. power distribution: CPS

1. Division I, consisting of 125 volt D.C. Battery 1A, 125 volt Battery charger 1A, 125 volt D.C. MCC 1A, and Distribution Panel.

2. Division II, consisting of 125 volt DC Battery 1B, 125 volt Battery Charger 1B, 125 volt D.C. MCC 1B, and Distribution Panel.

3. Division III, consisting of 125 volt DC Battery 1C, 125 volt Battery Charger 1C, 125 volt D.C. MCC 1C, and Distribution Panel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

c) 480 volt AC MCC's

1) Aux. Bldg. MCC's 1B1 — 1B4
2) SSW MCC 1B
3) DG MCC 1B
4) Containment Bldg. MCC's F1, F2, and H

c) 480 volt AC MCC's

1) Aux. Bldg. MCC's 1A1 — 1A4
2) SSW MCC 1A
3) DG MCC 1A
4) Containment Bldg. MCC's E1, E2, and G

INSERT 3.8.3.1.a.4

4. Reactor Protection System (RPS) 120V AC Solenoid Buses A and B from their associated inverters.

INSERT

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LIMITING CONDITION FOR OPERATION (Continued) No Change

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. | CPS

> 3. Insert Attached

b. For D.C. power distribution:

1. With either Division I or Division II of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following ~~24~~ hours.

2. With Division III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.3.1.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels. | CPS

4.8.3.1.2 Whenever an RPS Solenoid Bus is energized from the alternate source or a Bus Power Monitor is inoperable, verification shall be made once per 8 hours that the supply frequency is $\geq$ 57 Hz.

4.8.3.1.3 At least once per 31 days a CHANNEL FUNCTIONAL TEST shall be performed on each RPS Solenoid Bus Power Monitor.

4.8.3.1.4 At least once per 18 months a CHANNEL CALIBRATION shall be performed on each RPS Solenoid Bus Power Monitor and associated power supply Regulating Transformer. CPS

DEC 3 1982

Insert 3.8.3.1

3. For inoperable RPS Solenoid Bus inverters:

   a. With an RPS Sol. oid Bus inverter inoperable transfer the bus to the alternate power source provided the other RPS Solenoid Bus is not being supplied from the alternate source.

   b. With both RPS Solenoid Bus inverters inoperable de-energize one RPS Solenoid Bus.

   c. With the frequency of the 120V AC supply to the RPS Solenoid buses A or B $\leq$ 57 Hz, demonstrate the OPERABILITY of all equipment which could have been subjected to the abnormal frequency for all class IE loads connected to the associated buses, by performance of a CHANNEL FUNCTIONAL TEST or CHANNEL CALIBRATION, as required, within 24 hours.

ELECTRICAL POWER SYSTEMS

DISTRIBUTION - SHUTDOWN

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LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For A.C. power distribution, Division I or Division II, and when the HPCS system is required to be OPERABLE, Division III, with:

1. Division I consisting of:
   a) 4160 volt A.C. bus 1A1.
   b) 480 volt ~~A.C. MCCs A and 1A.~~ Unit Subs A and 1A.
   d) 120 volt A.C. distribution panels in 480 volt ~~MCCs A and 1A.~~ Auxiliary Building MCC 1A1 and Control Building MCC E2.
2. Division II consisting of:
   a) 4160 volt A.C. bus 1B1.
   b) 480 volt ~~A.C. MCCs B and 1B.~~ Unit Subs B and 1B.
   d) 120 volt A.C. distribution panels in 480 volt ~~MCCs B and 1B.~~ Auxiliary Building MCC 1B1 and Control Building MCC F2.
3. Division III consisting of:
   a) 4160 volt A.C. bus 1C1.
   b) 480 volt A.C. AB MCC 1C and AB MCC 1C1, and SSW MCC 1C.
   c) 120 volt A.C. distribution panels in 480 volt AB MCC 1C and AB MCC 1C1.

b. For D.C. power distribution, Division (I) or Division (II, and when the HPCS system is required to be OPERABLE, Division (III), with:

1. Division I consisting of 125 volt D.C. Batteries 1A, 125 volt Battery Charger 1A, 125 volt D.C. MCC-1A, and Distribution Panel.
2. Division 2 consisting of 125 volt D.C. Batteries 1B, 125 volt Battery Charger 1B, 125 volt D.C. MCC-1B, and Distribution Panel.
3. Division 3 consisting of 125 volt D.C. Batteries 1C, 125 volt Battery Charger 1C, 125 volt D.C. MCC-1C, and Distribution Panel.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

c) 480 volt A.C. MCC's
1) Aux. Bldg. MCC's 1B1 through 1B4
2) SSW MCC 1B
3) D.G. MCC 1B
4) Cont. Bldg. MCC's F1, F2, and H

c) 480 volt A.C. MCC's
1) Aux. Bldg. MCC's 1A1 through 1A
2) SSW MCC 1A
3) D.G. MCC 1A
4) Cont. Bldg. MCC's E1, E2, and

*When handling irradiated fuel in the secondary containment.

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LIMITING CONDITION FOR OPERATION (Continued) NO CHANGE

ACTION:

a. For A.C. power distribution:

   1. With less than Division I and/or Division II of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

b. For D.C. power distribution:

   1. With less than Division I and/or Division II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the Auxiliary Building and Enclosure Building and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside containment shall be de-energized*:

a. Circuit numbers (__, __, __ and __) in panel ( ).

b. Circuit numbers (__, __, __ and __) in panel ( ).

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.

DELETE THIS PAGE

*Except during entry into the containment.

**Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.~~2~~ 1 All containment penetration conductor overcurrent protective devices shown in Table 3.8.4.~~2~~ 1-1 shall be OPERABLE. CPS

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the containment penetration conductor overcurrent protective devices shown in Table 3.8.4.~~2~~ 1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and: CPS

1. For ~~(4.16)~~ 6.9 kV circuit breakers, de-energize the ~~(4.16)~~ 6.9 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. CPS

2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by (racking out the breaker) within 72 hours and verify the inoperable breaker(s) to be (racked out) at least once per 7 days thereafter.

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 ~~(4.16)~~ kV circuits which have their redundant circuit breakers tripped or to 480 volt circuits which have the inoperable circuit breaker (racked out). CPS

SURVEILLANCE REQUIREMENTS

4.8.4.~~2~~ 1 Each of the containment penetration conductor overcurrent protective devices shown in Table 3.8.4.~~2~~ 1-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (~~4-15~~ 6.9 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers ~~(of each voltage level)~~ and performing: CPS

   a) A CHANNEL CALIBRATION of the associated protective relays, and

   b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

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.

in accordance with the manufacturers recommendations that shall test

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current ~~with a value equal to 300% of~~ the pickup of the long time delay trip element and ~~150% of~~ the pickup of the short time delay trip element, and verify~~ing~~ that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current ~~equal to 120% of the pickup value of the element~~ in accordance with the manufacturers instructions and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested. CPS CPS

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.

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TABLE 3.8.4.~~2~~1-1

|CPS

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER(*) AND LOCATION	SYSTEM(S) AFFECTED
a. (~~4.16~~ 6.9) KV Circuit Breakers	
Insert A Attached	
b. (480) VAC ~~(Molded Case)~~ Circuit Breakers	
1. Type Molded Case	
Insert B1 Attached	
2. Type Switchgear	
Insert B2 Attached	

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(*List all primary and backup breakers.)

INSERT A

Reactor Recirc. Pump 1A
Penetration 1EE01E 1/c-1000 MCM per Ø

Normal Operation Protection

6.9kV Swgr. Location 121, AH (R,C); EL 781 FT.
Two (2) identical circuit breakers in series
with identical protective relays.
Westinghouse Type COM-5

Low Frequency Operation Protection

Two (2) identical 250 ampere molded case circuit breakers
in series.
Location 121, AH (R,C); EL 781 FT.

Reactor Recirc. Pump 1B
Penetration 1EE02E 1/c-1000MCM per Ø

Normal Operation Protection

6.9kV Swgr. Location 105, AH (R,C); EL 781 FT.
Two (2) identical circuit breakers in series with
identical protective relays.
Westinghouse Type COM-5

Low Frequency Operation Protection

Two (2) identical 250 ampere molded case circuit breakers
in series
Location 105, AH (R,C); EL 781 FT.

INSERT B1

Auxiliary Building MCC 1F (1AP41E)
Location 119, Y (R,C); EL 762 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
1E	100	350 MCM	RLC118 1LL18EA	1LL05B	1EE03E	Lighting
1D	80	350 MCM	RLC116 1LL16EA	1LL05A	1EE03E	Lighting
3D	15	#6	Sump Pump 1A 1RE03PA	1RE01A	1EE05E	Equip. Drn. Radwaste
5A	15	#6	Sump Pump 1RE05PA	1RE03A	1EE05E	Equip. Drn. Radwaste
5B	15	#6	Sump Pump 1A 1RF03PA	1RF07A	1EE05E	Floor Drn. Radwaste
5C	15	#6	Sump Pump 1RF07PA	1RF10A	1EE05E	Floor Drn. Radwaste
8B	15	#6	RWCU Vlv Mtr 1G33-F102	1RT22A	1EE05E	Reac. Water Cleanup
7A	15	#6	RWCU Vlv Mtr 16EE-F031	1RT09A	1EE05E	Reac. Water Cleanup
7B	15	#6	RWCU Vlv Mtr 1G33-F042A	1RT14A	1EE05E	Reac. Water Cleanup
7C	15	#6	RWCU Vlv Mtr 1G33-F044	1RT16A	1EE05E	Reac. Water Cleanup
2A	15	#6	Head Vent Vlv 1B21-F001	1NB01A	1EE07E	Reactor Venting
2B	15	#6	Head Vent Vlv 1B21-F005	1NB03A	1EE07E	Reactor Venting
2C	15	#6	Wtr Press Cont VLV 1C11-003	1RD03A	1EE07E	Cont. Rod Driver
3C	15	#6	Space Htr 1B33-C001A	1RR01G	1EE07E	Reac. Recirc.
4C	15	#6	Suct Vlv Mtr 1B33-F023A	1RR04A	1EE07E	Hoist
4A	15	#6	WINCH 1F42-E001	1FH06A	1EE07E	Hoist
6B	15	#6	Disc DC Vlv 1B33-F067A	1RR08A	1EE07E	Reac. Recirc.

INSERT B1 (Cont'd)

Auxiliary Building MCC 1F (1AP41E)

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
8A	15	#6	RWCU SUCT. VLV 1G33-F100	1RT20A	1EE07E	Reac. Water Cleanup
1B	(later)	#6	1F15-E005	1FH12A	1EE07E	Fuel Handling
1C	(later)	#6	Refuel Platform 1F16-E003EC	1FH13J	1EE07E	Fuel Handling
8C	15	#6	J/B 1HC69C	1HC62A	1EE07E	Hoists
8D	15	#6	Shield Door 1HC71G	1HC64A	1EE07E	Hoists
4B	100	350	OIL PMP MTR 1B33-D003A	1RR14A	1EE36E	Reac. Recirc.
5D	15	#6	Fan Motor 1B33-D003A	1RR20A	1EE36E	Reac. Recirc.
1F	15	#6	SRM/IRM Drives 1H22-P008	1NR02B	1EE05E	Control Rods
				1NR02D	1EE05E	Control Rods
				1NR02F	1EE05E	Control Rods
				1NR02H	1EE05E	Control Rods
				1NR02K	1EE05E	Control Rods
				1NR02M	1EE05E	Control Rods
				1NR02P	1EE05E	Control Rods
				1NR02R	1EE05E	Control Rods
				1NR02T	1EE05E	Control Rods
				1NR02V	1EE05E	Control Rods
				1NR02X	1EE05E	Control Rods
				1NR03A	1EE05E	Control Rods

INSERT
3/4 8-22C

INSERT B1 (Cont'd)

Auxiliary Building MCC 1G (1AP42E)
Location 106, Y (R,C); EL 762 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
1E	100	350 MCM	Ltg. Pnl 117 1LL7EA	1LL06B	1EE04E	Lighting
1D	100	350 MCM	Ltg. Pnl 115 1LL1[illegible]	1LL06A	1EE04E	Lighting
4A	15	#6	[illegible] ump 1RE03PB	1RE01E	1EE06E	Equip. Drn. Radwaste
4C	15	#6	Sump Pump 1RF03PB	1RF07E	1EE06E	Floor Drn. Radwaste
4D	15	#6	Sump Pump 1RF07PB	1RF10E	1EE06E	Floor Drn. Radwaste
6C	15	#6	VLV 1G33-F042B	1RT15A	1EE06E	Reac. Water Cleanup
5B	15	#6	AGITATOR 1G36-A001	1RT46A	1EE06E	Reac. Water Cleanup
4B	15	#6	Sump Pump 1RE05PB	1RE03E	1EE06E	Equip. Drn. Radwaste
5A	15	#6	FAN 1B33-D003B	1RR17A	1EE06E	Reac. Recirc.
1B	15	#6	HYD. SYS 1F42-D002	1FH03F	1EE08E	Fuel Handling
3A	15	#6	VENT VALVE 1B21-F002	1NB02A	1EE08E	Reactor Vent Valve
2D	15	#6	SPACE HTR. 1B33-C001B	1RR22G	1EE08E	Reac. Recirc.
3C	15	#6	SUCT. VALVE 1B33-F023B	1RR05A	1EE08E	Reac. Recirc.
6A	15	#6	DISCH. VLV. 1B33-F067B	1RR09A	1EE08E	Reac. Recirc.
3B	100	350	OIL PUMP 1B33-D003B	1RR15A	1EE04E	Reac. Recirc.
5D	15	#6	DEMIN. PUMP 1G36-C001B	1RT44A	1EE08E	Reac. Water Cleanup
5C	15	#6	AGITATOR 1G36-A002	1RT47A	1EE08E	Reac. Water Cleanup

INSERT B1 (Cont'd)

Auxiliary Building MCC 1A2 (1AP73E)
Location 121, V (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
1B	15	#6	RHR Valve 1E12-F037A	1RH63A	1EE09E	Resid. Ht. Removal

INSERT B1 (Cont'd)

Auxiliary Building MCC 1A1 (1AP72E)
Location 121, Y (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
2B	100	350	SLC 158 1LL58EA	1LL21B	1EE03E	Lighting
1D	50	#2	Drywell Fan 1VP01CA	1VP05A	1EE05E	Drywell HVAC
3B	90	350	Drywell Fan 1VP01CC	1VP07A	1EE05E	Drywell HVAC
1C	150	350	Comb. Gas Compressor 1HG02CA	1HG01A	1EE09E	H2 Recomb.
3A	100	350	Stby Liq Pmp 1C41-C001	1SC01A	1EE09E	Standby Liq. Cont.

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INSERT B1 (Cont'd)

Auxiliary Building MCC 1A3 (1AP74E)
Location 121, V (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
5C	15	#6	Sample Pump 1PS08P	1PS15K	1EE07E	Process Sampling
4B	15	#6	Sample Pump 1PS06P	1PS15D	1EE07E	Process Sampling
4D	15	#6	Sample Pump 1PS07P	1PS15G	1EE07E	Process Sampling
2D	15	#6	Sample Pump 1PS05P	1PS15A	1EE07E	Process Sampling
13B	15	#6	Shutoff Valve 1SM002A	1SM02A	1EE09E	Supp. Pool Make-up
13A	15	#6	Shutoff Valve 1SM001A	1SM01A	1EE09E	Supp. Pool Make-up
13C	15	#6	Spray Valve 1E12-F042A	1RH27A	1EE09E	Resid. Ht. Removal
1B	15	#6	Isol. Valve 1FP078	1FP67A	1EE09E	Fire Protection
14D	20	#6	Supply Fan 1VR08C	1VR17A	1EE37E	Contain. HVAC

INSERT B1 (Cont'd).

Auxiliary Building MCC 1A4 (1AP93E)
Location 121, Y (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
10B	15	#6	Spray Valve 1E12-F028A	1RH61A	1EE09E	Resid. Ht. Removal
7C	15	#6	Isol. Valve 1CY021	1CY09A	1EE09E	Cycled Conden.
10A	15	#6	Suct. Valve 1HG009A	1HG02A	1EE09E	H2 Recom.
9A	15	#6	Isol. Valve 1SX095A	1SX56A	1EE09E	Shutdown Serv. Water
9B	15	#6	Isol. Valve 1CC057	1CC17A	1EE09E	Component Cool Water
9C	15	#6	Isol. Valve 1CC128	1CC17F	1EE09E	Component Cool Water
10C	15	#6	Outlet Valve 1C41-F001A	1SC05A	1EE09E	Standby Liq. Cont.
7A	15	#6	Isol. Valve 1SX089A	1SX22A	1EE37E	Shutdown Serv. Water
7B	15	#6	Isol. Valve 1SX096A	1SX22L	1EE37E	Shutdown Serv. Water

INSERT B1 (Cont'd)

Auxiliary Building MCC 1B1 (1AP75E)
Location 105, X (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
2C	50	#2	Cool Fan 1VP01CB	1VP06A	1EE06E	Drywell HVAC
3A	90	350	Cool Fan 1VP01CD	1VP08A	1EE06E	Drywell HVAC
4A	100	350	Stby Pump 1CR1-CØ01B	1SC02A	1EE10E	Standby Liq. Cont.
2A	150	350	H2 Compr 1HG02CB	1HG05A	1EE11E	H2 Recomb. Contain.
2B	15	#6	Supply Fan 1VR11C	1VR18A	1EE11E	Contain. HVAC

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INSERT B1 (Cont'd)

Auxiliary Building MCC 1B2 (1AP76E)
Location 106, V (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
11C	15	#6	Isol. Valve 1SX095B	1S06A	1EE10E	Shutdown Serv. Water
2B	15	#6	Inlet Valve 1CC068	1CC08A	1EE10E	Component Cool Water
1B	15	#6	Inlet Valve 1CC065	1CC08D	1EE10E	Component Cool Water
2C	15	#6	Outlet Valve 1CC070	1CC09A	1EE10E	Component Cool Water
2A	15	#6	Outlet Valvle 1CC067	1CC09D	1EE10E	Component Cool Water
10C	15	#6	Sup Pool Vlv 1E12-F073A	1RH42A	1EE11E	Resid. Ht. Removal
11B	15	#6	Isol Valve 1SX095B	1SX57A	1EE11E	Shutdown Serv. Water
10A	15	#6	Suct. Valve 1HG009B	1HG06A	1EE11E	H2 Recomb.
11A	15	#6	Sup. Pool Valve 1E12-F073B	1RH43A	1EE11E	Resid. Ht. Removal
14B	15	#6	Spray Valve 1E12-F028B	1RH62A	1EE11E	Resid. Ht. Removal
10B	15	#6	Upper Pool Univ. 1E12-F037B	1RH64A	1EE11E	Resid. Ht. Removal

INSERT B1 (Cont'd)

Auxiliary Building MCC 1B3 (1AP77E)
Location 106, V (R,C); EL 781 FT

Each compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
2A	15	#6	Isol. Valve 1CC050	1CC12A	1EE10E	Component Cool Water
2B	15	#6	Isol. Valve 1CC053	1CC12D	1EE10E	Component Cool Water
3B	15	#6	Isol. Valve 1CC071	1CC13A	1EE10E	Component Cool Water
3C	15	#6	Isol. Valve 1CC074	1CC13D	1EE10E	Component Cool Water
3A	15	#6	Isol. Valve 1CC060	1CC16D	1EE10E	Component Cool Water
4A	15	#6	Isol. Valve 1CC127	1CC16L	1EE10E	Component Cool Water
4C	15	#6	Isol. Valve 1CY017	1CY06A	1EE10E	Cycled Condensate
5A	15	#6	Isol. Valve 1CY020	1CY06F	1EE10E	Cycled Condensate
5B	15	#6	Isol. Valve 1FC007	1FC05A	1EE10E	Fuel Pool Cooling
5C	15	#6	Isol. Valve 1FC037	1FC20A	1EE10E	Fuel Pool Cooling
10A	15	#6	Isol. Valve 1E51-F063	1RI02A	1EE11E	Reac. Inject.
14A	15	#6	RCIC Valve 1E51-F076	1RI15A	1EE11E	Reac. Inject.
10B	15	#6	Isol. Valve 1G33-F001	1RT05A	1EE11E	Reac. Water Cleanup
10C	15	#6	Isol. Valve 1G33-F020	1RT08A	1EE11E	Reac. Water Cleanup
11A	15	#6	Isol. Valve 1G33-F040	1RT13A	1EE11E	Reac. Water Cleanup
11B	15	#6	Isol. Valve 1G33-F028	1RT18A	1EE11E	Reac. Water Cleanup

INSERT B1 (Cont'd)

Auxiliary Building MCC 1B3 (1AP77E)

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
8A	15	#6	Isol. Valve 1E12-F009	1RH17A	1EE11E	Resid. Ht. Removal
8C	15	#6	Spray Valve 1E12-F042B	1RH28A	1EE11E	Resid. Ht. Removal
7B	15	#6	Isol. Valve OMC010	1MC03A	1EE11E	Condensate Make-up
7C	15	#6	Isol. Valve 1B21-F016	1NB05A	1EE11E	Reac. Water
12B	15	#6	Isol. Valve 1SX089B	1SX23A	1EE11E	Shutdown Serv. Water
13A	15	#6	Isol. Valve 1SX096B	1SX23L	1EE11E	Shutdown Serv. Water
13B	15	#6	Isol. Valve 1VQ006B	1VQ24A	1EE11E	Drywell Purge
7A	15	#6	Isol. Valve 1IA013B	1IA03A	1EE11E	Instr. Air
14B	15	#6	Isol. Valve 1WØ001B	1WØ14A	1EE11E	Chilled Water
14C	15	#6	Isol. Valve 1WØ002B	1WØ16A	1EE11E	Chilled Water
13C	15	#6	Isol. Valve 1VR002B	1VR09A	1EE11E	Contain. HVAC
6A	15	#6	Isol. Valve 1FP052	1FP64A	1EE11E	Fire Protection
6B	15	#6	Isol. Valve 1FP053	1FP65A	1EE11E	Fire Protection
11C	15	#6	Isol. Valve 1FP079	1FP68A	1EE11E	Fire Protection

INSERT B1 (Cont'd)

Auxiliary Building MCC 1B4 (1AP94E)
Location 105, X (R,C); EL 781 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
8C	15	#6	Shutoff Valve 1SM001B	1SM03A	1EE11E	Supp. Pool Make-up
9A	15	#6	Shutoff Valve 1SM002B	1SM04A	1EE11E	Supp. Pool Make-up
9B	15	#6	Isol. Valve 1FP050	1FP62A	1EE11E	Fire Protection
7A	15	#6	Isol. Valve 1VP005A	1VP11A	1EE11E	Drywell HVAC
7B	15	#6	Isol. Valve 1VP005B	1VP12A	1EE11E	Drywell HVAC
7C	15	#6	Isol. Valve 1VP014A	1VP15A	1EE11E	Drywell HVAC
8A	15	#6	Isol. Valve 1VP014B	1VP16A	1EE11E	Drywell HVAC

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INSERT B1 (Cont'd)

Auxiliary Building MCC 1H (1AP95E)
Location 119, Z (R,C); EL 762 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
7D	80	350	Welding 1EW02E	1EW01A	1EE03E	Welding
2C	15	#6	Supp. Pool Fill Valve 1SM004	1SM05A	1EE05E	Supp. Pool Make-up
3C	15	#6	RWCU 1WX01PA	1WX06A	1EE05E	Reac. Waste Cleanup
2A	15	#6	RWCU 1G33-F107	1RT33A	1EE05E	Hoists
3A	15	#6	RWCU 1G36-C001A	1RT43A	1EE05E	Reac. Water Cleanup
7B	30	#2	Monorail 1B21-E300	1HC13E	1EE05E	Hoists
7F	15	#6	Hatch Shield Door 1HC68G	1HC65A	1EE07E	Hoists
5A	20	#6	Circuit 7 1F42-E001	1FH06J	1EE07E	Fuel Handling
6B	15	#6	Refuel Plat 1F15-E005	1FH11A	1EE07E	Fuel Handling
4A	15	#6	Air Hand Fan 1WØ05SF	1WØ25G	1EE07E	Chilled Water
4B	40	#6	Air Hand Fan 1WØ05SH	1WØ25U	1EE07E	Chilled Water
4D	15	#6	Air Hand Fan 1WØ05SM	1WØ25U	1EE07E	Chilled Water
4C	15	#6	Air Hand Fan 1WØ05SK	1WØ27A	1EE07E	Chilled Water
3D	40	#2	Air Hand Fan 1WØ05SB	1SØ25A	1EE05E	Chilled Water
6A	100	350	Oil Pump 1B33-D003A	1RR19A	1EE36E	Reac. Recirc.

INSERT B1 (Cont'd)

Auxiliary Building MCC 1H (1AP95E)

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
2B	100	350	Mixing Htr. 1C41-D003	1SC03A	1EE36E	Standby Liq. Control
3B	30	350	Tnk Htr. 1C41-D002	1SC04A	1EE36E	Standby Liq. Control
7A	15	#6	Fan Mtr. 1B33-D003A	1RR21A	1EE36E	Reac. Recirc.
7C	15	#6	Area Coolers	1WO34C	1EE07E	Area Coolers
				1WO34D	1EE07E	Area Coolers
				1WO34E	1EE07E	Area Coolers
				1WO34F	1EE07E	Area Coolers
				1WO34P	1EE07E	Area Coolers
				1WO34Q	1EE07E	Area Coolers
				1WO34R	1EE07E	Area Coolers
				1WO34S	1EE07E	Area Coolers

INSERT B1 (Cont'd)

Auxiliary Building MCC 1I (1AP96E)
Location 167, Z (R,C); EL 762 FT

Each Compartment listed below has two (2) identical circuit breakers in series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
6C	80	350	Welding 1EW03E	1EW02A	1EE04E	Welding
6B	100	350	Welding 1EW06A	1EW06A	1EE04E	Welding
1B	15	#6	Drn. Valve 1G33-F101	1RT21A	1EE06E	Reac. Water Cleanup
1D	15	#6	Tank Pump 1G36-C002	1WX06D	1EE06E	Chemical Radwaste
6E	15	#6	Jib Crane 1HC65G	1HC13A	1EE06E	Hoists
3B	15	#6	Fan Motor 1B33-D003B	1RR18A	1EE06E	Reac. Recirc.
5C	15	#6	Oil Pump 1B33-D003B	1RR16A	1EE06E	Reac. Recirc.
1C	30	#2	Precoat Pump 1G36-C002	1RT45A	1EE06E	Reac. Water Cleanup
6F	30	#2	Crane 1B33-E300	1HC13C	1EE06E	Hoists
2A	40	#2	Fan Motor 1WØ05SC	1WØ26A	1EE06E	Chilled Water
5B	15	#6	Suct. Valve 1G33-F106	1RT24A	1EE08E	Reac. Water Cleanup
5A	15	#6	Bypass Valve 1G33-F104	1RT23A	1EE08E	Reac. Water Cleanup
2B	15	#6	Fan Motor 1WØ05SG	1WØ26G	1EE08E	Chilled Water
2C	40	#6	Fan Motor 1WØ05SJ	1WØ26N	1EE08E	Chilled Water
3A	15	#6	Fan Motor 1WØ05SN	1WØ26U	1EE08E	Chilled Water
2D	15	#6	Fan Motor 1WØ05SL	1WØ27G	1EE08E	Chilled Water

INSERT B1 (Cont'd)

Auxiliary Building MCC 1I (1AP96E)

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
6A	15	#6	Area Coolers	1WO35C	1EE08E	Area Coolers
				1WO35D	1EE08E	Area Coolers
				1WO35E	1EE08E	Area Coolers
				1WO35F	1EE08E	Area Coolers
				1WO35P	1EE08E	Area Coolers
				1WO35Q	1EE08E	Area Coolers
				1WO35R	1EE08E	Area Coolers
				1WO35S	1EE08E	Area Coolers

125V DC MCC 1D (1DC15E)
Location 25, U (R,C); EL 781 FT

Each compartment listed bwlow has two (2) identical circuit breakers in Series.

COMPT	CIR BKR TRIP	PENETRATION CABLE SIZE	EQUIPMENT SERVICE	CABLE NUMBER	PENETRATION NUMBER	SYSTEMS AFFECTED
4C	100	350	Emerg. Lighting	1LL22E	1EE04E	Emergency Lighting

INSERT B2

Polar Crane - Penetration 1EE03E 2-350MCM per Ø

Unit Substation 1A1 Compt. 7B
(R,C); EL 781 FT.

Primary Protection

BBE Solid State Trip Device Type	SS14
Current Senser	600A
L.T. Setting	1.1 X TAP
ST Setting	10 X TAP

Secondary Protection

Westinghouse Type CO-8 Relay

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3.8.4.2

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION (Optional-Bypassed) CPS

LIMITING CONDITION FOR OPERATION

3.8.4.~~3~~2 The thermal overload protection of each valve shown in Table 3.8.4.~~3~~2-1 shall be bypassed (continuously) ~~(or) (only under accident conditions) (, as applicable,)~~ by an OPERABLE bypass device integral with the motor starter. CPS

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, 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). CPS

SURVEILLANCE REQUIREMENTS

4.8.4.~~3~~2.1 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 are bypassed only under accident~~ conditions): CPS

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.)~~ CP

b. Following maintenance on the motor starter.

(4.8.4.~~3~~2.2 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.) CPS

CPS

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM APPLICANT

DRAFT

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, (continuously) bypass the inoperable thermal overload within 8 hours (; restore the inoperable thermal overload to OPERABLE status within 30 days) or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

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TABLE 3.8.4.~~3~~2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)(No)	SYSTEM(S) AFFECTED
1SX073A, 74A & 76A		
1SX105A & 107A		
1SX071A		
1SX071B, 073B, & 074B		
1SX076B, 105B, & 107B		
1SX003A, 004A, & 014A		
1SX008A, 011A, & 013D		
1SA003B, 004B, 011B, & 014B		
1SX008B & 013B		
1SX003C & 004C		
1SX008C, 013F, 014C		
1SX017A, 019A, 020A, & 063A		
1S017B, 019B, 020B, & 063B		
1FP051, 054, & 078		
1FP052, 053, & 079		
1SX006C		
1SX088A, 096A, & 097A		
1SX012A, 016A, 062A, & 082		
1SX089A & 095A and 1E12-F014A		
1FC008 & 036		
1HG001, 004, & 009A		
1IA012A & 012B		
1CY016 & 1CY021		
1SF001 & 004		
0MC009		
1FC015A		
1FC011A		
1 VP004A, 004B, 015A, & 015B		
1 WO001A & 002A		
1VQ 006A & IVR002A		
1CC075A & 076A		
ICC049, 054, 072, & 073		
1SM001A & 002A		
1SX088B, 1SX089B, 096A & 096B		
1SX012B, 016B, 062B, 082B		
1E12-F014B and 1SX095B & 097B		
1HG005, 008, & 009B		
1IA013A & 013B		
1CY017 & 020		

Insert Attached

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Valve No.	Bypass	Direction	System(s) Affected
1B21-F016	Continuous	Close	Nuclear Boiler
1B21-F019	Continuous	Close	Nuclear Boiler
1B21-F065A	Continuous	Open/Close	Nuclear Boiler
1B21-F065B	Continuous	Open/Close	Nuclear Boiler
1B21-F067A	Continuous	Close	Nuclear Boiler
1B21-F067B	Continuous	Close	Nuclear Boiler
1B21-F067C	Continuous	Close	Nuclear Boiler
1B21-F067D	Continuous	Close	Nuclear Boiler
1B21-F068	Continuous	Close	Nuclear Boiler
1B21-F098A	Continuous	Close	Nuclear Boiler
1B21-F098B	Continuous	Close	Nuclear Boiler
1B21-F098C	Continuous	Close	Nuclear Boiler
1B21-F098D	Continuous	Close	Nuclear Boiler
1CC049	Continuous	Close	Component Cool Water
1CC050	Continuous	Close	Component Cool Water
1CC053	Continuous	Close	Component Cool Water
1CC054	Continuous	Close	Component Cool Water
1CC057	Continuous	Close	Component Cool Water
1CC060	Continuous	Close	Component Cool Water
1CC065	Continuous	Close	Component Cool Water
1CC068	Continuous	Close	Component Cool Water
1CC071	Continuous	Open/Close	Component Cool Water
1CC072	Continuous	Open	Component Cool Water
1CC073	Continuous	Open	Component Cool Water
1CC074	Continuous	Open/Close	Component Cool Water
1CC075A	Continuous	Close	Component Cool Water
1CC075B	Continuous	Close	Component Cool Water
1CC076A	Continuous	Close	Component Cool Water
1CC076B	Continuous	Close	Component Cool Water
1CC127	Continuous	Close	Component Cool Water
1CC128	Continuous	Close	Component Cool Water
1CC264	Continuous	Close	Component Cool Water
1CC265	Continuous	Close	Component Cool Water
1CY016	Continuous	Close	Cycled Condensate
1CY017	Continuous	Close	Cycled Condensate
1CY020	Continuous	Close	Cycled Condensate
1CY021	Continuous	Close	Cycled Condensate
1C41-F001A	Continuous	Open	Standby Liquid Control
1C41-F001B	Continuous	Open	Standby Liquid Control
1E12-F003A	Continuous	Open	Residual Heat Removal
1E12-F003B	Continuous	Open	Residual Heat Removal
1E12-F004A	Continuous	Open/Close	Residual Heat Removal
1E12-F004B	Continuous	Open/Close	Residual Heat Removal
1E12-F006A	Continuous	Close	Residual Heat Removal
1E12-F006B	Continuous	Close	Residual Heat Removal
1E12-F008	Continuous	Open/Close	Residual Heat Removal
1E12-F009	Continuous	Close	Residual Heat Removal
1E12-F011A	Continuous	Close	Residual Heat Removal
1E12-F011B	Continuous	Close	Residual Heat Removal
1E12-F014A	Continuous	Open/Close	Residual Heat Removal
1E12-F014B	Continuous	Open/Close	Residual Heat Removal

INSERT (Cont'd)

Valve No.	Bypass	Direction	System(s) Affected
1E12-F021	Continuous	Close	Residual Heat Removal
1E12-F023	Continuous	Open/Close	Residual Heat Removal
1E12-F024A	Continuous	Open/Close	Residual Heat Removal
1E12-F024B	Continuous	Open/Close	Residual Heat Removal
1E12-F026A	Continuous	Close	Residual Heat Removal
1E12-F026B	Continuous	Close	Residual Heat Removal
1E12-F027A	Continuous	Open/Close	Residual Heat Removal
1E12-F027B	Continuous	Open/Close	Residual Heat Removal
1E12-F028A	Continuous	Open/Close	Residual Heat Removal
1E12-F028B	Continuous	Open/Close	Residual Heat Removal
1E12-F037A	Continuous	Open/Close	Residual Heat Removal
1E12-F037B	Continuous	Open/Close	Residual Heat Removal
1E12-F040	Continuous	Close	Residual Heat Removal
1E12-F042A	Continuous	Open/Close	Residual Heat Removal
1E12-F042B	Continuous	Open/Close	Residual Heat Removal
1E12-F042C	Continuous	Open/Close	Residual Heat Removal
1E12-F047A	Continuous	Open	Residual Heat Removal
1E12-F047B	Continuous	Open	Residual Heat Removal
1E12-F048A	Continuous	Open/Close	Residual Heat Removal
1E12-F048B	Continuous	Open/Close	Residual Heat Removal
1E12-F049	Continuous	Close	Residual Heat Removal
1E12-F052A	Continuous	Close	Residual Heat Removal
1E12-F052B	Continuous	Close	Residual Heat Removal
1E12-F053A	Continuous	Open/Close	REsidual Heat Removal
1E12-F053B	Continuous	Open/Close	Residual Heat Removal
1E12-F064A	Continuous	Open/Close	Residual Heat Removal
1E12-F064B	Continuous	Open/Close	Residual Heat Removal
1E12-F064C	Continuous	Open/Close	Residual Heat Removal
1E12-F068A	Continuous	Open	Residual Heat Removal
1E12-F068B	Continuous	Open	Residual Heat Removal
1E12-F073A	Continuous	Open/Close	Residual Heat Removal
1E12-F073B	Continuous	Open/Close	Residual Heat Removal
1E12-F074A	Continuous	Open/Close	Residual Heat Removal
1E12-F074B	Continuous	Open/Close	Residual Heat Removal
1E12-F087A	Continuous	Close	Residual Heat Removal
1E12-F087B	Continuous	Close	Residual Heat Removal
1E12-F094	Continuous	Open/Close	Residual Heat Removal
1E12-F096	Continuous	Open/Close	Residual Heat Removal
1E12-F105	Continuous	Open/Close	Residual Heat Removal
1E21-F001	Continuous	Open/Close	Condensate Polishing
1E21-F005	Continuous	Open/Close	Condensate Polishing
1E21-F011	Continuous	Open/Close	Condensate Polishing
1E21-F012	Continuous	Close	Condensate Polishing
1E22-F001	Continuous	Open/Close	High Press Core Spray
1E22-F004	Continuous	Open/Close	High Press Core Spray
1E22-F010	Continuous	Close	High Press Core Spray
1E22-F011	Continuous	Close	High Press Core Spray
1E22-F012	Continuous	Open/Close	High Press Core Spray
1E22-F015	Continuous	Open/Close	High Press Core Spray
1E22-F023	Continuous	Close	High Press Core Spray

INSERT (Cont'd)

Valve No.	Bypass	Direction	System(s) Affected
1E32-F001A	Continuous	Close	Isolation Valve Seal
1E32-F001E	Continuous	Close	Isolation Valve Seal
1E32-F001J	Continuous	Close	Isolation Valve Seal
1E32-F001N	Continuous	Close	Isolation Valve Seal
1E32-F002A	Continuous	Close	Isolation Valve Seal
1E32-F002E	Continuous	Close	Isolation Valve Seal
1E32-F002J	Continuous	Close	Isolation Valve Seal
1E32-F002N	Continuous	Close	Isolation Valve Seal
1E32-F003A	Continuous	Close	Isolation Valve Seal
1E32-F003E	Continuous	Close	Isolation Valve Seal
1E32-F003J	Continuous	Close	Isolation Valve Seal
1E32-F003N	Continuous	Close	Isolation Valve Seal
1E32-F006	Continuous	Close	Isolation Valve Seal
1E32-F007	Continuous	Close	Isolation Valve Seal
1E32-F008	Continuous	Close	Isolation Valve Seal
1E32-F009	Continuous	Close	Isolation Valve Seal
1E51-F010	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F013	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F019	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F022	Continuous	Open/close	Reac. Core Isol. Cool
1E51-F031	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F045	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F046	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F059	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F063	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F064	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F068	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F076	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F077	Continuous	Open/Close	Reac. Core Isol. Cool
1E51-F078	Continuous	Open/Close	Reac. Core Isol. Cool
1FC007	Continuous	Close	Fuel Pool Cool & Clean
1FC008	Continuous	Close	Fuel Pool Cool & Clean
1FC011A	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC011B	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC015A	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC015B	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC016A	Continuous	Close	Fuel Pool Cool & Clean
1FC016B	Continuous	Close	Fuel Pool Cool & Clean
1FC024A	Continuous	Close	Fuel Pool Cool & Clean
1FC024B	Continuous	Close	Fuel Pool Cool & Clean
1FC026A	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC026B	Continuous	Open/Close	Fuel Pool Cool & Clean
1FC036	Continuous	Close	Fuel Pool Cool & Clean
1FC037	Continuous	Close	Fuel Pool Cool & Clean
1FP050	Continuous	Close	Fire Protection
1FP051	Continuous	Close	Fire Protection
1FP052	Continuous	Close	Fire Protection
1FP053	Continuous	Close	Fire Protection
1FP054	Continuous	Close	Fire Protection
1FP078	Continuous	Close	Fire Protection
1FP079	Continuous	Close	Fire Protection
1FP092	Continuous	Close	Fire Protection

INSERT (Cont'd)

Valve No.	Bypass	Direction	System(s) Affected
1G33-F001	Continuous	Close	React. Wtr. Clean Up
1G33-F004	Continuous	Close	React. Wtr. Clean Up
1G33-F028	Continuous	Close	React. Wtr. Clean Up
1G33-F034	Continuous	Close	React. Wtr. Clean Up
1G33-F039	Continuous	Close	React. Wtr. Clean Up
1G33-F040	Continuous	Close	React. Wtr. Clean Up
1G33-F053	Continuous	Close	React. Wtr. Clean Up
1G33-F054	Continuous	Close	React. Wtr. Clean Up
1HG001	Continuous	Open	H2 Recombining
1HG004	Continuous	Open/Close	H2 Recombining
1HG005	Continuous	Open/Close	H2 Recombining
1HG008	Continuous	Open/Close	H2 Recombining
1HG009A	Continuous	Open/Close	H2 Recombining
1HG009B	Continuous	Open/Close	H2 Recombining
1IA012A	Continuous	Close	Instrument Air
1IA012B	Continuous	Close	Instrument Air
1IA013A	Continuous	Close	Instrument Air
1IA013B	Continuous	Close	Instrument Air
OMC009	Continuous	Open	Make Up Condensate Storage
OMC010	Continuous	Open	Make Up Condensate Storage
1SF001	Continuous	Close	Suppression Pool & Clean Up
1SF002	Continuous	Close	Suppression Pool & Clean Up
1SF004	Continuous	Close	Suppression Pool & Clean Up
1SM001A	Continuous	Open	Suppression Pool Make-up
1SM001B	Continuous	Open	Suppression Pool Make-up
1SM002A	Continuous	Open	Suppression Pool Make-up
1SM002B	Continuous	Open	Suppression Pool Make-up
1SX003A	Continuous	Open	Shutdown Service Water
1SX003B	Continuous	Open	Shutdown Service Water
1SX003C	Continuous	Open	Shutdown Service Water
1SX004A	Continuous	Open	Shutdown Service Water
1SX004B	Continuous	Open	Shutdown Service Water
1SX004C	Continuous	Open	Shutdown Service Water
1SX006C	Continuous	Open	Shutdown Service Water
1SX008A	Continuous	Open/Close	Shutdown Service Water
1SX008B	Continuous	Open/Close	Shutdown Service Water
1SX008C	Continuous	Open/Close	Shutdown Service Water
1SX011A	Continuous	Open/Close	Shutdown Service Water
1SX011B	Continuous	Open/Close	Shutdown Service Water
1SX012A	Continuous	Open/Close	Shutdown Service Water
1SX012B	Continuous	Open/Close	Shutdown Service Water
1SX013D	Continuous	Open/Close	Shutdown Service Water
1SX013E	Continuous	Open/Close	Shutdown Service Water
1SX013F	Continuous	Open/Close	Shutdown Service Water
1SX014A	Continuous	Close	Shutdown Service Water
1SX014B	Continuous	Close	Shutdown Service Water
1SX014C	Continuous	Close	Shutdown Service Water

INSERT (Cont'd)

Valve No.	Bypass	Direction	System(s) Affected
1SX016A	Continuous	Open/Close	Shutdown Service Water
1SX016B	Continuous	Open/Close	Shutdown Service Water
1SX017A	Continuous	Open/Close	Shutdown Service Water
1SX017B	Continuous	Open/Close	Shutdown Service Water
1SX19A	Continuous	Open/Close	Shutdown Service Water
1SX19B	Continuous	Open/Close	Shutdown Service Water
1SX020A	Continuous	Close	Shutdown Service Water
1SX020B	Continuous	Close	Shutdown Service Water
1SX062A	Continuous	Open/Close	Shutdown Service Water
1SX062B	Continuous	Open/Close	Shutdown Service Water
1SX063A	Continuous	Open	Shutdown Service Water
1SX063B	Continuous	Open	Shutdown Service Water
1SX071A	Continuous	Open/Close	Shutdown Service Water
1SX071B	Continuous	Open/Close	Shutdown Service Water
1SX073A	Continuous	Open/Close	Shutdown Service Water
1SX073B	Continuous	Open/Close	Shutdown Service Water
1SX074A	Continuous	Open/Close	Shutdown Service Water
1SX074B	Continuous	Open/Close	Shutdown Service Water
1SX076A	Continuous	Open/Close	Shutdown Service Water
1SX076B	Continuous	Open/Close	Shutdown Service Water
1SX082A	Continuous	Close	Shutdown Service Water
1SX082B	Continuous	Close	Shutdown Service Water
1SX088A	Continuous	Open/Close	Shutdown Service Water
1SX088B	Continuous	Open/Close	Shutdown Service Water
1SX089A	Continuous	Open/Close	Shutdown Service Water
1SX089B	Continuous	Open/Close	Shutdown Service Water
1SX095A	Continuous	Open	Shutdown Service Water
1SX095B	Continuous	Open	Shutdown Service Water
1SX096A	Continuous	Open/Close	Shutdown Service Water
1SX096B	Continuous	Open/Close	Shutdown Service Water
1SX097A	Continuous	Open/Close	Shutdown Service Water
1SX097B	Continuous	Open/Close	Shutdown Service Water
1SX105A	Continuous	Open/Close	Shutdown Service Water
1SX105B	Continuous	Open/Close	Shutdown Service Water
1SX107A	Continuous	Open/Close	Shutdown Service Water
1SX107B	Continuous	Open/Close	Shutdown Service Water
1SX173A	Continuous	Open/Close	Shutdown Service Water
1SX173B	Continuous	Open/Close	Shutdown Service Water
2SX073A	Continuous	Open/Close	Shutdown Service Water
2SX073B	Continuous	Open/Close	Shutdown Service Water
2SX076A	Continuous	Open/Close	Shutdown Service Water
2SX076B	Continuous	Open/Close	Shutdown Service Water
2SX107A	Continuous	Open/Close	Shutdown Service Water
2SX107B	Continuous	Open/Close	Shutdown Service Water
1VP004A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP004B	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP005A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP005B	Continuous	Close	Drywell Cooling - Plant Chilled Water

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3/4 8-24E

INSERT (Cont'd)

Valve No.	Bypass	Direction	System(s) Affected
1VP014A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP014B	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP015A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VP015B	Continuous	Close	Drywell Cooling - Plant Chilled Water
1VQ006A	Continuous	Close	Drywell Purge - Contain HVAC
1VQ006B	Continuous	Close	Drywell Purge - Contain HVAC
1VR002A	Continuous	Close	Drywell Purge - Contain HVAC
1VR002B	Continuous	Close	Drywell Purge - Contain HVAC
1W0001A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1W0001B	Continuous	Close	Dryweall Cooling - Plant Chilled Water
1W0002A	Continuous	Close	Drywell Cooling - Plant Chilled Water
1W0002B	Continuous	Close	Drywell Cooling - Plant Chilled Water

INSERT

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TABLE 3.8.4.~~3~~2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)(No)	SYSTEM(S) AFFECTED
1SF002		
1FC015B, 016B, 024B, & 026B		
1FC011B		
1FC007		
1FC037		
0MC010		
1VP005A, 005B, 014A, & 014B		
1WO001B & 002B		
1VQ006B & 1VR002B		
1CC075B & 076B		
1CC065, 067, 068, & 070		
1CC050, 053, 072, & 128		
1CC057 & 060		
1CC071 & 074		

3/4.8.4.3 DRAFT | CPS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.~~4~~ 3 ~~Two~~ One RPS electric power monitoring channels for ~~the~~ each inservice RPS ~~MG set,~~ Special solenoid power supply or alternate power supply shall be OPERABLE. | CPS

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~~ the RPS special solenoid electric power monitoring channels for an inservice RPS ~~MG set~~ special solenoid power supply or alternate power supply inoperable, restore ~~at least one electric~~ the power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS ~~MG set~~ Special solenoid power supply or alternate power supply from service. | CPS

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS special solenoid electric power monitoring channels shall be determined OPERABLE: | CPS

a. At least once per six months by performance of a CHANNEL FUNCTIONAL TEST; and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage ≤ (~~132~~ *) VAC (Bus A), ≤ * (Bus B),
2. Under-voltage ≥ (~~108~~ *) VAC, ~~and~~ (Bus A), ≥ * VAC (Bus B), and
3. Under-frequency > (~~57~~ 57) Hz., -0 + 2% | CPS

* To be determined later by on-site measurements | CPS

NEW | CPS

ELECTRICAL POWER SYSTEMS

3/4.8.4.4 NON-SAFEFTY LOADS ON EMERGENCY SOURCES

LIMITING CONDITION FOR OPERATION

3.8.4.4 All over-current devices connecting class 1E sources to non-safety loads shown in Table 3.8.4.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION: Insert Attached

SURVEILLANCE REQUIREMENTS

4.8.4.4 Each over-current protective device shown in Table 3.8.4.4-1 shall be demonstrated OPERABLE:

a. At least once per 18 months.

1. By selecting and functionally testing a representative sample of at least 10% of each type of circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall follow manufacturer's instructions and shall test the long time, and instantaneous elements for pickup and time delay, where applicable. 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.

2. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

a. With one or more of the above required containment penetration conductor overcurrent devices shown in Table 3.8.4.1-1 and/or fuses tested pursuant Specification 4.8.4.1.a.2 inoperable:

   1. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping, racking out, or removing the alternate device or racking out or removing the inoperable device within 72 hours, and

   2. Verify at least once per 7 days thereafter the alternate device is tripped, racked out, or removed, or the device is racked out or removed.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices which have the inoperable device racked out or removed or, which have the alternate device tripped, racked out, or removed.

TABLE 3.8.4.4-1

NEW

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Auxiliary Building MCC 1A1 (1AP72E)
Location 121, Y (R,C); EL 781 FT

Each Service listed below has two (2) qualified interrupting devices in series.

COMPT/CKT	CIR BKR TRIP	SERVICE
2B	100	Standby Lighting Cabit 158 1LL58EA
9B/1	30A	Operations Radio Feed
8D	30A	Operations Radio Feed
9B/2	30	Telephone Feed
8D	30	Telephone Feed
9B/48	20	Emergency Notification Feed
8D	20	Emergency Notification Feed
9B/46	20	Upper and Lower Personnel Airlock Lights

Auxiliary Building MCC 1A3 (1AP74E)
Location 121, V (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT/CKT	CIR BKR TRIP	SERVICE
14CL	100	Standby Lighting Cabinet 161 1LL61EA
10AL	100	Standby Lighting Cabinet 161 1LL61EA
14CR	100	Standby Lighting Cabinet 170 1LL70EA
10CR	100	Standby Lighting Cabinet 170 1LL70EA

NEW | CPS

TABLE 3.8.4.4-1

(Continued)

Auxiliary Building MCC 1B1 (1AP75E)
Location 105, X (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT/CKT	CIR BKR TRIP	SERVICE
7B/1	30	Radio Base Station "B"
7AR	30	

Auxiliary Building MCC 1B1 (1AP76E)
Location 106, V (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT	CIR BRK TRIP	FUSE SIZE	SERVICE
14AL	100	100	Standby Lighting Cabinet 159 1LL59EA
14AR	100	100	Standby Lighting Cabinet 160 1LL60EA

Auxiliary Building DC MCC 1A (1DC13E)
Location 117, V (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT/CKT	CKT BRK TRIP	FUSE SIZE	SERVICE
11A/7	100	----	Emergency Lighting Cab 164 1LL64E
13C	100	----	Emergency Lighting Cab 164 1LL64E

Table 3.8.4.4-1

(Continued)

NEW | CPS

Auxiliary Building DC MCC 1B (1DC14E)
Location 107, U-V, (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT/CKT	CKT BRK TRIP	FUSE SIZE	SERVICE
4A/5	100	100	Emergency Lighting Cab. 165 1LL65E
4A/6	100	100	Emergency Lighting Cab. 166 1LL66E

Auxiliary Building DC MCC 1D (1DC15E)
Location 124, T, (R,C); EL 781 FT

Each Service listed below has two qualified interrupting devices in series.

COMPT/CKT	CRT BKR TRIP	FUSE SIZE	SERVICE
3A/1	100	----	Emergency Lighting Cab. 163 1LL63E
4C	100	----	Emergency Lighting Cab. 163 1LL63E

| CPS

NEWS | CPS

ELECTRICAL POWER SYSTEMS

3/4.8.4.5 REDUNDANT FAULT PROTECTION FOR PGCC FIRE PROTECTION, COMMUNICATION, RPS, and MSIV CIRCUITS

LIMITING CONDITION FOR OPERATION

3.8.4.5 All over-current devices connecting class 1E sources to non-safety loads shown in Table 3.8.4.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION: Insert Attached

SURVEILLANCE REQUIREMENTS

4.8.4.5 Each over-current protective device shown in Table 4.8.4.5-1 shall be demonstrated OPERABLE:

a. At least once per 18 months.

1. By selecting and functionally testing a representative sample
of at least 10% of each type of circuit breakers. Circuit
breakers selected for functional testing shall be selected on
a rotating basis. Testing of these circuit breakers shall follow
manufacturer's instructions and shall test the long time, and
instantaneous elements for pickup and time delay, where
applicable. 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.

2. By selecting and functionally testing a representative sample
of each type of fuse on a rotating basis. Each representative
sample of fuses shall include at least 10% of all fuses of that
type. The functional test shall consist of a non-destructive
resistance measurement test which demonstrates that the fuse
meets its manufacturer's design criteria. Fuses found inoperable
during these functional testing shall be replaced with OPERABLE
fuses prior to resuming operation. For each fuse found inoperable
during these functional tests, an additional representative
sample of at least 10% of all fuses of that type shall be
functionally tested until no more failures are found or all
fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

a. With one or more of the above required containment penetration conductor overcurrent devices shown in Table 3.8.4.1-1 and/or fuses tested pursuant Specification 4.8.4.1.a.2 inoperable:

1. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping, racking out, or removing the alternate device or racking out or removing the inoperable device within 72 hours, and

2. Verify at least once per 7 days thereafter the alternate device is tripped, racked out, or removed, or the device is racked out or removed.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices which have the inoperable device racked out or removed or, which have the alternate device tripped, racked out, or removed.

TABLE 3.8.4.5-1

REDUNDANT FAULT PROTECTION FOR PGCC
FIRE PROTECTION, COMMUNICATIONS,
RPS and MSIV CIRCUITS

1. 120 VAC PGCC Fire Protection Cables in Class 1E Ducts

   (Later)

2. 120 VAC PGCC Communications Cables in Class 1E Ducts

   (Later)

3. 120 VAC RPS Scram Solenoid Supply Circuits

   (Later)

4. 120 VAC MSIV Solenoid Supply Circuits

   (LATER)

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3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. ~~Fuel grapple position.~~

| CPS

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.

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SURVEILLANCE REQUIREMENTS

NO CHANGE

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.

DRAFT

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Prior to and during the time any control rod is withdrawn# and shutdown margin demonstrations are in progress, ~~either~~

~~1)~~ The "shorting links" removed from the RPS circuitry. ~~, or~~

~~2) The rod pattern control system OPERABLE per Specification 3.1.4.2.~~

CPS

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.~~ CPS

DRAFT

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. Prior to and at least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps:

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

3. At least once per 24 hours.

d. Verifying, within 8 hours prior to and at least once per 12 hours during, that the RPS circuitry "shorting links" have been removed ~~(or that the rod pattern control system is OPERABLE)~~ during: CPS

1. The time any control rod is withdrawn,## or

2. Shutdown margin demonstrations.

~~## Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ CPS

DRAFT

REFUELING OPERATIONS

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

No Change

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.

DRAFT

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

No Change

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.

DRAFT

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. | CPS

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.* | CPS

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.* | CPS

*Except movement of incore instrumentation and control rods with their normal drive system.

DRAFT

3/4.9.6 FUEL HANDLING EQUIPMENT

FUEL HANDLING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.1 The fuel handling platform shall be OPERABLE and used for handling of fuel assemblies or control rods during operation associated with the IFTS or the spent fuel storage pool, with only the main hoist used for moving irradiated fuel. CPS

APPLICABILITY: During handling of fuel assemblies of control rods during operations associated with the IFTS or the spent fuel storage pool.

ACTION:

With the requirements for fuel handling platform OPERABILITY not satisfied, suspend use of any inoperable platform equipment from operations involving the handling of control rods and fuel assemblies during operations associated with the IFTS or the spent fuel storage pool after placing the load in a safe location. The provisions of Specification 3.0.3 or 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each fuel handling platform crane or hoist used for handling of control rods or fuel assemblies during operations associated with the IFTS or the spent fuel storage pool shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff when the load exceeds:

   1. 1000 ± 50 pounds for the fuel hoist.

   2. 1000 ± 50 pounds for the auxiliary hoist with the Load Selector on 1000 pounds.

   3. 500 ± 50 pounds for the auxiliary hoist with the Load Selector on 500 pounds.

b. Demonstrating operation of the fuel hoist loaded interlock when the load exceeds 350 ± 50 pounds.

c. Demonstrating operation of the downtravel stop when downtravel below the platform rails exceeds:

   1. _______ feet for the fuel hoist.

   2. _______ feet for the auxiliary hoist.

CPS

Insert Attached

Insert 4.9.6.1

a. Demonstrating operation on the slack cable cutoff on the main hoist when the total cable load is 50±10 pounds.

b. Demonstrating operation of the grapple engaged loaded interlock on the main hoist before the total cable load exceeds 400 pounds.

c. Demonstrating operation of the jam cutoff on the main hoist before the total cable load exceeds 1150 pounds.

d. Demonstrating operation of the primary and redundant overload cutoff on the auxiliary hoist before the load exceeds 550 pounds with the load override switch at the 500 pound position.

e. Demonstrating operation of the primary and redundant overload cutoff on the auxiliary hoist before the load exceeds 1050 pounds with the load override switch at the 1000 pound position.

f. Demonstrating operation of the downtravel cutoff on the main hoist when the bottom of the grapple is 4.0 inches below the top of the fuel assembly handles in the reactor core.

DRAFT

SURVEILLANCE REQUIREMENTS (Continued)

g ~~d.~~ Demonstrating operation of the fuel hoist and auxiliary hoist uptravel stops when the grapple is less than or equal to 8 feet below the platform rails.

~~e. Demonstrating operation of the fuel hoist slack cable cutoff when the load is less than 50 ± 10 pounds.~~ CPS

h ~~f.~~ Demonstrating operation of the fuel hoist disengaged interlock that prevents hook disengagement when the fuel hoist is loaded.

i ~~g.~~ Demonstrating operation of the fuel hoist raise power cutoff when the hoist is loaded and the hook disengaged.

j ~~h.~~ Demonstrating operation of the fuel hoist raise power cutoff when the fuel handling platform area radiation monitor dose rate exceeds ______ MR/HR.

4.9.6.1.2

The auxiliary hoist load override switch shall be verified to be in the 500 pound position within 2 hours and at least once per 12 hours during hoist operation, except when engaged in new fuel movement in which case the switch may be in the 1000 pound position. CPS

DRAFT

REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.2 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.

Replace with attached revised page

SURVEILLANCE REQUIREMENTS

4.9.6.2 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 hoists when the load exceeds (500 ± 50) pounds.

c. Demonstrating operation of the uptravel mechanical stop on the frame mounted and monorail hoists when uptravel brings the top of (active) fuel assembly to (8) feet below the (normal fuel storage pool) water level.

d. Demonstrating operation of the downtravel mechanical cutoff on the main hoist when grapple hook down travel reaches (4) inches below fuel assembly handle.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than (50 ± 10) pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds (485 ± 50) pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds (550 ± 50) pounds.

DRAFT

REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.2 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel with only the main hoist used for handling fuel assemblies

CPS

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

4.9.6.2 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. In the containment fuel pool, reactor cavity or reactor pressure vessel by:

1. Demonstrating operation of the slack cable cutoff on the main hoist when the total cable load is 50± 10 pounds.

2. Demonstrating operation of the grapple engaged loaded interlock on the main hoist before the total cable load exceeds 535 pounds.

3. Demonstrating operation of the jam cutoff on the main hoist before the total cable load exceeds 1250 pounds.

4. Demonstrating operation of the primary and redundant overload cutoff on the auxiliary hoists before the load exceeds 550 pounds.

5. Demonstrating operation of the main hoist and auxiliary hoist up-travel stops when the grapple is lower than or equal to 8 feet below the platform rails.

CPS

INSERT

b. In or over the reactor pressure vessel by:

1. Demonstrating operation of the downtravel cutoff on ~~the main~~ hoist when the bottom of the grapple is 4.0 inches below the top of the fuel assembly handles in the reactor core.

2. Demonstrating operation of the primary and redundant fuel load interlocks on the main hoist before the total cable load exceeds 600 pounds.

CPS

NEW

REFUELING OPERATIONS

AUXILIARY PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.3 The auxiliary platform shall be OPERABLE.

APPLICABILITY: During handling of control rods with the auxiliary platform.

ACTION:

With the requirements for auxiliary platform OPERABILITY not satisfied, suspend use of the auxiliary platform after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.3 The auxiliary platform hoist shall be demonstrated OPERABLE within 7 days prior to the handling of control rods by:

a. DEMONSTRATING OPERATION OF THE OVERLOAD CUTOFF BEFORE THE LOAD EXCEEDS 500 pounds.

b. DEMONSTRATING OPERATION OF THE AUXILIARY PLATFORM HOIST UPTRAVEL STOPS WHEN THE GRAPPLE IS LOWER THAN OR EQUAL TO 8 FEET BELOW THE PLATFORM RAILS.

CPS

DRAFT

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of (~~1100~~ LATER) pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage or upper containment fuel pool racks. ICPS

APPLICABILITY: With fuel assemblies in the spent fuel storage or upper containment fuel 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

4.9.7.1 Crane physical stops which prevent (   ) crane travel with loads in excess of (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.

4.9.7.2 Crane interlocks (and physical stops) which prevent containment polar crane travel with loads in excess of (1100) pounds over fuel assemblies in the upper containment fuel pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

DRAFT

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

NO CHANGE

3.9.8 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

No Change

DRAFT

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage and upper containment fuel pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage or upper containment fuel pools.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage or upper containment fuel pool areas, as applicable after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage and upper containment fuel pools shall be determined to be at least at its minimum required depth at least once per 7 days.

No Change

DRAFT

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements

DRAFT

SURVEILLANCE REQUIREMENTS

No Change

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

DRAFT

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f.

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.

All fuel loading operations shall be suspended.

DRAFT

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f.

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.

All fuel loading operations are suspended.

DRAFT

No Change

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 23 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY wtihin 4 hours.

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

SURVEILLANCE REQUIREMENTS

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

*The shutdown cooling loop may be removed from operation for up to 2 hours per 8-hour period.

DRAFT

LOW WATER LEVEL

No Change

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each train consisting of at least:

a. One OPERABLE RHR pump, and

b. 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.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

DRAFT

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access doors of all rooms through which the transfer system penetrates are closed and locked.

b. All access door interlocks are OPERABLE.

c. The Versa blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. All IFTS primary and secondary carriage position and liquid level indicators are OPERABLE.

e. The keylock switch which provides IFTS access control-transfer system lockout is OPERABLE.

(f. All flashing lights outside of access doors are OPERABLE.)

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

Replace with revised Page

SURVEILLANCE REQUIREMENTS

4.9.12 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

a. All access door interlocks are OPERABLE.

b. The Versa blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.

c. All IFTS primary and secondary carriage position and level indicators are OPERABLE.

d. The keylock switch which provides IFTS access control-transfer system lockout is OPERABLE.

(e. All flashing lights outside of access doors are OPERABLE.)

DRAFT

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access doors* to all rooms through which the transfer system penetrates are closed and locked.

b. All access door interlocks and palm switches are OPERABLE.

c. The blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. All IFTS primary and secondary carriage position and liquid level indicators are OPERABLE.**

e. Any keylock switch that provides IFTS access control-transfer system lockout is OPERABLE.

f. All warning lights outside the access doors are OPERABLE.

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

CPS

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

a. All access door interlocks and palm switches are OPERABLE.

b. The blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.

c. All IFTS primary and secondary carriage position and level indicators are OPERABLE.

d. Any keylock switch that provides IFTS access control-transfer system lockout is OPERABLE.

e. All warning lights outside the access doors are OPERABLE.

INSERT. Attached

INSERT

4.9.12.2 Within 1 hour prior to the startup of the IFTS, verify that no personnel are in areas immediately adjacent to the IFTS and that all access doors to rooms through which the IFTS penetrates are closed and locked.

CPS

*The definition of a door shall include those shield or removable plugs necessary for containment inspection areas adjacent to the IFTS or an IFTS maintenance access area.

**If any single primary or secondary carriage position or liquid level sensor is not OPERABLE, operation of the system may continue. The system shall be restored to OPERABLE condition at the first reasonable opportunity but prior to any subsequent refueling outage.

DRAFT

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

NO CHANGE

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3, 3.6.2.1, 3.6.2.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the containment and drywell air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

DRAFT

3/4.10.2 ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod pattern control system (RPCS) per Specification 3.1.4.2 may be suspended by means of the individual rod position bypass switches for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RPCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RPCS are bypassed, verify:

a. With 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed, that movement of the control rods from (75)% ROD DENSITY to the RPCS low power setpoint is limited to the established control rod sequence for the specified test, and

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

DRAFT

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE per Specification 3.9.2 with:

   S ~~1.~~ The RPS circuitry "shorting links" removed ~~(, or)~~.

   ~~2. The rod pattern control system OPERABLE per Specification 3.1.4.2).~~ CPS

b. Conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2, with the:

   ~~1.~~ RPS circuitry "shorting links" removed. ~~(, or)~~

   ~~(2. The rod pattern control system OPERABLE).~~ CPS

b. A second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c.. No other CORE ALTERATIONS are in progress.

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3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

NO CHANGE

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched flow may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

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3/4.10.5 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

NO CHANGE

3.10.5 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.5 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

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NEW CPS Section

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1 ~~5.1-3~~) 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. 85

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 are not applicable.~~ CPS

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.

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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~~[a]~~ [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]~~	~~Continuous[f]~~	~~W Composite[f]~~	~~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 Composite[f]~~	~~Sr-89, Sr-90~~	~~5×10^{-8}~~
			~~Fe-55~~	~~1×10^{-6}~~

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CPS

DRAFT

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 ~~5.1-3~~ 5.1.3-1) shall be limited: CPS

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, 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.*~~ CPS

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents to UNRESTRICTED AREAS 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. CPS

~~*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.~~ CPS

DRAFT

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

Insert Attached

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.~~ CPS

APPLICABILITY: At all times.

ACTION:

and any portion of the liquid radwaste treatment system not in operation CPS

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

4.11.1.3.1 Doses due to liquid releases ~~from each reactor unit~~ to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM. CPS

4.11.1.3.2 The installed liquid radwaste treatment system shall be demonstrated OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2 CPS

Insert 3.11.1.3

The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the releases of radioactivity when the projected dose due to the liquid effluent to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

RADIOACTIVE EFFLUENTS

STET

~~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.)~~

CPS

LIMITING CONDITION FOR OPERATION

Insert Attached

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 ___ curies, excluding tritium and dissolved or entrained noble gases.~~

~~a. _______~~
~~b. _______~~
~~c. _______~~
~~d. Outside temporary tank~~

CPS

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 and 3.0.4 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 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.~~

CPS

Insert

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tanks shall be limited to the limits calculated in the ODCM such that a complete release of the tank contents would not result in a concentration at the nearest offsite potable water supply that would exceed the limits specified in 10CFR20 Appendix B Table II.

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. CPS

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 6.9.1.9.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. CPS

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Insert Attached Page

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Offgas Treatment System (Pretreatment)*	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE	P Each PURGE[c] Grab Sample	P Each PURGE[c]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. (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}
			H-3	1×10^{-6}
D. All Release Types as listed in A, B, C above.	Continuous[f]	W[g] Charcoal Sample	I-131	1×10^{-12}
	Continuous[f]	W[g] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

*If the plant uses storage tanks, each tank shall be sampled prior to release and the sample analyzed

CPS

Insert to:

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. Containment Building PURGE and VENT	P Each PURGE and VENT Grab Sample	P Each PURGE and VENT	Principal Gamma Emitters[b]	1×10^{-4}
		M	H-3	1×10^{-6}
B. Station HVAC Exhaust	$M^{c,d}$ Grab Sample	$M^{c,d}$	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Standby Gas Treatment System Exhaust, when flow exists	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
D. All release Types as Listed in A, B, C above.	Continuous[e]	W^f Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[e]	W^f Particulate Sample	Principal Gamma Emitters[b]	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6} (Xe-133 equivalent)

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.

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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~~ 7. CPS

[c]Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within one hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor or 3.

~~[d]Not Applicable.~~ CPS

d ~~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. CPS

e ~~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. CPS

f ~~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 inc-eased 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. CPS

DRAFT

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~~ 5.1.2-1) 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, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DRAFT

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM | CPS

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following: | CPS

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, 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. | CPS

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days. | CPS

DRAFT

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 Insert Attached ~~The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.~~

APPLICABILITY: ~~Whenever the main condenser air ejector (evacuation) system is in operation.~~ Insert Attached

ACTION: Insert Attached

~~a. With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, 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. Identification of the inoperable equipment or subsystems and the reason for inoperability,
2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and
3. Summary description of action(s) taken to prevent a recurrence.

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

CPS

SURVEILLANCE REQUIREMENTS

4.11.2.4 ~~The readings of the relevant instruments shall be checked every 12 hours when the main condenser air ejector is in use to ensure that the gaseous radwaste treatment system is functioning.~~

The GASEOUS RADWASTE (OFFGAS) TREATMENT SYSTEM Shall be verified to be in either the normal or charcoal bypass mode at least once per 7 days whenever the main condenser steam jet air ejector system is in operation.

CPS

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE (OFFGAS) TREATMENT SYSTEM shall be in operation in either the normal or charcoal bypass mode. The charcoal bypass mode shall not be used unless the Post-Treatment Air Ejector Off-Gas Radiation Monitor is OPERABLE as specified in Table 3.3.7.12-1.

APPLICABILITY: Whenever the main condenser steam jet air ejector system is in operation.

ACTION:

a. With the GASEOUS RADWASTE (OFFGAS) TREATMENT SYSTEM not used in the normal mode for more than 7 days, 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:

Insert

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RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.5 Insert Attached ~~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 5.1.3-1) would exceed 0.3 mrem to any organ in a 31 day period.~~

~~APPLICABILITY: At all times other than when the VENTILATION EXHAUST TREATMENT system is undergiong routine maintenance.~~

~~ACTION:~~

~~a. With gaseous waste being discharged~~ ~~from the ventilation exhaust~~ ~~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~~ the inoperable equipment or subsystems, and the reason for the inoperability,

2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3. Summary description of action(s) taken to prevent a recurrence.

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

CPS

SURVEILLANCE REQUIREMENTS

~~4.11.2.5.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days in accordance with the ODCM,~~ ~~when the ventilation exhaust treatment system is not in use.~~

Insert Attached

CPS

LIMITING CONDITION FOR OPERATION

3.11.2.5 The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and 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.

APPLICABILITY: At all times.

ACTION:

a. With the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days, or 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 which includes the following information:

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 Doses due to gaseous release from each reactor unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

SURVEILLANCE REQUIREMENTS

4.11.2.5.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

DRAFT

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE ~~(Systems designed to withstand a hydrogen explosion)~~

~~Appropriate alternatives to the ACTIONS below can be accepted if they provide incentive for timely repair of monitors and for compliance with GDC 3 (Fire Protection).~~ | CPS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen ~~or oxygen~~ in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume. | CPS

APPLICABILITY: ~~At all times.~~ Whenever the main-condenser steam jet air ejector system is in operation.

ACTION:

a. With the concentration of hydrogen ~~or oxygen~~ in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

~~b. With continuous monitors inoperable, utilize grab sampling procedures for a period not to exceed 30 days.~~ | CPS

b ~~c~~. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable. | CPS

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen ~~or oxygen~~ in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main condenser offgas treatment system with the hydrogen or oxygen monitors required OPERABLE by Table 3.3.7.12-1 of Specification 3.3.7.12. | CPS

~~[illegible]~~ | CPS

Not Applicable to CPS

DRAFT

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE (Systems not designed to withstand a hydrogen explosion)

Appropriate alternatives to the ACTIONS below can be accepted if they provide incentive for timely repair to monitors and for comliance with GDC 3 (Fire Protection).

LIMITING CONDITION FOR OPERATION

3.11.2.6A The concentration of hydrogen and/or oxygen in the main condenser offgas treatment system shall be limited to less than or equal to 2% by volume.

whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen and/or oxygen in the main condenser offgas treatment 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 main condenser offgas treatment 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% 4% by volume without delay within 48 hours.

c. With continuous monitors inoperable, utilize grab sampling procedures for a period not to exceed 30 days.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6A The concentrations of hydrogen and/or oxygen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main condenser offgas treatment system whenever the main condenser evacuation system is in operation with the hydrogen and/or oxygen monitors required OPERABLE by Table 3.3.7.12-1 of Specification 3.3.7.12.

Not Applicable to CPS

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The gross radioactivity (~~beta and/or~~ gamma) rate of noble gases~~*~~ measured at the main condenser air ejector shall be limited to less than or equal to 289 millicuries/sec ~~100 microcuries/sec per MW~~ (after 30 minutes decay).

APPLICABILITY: At all times.

ACTION:

With the gross radioactivity (~~beta and/or~~ gamma) rate of noble gases at the main condenser air ejector exceeding 289 millicuries/sec ~~100 microcuries/sec per MW~~, (after 30 minutes decay), restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases at ~~(near)~~ the discharge ~~outlet~~ of the main condenser air ejector shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.7.2 The gross radioactivity (~~beta and/or~~ gamma) rate of noble gases~~*~~ from the main condenser air ejector shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the main condenser air ejector:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Off-Gas Radiation Pretreatment ~~Condenser Air Ejector Noble Gas Activity~~ Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

~~*Plants using gamma-scintillation detector(s) to measure the Kr-85m, 87, 88 and Xe-133, 135, 138 contribution after 30 minutes decay may substitute the words "release rate of the sum of the activities from the noble gases" for the words "gross radioactivity rate of noble gases" in this specification.~~

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with a ~~the~~ PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site. |CPS

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, ~~boric acid solutions,~~ and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM. |CPS

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transporation and disposal requirements.

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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 mrem~~s~~ to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem~~s~~. 8

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 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from unit operation ~~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. 8

New CPS Section

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report per Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. |CPS

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.7 ~~8,~~ identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s). CPS

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

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12.1-1, the detection capabilities required by Table 4.12.1-1.

~~4.12.1.2 Cumulative potential dose contributions for the current calendar year from radionuclides detected in environmental samples shall be determined in accordance with the methodology and paramaters in the ODCM.~~ CPS

TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[(a)]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION[(b)]	40 routine monitoring stations with two or more dosimeters or with one instrument for measuring and recording dose rate continuously placed as follows: (1) An inner ring of stations, one in each meteorological sector, in the general area of the SITE BOUNDARY with the exception of the W sector; (2) An outer ring of stations, one in each meteorological sector of NE, ENE, E, ESE, SE, in the 6.5 to 8 ~~mile~~ km range from the site; (3) The balance of the stations of the stations placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly.	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: a. 3 samples from close to the 3 SITE BOUNDARY locations (in different sectors) in one of the highest calculated annual average groundlevel χ/Q. b. 1 sample from the vicinity community having one of the highest calculated annual highest groundlevel χ/Q.	Continual 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.

CPS

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TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATION(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE (Continued)			
	1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction c		
3. WATERBORNE			
a. Surface f	1 sample upstream 1 sample downstream	Composite sample over one-month period. g	Gamma isotopic analysis e monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources only if likely to be affected. h	Quarterly.	Gamma isotopic e and tritium analysis quarterly.
c. Drinking	1 samples of each of 1 to 3 of the nearest water supplies that could be affected by its its discharge	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.
	1 sample from a control location.		
d. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis e semiannually.

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TABLE 3.12.1-1 (Continued)

Table Notation

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~7~~ by. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. Pursuant to Specification 6.9.1.~~8~~ 7, 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). |CPS |CPS

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.

[c]Insert Attached ~~Methodology to guarantee complete recovery of radioiodine shall be described in the ODCM.~~ |CPS

[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 the discharge line.

pg 3/4 12-7

Insert

[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 provi[de] valid background data may be substituted.

TABLE 3.12.1-1 (Continued)

Table Notation

[g]A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

[h]Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[i]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[j]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

ANALYSIS	WATER (pCi/L)	AIRBORN PARTICULATE or GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400(a)				
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(a)			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.

(a)Total for parent and daughter.

CPS

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b][c]

ANALYSIS	WATER (pCi/L)	AIRBORNE PARTICULATE OR GAS (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENTS (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30					
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-140	60			60		
La-140	15			15		

/CPS

TABLE 4.12.1-1 (Continued)

TABLE NOTATIONS

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable at 95% confidence level, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating report pursuant to Specification 6.9.1.~~7~~6 | CPS

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13, Rev. 1, July, 1977, except for the specification regarding energy dependence. Correction factors shall be provided for energy ranges not meeting the energy dependence specification. | CPS

[c]The LLD is defined, for purpose 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. | CPS

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),

8 is the sample size (in units of mass or volume),

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection (or end of the sample collection period) and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

Table 4.12.1-1 (Continued)

TABLE NOTATIONS

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~7~~ 6. | CPS

[d] LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used. | CPS

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment, greater than the values currently being calculated in Specification 3/4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7~~8~~. |CPS

b. With a land use census identify a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s) (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.7~~8~~, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6~~7~~. |CPS

*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 - item 4.c. shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~7~~6. |CPS

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1~~7~~6. |CPS

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BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not not part of these Technical Specifications.

NO CHANGE

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 requires two 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 3.6.7.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

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BASES

NO CHANGE

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.

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NO CHANGE

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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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. CPS

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

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3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.06 during the limiting power transient analyzed in Section (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 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. CPS

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

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BASES

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. CPS

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

The rod withdrawal limiter system input power signal orginates from the first stage turbine pressure. When operating with the steam bypass valves open, this signal indicates a core power level which is less than the true core power. Consequently, near the low power setpoint and high power setpoint of the rod pattern control system, the potential exists for nonconservative control rod withdrawals. Therefore, when operating at a sufficiently high power level, there is a small probability of violating fuel Safety Limits during a licensing basis rod withdrawal error transient. To ensure that fuel Safety Limits are not violated, ~~this~~ specification 3.1.4 prohibits control rod withdrawal when a biased power signal exists and core power exceeds the specified level. CPS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RPCS to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

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BASES

CONTROL ROD PROGRAM CONTROLS (Continued)

The RPCS provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.~~2~~ 4a of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3. CPS

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core INSERT ATTACHED ~~in approximately 90 to 120 minutes. A minimum available quantity of 3470 gallons of sodium pentaborate solution containing a minimum of 4246 lbs. of sodium pentaborate is required to meet a shutdown requirement of 3% Δk/k. There is an additional allowance of 150 ppm in the reactor core to account for imperfect mixing and the filling of other piping systems connected to the reactor vessel. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.~~ CPS

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

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972
2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972
3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

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and other piping systems connected to the 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 3542 gallons of sodium pentaborate solution containing a minimum of 4246 lbs of sodium pentaborate. This quantity of solution is a net amount which is above the pump suction, thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm per pump provides a negative reactivity inspection 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.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times (1.02) is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3. CPS

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. 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.

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BASES

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 scram ~~setting~~ trip setpoint and the upscale flow biased neutron flux-~~high~~ control rod block ~~functions of the APRM instruments~~ trip setpoint must be adjusted to ensure that the MCPR does not become less than 1.06 or that $\geq$ 1% plastic strain does not occur in the degraded situation. The scram ~~settings~~ and rod block ~~settings~~ setpoints 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) (peak power distribution to ensure than an LHGR transient would not be increased in degraded conditions). (The method used to determine the design TPF shall be consistent with the method used to determine the MTPF).~~ Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

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Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3015 Mwt* which corresponds to (105)% of rated steam flow | CPS

Vessel Steam Output .... 13.08 x 10^6 lbm/hr which corresponds to (105)% of rated steam flow | CPS

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks ~~0.1~~ 0.09 ft^2. | CPS

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	~~1.20~~ 1.17

| CPS

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistant with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figure 3.2.3-1. ~~The power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits.~~ CPS

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4] The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level.

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BASES

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial 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.~~ CPS

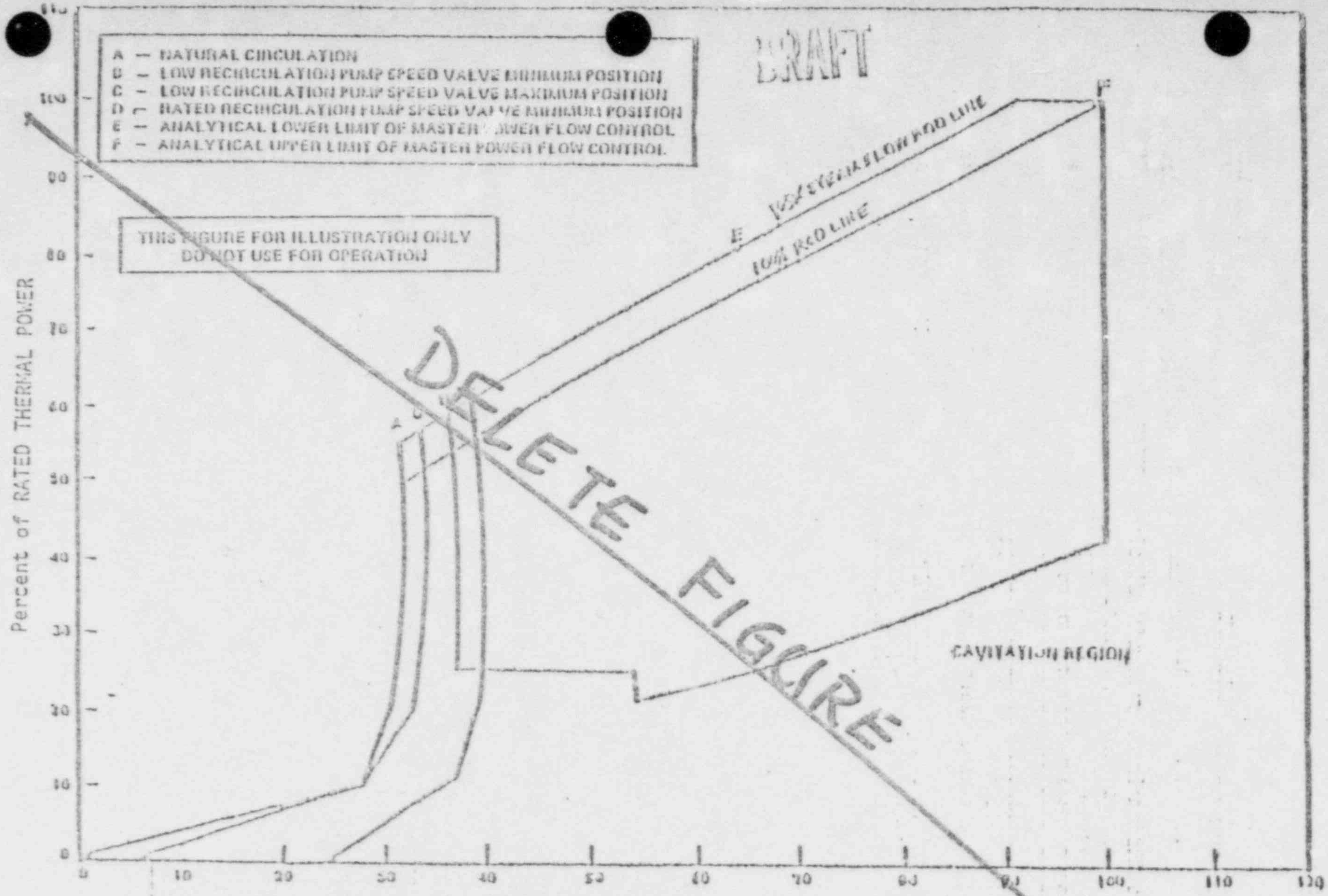
References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.

3. Qualification of the One Dimensional Core Transient Model For Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program For The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.



Power - Flow Operating Map

Figure B 3/4 2.3-1

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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.~~ CPS

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The normal channel coincident trip logic for the RPS is two-out-of-four. There are four redundant trip channels for each reactor trip function. An RPS trip channel may have inputs from more than one sensor. For each trip function using coincident trip logic, there are four two-out-of-four trip circuits, one located in each of the four RPS divisions. Whenever two or more channels of any reactor trip function indicate a trip condition, the four two-out-of-four coincident logics will trip in all four divisions and will produce a reactor scram.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION (Continued)

The non-coincidant NMS reactor trip function is enabled only when the "shorting links" are removed such that when any OPERABLE SRM, IRM or APRM channel indicates a trip condition, the non-coincident trip logic in each of the four divisions will trip and 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. CPS

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 respons times.

Because the trip logic of the solid state reactor protection system results in a trip of all four divisions and full reactor scram if the logic is satisfied for the coincident logic reactor trip functions or the non-coincident NMS reactor trip function, the REACTOR PROTECTION SYSTEM RESPONSE TIME tests of the various reactor trip functions can only be performed during shutdown. All four divisional logic response times are therefore checked for every response test of two RPS channels of each function. Each function has four logic trains through two four coincident ~~(or non-coincident)~~ logic circuits located one in each division. There are four coincident ~~(or non-coincident)~~ logic circuits associated with each reactor trip function each of which will cause the trip of one RPS division logic. CPS

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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.

The 2-out-of-4 logic for the MSIV isolation functions is identical to the logic of the RPS and ~~the~~ instrumentation system response time is ~~also identical.~~ treated in an identical manner. 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 for the establishment of emergency power. 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.~~ CPS

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.~~ Insert Attached CPS

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.~~ Insert Attached CPS

Insert 3/4.3.2 and 3/4.3.3

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.

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BASES

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979, and Section 5.8 ~~of Appendix (___)~~ of the FSAR. | CPS

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.~~ Reactor Protection System. | CPS 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~~ the 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.~~ four RPS logic divisions. Similarly, a (position switch) for each of ~~two~~ the 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.~~ four RPS logic divisions. For each EOC-RPT system, the sensor relay contacts are arranged to form ~~a 2-out-of-2~~ an any 2 logic for the fast closure of turbine control valves and ~~a 2-out-of-2~~ an any 2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps. | CPS

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)~~ 40% of RATED THERMAL POWER are annunciated in the control room. | CPS

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)~~ 140 ms~~, less the time allotted for sensor response, i.e., (10)ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., (83)ms, and plant pre-operational test results).~~

Insert Attached

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. | CPS

Insert 3/4 3.4

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.

Insert
B 3/4 3-4B

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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.~~ | CPS

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.~~ Insert Attached | CPS

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided and 3/4.3 Instrumentation consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits~~.~~ The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block. | CPS

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.~~ Insert Attached | CPS

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, ~~41~~, 60, 61, 63 and 64. | CPS

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

Insert 3/1.3.5 and 3/4.3.6

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.

MONITORING INSTRUMENTATION (Continued)

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~~ May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980) | CPS

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room ventilation system will automatically be placed in the chlorine 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). | CPS

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BASES

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warrented for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. (The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.) CPS

Insert 3/4.3.7.11, 3/4.3.7.12 Attached

~~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.~~ CPS

3/4.3.~~9~~8 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action of ~~the containment spray system and~~ the feedwater system/main turbine trip system in the event of failure of the feedwater controller under maximum demand. CPS

Insert Attached

DRAFT

3/4.3.~~8~~7.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. 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.~~8~~7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (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.

CPS

Insert

3/4.3.8 PLANT SYSTEMS ACTUATION INSTRUMENTATION (cont.)

The plant systems actuation instrumentation ~~also~~ initiates action to mitigate the consequences of accidents that are beyond the ability of the operator to control. The LPCI mode of the RHR system is automatically initiated on a high drywell pressure signal and/or a low reactor water level, level 1, signal. The containment spray system will then actuate automatically following high drywell and high containment pressure signals. A 10-minute time delay exists between initiation of LPCI and containment spray actuation. A high reactor water level, level 8, signal will actuate the feedwater system/main turbine trip system.

CPS

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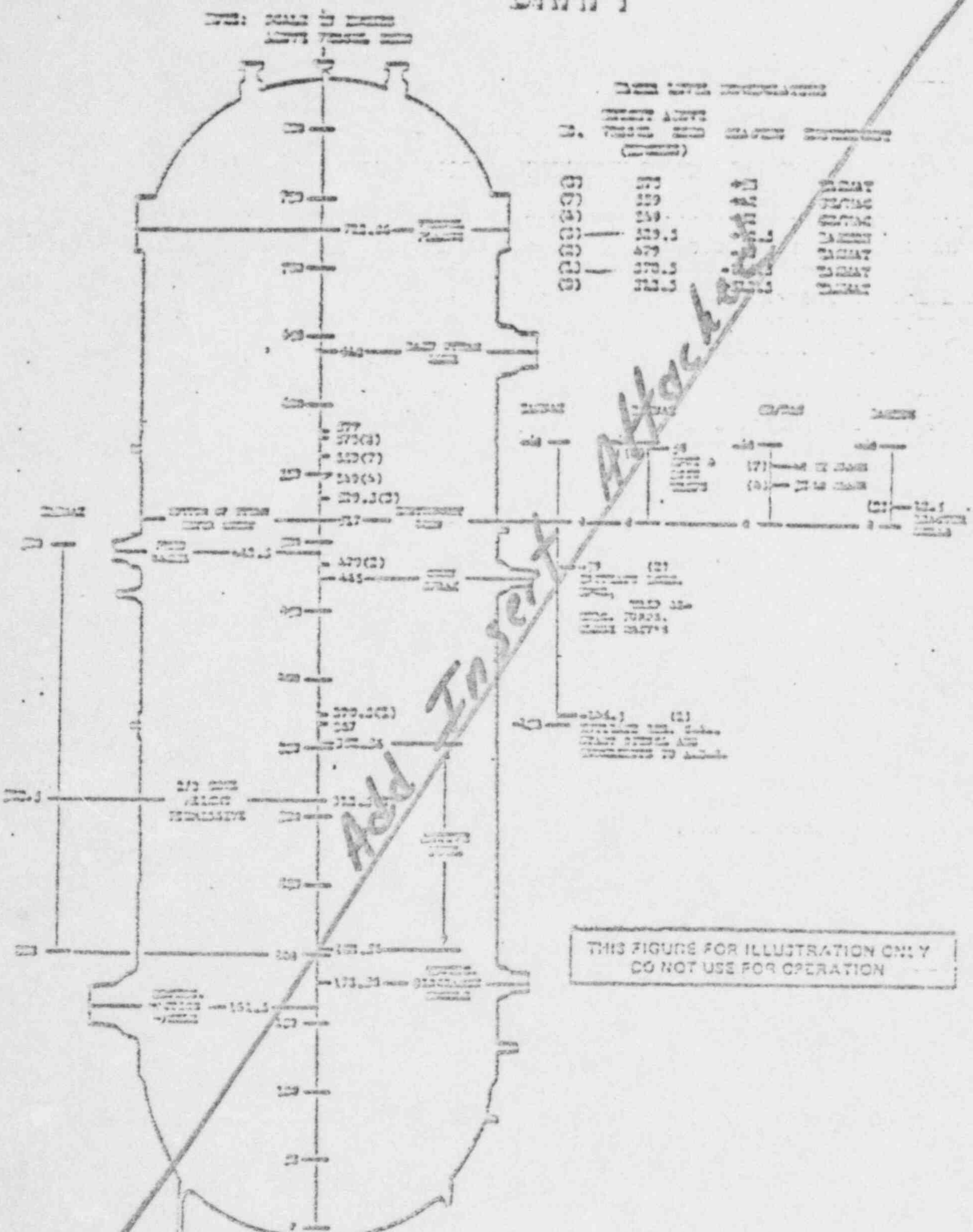
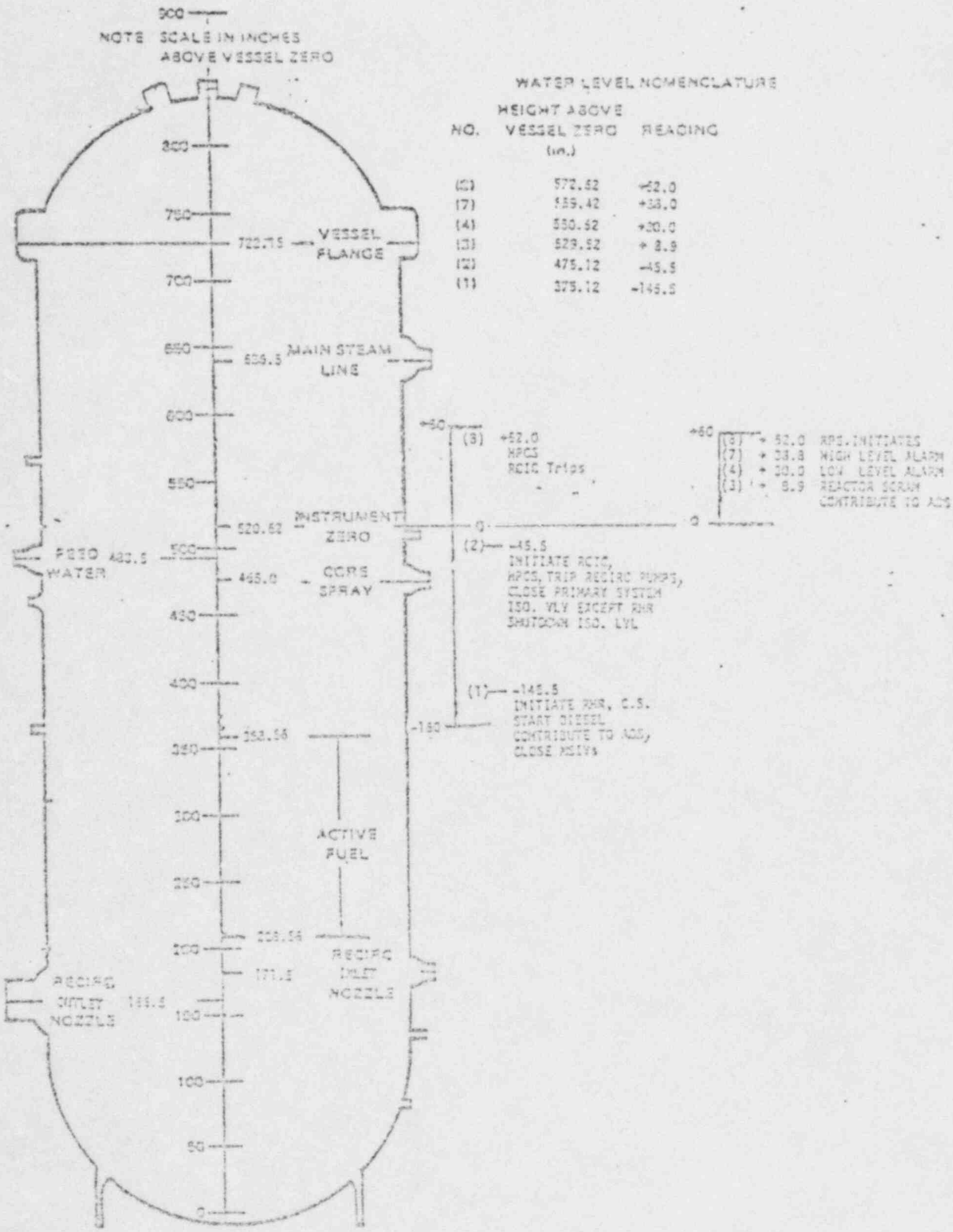


Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (in.)	READING
(8)	572.62	+52.0
(7)	559.42	+38.0
(4)	550.62	+30.0
(3)	529.52	+ 8.9
(2)	475.12	-45.5
(1)	375.12	-145.5

Bases Figure B3/4.3-1. Reactor Vessel Water Level

DRAFT

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 ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

50

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 (70)°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F. CPS

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 5 SRVs operating in the relief mode and 6 SRVs operating in the safety mode is acceptable. CPS

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

DRAFT

No Change

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.

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.

DRAFT

BASES

No Change

3/4.4.5 SPECIFIC ACTIVITY

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

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to (20) following a postulated (steam line rupture). The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.3 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.

Insert Attached

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

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

CPS

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and ~~the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~ Figure B 3/4.4.6-2. The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes an assumed ~~predicted~~ adjustments ~~for this~~ shift in RT_{NDT} for the end of life fluence, ~~as well as adjustments for possible errors in the pressure and temperature sensing instruments.~~

CPS

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and ~~10 CFR 50, Appendix H,~~ irradiated flux wires ~~reactor vessel material specimens~~ 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 ~~specimens~~ can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire ~~specimen~~ data and Bases ~~recommendations of Regulatory Guide 1.99, Revision 1.~~ Figure B 3/4.4.6-2.

CPS

Insert 3/4.4.6

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis of compliance are more fully discussed in FSAR subsection 5.2.3.3.1 entitled "Fracture Toughness."

DRAFT

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.

Insert Attached

CPS

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment; however, single fa..ure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1975 Edition and Addenda through Winter 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 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.

Insert to 3/4.4.6

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10CFR 50.

DRAFT

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	TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	TRANS

~~Add Attached Table~~

/CPS

BASES TABLE 3/4.4.6-1

LIMITING REACTOR VESSEL TOUGHNESS VALUES (CLINTON 1)

I. BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT#-SLAB# OR HEAT#/LOT#	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (°F)	MAX. * ART_{NDT} (°F)	MIN. EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT} (°F)
PLATE	SA-533 GR.B, CL.1	C 4380-2	0.07	0.013	-20	52	84	32
WELD	N/A	76492/ L430B27AE	0.10	0.016	-30	80	75	50

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

I. NON-BELTLINE COMPONENT	MT'L TYPE OR WELD STEAM I.D.	HEAT # - SLAB # OR HEAT #/LOT #	HEAT # - SLAB # HIGHEST STARTING RT_{NDT} (°F)
SHELL RING	SA-533 GR.B CL.1	C4240-2 A2758-1	-10
BOTTOM HEAD DOME	"	A2757-1	-10
BOTTOM HEAD TORUS	"	C4027-1	+10
TOP HEAD DOME	"	C4374-3	-40
TOP HEAD TORUS	"	A2879-2	-10
TOP HEAD FLANGE	SA-508, CL. 2	CCZ 41-5478 SER 915	-40
VESSEL FLANGE	"	CWS 51-5218 SER 878	0
FEEDWATER NOZZLE	"	Q2AL10W	-20

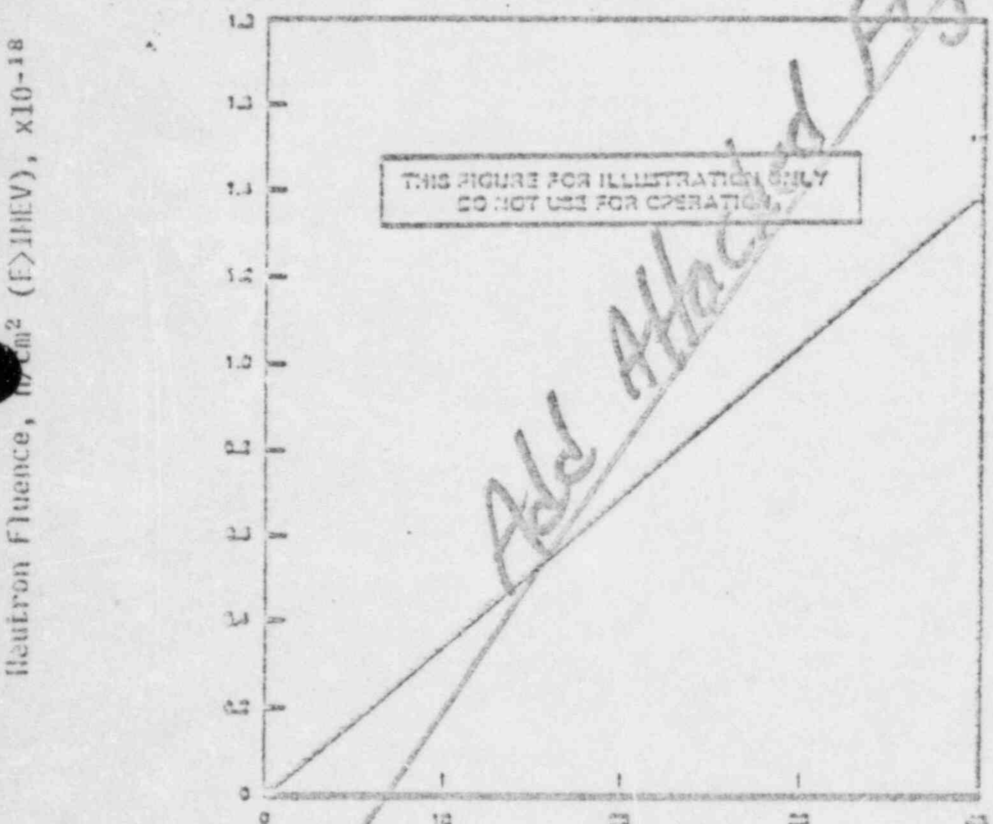
BASES TABLE 3/4.4.6-1 (Continued)

LIMITING REACTOR VESSEL TOUGHNESS VALUES (CLINTON 1)

I. NON-BELTLINE COMPONENT	MT'L TYPE OR WELD STEAM I.D.	HEAT # - SLAB # OR HEAT #/LOT #	HEAT # - SLAB # HIGHEST STARTING RT_{NDT} (°F)
WELD	PER GE PURCH. SPEC.	NO CMV'S AVAILABLE	0 (PER PURCH. SPEC. REQUIREMENTS)
CLOSURE STUDS	SA-540 GR. B23	11312	LST +10

DRAFT

Add Attached Figure

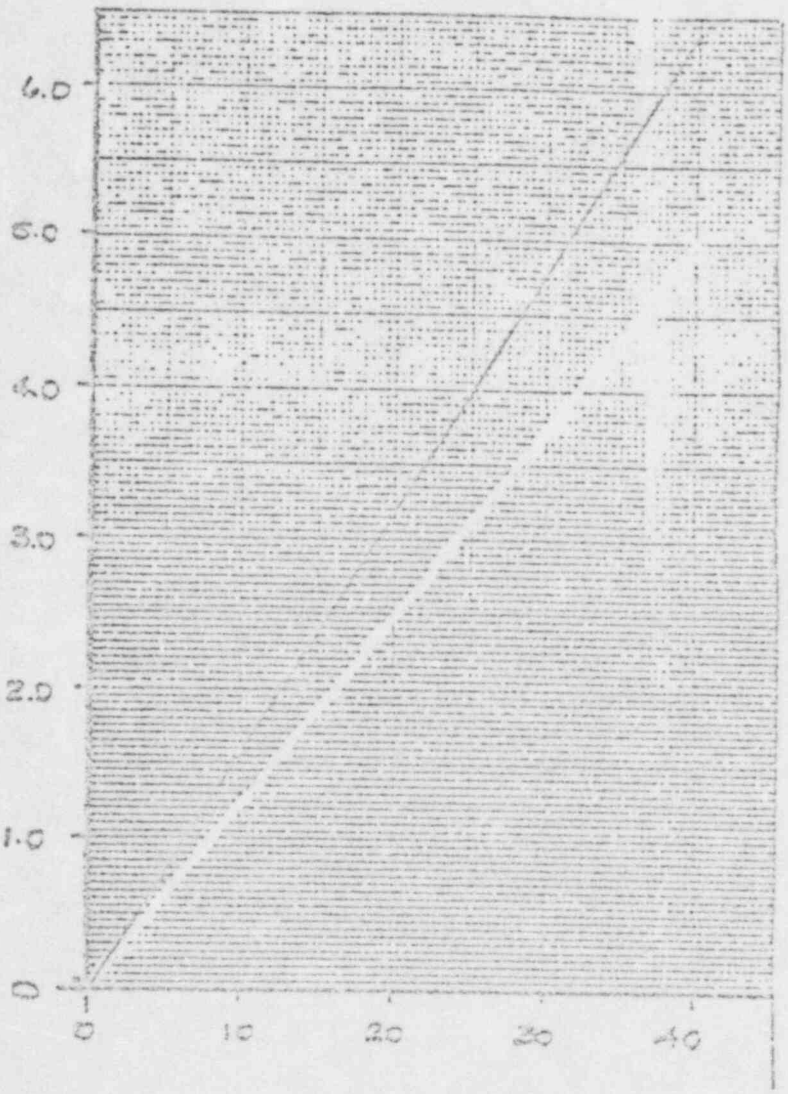


Service Life (Years*)

Fast Neutron Fluence (E>1 Mev) at ¼ T As a Function of Service Life*

Bases Figure B 3/4.4.6-1

* At (90)% of RATED THERMAL POWER and (90)% availability



Service Life (Years*)

Fast Neutron Fluence (E>1 Mev) at ¼ T As a Function of Service Life*

Bases Figure B 3/4.4.6-1**

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* At (90)% of RATED THERMAL POWER and (90)% availability

** Typical Figure; to be modified per specification 4.4.6.1.4.

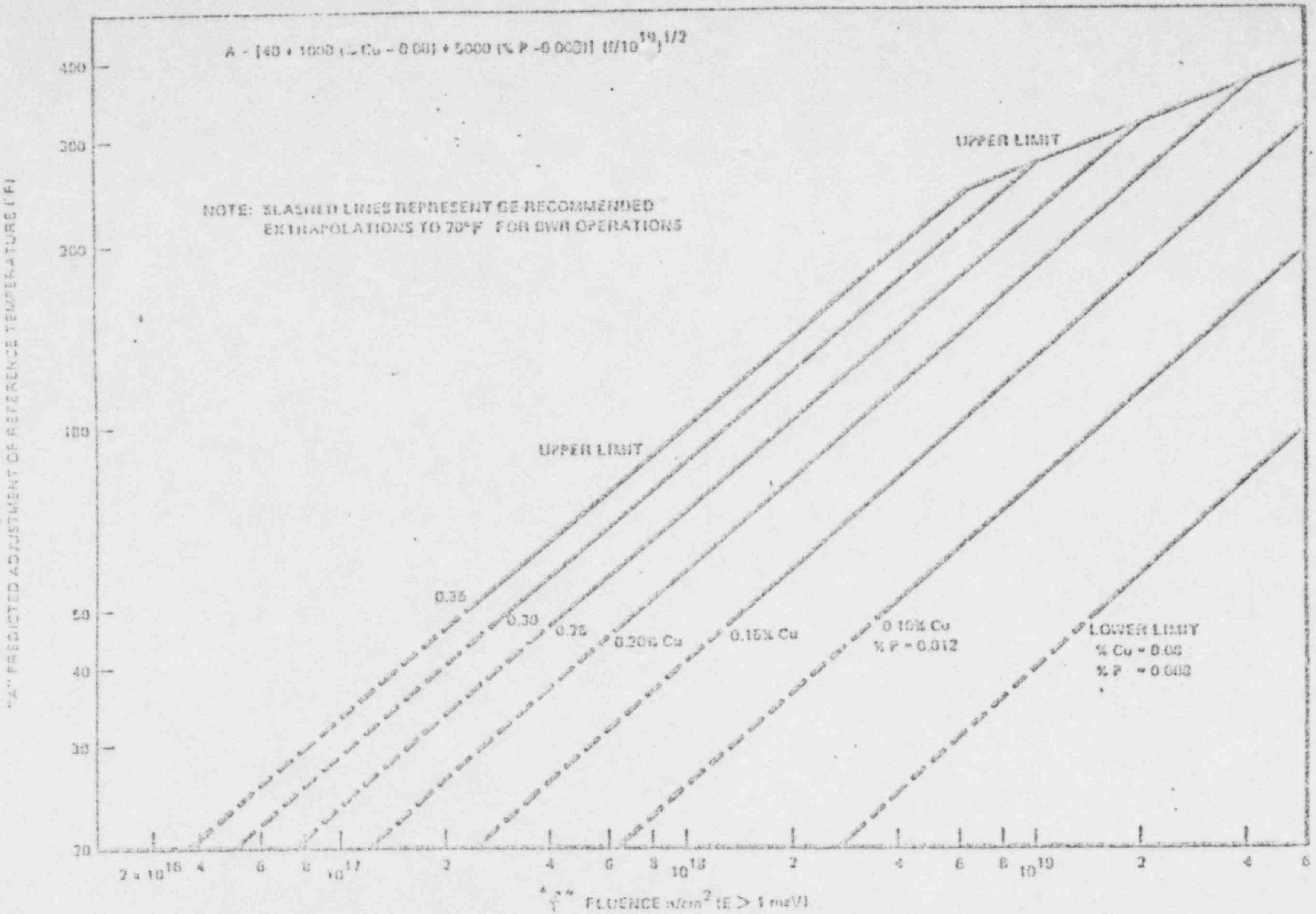

Figure B.3/4.4.6-2. Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature

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 folowing a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

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

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

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

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

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1140 psid, differential pressure between reactor vessel and HPCS suction source, to (0) psid. CPS

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to (467/1400/4900) gpm at differential pressures of 1177/1147/200 psid. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, (but no credit is taken in the safety analyses for the condensate storage tank water). CPS CPS

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ECCS-OPERATING and SHUTDOWN (Continued)

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. |CPS

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.~~ |CPS

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 ~~113~~ 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS. |CPS

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITIONS 4 and 5 the suppression ~~chamber~~ pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention plus a (2' 4") safety margin for conservatism. |CPS

3/4.6 CONTAINMENT SYSTEMS

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BASES

3/4.6.1 CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 9.0 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. CPS

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. CPS

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50, with the exception of exemption(s) granted for main steam isolation valve leak testing and ANSI/ANS 56.8 - 1981, ~~and testing the airlocks after each opening.~~

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the 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.

The main steam lines are serviced by a leakage control system which collects and processes any leakage through the main steam isolation valves. For this reason the leakage testing requirements for these penetrations differ from the requirements placed on other penetrations in 10 CFR 50, Appendix

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main streamline 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 MSIV-LCS motor operated boundary valve 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 MSIV's when isolation of the primary system and containment is required.

3/5.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 15 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 CONTAINMENT INTERNAL PRESSURE

of 3 psig.

design basis

The limitations on containment to secondary containment differential pressure ensure that the containment peak pressure ~~of 9.0 psig~~ does not exceed the design pressure of 15.0 psig during ~~steam line break~~ conditions or that the external pressure differential does not exceed the design maximum external pressure ~~differential of +(0.3) psid (or the differential at which water would overflow the wier wall into the drywell of ____ psid). The limit of -0.1 to +1.5 psid for initial positive containment to secondary containment pressure will limit the containment pressure to (12.0) psid which is less than the design pressure and is consistent with the safety analysis.~~

CRS

3/4.6.1.7 CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 185°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 (DRYWELL AND) CONTAINMENT PURGE SYSTEM

(The (20) inch (drywell and) containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a (LOCA or steam line break accident). Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the (20) inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.)

(The use of the (drywell and) containment purge lines is restricted to the (6) inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the (6) inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the (6) inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations.")

(Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.)

~~3/4.6.1.9 AND 3/4.6.1.10 WATER AND AIR POSITIVE SEAL ISOLATION VALVE LEAKAGE CONTROL SYSTEMS (Optional)~~

~~The OPERABILITY of the water and air positive seal isolation valve leakage control systems 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.)~~

CRS

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by containment spray. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

The limitation on drywell bypass leakage rate is based on having containment spray OPERABLE. It ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the containment exceeding its design pressure of 15.0 psig. The integrated drywell leakage limit ~~value~~ is ~~limited to~~ 10% of the design drywell leakage rate capability to account for uncertainties in test method and conditions and degradations between tests. The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and containment will result in a significant pressure buildup in the containment due to ~~this~~ bypass leakage. CPS

~~3/4.6.2.3 DRYWELL AIR LOCKS~~

~~The limitations on closure for the drywell air locks are required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the drywell.~~ CPS

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak pressure of 18.9 ~~22.0~~ psid does not exceed the design pressure of 30.0 psid. ~~and that the containment peak pressure of 9.0 psig does not exceed the design pressure of 15.0 psig during steam line break conditions. The maximum external drywell pressure differential is limited to +(0.1) psid, well below the (2.3) psid at which suppression pool water will be forced over the wier wall and into the drywell. The limit of (1.5) psid for initial positive drywell to containment pressure will limit the drywell pressure to (22.0) psid which is less than the design pressure and is consistent with the safety analysis.~~ CPS

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330 °F during LOCA ~~(steam line break)~~ conditions and is consistent with the safety analysis.

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BASES

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of (30) psig and (15) psig, respectively, during primary system blowdown from full operating pressure. CPS

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from (1050) psig. Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of (15) psig. Similarly the drywell pressure remains below the design pressure of (30) psig. The maximum and minimum water volumes for the suppression pool are ~~(132,851)~~ 150,230 cubic feet and ~~(136,140)~~ 146,400 cubic feet, respectively. These values include the water volume of the containment pool, horizontal vents, and weir annulus. Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes. CPS

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation without excessive load on the containment pool walls will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

In addition to the limits on temperature of the suppression pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

The (containment) ~~(and) (drywell)~~ spray system consists of two 100% capacity trains, each with three spray rings located at different elevations about the inside circumference of the (containment) ~~(and) (drywell)~~. RHR A pump supplies one train and RHR pump B supplies the other. RHR pump C cannot supply the spray system. Dispersion of the flow of water is effected by 350 nozzles in each train, enhancing the condensaton of water vapor in the (containment) ~~(and) (drywell)~~ volume and preventing overpressurization. Heat rejection is through the RHR heat exchangers. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control. CPS

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F

DEPRESSURIZATION SYSTEMS (Continued)

considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the standby service water is accomplished in the RHR heat exchangers.

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow through two 100% capacity dump lines following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes, ensuring the long term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. The minimum freeboard distance above the suppression pool high water level to the top of the weir wall is adequate to preclude flooding of the drywell in the event of an inadvertent dump. During refueling, neither automatic nor manual action can open the make-up dump valves.

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment (and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50). Containment and drywell isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The operability of the drywell isolation valves ensures that the drywell atmosphere will be directed to the suppression pool for the full spectrum of pipe breaks inside the drywell (and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50). Since the allowable value of drywell leakage is so large, individual drywell penetration leakage is not measured. By checking valve operability on any penetration which could contribute a large fraction of the design leakage, the total leakage is maintained at less than the design value.

DRAFT

BASES

3/4.6.5 DRYWELL TO CONTAINMENT ~~POST-LOCA~~ VACUUM RELIEF VALVES ~~BREAKERS~~

~~Drywell post-LOCA vacuum breakers are provided on the drywell to prevent drywell flooding due to an instrument break anywhere inside containment, and to act in conjunction with the combustible gas control system to provide mixing concentration from exceeding 4% by volume of the drywell following a LOCA.~~ 2

Four vacuum breaker lines, with two valves in series in each line are provided. ~~Any (three) vacuum breaker lines can provide full vacuum relief capability.~~

3/4.6.6 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building and associated structures provides secondary containment during normal operation when the containment is sealed and in service. At other times, the containment may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the Reactor Building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters.

Drywell vacuum relief valves are provided on the drywell to pass sufficient quantities of gas from the containment to the drywell to prevent an excess negative pressure from developing in the drywell.

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BASES

3/4.6.7 ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the containment below its flammable limit during post-LOCA conditions. Either containment ~~and drywell~~ hydrogen recombiner system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. The containment ~~and drywell~~ 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. ICPS ICPS

Two 100% drywell purge systems are the primary means of H_2 control within the drywell, purging hydrogen produced following a LOCA into the containment volume. Hydrogen generated from the metal-water reaction and radiolysis is assumed to evolve to the drywell atmosphere and form a homogenous mixture through natural forces and mechanical turbulence (ECCS pipe break flow). The drywell purge system forces drywell atmosphere through ~~the horizontal vents~~ a sparger in the suppression pool and into the containment and as a result no bypass path exists. ICPS

The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

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3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SHUTDOWN SERVICE WATER SYSTEM

The OPERABILITY of the shutdown service water system 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. CPS

3/4.7.2 CONTROL ROOM VENTILATION SYSTEM

The OPERABILITY of the control room ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds ~~130~~ 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system. CPS

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds ~~130~~ 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

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

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed, nominal time less 25%, may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those snubbers which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression $35\ (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers will require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records ,i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . . . The requirement to monitor the snubber service life is included to enusre that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

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BASES

NO CHANGE

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinkler systems, 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 suppressi n, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight and pressure of the tanks.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

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BASES

3/4.7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of ( )°F.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the (feedwater controller failure) analysis in FSAR Chapter 15. CPS

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least Division I or II of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source. Division III supplies the high pressure core spray (HPCS) system only.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator 1A or 1B is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator 1A or 1B as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period diesel generator 1A or 1B is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, 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.

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BASES

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."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's 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.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by ~~either de-energizing circuits not required during reactor operation or~~ demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers or fuses by periodic surveillance. (INSERT)

CPS

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are than tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The ~~(OPERABILITY) (or)~~ (bypassing) of the motor operated valves thermal overload protection (continuously) ~~(or) (during accident conditions) (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 ~~(OPERABILITY) (or)~~ (bypassing) of the thermal overload protection (continuously) ~~(and) (or) (during accident conditions)~~ are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

CP

(INSERT)

The design used at Clinton provides two (2) qualified devices in series for all penetrations. Each qualified device is sized to protect the penetration.

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3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality 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 FUEL HANDLING EQUIPMENT

The OPERABILITY requirements ensure that the appropriate fuel handling equipment will be used for handling control rods and fuel assemblies during operations associated with the IFTS and the spent fuel storage pool and within the reactor pressure vessel, that each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and that the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. CPS

/or

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BASES

No CHANGE

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the pools ensures that in the event this load is dropped 1) the activity release will be limited to that contained in 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 AND UPPER CONTAINMENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the 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 and in operation or that an altenate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

The purpose of the inclined fuel transfer system specification is to control personnel access to those potentially high radiation areas immediated adjacent to the system and to assure safe operation of the system.

DRAFT

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

No Change

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

The requirements for PRIMARY CONTAINMENT INTEGRITY and DRYWELL INTEGRITY are not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD PATTERN CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limits for dissolved or entrained noble gases were determined by converting their MPC's in air to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

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 will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which 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 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 consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

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 WASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 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.

> Insert Attached

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. The annual dose limits are the dose associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC 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)(1)). For individuals who may at times be within the SITE BOUNDARY, the occupancy of the individual will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems year.

This specification applies to the release of gaseous effluents from all reactors at the site.

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 Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

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.

Insert
B 3/4 11-2 A

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 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 established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - IODINE-131, 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 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 methods 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,

DOSE-IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

tritium and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in areas at and beyond the SITE BOUNDARY. The pathways which were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 and 3/4.11.2.5 GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

~~3/4.11.2.7 MAIN CONDENSER~~

~~Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion~~ area boundary ~~will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.~~ CPS

~~3/4.11.2.8 VENTING OR PURGING~~

~~This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas.~~ CPS

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. 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, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 CFR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. 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 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12.1 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 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 modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually); Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census. The best information from the door-to-door survey, aerial survey or consulting with local agricultural authorities or any combination of these methods 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 500 m^2 | CPS ~~square feet~~ provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine

BASES

LAND USE CENSUS (Continued)

this minimum garden size, the following assumptions were used: (1) that 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/square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an 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 reasonably valid for the purpose of Section IV.B.2 of Appendix I to 10 CFR Part 50.

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SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

INSERT 5.1.3 |CPS

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment is a (steel lined, reinforced concrete structure composed of a vertical right cylinder and a hemispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head which contains an (approximately) 20 feet deep water filled suppression pool connected to the drywell through a series of horizontal vents. The containment has a minimum net free air volume of (1,400,000) cubic feet. The drywell has a minimum net free air volume of (270,000) cubic feet.) |CPS |CPS |CPS

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell (30) psig. |CPS
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

(c. Maximum external to internal differential pressure:
   1. Drywell (21) psid. |CPS
   2. Containment 3.0 psid.) |CPS

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Fuel Building, the ECCS pump rooms and the containment gas control boundary, including extension, and has a minimum free volume of 2,000,000 cubic feet.

Insert 5.1.3

UNRESTRICTED AREA BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The gaseous effluent and liquid effluent release points are shown in Figure 5.1.3-1.

Replace with attached figure | CPS

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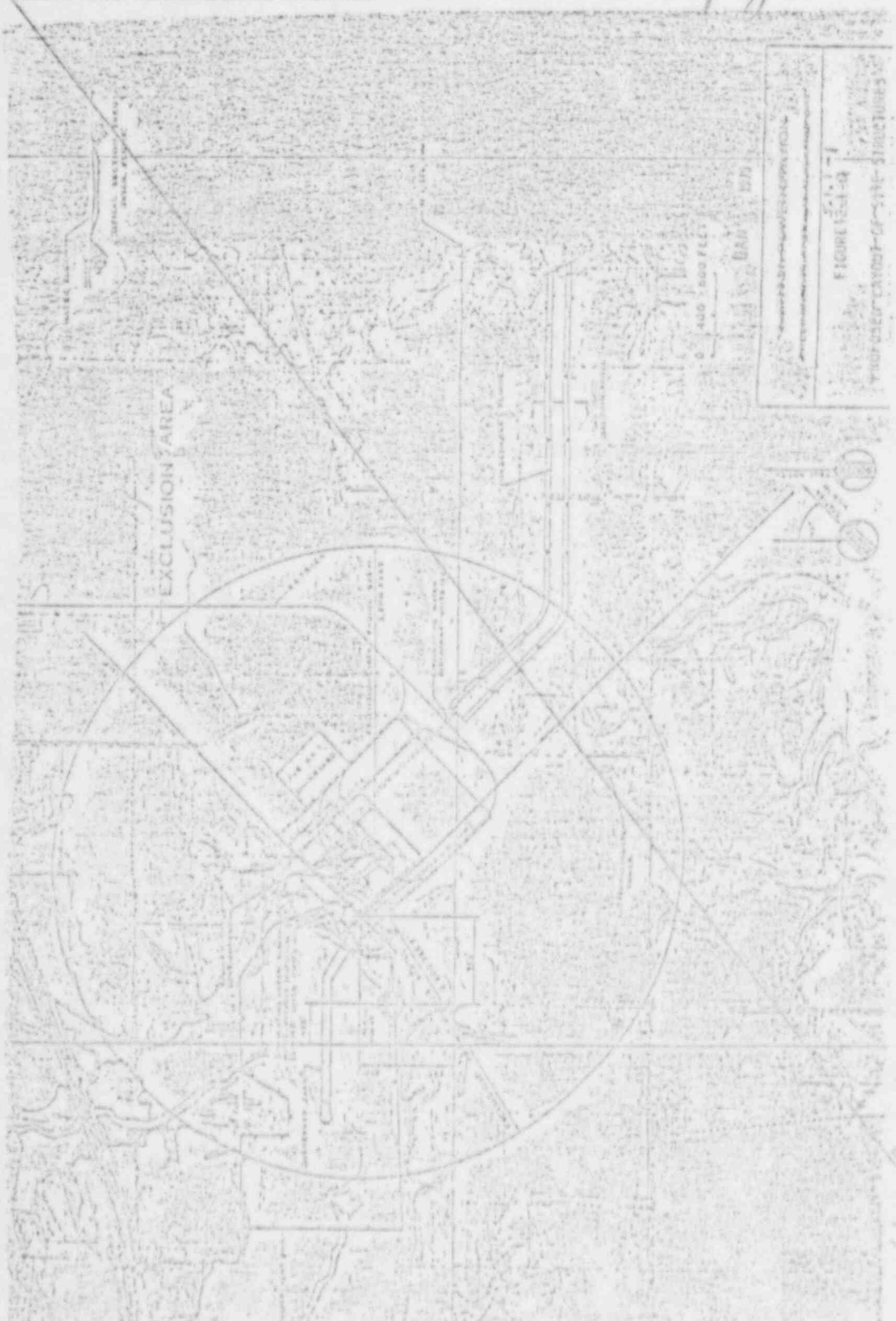
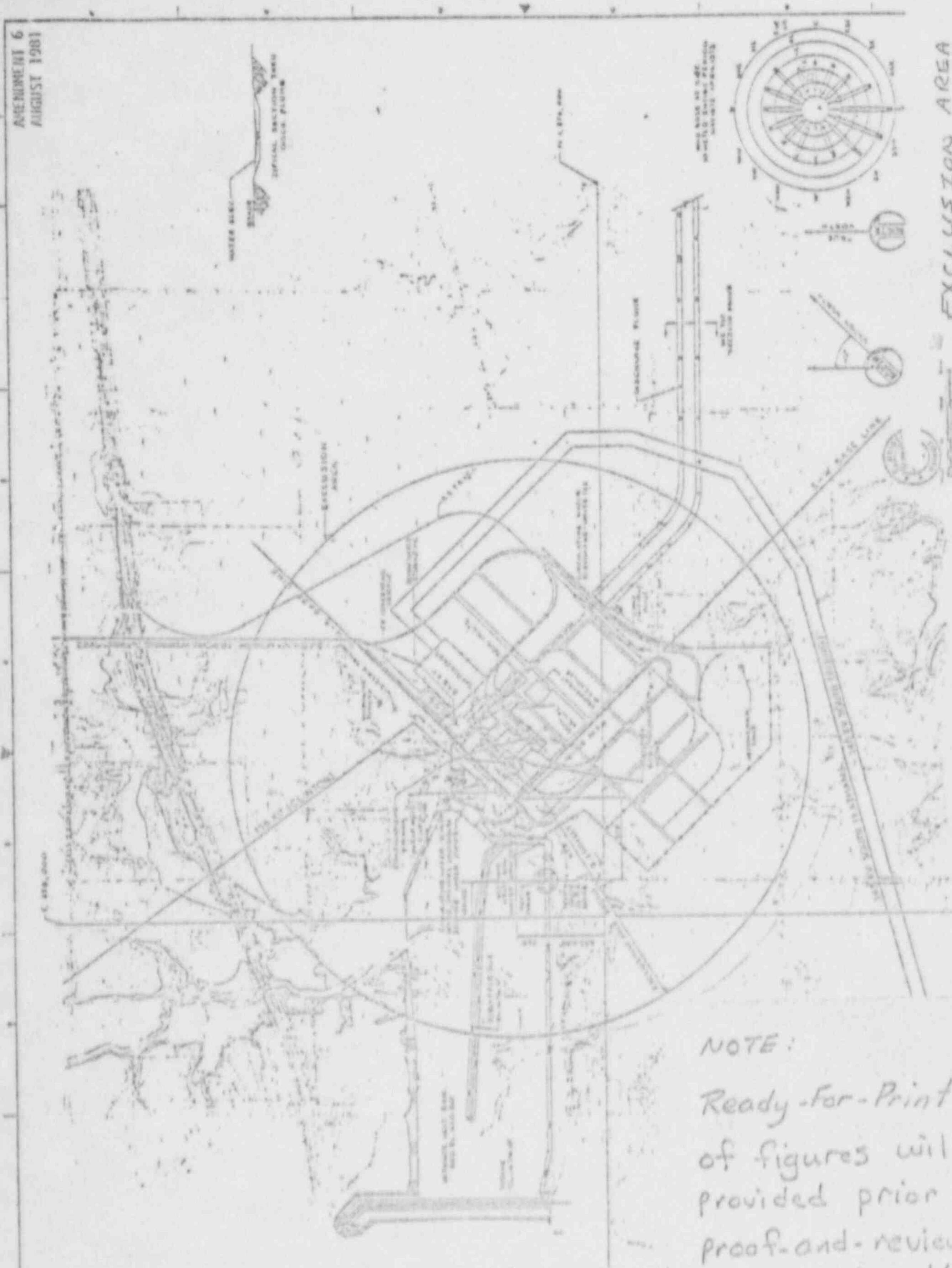


EXCLUSION AREA

FIGURE 5.1.1-1

Need better figure



EXCLUSION AREA

Figure 5.1.1-1

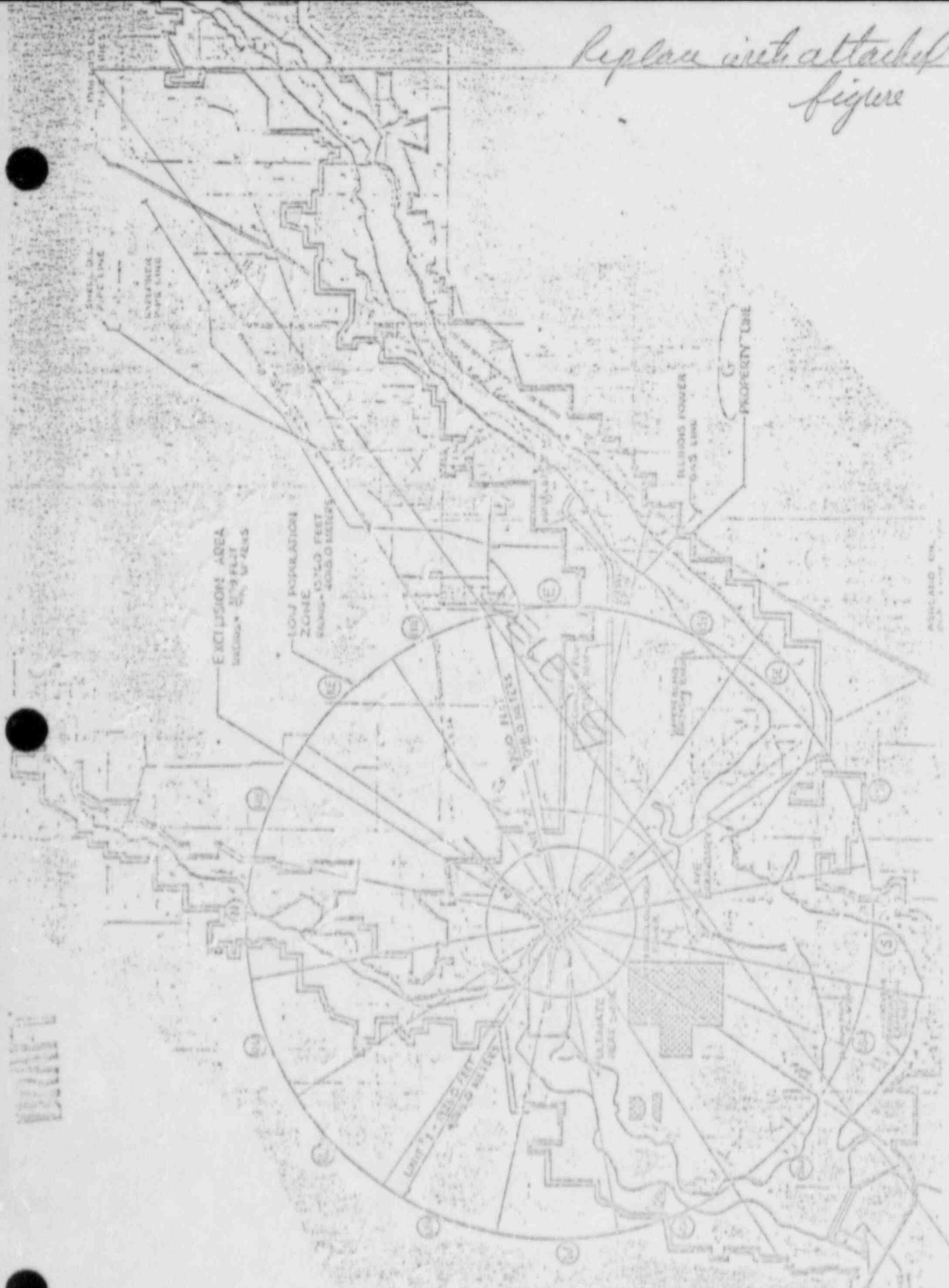
NOTE:

Ready-for-Print copy of figures will be provided prior to proof-and-review or no later than 3 months prior to fuel load.

Insert 5-2A

CLINTON-1

Replace with attached figure | CB



LOW POPULATION ZONE

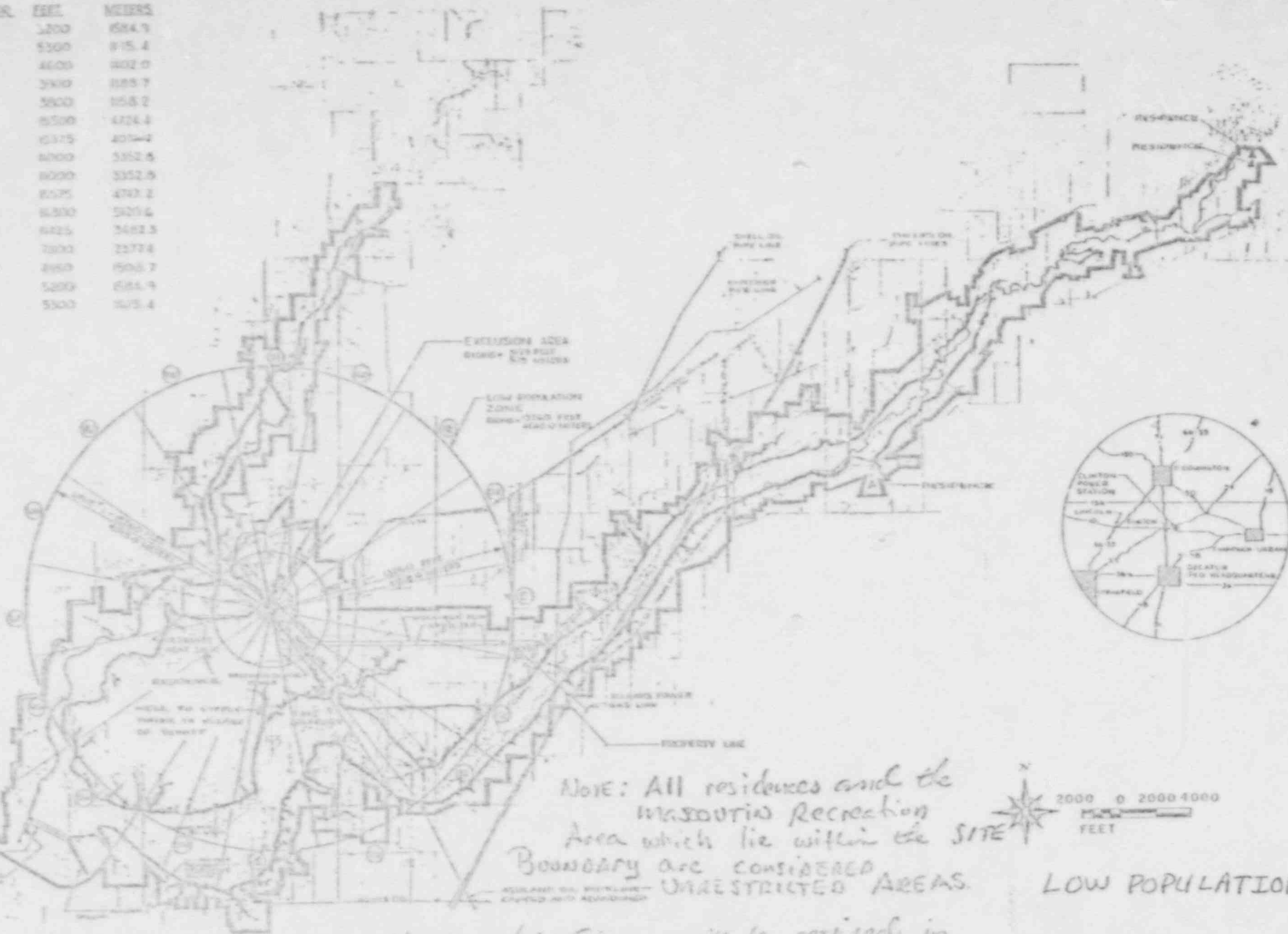
FIGURE 5.1.2-1

CLINTON-1

DISTANCE TO NEA[illegible] SITE BOUNDARY WITHIN SECTOR

SECTOR	FEET	METERS
N	5200	1584.9
NNE	5300	1615.4
NE	4600	1402.0
ENE	3900	1188.7
E	3800	1158.2
ESE	15500	4724.4
SE	13375	4076.7
SSE	11000	3352.8
S	11000	3352.8
SSW	15575	4747.2
SW	16800	5120.6
WSW	11425	3482.3
W	7800	2377.4
WNW	4940	1505.7
NW	5200	1584.9
NNW	5300	1615.4

Insert 5-3A



NOTE: All residences and the Mascoutin Recreation Area which lie within the SITE Boundary are considered UNRESTRICTED AREAS.

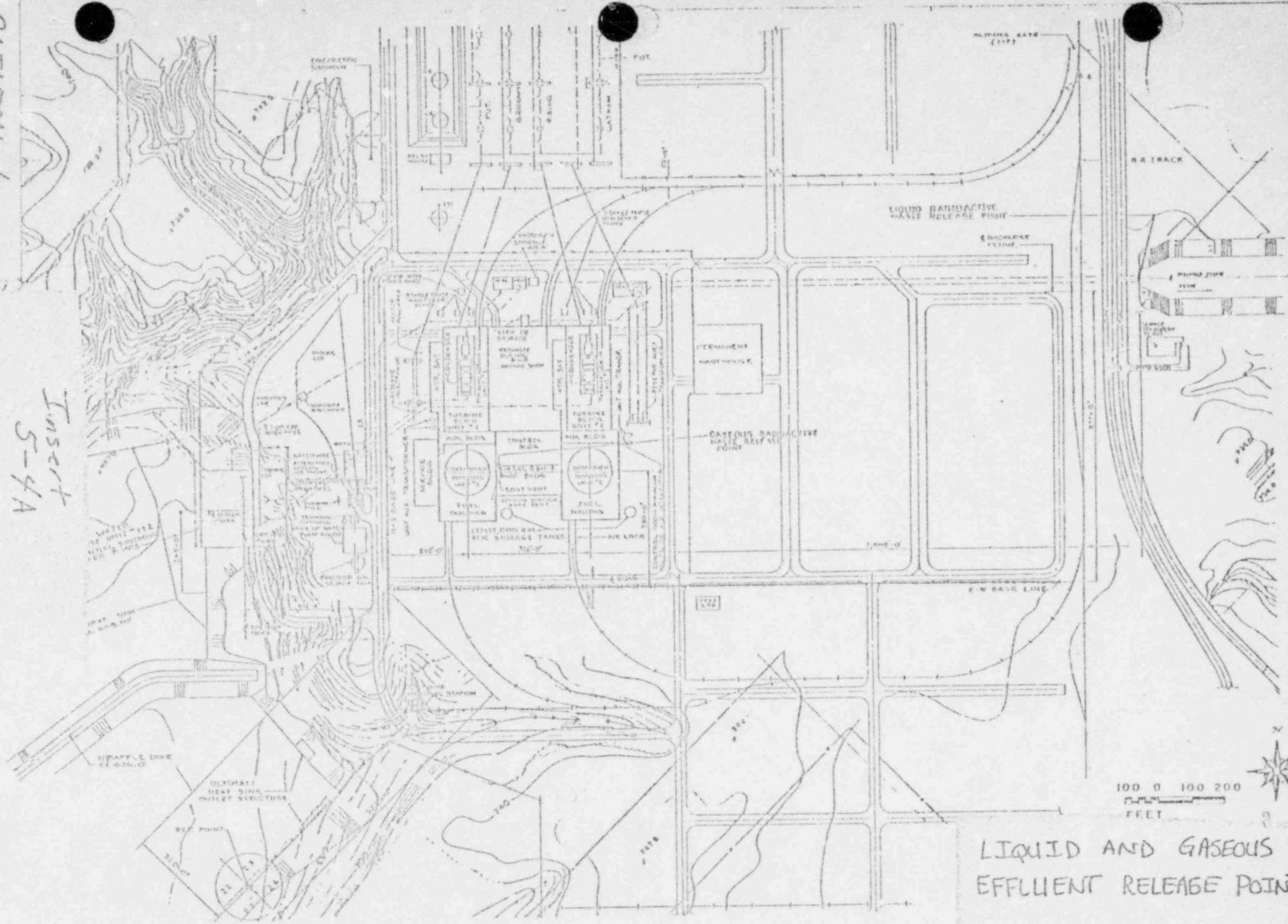
NOTE: this Figure will be revised in accordance with the methodology to be determined within the ODCM.

LOW POPULATION ZONE

Figure 5.1.2-1

CLINTON-1

Insert 5-4A



LIQUID AND GASEOUS EFFLUENT RELEASE POINT

Figure 5.1.3-1

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 624 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a ~~maximum~~ core average enrichment of ~~2.12~~ 1.706 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of ( ) weight percent U-235. |CPS

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 145 control rod assemblies, each consisting of a cruciform array of stainless steel tubes, containing ~~144~~ 143.70 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath. |CPS

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of ~~(575)~~ °F. |CPS

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 16,000 cubic feet at a nominal steam dome saturation temperature of 54~~0~~9°F. |CPS

 |CPS

DRAFT

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, (which includes a conservative allowance of (2.6)% delta k/k for uncertainties) (including all calculational uncertainties and biases) as described in Section 9.1 of the FSAR.

b. A nominal 6.687 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooding is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 754'0".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2512 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.

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TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	80 step change cycles	Loss of ~~(all)~~ feedwater heaters
	(180) reactor trip cycles	(100)% to (0)% of RATED THERMAL POWER
	(40) hydrostatic pressure or leak tests	Pressurized to ≥ (930) psig and ≤ (1250) psig

CPS

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SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Power Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President ( ) shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room.

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Protection Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

ADMINISTRATIVE CONTROLS

UNIT STAFF (continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel). CPS

~~[The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).]~~ CPS

[Adequate shift coverage shall be maintained without routine heavy use of overtime. ~~The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating.~~ However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Power Plant Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Power Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.] CPS

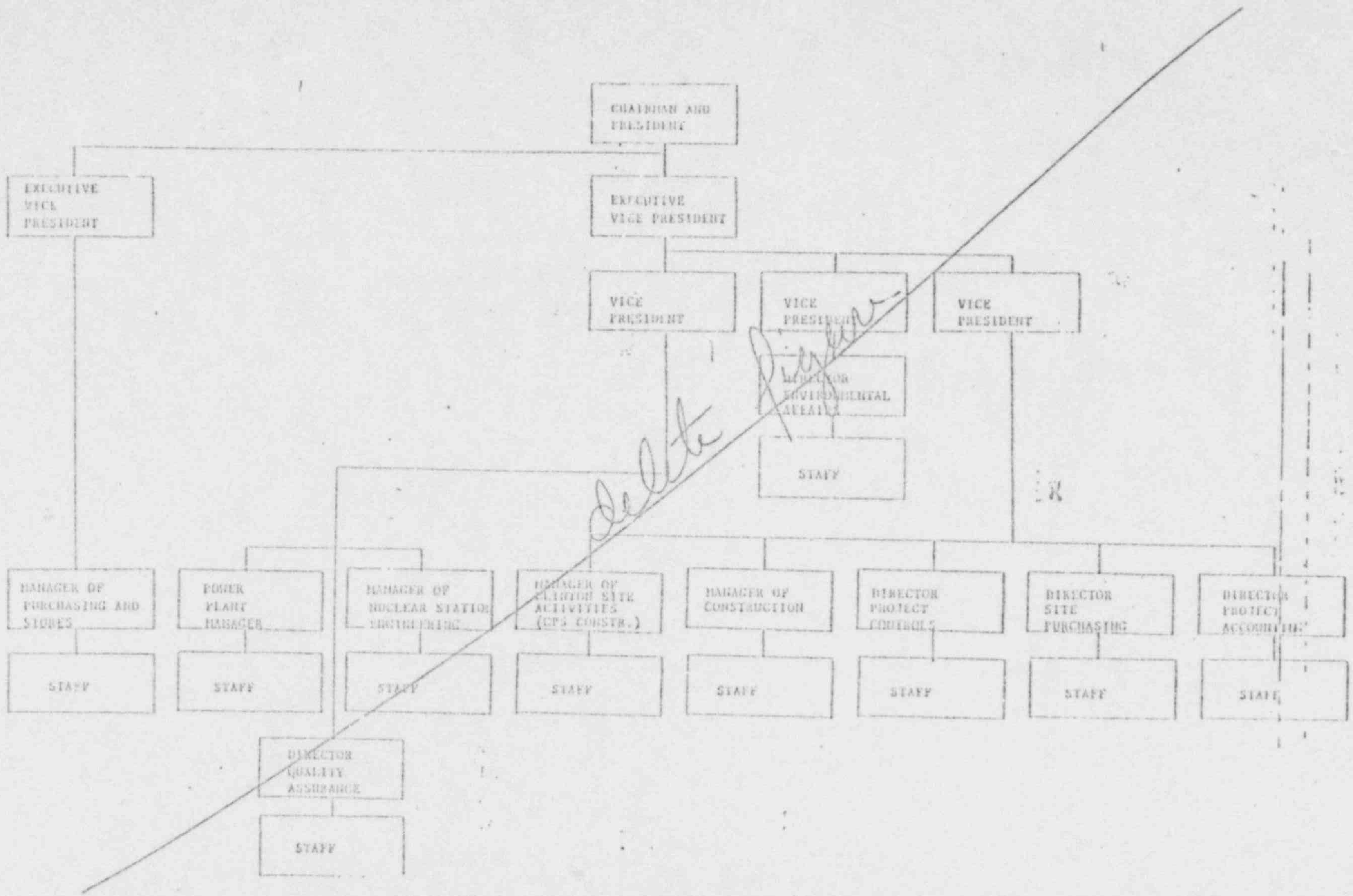

FIGURE 6.2.1-1

OFFSITE ORGANIZATION

Please refer to CPS-FSAR
Figures 13.1-1A, 13.1-1B
and 13.1-2 until proof-
and-review stage of CPS-TS.

OFFSITE ORGANIZATION

Figure 6.2.1-1

Please refer to CPS-FSAR Figure 13.1-3 until proof-and-review stage of CPS-TS

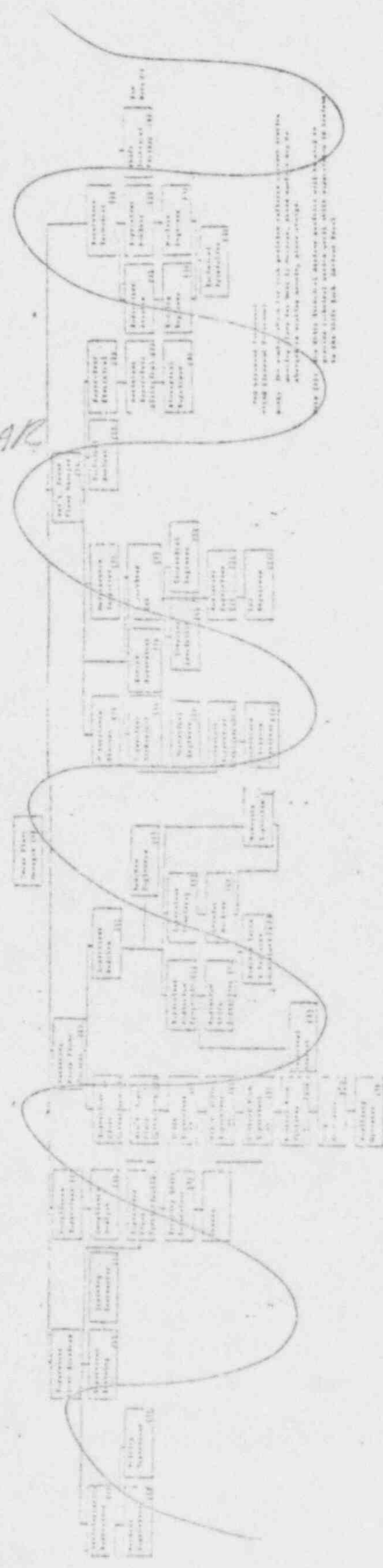

FIGURE 6.2.2-1

UNIT ORGANIZATION

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Operator license on Unit 1.
SRO - Individual with a Senior Operator license on Unit 1.
RO - Individual with a Operator license on Unit 1.
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the ~~Executive Vice President.~~ Director - Nuclear Safety and Engineering Analysis, Manager - Nuclear Station Engineering, and Members of the NRAG.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least ~~2~~ 3 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field. CPS

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical. CPS

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to ~~(a high level corporate official in a technically oriented position who is not in the management chain for power production).~~ the Director - Nuclear Safety and Engineering Analysis, Manager - Nuclear Station Engineering and Members of the NRAG. CPS

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 except for the Rad Chem Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Supervisor of Training, shall meet or exceed the requirements and recommendations of Section 5.5 of ~~(an~~ ANSI Standard N18.1-1971 ~~acceptable to the NRC staff)~~ and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience. CPS

6.5 REVIEW AND AUDIT

6.5.1 FACILITY REVIEW GROUP (FRG)

FUNCTION

6.5.1.1 The FRG shall function to advise the Power Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The FRG shall be composed of the:

Chairman:	Assistant Power Plant Manager - Operations
Member:	Assistant-Power Plant Manager - Maintenance
Member:	Supervisor-Plant Operations
Member:	Supervisor-Technical
Member:	Supervisor-C&I
Member:	Supervisor-Rad ~~Chem~~ & Protection
~~Member:~~	~~Supervisor-Radiation Protection~~
Member:	Supervisor-Nuclear

CPS

CPS

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the FRG Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in FRG activities at any one time.

MEETING FREQUENCY

6.5.1.4 The FRG shall meet at least once per calendar month and as convened by the FRG Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRG necessary for the performance of the FRG responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The FRG shall be responsible for:

a. Review of ~~(1)~~ all administrative ~~proposed~~ procedures ~~required by Specification 6.8~~ and changes thereto, ~~(2) all proposed programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Power Plant Manager to affect nuclear safety;~~ CPS

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Executive Vice President and to the Nuclear Review and Audit Group;

f. Insert Attached

g ~~f.~~ Review of events requiring 24-hour written notification to the Commission;

h ~~g.~~ Review of unit operations to detect potential hazards to nuclear safety;

i ~~h.~~ Performance of special reviews, investigations, or analyses and reports thereon as requested by the Power Plant Manager or the Nuclear Review and Audit Group;

j ~~i.~~ Review of the Security Plan and implementing procedures and submittal of recommended changes to the Nuclear Review and Audit Group; and

k ~~j.~~ Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Nuclear Review and Audit Group.

l. m. } Insert Attached

CPS

6.5.1.7 The FRG shall:

a. Recommend in writing to the Power Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a through d prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a through ~~e~~ fe constitutes an unreviewed safety question. CPS

c. Provide written notification within 24 hours to the Executive Vice President and the Nuclear Review and Audit Group of disagreement between the FRG and the Power Plant Manager; however, the Power Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

Insert 6.5.1.6

f. Review of the safety evaluations for procedures, tests, and experiments, completed under the provisions of 10 CFR 50.59.

l. 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 the reports to the Vice President (NUCLEAR) and to the Nuclear Review and Audit Group (NRAG).

m. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

RECORDS

6.5.1.6 The FRG shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the ~~Executive~~ Vice President (NUCLEAR) and the Nuclear Review and Audit Group.

6.5.2 NUCLEAR REVIEW AND AUDIT GROUP (NRAG)

FUNCTION

6.5.2.1 The NRAG shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NRAG shall report to and advise the ~~Executive~~ Vice President (NUCLEAR) on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The NRAG shall be composed of the:

Director:	Executive Vice President
Member:	Manager-Nuclear Station Engineering Dept.
Member:	Power Plant Manager
Member:	Director-Design Engineering
Member:	Director-Nuclear Safety and Engineering Analysis
Member:	Supervisor-Technology Assessment
Member:	Supervisor-Compliance (NSED)

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NRAG Director to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NRAG activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NRAG Director to provide expert advice to the NRAG.

MEETING FREQUENCY

6.5.2.5 The NRAG shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NRAG necessary for the performance of the NRAG review and audit functions of these Technical Specifications shall consist of the Director or his designated alternate and at least four NRAG members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NRAG shall review:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. ALL REPORTABLE EVENTS ~~Events requiring 24-hour written notification to the Commission;~~ CPS

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the FRG.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the NRAG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire unit staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e, f Insert Attached

~~e.~~ h The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

i-m Insert Attached

~~f. The fire protection equipment and program implementation, and audit shall be performed at least once per 12 months utilizing either (a) qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least once every third year; and~~

g. Any other area of unit operation considered appropriate by the NRAG or the Executive Vice President.

AUTHORITY 6.5.2.9 Insert Attached

RECORDS

6.5.2.~~9~~ 10 Records of NRAG activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NRAG meeting shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

Insert 6.5.2.8

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedurer at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, December, 1977, at least once per 12 months.

AUTHORITY ,NRHG

6.5.2.9 The SRC shall report to and advise the ~~Senior~~ Sr Vice President, (Nuclear) on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

Insert
6-11A

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

Procedures and programs required by Technical Specification 6.8 and other procedures which affect plant nuclear safety as determined by the *Power Plant Manager*, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows:

6.5.3.1 Technical Review

a. Each such procedure, program or procedure change shall be independently reviewed by an individual knowledgeable in the area affected other than the individual who prepared the procedure, program or procedure change. The Power Plant Manager shall approve all plant procedures, programs, and changes thereto.

b. Individuals responsible for reviews performed in accordance with item 6.5.3.1a. above shall be members of the plant staff previously designated by the Power Plant Manager. 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 review personnel of the appropriate discipline.

Individuals performing these reviews shall meet or exceed the qualifications stated in section 4.4 of ANSI N18.1-1971 for the appropriate discipline.

c. When required by 10 CFR 50.59, a safety evaluation to determine whether or not an unreviewed safety question is involved shall be included in the procedure review. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions shall be obtained prior to the Power Plant Manager approval for implementation.

d. Written records of reviews performed in accordance with item 6.5.3.1a. above, including recommendations for approval or disapproval, shall be prepared and maintained.

All items not reviewed in accordance with item 6.5.3.1a. above shall be reviewed by PORC.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the FRG and submitted to the NRAG and the Vice President (Nuclear).

CPS

~~6.6 REPORTABLE OCCURRENCE ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:~~

~~a. The Commission shall be notified and a report submitted pursuant to the requirements of Specification 6.9, and~~

~~b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the FRG, and the results of this submitted to the NRAG and the Executive Vice President.~~

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President and the NRAG shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRG. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRAG, and the Executive Vice President within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

~~b. The applicable procedures required to implement the requirements of NUREG-0737.~~ CPS

b ~~c.~~ Refueling operations.

c ~~d.~~ Surveillance and test activities of safety-related equipment.

d ~~e.~~ Security Plan implementation.

e ~~f.~~ Emergency Plan implementation.

f ~~g.~~ Fire Protection Program implementation.

g-i. Insert Attached

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed ~~by the FRG~~ in accordance with 6.5.1.6 or 6.5.3 as appropriate and shall be approved by the Power Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed ~~by the FRG~~ in accordance with 6.5.1.6 or 6.5.3 as appropriate, and approved by the Power Plant Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the LPCS, HPCS, RHR, RCIC pump side and turbine exhaust, suppression pool cleanup and return, and shutdown service water supply and return. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

6.8.1 INSERT

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sample and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS ~~AND REPORTABLE OCCURRENCES~~

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

ADMINISTRATIVE CONTROLS

STARTUP REPORT (Continued)

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refuelings). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

6.9.1.6 Insert Attached

~~(b. Documentation of all challenges to main steam line safety/relief valves.)~~

6.9.1.7 Insert Attached

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience (, including documentation of all challenges to the main steam system safety/relief valves,) shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report. Insert Attached

~~REPORTABLE OCCURRENCES~~

~~6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a Licensee Event Report shall be completed and reference shall be made to the original report date.~~

~~*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

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6.9.1.6 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 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

~~*A single submittal may be made for a multiple unit station.~~

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**One map shall cover stations near the site boundary; a second shall include the more distant stations.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT* 9

6.9.1.~~11~~ 7 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 with the date of initial criticality.

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 due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall also include once a year 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

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~~*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~

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**In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

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. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

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Insert 6.9.1.8

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days of when the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the FRG.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office of the NRC or his designee no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the Limiting Safety System Setting in the Technical Specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a Limiting Condition for Operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the Safety Analysis Report.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the Safety Analysis Report.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

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PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Failure of main steam line safety/relief valves.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Regional Administrator of the Regional Office of the NRC within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

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a. Reactor protection system or engineered safety features instrumentation settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in Specification 6.9.1.8c designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE ~~OCCURRENCES submitted to the Commission~~ EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

RECORD RETENTION (Continued)

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the FRG and the NRAG.

l. Records of the service lives of all ~~hydraulic and mechanical~~ snubbers ~~listed on Table(s) 3.7.5-1 (and 3.7.5-2)~~ including the date at which the service life commences and associated installation and maintenance records required by Specification 3/4.7.4.

m. Insert Attached

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

Insert 6.10.3

m. Records of analysis required by the radiological environmental monitoring program.

HIGH RADIATION AREA (Continued)

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Radiation Protection Supervisor. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

*Measurements made at 18 inches from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

a ~~1.~~ Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1 ~~a.~~ Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

2 ~~b.~~ A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3 ~~c.~~ Documentation of the fact that the change has been reviewed and found acceptable by the ~~(URG)~~ FRG.

b ~~2.~~ Shall become effective upon review and acceptance by the ~~(URG)~~ FRG.

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6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

a ~~1.~~ Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1 ~~a.~~ Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2 ~~b.~~ A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3 ~~c.~~ Documentation of the fact that the change has been reviewed and found acceptable by the ~~(URG)~~ FRG.

b ~~2.~~ Shall become effective upon review and acceptance by the ~~(URG)~~ FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS~~*~~

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

a ~~1~~. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~(Unit Review Group)~~ FRG. The discussion of each change shall contain:

1 ~~a~~. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

2 ~~b~~. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3 ~~c~~. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

4 ~~d~~. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5 ~~e~~. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6 ~~f~~. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7 ~~g~~. An estimate of the exposure to plant operating personnel as a result of the change; and

8 ~~h~~. Documentation of the fact that the change was reviewed and found acceptable by the ~~(URG)~~ FRG.

b ~~2~~. Shall become effective upon review and acceptance by the ~~(URG)~~ FRG.

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~~*Licensee may choose to submit the information called for in this Specification as part of the annual FSAR update.~~