FEB 06 1985

Docket No.: 50-387

Mr. Norman W. Curtis
Vice President
Engineering and Construction - Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentwon, Pennsylvania 18101

Dear Mr. Curtis:

SUBJECT: AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-14, SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. The amendment is in response to your letters dated May 18, 1984, and September 20, 1984. This amendment revises the Unit 1 Technical Specifications to reflect the Susquehanna Unit 2 Technical Specifications. These changes are administrative in nature.

A copy of the related safety evaluation supporting Amendment No. 29 to Facility Operating License NPF-14 is enclosed.

Sincerely,

Original signed by:

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

Enclosures:
1. Amendment No. 29 to NPF-14
2. Safety Evaluation
3. Federal Register Notice

See Licence Amdt Jacket.

cc w/enclosures:
See next page

DL:LB#2	DL:LB#2	OELD	DL:LB#2	AD/L:DL
EHylton	MCampagnone:pob	JGray	ASchwencer	TNovak
01/23/85	01/23/85	01/24/85	01/25/85	01/25/85

with noted changes to amtd. & post notice

8502110600 850206
PDR ADOCK 05000387
P PDR

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: FEB 06 1985

Previous concurrences concurred on by*:

With noted change above to
amend & to post notice. (JGray)

DL:LB#2	DL:LB#2	DL:LB#2	OELD	DL:AD/L
*MJCampagnone:pob	*EHylton	*ASchwencer	*JGray	TNovak
01/23/85	01/23/85	01/23/85	01/24/85	01/26/85

3. This amendment is effective as of ~~thirty (30) days from~~ the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance:

Spelling
Change

DL:LB#2	DL:LB#2	DL:LB#2	OELD	DL:AD/L
MJCampagnone:pob	EHylton	ASchwencer	J.GRAY	JNovak
01/[illegible]/85	01/23/85	01/23/85	01/24/85	01/[illegible]/85

with noted change above to amend.

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

FEB 06 1985

Docket No.: 50-387

Mr. Norman W. Curtis
Vice President
Engineering and Construction - Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentwon, Pennsylvania 18101

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

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3. Federal Register Notice

cc w/enclosures:
See next page

Susquehanna

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U. S. Environmental Protection Agency
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Curtis Building
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

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

PENNSYLVANIA POWER & LIGHT COMPANY
ALLEGHENY ELECTRIC COOPERATIVE, INC.
DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated May 18, 1984, as supplemented by September 20, 1984, letter complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 29 the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facilities in accordance with the Technical Specifications and the Environmental Protection Plan.

8502110604 850206
PDR ADOCK 05000387
P PDR

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: FEB 06 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 29
FACILITY OPERATING LICENSE NO. NPF-14
DOCKET NO. 50-387

Replace the following pages of the Appendix "A" Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of changes.

REMOVE	INSERT
i	i
ii	ii
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viii	viii
ix	ix
x	x
xvii	xvii
xviii	xviii
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PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

1.30 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

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

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

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

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

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

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

1.32 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.

RATED THERMAL POWER

1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

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

REPORTABLE EVENT

1.35 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50. 26

ROD DENSITY

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

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

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

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

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

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

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

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL CONDITIONS

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

#The reactor mode switch may be placed in the 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.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
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*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.

2.0 INTRODUCTION

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

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 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 x 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.

2.1.2 THERMAL POWER, High Pressure and High Flow

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

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

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

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

---

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

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

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

MITING 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 rods in all directions.

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

         1) Electrically, or

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

      c) Comply with Surveillance Requirement 4.1.1.c.

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

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

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

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

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

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

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

      a) Electrically, or

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

---

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

   a) Electrically, or

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

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

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

a. At least once per 7 days, and

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

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

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

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

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

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

   a) Restore the inoperable accumulator to OPERABLE status, or

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

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

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

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

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

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

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

b. In OPERATIONAL CONDITION 5*:

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

   a) Electrically, or

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

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

*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

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

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

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

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

2. Verifying that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point for greater than or equal to 10 minutes with no control rod drive pump operating.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

   1. Determine the position of the control rod by:

      a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,

      b) Returning the control rod, by single notch movement, to its original position, and

      c) Verifying no control rod drift alarm at least once per 12 hours, or

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

   3. When THERMAL POWER is:

      a) Within the low power setpoint of the RSCS:

         1) Declare the control rod inoperable, and

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

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

         1) Electrically, or

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

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

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

---

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

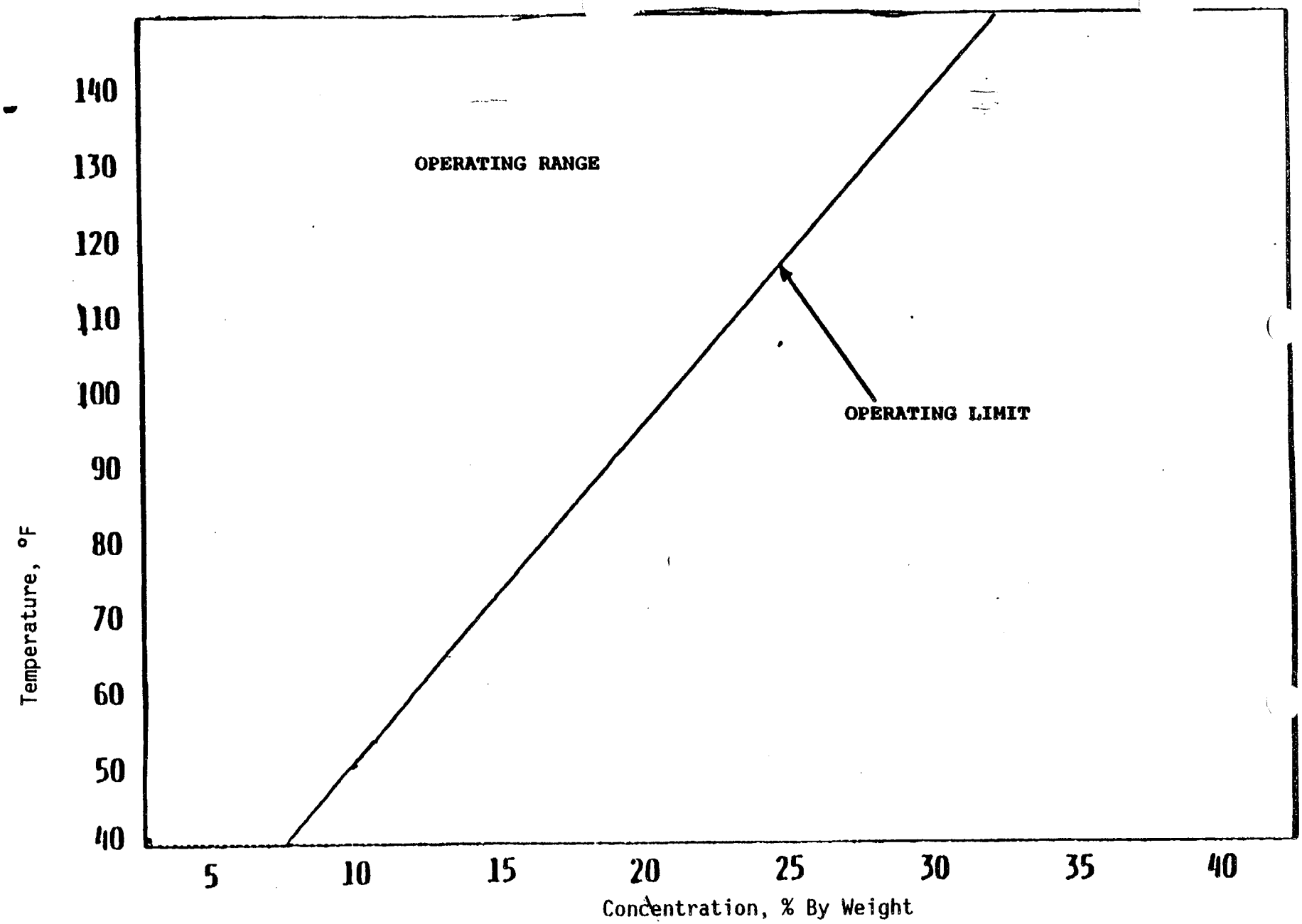
SURVEILLANCE REQUIREMENTS

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

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

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

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



SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

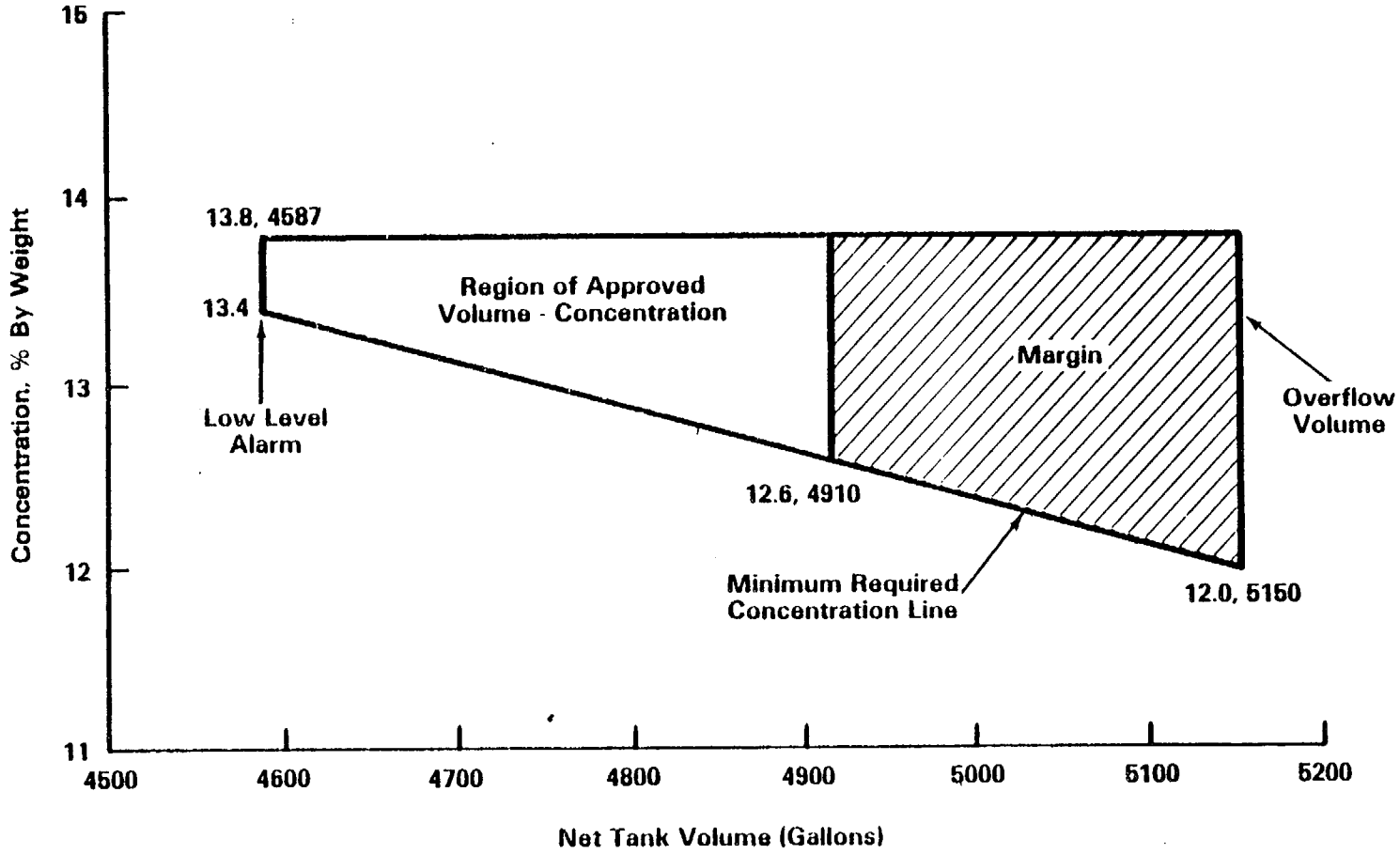


FIGURE 3.1.5-2

SODIUM PENTABORATE

SOLUTION CONCENTRATION

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

d. The provisions of Specification 4.0.4 are not applicable.

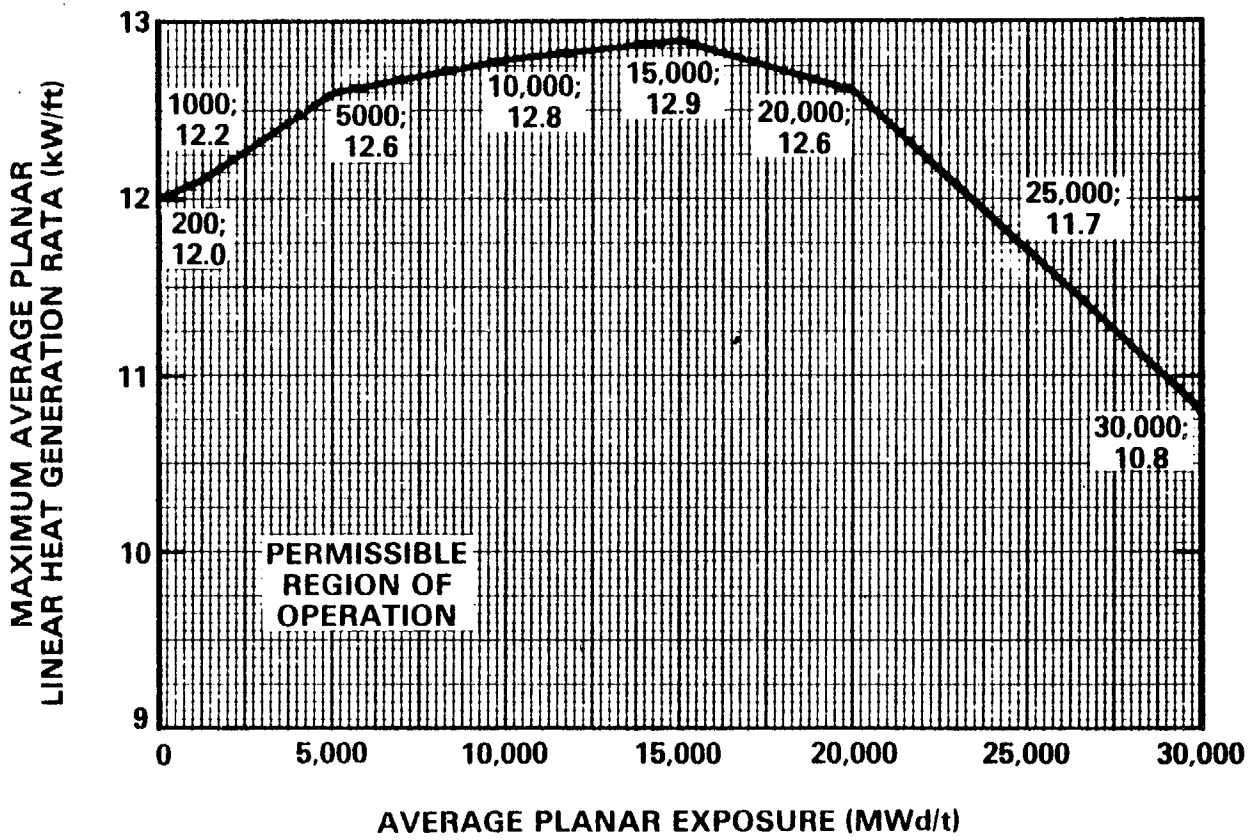


FIGURE 3.2.1-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CR183 (LOW ENRICHMENT)

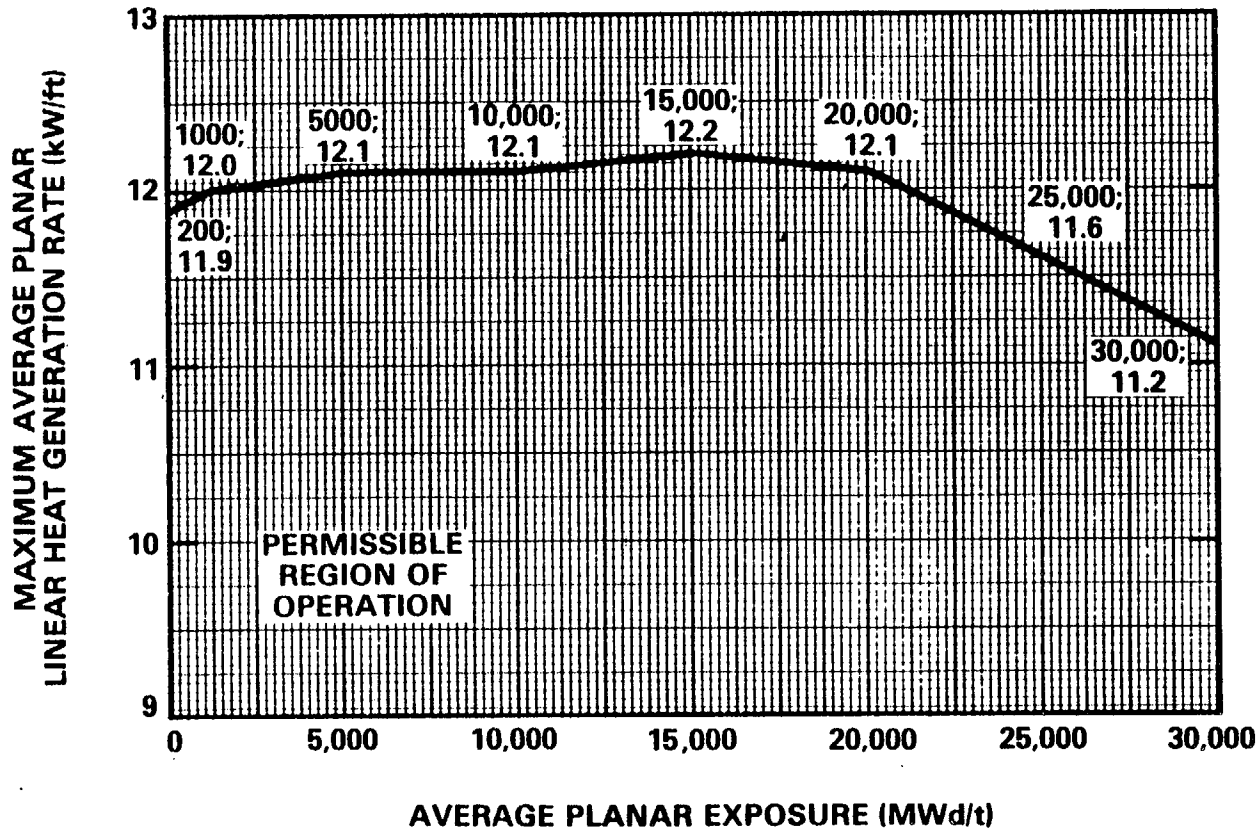


FIGURE 3.2.1-2

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CR233 (MEDIUM ENRICHMENT)

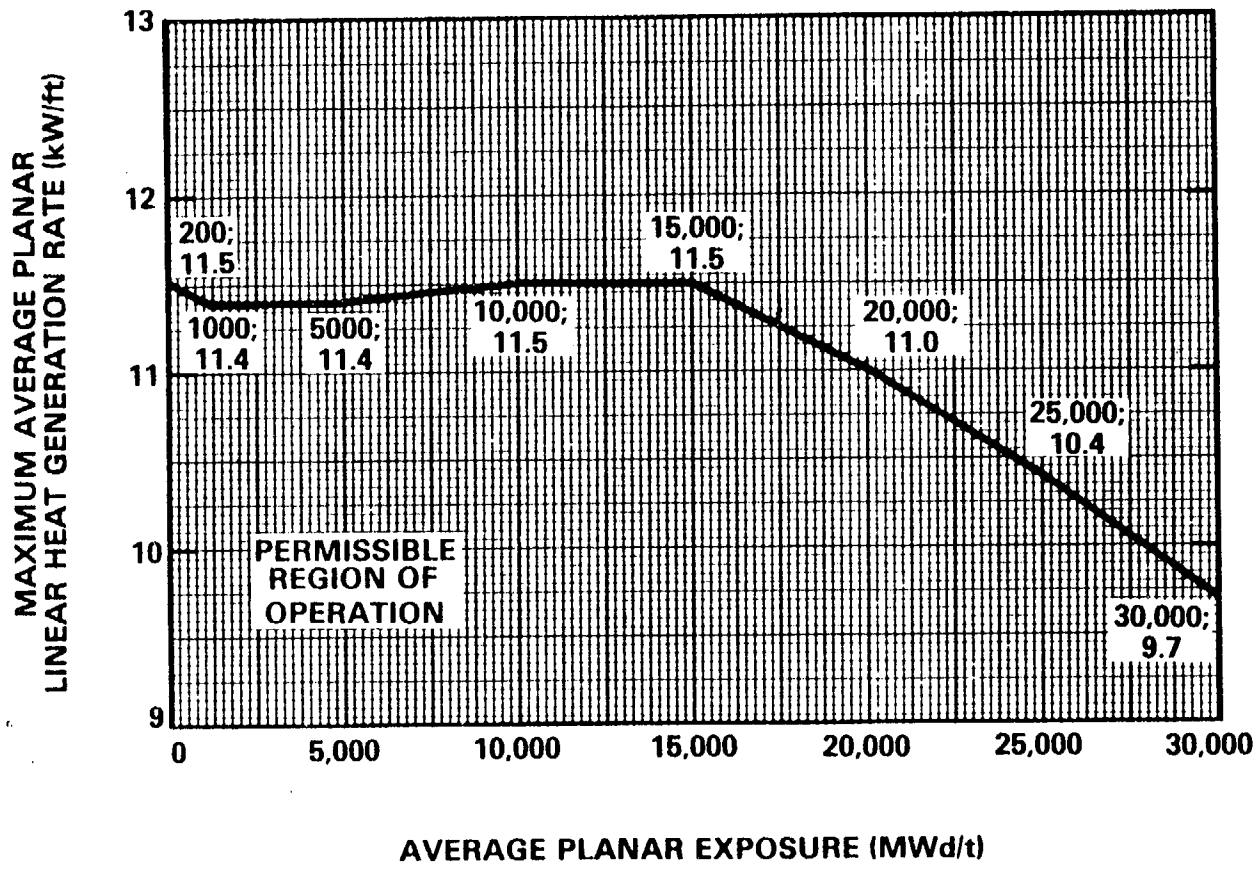


FIGURE 3.2.1-3

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CR711 (NATURAL ENRICHMENT)

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

Trip Setpoint	Allowable Value
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.0.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

---

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% 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

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit determined from 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 and the turbine bypass system is OPERABLE per Specification 3.7.8, with:

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

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

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

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

where:

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

N_i = number of active control rods measured in the ith surveillance tests,

τ_i = average scram time to notch 39 of all rods measured in the ith 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.

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 1 hour, MCPR is determined to be greater than or equal to the MCPR limit as a function of average scram time as 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 turbine bypass system inoperable per Specification 3.7.8, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that within 1 hour, MCPR is determined to be greater than or equal to the MCPR limit as a function of average scram time as shown in Figure 3.2.3-1, turbine bypass inoperable curve, times the K_f shown in Figure 3.2.3-2.

c. With MCPR less than the applicable MCPR limit determined from Figure 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.

d. The provisions of Specification 4.0.4 are not applicable.

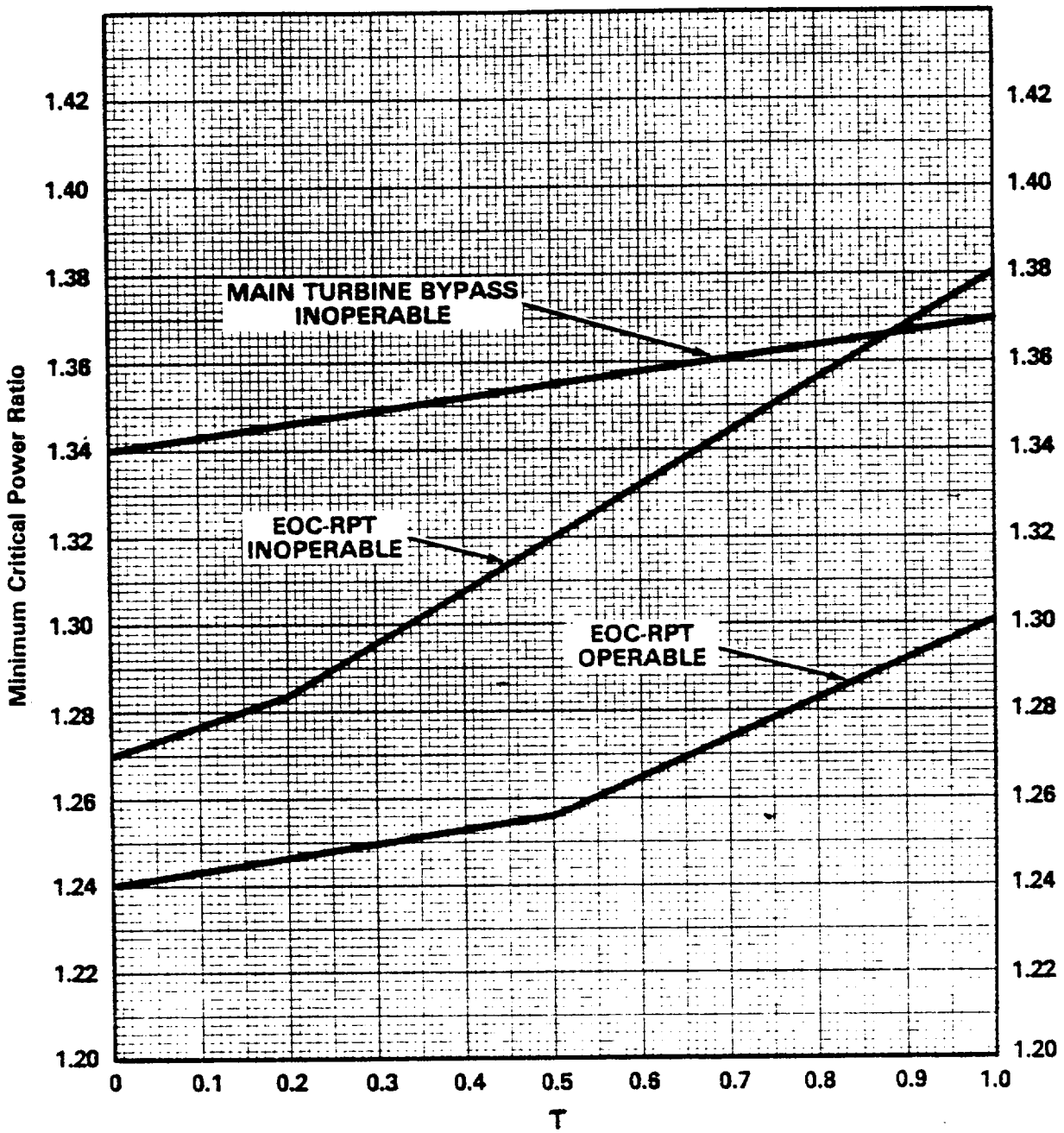


FIGURE 3.2.3-1

MINIMUM CRITICAL POWER RATIO (MCPR)
VERSUS τ AT RATED FLOW

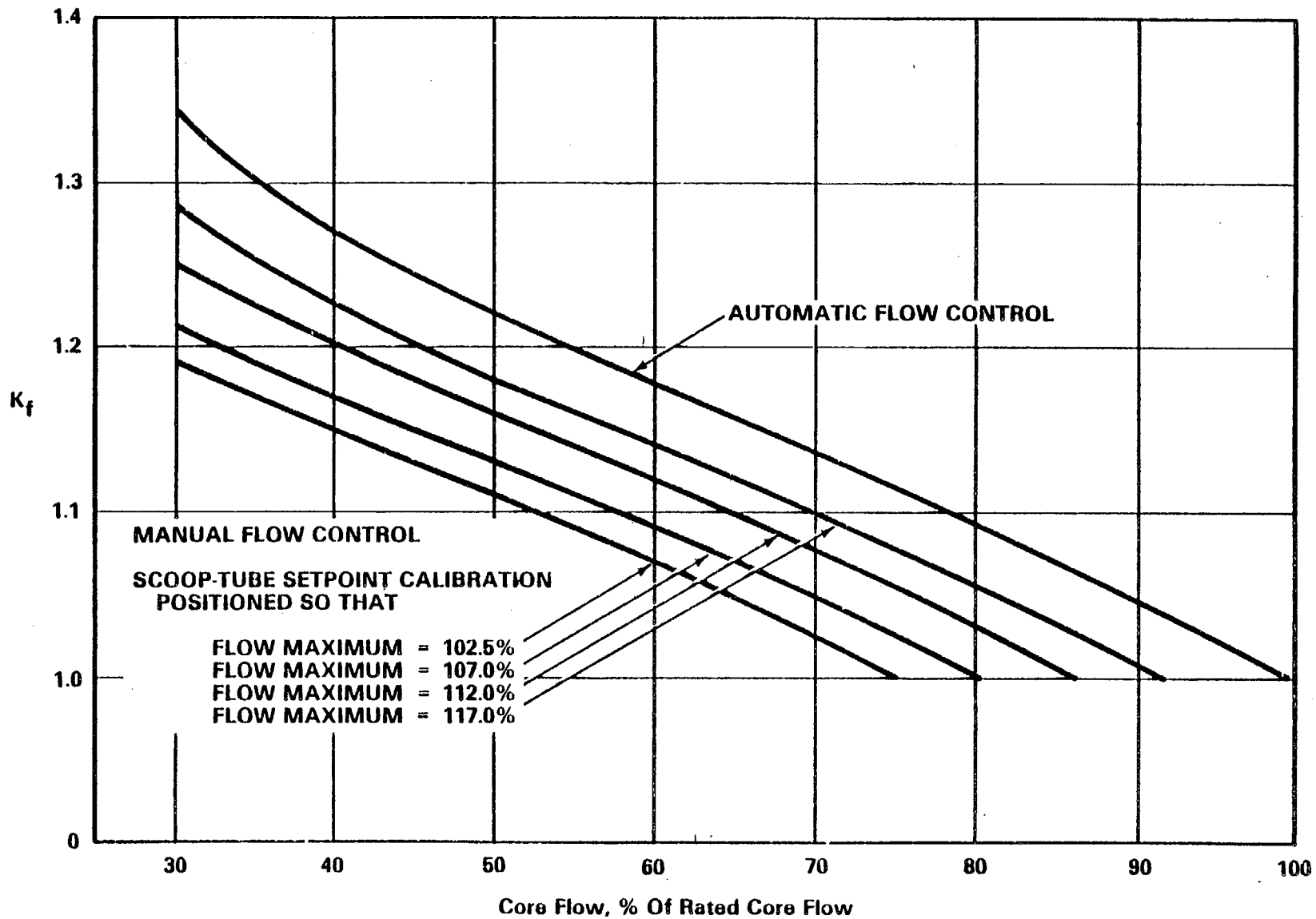


FIGURE 3.2.3-2

K_f FACTOR

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

---

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

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

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

TABLE 3.1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

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

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

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

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

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

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure until the function is automatically bypassed within 2 hours.

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

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

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

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

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

(b) This function is automatically bypassed when the reactor mode switch is in the Run position.

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

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

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

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

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

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

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

(j) This function shall be automatically bypassed when turbine first stage pressure is less than 108 psig or 17% of the value of first stage pressure in psia at valves wide open (V.W.O) steam flow, equivalent to THERMAL POWER of about 24% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

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

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant.
#Measured from actuation of fast-acting solenoid.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

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

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

*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. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, for the HPCI and RCIC systems, the inoperable channel shall be restored to OPERABLE status within 8 hours or the ACTION required by Table 3.3.2-1 for that trip Function shall be taken.

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

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	A	2	1, 2, 3	20
2) Low Low, Level 2	B	2	1, 2, 3	20
3) Low Low Low, Level 1	X	2	1, 2, 3	20
b. Drywell Pressure - High	Y,Z,X	2	1, 2, 3	20
c. Manual Initiation	NA	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	Y [c]	2	1, 2, 3 and *	25
b. Drywell Pressure - High	Y,Z [c]	2	1, 2, 3	25
c. Refuel Floor High Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
d. Railroad Access Shaft Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
e. Refuel Floor Wall Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
f. Manual Initiation	NA	1	1, 2, 3 and *	24

4 23 6

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION	
3. MAIN STEAM LINE ISOLATION					
a. Reactor Vessel Water Level - Low, Low, Level 2	B	2	1, 2, 3	21	
b. Main Steam Line Radiation - High	C	2	1, 2, 3	21	
c. Main Steam Line Pressure - Low	P	2	1	22	
d. Main Steam Line Flow - High	D	2/line	1, 2, 3	20	
e. Condenser Vacuum - Low	UA	2	1, 2, 3	21	
f. Main Steam Line Tunnel Temperature - High	E	2/line	1, 2, 3	21	
g. Main Steam Line Tunnel Δ Temperature - High	E	2	1, 2, 3	21	6
h. Manual Initiation	NA	1	1, 2, 3	24	
4. REACTOR WATER CLEANUP SYSTEM ISOLATION					
a. RWCU Δ Flow - High	J	1	1, 2, 3	23	6
b. RWCU Area Temperature - High	W	3	1, 2, 3	23	
c. RWCU Area Ventilation Δ Temp. - High	W	3	1, 2, 3	23	6
d. SLCS Initiation	(d)	NA	1, 2, 3	23	
e. Reactor Vessel Water Level - Low Low, Level 2	B	2	1, 2, 3	23	
f. RWCU Flow - High	J	1	1, 2, 3	23	6
g. Manual Initiation	NA	1	1, 2, 3	24	

TABLE 3.3. (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	A	2	1, 2, 3	26
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	UB	1	1, 2, 3	26
c. RHR Equipment Area Δ Temperature - High	M	1	1, 2, 3	26
d. RHR Area Cooler Temperature - High	M	1	1, 2, 3	26
e. RHR Flow - High	M	1	1, 2, 3	26
f. Manual Initiation	NA	1	1, 2, 3	24

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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

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

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

NOTES

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

** Actuates dampers shown in Table 3.6.5.2-1.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves which are actuated by these isolation signals.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel or trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

(c) Also starts the standby gas treatment system.

(d) Closes only RWCU system inlet outboard valve.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 13.0 inches*	≥ 11.5 inches
2) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Manual Initiation	NA	NA
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Refuel Floor High Exhaust Duct Radiation - High	≤ 2.5 mR/hr.**	≤ 4.0 mR/hr.**
d. Railroad Access Shaft Exhaust Duct Radiation - High	≤ 2.5 mR/hr.**	≤ 4.0 mR/hr.**
e. Refuel Floor Wall Exhaust Duct Radiation - High	≤ 2.5 R/hr.**	≤ 4.0 mR/hr.**
f. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45.0 inches
b. Main Steam Line Radiation - High	≤ 3 X full power background	≤ 3.6 X full power background
c. Main Steam Line Pressure - Low	≥ 861 psig	≥ 841 psig
d. Main Steam Line Flow - High	≤ 107 psid	≤ 110 psid

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
MAIN STEAM LINE ISOLATION (Continued)		
e. Condenser Vacuum - Low	≥ 9.0 inches Hg vacuum	≥ 8.8 inches Hg vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
g. Main Steam Line Tunnel Δ Temperature - High	≤ 99°F	≤ 108°F
h. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. RWCU Δ Flow - High	≤ 60 gpm	≤ 80 gpm
b. RWCU Area Temperature - High	≤ 147°F or 118.3°F#	≤ 154°F or 125.3°F#
c. RWCU/Area Ventilation Δ Temperature - High	≤ 69°F or 35.3°F#	≤ 78°F or 44.3°F#
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
f. RWCU Flow - High	≤ 426 gpm	≤ 436 gpm
g. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure - High	≤ 177" H_2O**	≤ 189" H_2O**
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig

TABLE 3.3. (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION Continued)		
d. RCIC Equipment Room Temperature - High	≤ 167°F**	≤ 174°F**
e. RCIC Equipment Room Δ Temperature - High	≤ 89°F	≤98°F
f. RCIC Pipe Routing Area Temperature - High	≤ 167°F##	≤ 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##
h. RCIC Emergency Area Cooler Temperature - High	≤ 147°F	≤ 154°F
i. Manual Initiation	NA	NA
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 350 inches H_2O**	≤ 367 inches H_2O** 15
b. HPCI Steam Supply Pressure - Low	≥ 104 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	≤ 167°F	≤ 174°F
e. HPCI Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F
f. HPCI Emergency Area Cooler Temperature - High	≤ 147°F	≤154°F
g. HPCI Pipe Routing Area Temperature - High	≤ 167°F##	≤ 174°F##
h. HPCI Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##
i. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches*	≥ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 98 psig	≤ 108 psig
c. RHR Equipment Area Δ Temperature - High	≤ 89°F**	≤ 90.5°F**
d. RHR Area Cooler Temperature - High	≤ 167°F**	≤ 170.5°F**
e. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
f. 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.
#Lower setpoints for TSH-G33-N600 E, F and TSH-G33-N602 E, F.
##15 minute time delay.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	≤10[a]
2) Low Low, Level 2	≤1.0*/≤10[a]**
3) Low Low Low, Level 1	≤ 10[a]
b. Drywell Pressure - High	≤10[a]
c. Manual Initiation	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	≤10[a]
b. Drywell Pressure - High	≤10[a]
c. Refuel Floor High Exhaust Duct Radiation - High[b]	≤10[a]
d. Railroad Access Shaft Exhaust Duct Radiation - High[b]	≤10[a]
e. Refuel Floor Wall Exhaust Duct Radiation -High[b]	≤10[a]
f. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level- Low Low, Level 2	≤10[a]
b. Main Steam Line Radiation - High[b]	≤1.0*/≤10[a]**
c. Main Steam Line Pressure - Low	≤1.0*/≤10[a]**
d. Main Steam Line Flow-High	≤0.5*/≤10[a]**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	≤10[a]##
b. RWCU Area Temperature - High	NA
c. RWCU Area Ventilation Temperature ΔT - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≤10[a]
f. RWCU Flow - High	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure - High	≤10[a]###
b. RCIC Steam Supply Pressure - Low	≤10[a]
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA
e. RCIC Equipment Room Δ Temperature - High	NA
f. RCIC Pipe Routing Area Temperature - High	NA
g. RCIC Pipe Routing Area Δ Temperature - High	NA
h. RCIC Emergency Area Cooler Temperature - High	NA
i. Manual Initiation	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Flow - High	≤10[(a)]####
b. HPCI Steam Supply Pressure - Low	≤10[(a)]
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. HPCI Equipment Room Δ Temperature - High	NA
f. HPCI Emergency Area Cooler Temperature - High	NA
g. HPCI Pipe Routing Area Temperature - High	NA
h. HPCI Pipe Routing Area Δ Temperature - High	NA
i. Manual Initiation	NA
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 3	≤10[(a)]
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
c. RHR Equipment Area Δ Temperature - High	NA
d. RHR Area Cooler Temperature - High	NA
e. RHR Flow - High	NA
f. 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 delay for diesel generator starting assumed in the accident analysis.

(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 for MSIV Valves.

**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 Table 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.

##With time delay of 45 seconds.

###With time delay of 3 seconds.

####With time delay of 3 seconds.

TABLE 3.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1) Low, Level 3	S	M	R	1, 2, 3
2) Low Low, Level 2	S	M	R	1, 2, 3
3) Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	R	1, 2, 3
c. Manual Initiation	NA	R	NA	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and *
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Refuel Floor High Exhaust Duct Radiation - High	S	M	R	1, 2, 3 and *
d. Railroad Access Shaft Exhaust Duct Radiation - High	S	M	R	1, 2, 3 and *
e. Refuel Floor Wall Exhaust Duct Radiation - High	S	M	R	1, 2, 3 and *
f. Manual Initiation	NA	R	NA	1, 2, 3 and *

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	NA	M	Q	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	NA	M	Q	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	S	M	R	1, 2, 3
b. RWCU Area Temperature - High	NA	M	Q	1, 2, 3
c. RWCU Area Ventilation Δ Temperature - High	NA	M	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f. RWCU Flow - High	S	M	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/loop	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	1/loop	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
g. Manual Initiation	1/valve	1, 2, 3	33

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv ESS Bus Under-voltage (Loss of Voltage, <20%)	1/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35
b. 4.16 kv ESS Bus Under-voltage (Degraded Voltage, <65%)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36
c. 4.16 kv ESS Bus Under-voltage (Degraded Voltage <84%)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
(b) One trip system. Provides signal to HPCI pump suction valves only.
(c) Two out of two logic.
* When the system is required to be OPERABLE per Specification 3.5.2.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.
** Required when ESF equipment is required to be OPERABLE.
\#\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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

a. For one trip system, place the inoperable trip system in the tripped condition within 1 hour* or declare the associated ECCS inoperable.

b. For both trip systems, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within 1 hour.

ACTION 33 - 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 ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 1 hour* or declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 36 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour;* operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

TRIP FUNCTION	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	
a. Reactor Vessel Water Level-Low Low Low, Level 1	≤27
b. Drywell Pressure-High	≤27
c. Reactor Vessel Steam Dome Pressure-Low	≤27
d. Manual Initation	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Reactor Vessel Water Level-Low Low Low, Level 1	≤40
b. Drywell Pressure-High	≤40
c. Reactor Vessel Steam Dome Pressure-Low	
1) System Initiation	≤40
2) Recirculation Discharge Valve Closure	≤40
d. Manual Initiation	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	
a. Reactor Vessel Water Level - Low Low, Level 2	≤30
b. Drywell Pressure - High	≤30
c. Condensate Storage Tank Level-Low	NA
d. Reactor Vessel Water Level-High, Level 8	NA
e. Suppression Pool Water Level-High	NA
f. Manual Initiation	NA
4. AUTOMATIC DEPRESSURIZATION SYSTEM	
a. Reactor Vessel Water Level-Low Low Low, Level 1	NA
b. Drywell Pressure-High	NA
c. ADS Timer	NA
d. Core Spray Pump Discharge Pressure-High	NA
e. RHR LPCI Mode Pump Discharge Pressure-High	NA
f. Reactor Vessel Water Level-Low, Level 3	NA
g. Manual Initiation	NA
5. LOSS OF POWER	
a. 4.16 kV ESS Bus Undervoltage (Loss of Voltage <20%)	NA
b. 4.16 kV ESS Bus Undervoltage (Degraded Voltage <65%)	NA
c. 4.16 kV ESS Bus Undervoltage (Degraded Voltage <84%)	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Steam Dome Pressure - Low	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Steam Dome Pressure - Low	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3
d. Suppression Pool Water Level - High	NA	M	Q	1, 2, 3
e. Reactor Vessel Water Level - High, Level 8	NA	M	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

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

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

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTION STATEMENTS

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

a. For one trip system, place the inoperable channel and/or that trip system in the tripped condition within 1 hour or declare the RCIC system inoperable.

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

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

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 1 hour or declare the RCIC system inoperable.

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

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

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

ACTION 61 - With the number of OPERABLE Channels:

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

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

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

NOTES

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

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

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

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

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

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

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

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + 40%	$\leq$ 0.66 W + 43%
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale	$\leq$ 0.58 W + 50%*	$\leq$ 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 4 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps**	$\geq$ 2 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	$\leq$ 44 gallons	$\leq$ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 11% flow deviation

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

** [illegible] initial loading and startup the count rate may [illegible] 0.5 cps.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b],M	Q	1*
b. Inoperative	NA	S/U[b],M	NA	1*
c. Downscale	NA	S/U[b],M	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	S	S/U[b],M	R	1
b. Inoperative	NA	S/U[b],M	NA	1, 2, 5
c. Downscale	S	S/U[b],M	R	1
d. Neutron Flux - Upscale, Startup	S	S/U[b],M	R	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W	NA	2, 5
b. Upscale	NA	S/U[b],W	Q	2, 5
c. Inoperative	NA	S/U[b],W	NA	2, 5
d. Downscale	NA	S/U[b],W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W	NA	2, 5
b. Upscale	NA	S/U[b],W	Q	2, 5
c. Inoperative	NA	S/U[b],W	NA	2, 5
d. Downscale	NA	S/U[b],W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b],M	Q	1
b. Inoperative	NA	S/U[b],M	NA	1
c. Comparator	NA	S/U[b],M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

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

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

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Main Control Room Outside Air Intake Radiation Monitor	2/intake	1,2,3,5 and *	$\leq$ 5 mR/hr	0.01 to 100 mR/hr	70
2. Area Monitors					
a. Criticality Monitors					
1) New Fuel Storage Vault	1	(a)	$\leq$ 15 mR/hr	10^{-2} to 10^{2} mR/hr	71
2) Spent Fuel Storage Pool	1	(b)	$\leq$ 15 mR/hr	10^{-1} to 10^{3} mR/hr	71

* When irradiated fuel is being handled in the secondary containment.
(a) With fuel in the new fuel storage vault.
(b) With fuel in the spent fuel storage pool.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION STATEMENTS

ACTION 70 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the isolation mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the isolation mode of operation within 1 hour.

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

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Control Room Outside Air Intake Radiation Monitor	S	M	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	(a)
2) Spent Fuel Storage Pool	S	M	R	(b)

(a) With fuel in the new fuel storage vault.
(b) With fuel in the spent fuel storage pool.
* When irradiated fuel is being handled in the secondary containment.

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status. |26

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

SURVEILLANCE REQUIREMENTS

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

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.05 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety. |26

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Accelerometers and Triggers		
a. Reactor Equipment, Unit 1	-0.25 to +0.25g	1
b. Reactor Bldg. Floor (RHR), Unit 1	"	1
c. ESSW Pumphouse Floor[a]	"	1
d. Containment Foundation, Unit 1[a]	"	1
e. Containment Structure, Unit 1	"	1
2. Peak Accelerographs		
a. Reactor Equipment, Unit 2	-0.5 to +0.5g	1
b. Reactor Piping, Unit 2	"	1
c. RHR Pump Room, Unit 2	"	1
3. Response-Spectrum Analyzer[c]/Recorders[b]	1-32 Hz	1

[a] The Unit 1 Containment Foundation, Unit 2 Containment Foundation and ESSW Pumphouse Floor Accelerometer Channels have associated triggers. These triggers provide control room indication and annunciation.

[b] Receive signals from respective accelerometers.

[c] One instrument: on-line capability, with control room annuciation for triggered channels; off-line capability for all channels.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Accelerometers and Triggers		
a. Reactor Equipment, Unit 1	SA	R
b. Reactor Bldg. Floor (RHR), Unit 1	SA	R
c. ESSW Pumphouse Floor[a]	SA	R
d. Containment Foundation, Unit 1[a]	SA	R
e. Containment Structure, Unit 1	SA	R
2. Peak Accelerographs		
a. Reactor Equipment, Unit 2	NA	R
b. Reactor Piping, Unit 2	NA	R
c. RHR Pump Room, Unit 2	NA	R
3. Response-Spectrum Analyzer/Recorders	SA	R

[a] The Unit 1 Containment Foundation and the ESW Pumphouse Floor are the triaxial accelerometer channels with triggers. The triggers are functionally tested and calibrated with their associated channels.

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more meteorological monitoring instrumentation channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status. 26

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Steam Dome Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Suppression Chamber Water Level	M	R
4. Suppression Chamber Water Temperature	M	R
5. Suppression Chamber Air Temperature	M	R
6. Primary Containment Pressure	M	R
7. Drywell Temperature	M	R
8. Drywell Oxygen/Hydrogen Analyzer	M	Q*
9. Safety/Relief Valve Position Indicators	M	R
10. Containment High Radiation	M	R
11. Noble gas monitors		
a. Reactor Bldg. Vent	M	R
b. SGTS Vent	M	R
c. Turbine Bldg. Vent	M	R

*For hydrogen analyzer, use sample gas containing one volume percent hydrogen, balance nitrogen.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least three source range monitor channels shall be OPERABLE.

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 two or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 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.
***For initial loading and startup the count rate may be less than 3 cps if the following conditions are met; (1) the signal to noise ratio is greater than 2.0, (2) the signal is greater than 1/2 cps, and (3) a counting interval sufficient to accumulate at least 500 counts is employed.

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

a. Recalibration of the LPRM detectors, and

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 The chlorine detection system shall be OPERABLE with two independent chlorine detectors with their alarm/trip saturation point adjusted to actuate within 5 seconds at a chlorine concentration of ≤ 5 ppm.

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one chlorine detector inoperable, restore the inoperable detector to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency outside air supply subsystem in the recirculation mode of operation.

b. With the chlorine detection system otherwise inoperable, within 1 hour initiate and maintain operation of at least one control room emergency outside air supply subsystem in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 The above required chlorine detection system shall be demonstrated OPERABLE by performance of a:

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

b. Detector functional test at least once per 18 months by introducing a measured amount of chlorine into each detector, equivalent to a chlorine concentration of ≤ 5 ppm, and verifying the elapsed time to actuate each detector is less than or equal to 5 seconds.

TABLE 3.3.7.10-1 (Continued)

TABLE NOTATION

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

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

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 101 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcurie/mL.

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

TABLE 4.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	P*	P	R(3)	Q(1)
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D	M	R(3)	Q(2)
b. RHR Service Water System Effluent Line	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Discharge Line	D(4)	N.A.	R	Q

*Perform CHANNEL CHECK at least once per 24 hours if discharge valve interlocks referenced in Table 3.3.7.10-1 are not functioning.

TABLE 4.3.7.10-1 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

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

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

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

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.11-1

ACTION:

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

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

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable. 26

SURVEILLANCE REQUIREMENTS

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

BLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	110
2. REACTOR BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1##	*	111
b. Iodine Monitor	1	*	112
c. Particulate Monitor	1	*	112
d. Effluent System Flow Rate Monitor	1	*	113
e. Sampler Flow Rate Monitor	1	*	113

TABLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1##	*	114
b. Iodine Monitor	1	*	112
c. Particulate Monitor	1	*	112
d. Effluent System Flow Rate Monitor	1	*	113
e. Sampler Flow Rate Monitor	1	*	113
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (Prior to Input to Holdup System)			
a. Noble Gas Activity Monitor	1	**	115
5. STANDBY GAS TREATMENT SYSTEM MONITOR			
a. Noble Gas Activity Monitor	1##	#	116
b. Iodine Monitor	1	#	112
c. Particulate Monitor	1	#	112
d. Effluent System Flow Rate Monitor	1	#	113
e. Sampler Flow Rate Monitor	1	#	113

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

#During operation of the standby gas treatment system.

##Low-range and mid-range channels of monitor only

* At all times.

** During operation of the main condenser air ejector and offgas treatment system.

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 111 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 112 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11.2.1.2-1.

ACTION 113 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 114 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours and provided the mechanical vacuum pumps are not operated.

ACTION 115 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases to the environment may continue for up to 72 hours provided:

a. The offgas system is not bypassed, and

b. The Turbine Building vent noble gas activity monitor is OPERABLE;

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

ACTION 116 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend release of radioactive effluent via this pathway.

TABLE 4.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.7.[illegible]-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Monitor	W	M	R(2)	Q(1)	*
c. Particulate Monitor	W	M	R(2)	Q(1)	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	**
5. STANDBY GAS TREATMENT SYSTEM MONITOR					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	#
b. Iodine Monitor	W	M	R(2)	Q(1)	#
c. Particulate Monitor	W	M	R(2)	Q(1)	#
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	#
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	#

TABLE 4.3.7.11-1 (Continued)

TABLE NOTATION

During operation of the standby gas treatment system.

* At all times.

** During operation of the main condenser air ejector and offgas treatment system.

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

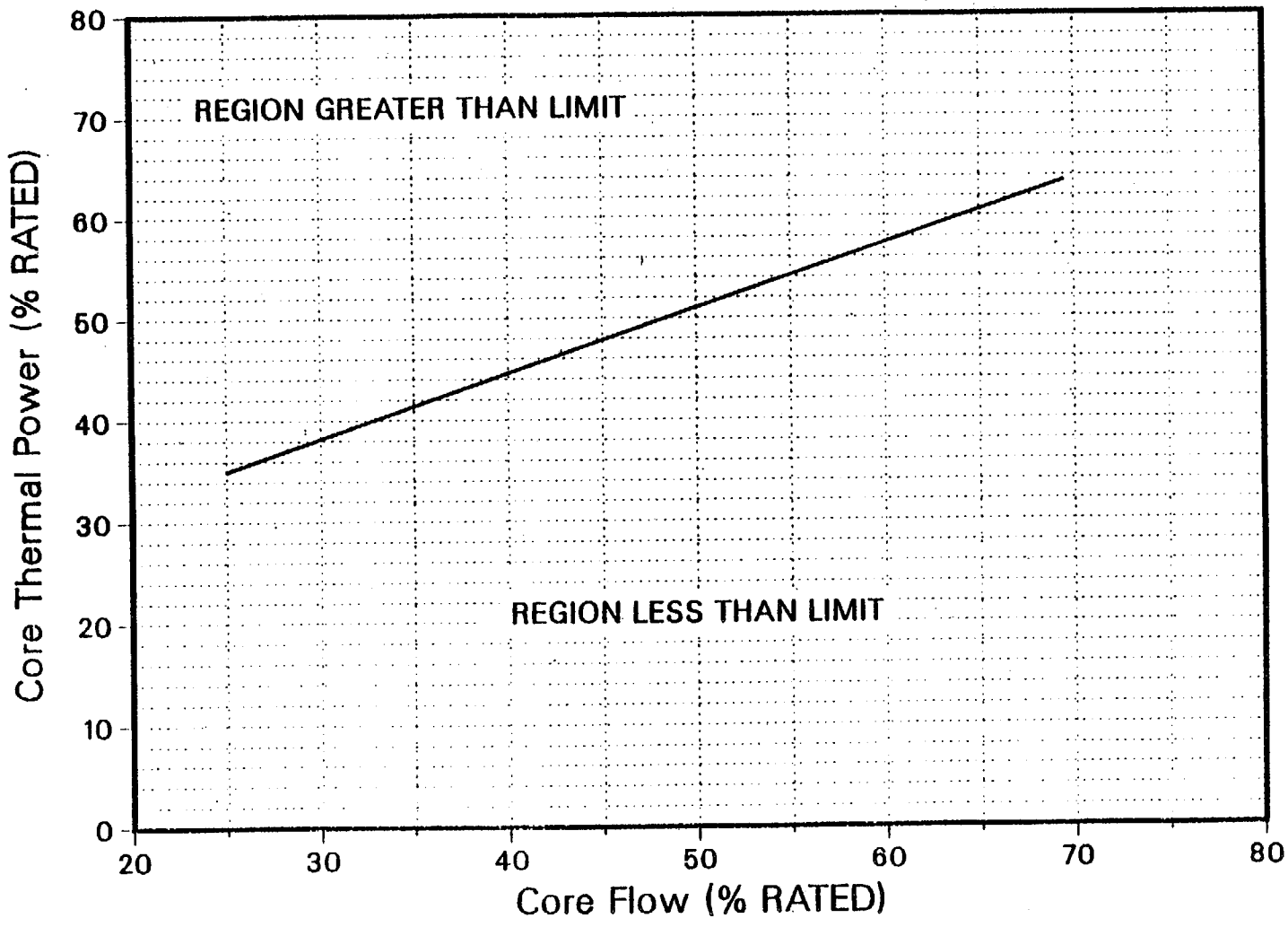
1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(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.

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

1. 0.5 volume percent hydrogen, balance nitrogen, and
2. 4 volume percent hydrogen, balance nitrogen.

Figure 3.4.1.1-1
THERMAL POWER LIMITATIONS



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 pumps are operating at the same speed:

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-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%.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 10 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:* **

2 safety-relief valves @ 1146 psig ±1%
4 safety-relief valves @ 1175 psig ±1%
4 safety-relief valves @ 1185 psig ±1%
3 safety-relief valves @ 1195 psig ±1%
3 safety-relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.2 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.25 by performance of a:

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

b. Calibration in accordance with procedures prepared in conjunction with its manufacturer's recommendations at least once per 18 months.##

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.
**Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.
##The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 At least the following reactor coolant system leakage detection systems shall be OPERABLE:

a. One primary containment atmosphere particulate radioactivity monitoring system channel aligned to the drywell. 15

b. Two drywell floor drain sump level monitoring system channels, and

c. One primary containment atmosphere gaseous radioactivity monitoring system channel aligned to the drywell. 15

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

1ST ISOLATION VALVE(S) NUMBER(S)	2ND ISOLATION VALVE(S) NUMBER(S)	ALARM SETPOINT (psig)	SERVICE
HV-E211F006A HV-E211F037A	HV-E211F005A	≤ 460	Core Spray Injection
HV-E211F006B HV-E211F037B	HV-E211F005B	≤ 460	Core Spray Injection
HV-E111F050A HV-E111F122A	HV-E111F015A	≤ 400	LPCI Injection
HV-E111F050B HV-E111F122B	HV-E111F015B	≤ 400	LPCI Injection
HV-E111F022	HV-E111F023	≤ 400	Head Spray
HV-E111F009	HV-E111F008	≤ 142	Shutdown Cooling

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, for conductivity and chloride concentration, for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN as rapidly as practical within the cooldown rate limit.

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

c. At all other times:

1. With the conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours.

2. With the chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 24 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

3. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

b. Analyzing a sample of the reactor coolant:

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, obtaining an in-line conductivity measurement at least once per:

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

2. 24 hours at all other times.

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

1. 7 days, and

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

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	PH
1	≤ 0.2 ppm	≤ 1.0	5.6 ≤ pH ≤ 8
2 and 3	≤ 0.1 ppm	≤ 2.0	5.6 ≤ pH ≤ 8.6
At all other times	≤ 0.5 ppm	≤ 10.0	5.3 ≤ pH ≤ 8.6

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 microcurie 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 microcurie 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 microcurie 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 microcurie 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 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. 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 microcurie per gram DOSE EQUIVALENT I-131 together with the following additional information. 26

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. In OPERATIONAL CONDITION 1 or 2, with:

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

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.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1-1 and 3.4.6.1-2, as applicable, for hydrostatic or leak testing, heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS, and operations with a critical core other than low power PHYSICS TESTS, with:

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

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

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

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

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1 and 3.4.6.1-2, as applicable, at least once per 30 minutes.

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.

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR shutdown cooling permissive setpoint.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

[#] One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

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

[##] The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

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

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system or 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.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, 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.

e. With a CSS header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or determine the ECCS header ΔP locally at least once per 12 hours; otherwise, declare the CSS inoperable.

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 usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water by:

1. Venting at the high point vents

2. Performing a CHANNEL FUNCTIONAL TEST of the condensate transfer pump discharge low pressure alarm instrumentation.

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

2. For the CSS, performance of a CHANNEL FUNCTIONAL TEST of the core spray header ΔP instrumentation.

3. For the LPCI system, verifying that at least one LPCI system subsystem cross-tie valve is closed with power removed from the valve operator.

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

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

1. The two CSS pumps in each subsystem together develop a total flow of at least 6350 gpm against a test line pressure of greater than or equal to 269 psig, corresponding to a reactor vessel steam dome pressure of $\geq$ 105 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 12,200 gpm against a test line pressure of $\geq$ 204 psig, corresponding to a reactor vessel to primary containment differential pressure $\geq$ 20 psid. 6

3. The HPCI pump develops a flow of at least 5000 gpm against a test line pressure of $\geq$ 1266 psig when steam is being supplied to the turbine at 920^{+140}_{-20} psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, 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.

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

**Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

SURVEILLANCE REQUIREMENTS (Continued)

2. For the HPCI system, verifying that the system develops a flow of at least 5000 gpm against a test line pressure of 210 ± 15 psig when steam is being supplied to the turbine at 150 ± 15 psig.*

3. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ 1 psid.

4. Verifying that the suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

5. Performing a CHANNEL CALIBRATION of the condensate transfer pump discharge low pressure alarm instrumentation and verifying the low pressure alarm setpoint to be ≥ 113 psig.

d. For the ADS:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

2. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

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

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

c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm systems and verifying air alarm setpoint of 2070 ± 35 psig on decreasing pressure.

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

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following subsystems shall be OPERABLE:

a. Core spray system (CSS) subsystems with a subsystem comprised of:

1. Two OPERABLE CSS pumps, and
2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water, equivalent to a level of 49%.

b. Low pressure coolant injection (LPCI) system subsystems with a subsystem comprised of:

1. At least one OPERABLE LPCI pump, and
2. An OPERABLE flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION

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

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

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 45.0 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

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

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

*See Special Test Exception 3.10.1

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

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 1.0 percent by weight of the containment air per 24 hours at P_a, 45.0 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 3.6.3-1, except for main steam line isolation valves*, main steam line drain valves* and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 45.0 psig.

c. *Less than or equal to 46 scf per hour for all four main steam lines through the isolation valves when tested at P_t, 22.5 psig.

d. *Less than or equal to 1.2 scf per hour for any one main steam line drain valve when tested at P_a, 45.0 psig.

e. A combined leakage rate of less than or equal to 3.3 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 49.5 psig.

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

ACTION:

With:

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

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

c. The measured leakage rate exceeding 46 scf per hour for all four main steam lines through the isolation valves, or

d. The measured leak rate exceeding 1.2 scf per hour for any one main steam line drain valve, or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 3.3 gpm,

*Exemption to Appendix "J" of 10 CFR 50.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 45.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. 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/hour when the gap between the door seals is pressurized to 10 psig.

b. By conducting an overall air lock leakage test at P_a, 45.0 psig, and verifying that the overall air lock leakage rate is within its limit:

1. After each opening, unless performed within the previous 6 months#, but at least once per 18 months*, and

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

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

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested upon initial entry after the primary containment has been de-inerted.

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

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

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting the blower(s) from the control room and operating the blower(s) for at least 15 minutes.

   2. Energizing the heaters and verifying a temperature rise indicating heater operation on downstream piping.

b. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed, each automatic valve actuates to its correct position and the blower starts.

   2. Verifying that the blower develops at least the below required vacuum at the rated capacity:

      a) Inboard valves, 15" H_2O at 100 scfm.
      b) Outboard valves, 60" H_2O at 200 scfm.

c. By verifying the operating instrumentation to be OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,
   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   3. CHANNEL CALIBRATION at least once per 18 months.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the required inspection shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken. 26

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression chamber water level indicator OPERABLE and/or with only five pairs of suppression pool water temperature indicators OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.

d. With no suppression chamber water level indicators OPERABLE and/or with only four pairs of suppression pool water temperature indicators OPERABLE, restore at least one water level indicator and at least five pairs of water temperature indicators to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 90°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 90°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER after suppression chamber average water temperature has exceeded 90°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 90°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least two suppression chamber water level indicators and at least sixteen surface water temperature indicators, at least one pair in each suppression pool sector, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months,

with the water level and temperature alarm setpoint for:

1. High water level $\leq$ 23'9",

2. Low water level $\geq$ 22'3", and

3. High water temperature:

   a) First setpoint, $\leq$ 90°F,
   b) Second setpoint, $\leq$ 105°F,
   c) Third setpoint, $\leq$ 110°F, and
   d) Fourth setpoint, $\leq$ 120°F.

d. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of at least 4.3 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. Two OPERABLE RHR pumps, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool 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.

b. With both suppression pool spray loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of 10,000 +0, -250 gpm per Specification 4.6.2.3.b.

c. At least once per 5 years by performing an air or water flow test through the spray header and each nozzle to verify that the flow path is unobstructed.

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

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. Two OPERABLE RHR pumps; and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of 10,000 +0, -250 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

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

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building, Zone I (Unit 1) Exhaust System Dampers (HD-17576A&B)	3.0
2. Reactor Building, Zone III (Unit 1) Exhaust System Dampers (HD-17502A&B)	6.0
3. Reactor Building, Zone III (Unit 2) Exhaust System Dampers (HD-27502A&B)	6.0

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

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

   2. In Operational Condition *, 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

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two drywell and two suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or one suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by energizing the recombiner system to at least 10 kw for $\geq$ 5 minutes.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required energization. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials, etc.

DRYWELL AIR FLOW SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 Three independent drywell air flow unit cooler subsystems (1V414A, B, 1V415A, B and 1V416A, B) shall be OPERABLE at low speed with each subsystem consisting of 2 unit cooler fans.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one fan of one or more unit cooler subsystems inoperable at low speed, restore the inoperable fan(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 Each drywell air flow unit cooler subsystem required above shall be demonstrated OPERABLE at least once per 92 days by:

a. Starting each fan at low speed from the control room, and

b. Verifying that each fan operates for at least 15 minutes.

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two independent residual heat removal service water (RHRSW) system subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE RHRSW pump*, and

b. An OPERABLE flow path capable of taking suction from the spray pond and transferring the water through one RHR heat exchanger.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

1. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump* within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both RHRSW subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 5 with one RHRSW subsystem, which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE 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.

*May not be a pump required for Unit 2.

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

EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent emergency service water system loops shall be OPERABLE with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and
b. An OPERABLE flow path capable of taking suction from the spray pond and transferring the water to the associated safety related equipment.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

   1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 7 days or be in a least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With two emergency service water pumps inoperable, restore at least one inoperable pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   3. With one emergency service water system loop inoperable, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With only one emergency service water pump and its associated flowpath OPERABLE, restore at least two pumps and their associated flowpaths to OPERABLE status within 72 hours or:

   1. In OPERATIONAL CONDITION 4 or 5, declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

   2. In Operational Condition *, declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency service water system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety-related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each pump starts automatically when its associated diesel generator starts.

*When handling irradiated fuel in the secondary containment.

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 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcurie 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 (tritium), and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcurie of removable contamination.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM-D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.6.1.3 The diesel driven fire pump diesel 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, is greater than or equal to 1.200,

3. 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 oncer per 18 months by verifying that:

1. The cell and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. Terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following spray and sprinkler systems shall be OPERABLE:

a. RCIC Pump Room, Unit 1
b. HPCI Pump Room, Unit 1
c. Upper Cable Spreading Room, Unit 1
d. Lower Cable Spreading Room, Unit 1
e. Diesel Generator A Room
f. Diesel Generator B Room
g. Diesel Generator C Room
h. Diesel Generator D Room
i. Fire Zones 1-3A and 1-3B
j. Fire Zone 1-4A
k. Fire Zone 1-5A
l. Fire Zone 1-2B 10

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 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. 26

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 a test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each deluge nozzle's spray area to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water flow test through each open head spray and sprinkler header and verifying each open head spray and sprinkler nozzle is unobstructed.

CO_2 SYSTEMS

LIMITING CONDITIONS

3.7.6.3 The following low pressure CO_2 systems shall be OPERABLE:

a. Control Room Under Floor, Unit 1
b. Control Room Under Floor, Unit 2
c. Lower Relay Room, Unit 1#
d. Upper Relay Room, Unit 1#
e. North Cable Chase
f. Center Cable Chase
g. Room C-411 Soffit
h. Control Room Soffit, Unit 1
i. Control Room Soffit, Unit 2

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 1 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 low pressure 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 (90)% and pressure to be greater than 270 psig, and

b. At least once per 18 months by:

1. Verifying the system valves and associated ventilation dampers and fire door release mechanisms actuate manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each accessible nozzle by performance of a "Puff Test."

#Accessible nozzles.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The Halon systems in the following panel modules shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure:

10700	10701	10702	10703	10704	10705
10706	10730	10731	10732		

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 1 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. 26

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. Performance of a flow test through accessible headers and nozzles to assure no blockage.

2. Performance of a functional test of the general alarm circuit and associated alarm and interlock devices.

FIRE HOSE STATIONS

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 3.7.6.5-1 inoperable, route an additional fire hose of equal or greater diameter to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours. 26

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 for all fire hose stations, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings for all fire hose stations.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7.6.5-1

FIRE HOSE STATIONS

LOCATIONS	COLUMN	HOSE RACK NUMBER
a. Control Structure		
El. 697'-0"	L-26	1HR-171
El. 697'-0"	L-32	2HR-171
El. 714'-0"	L-26	1HR-162
El. 714'-0"	L-31	2HR-162
El. 729'-0"	L-25.9	1HR-158
El. 729'-0"	L-32.1	2HR-158
El. 754'-0"	L-26	1HR-136
El. 754'-0"	L-32	2HR-136
El. 771'-0"	L-26	1HR-125
El. 771'-0"	L-31	2HR-125
b. Reactor Building		
El. 645'-0"	R-29	1HR-271
El. 645'-0"	P-20.6	1HR-272
El. 645'-0"	U-22	1HR-273
El. 645'-0"	R-37.4	2HR-271
El. 645'-0"	U-30.5	2HR-272
El. 645'-0"	R-30	2HR-273
El. 670'-0"	O-27.5	1HR-261
El. 670'-0"	Q-29	1HR-262
El. 670'-0"	T-22	1HR-263
El. 683'-0"	Q-27.5	1HR-251
El. 683'-0"	Q-20.6	1HR-252
El. 683'-0"	T-22	1HR-253
El. 719'-1"	Q-27.5	1HR-241
El. 719'-1"	S-27.5	1HR-242
El. 719'-1"	Q-20.6	1HR-243
El. 719'-1"	T-20.6	1HR-244
El. 719'-1"	T-23.5	1HR-245
El. 749'-1"	S-27.5	1HR-231
El. 749'-1"	S-27.5	1HR-231
El. 749'-1"	Q-20.6	1HR-232
El. 749'-1"	T-20.6	1HR-233
El. 779'-1"	P-26.5	1HR-221
El. 779'-1"	S-26.5	1HR-222
El. 779'-1"	Q-22	1HR-223
El. 779'-1"	U-20.6	1HR-224
El. 779'-1"	T-23.3	1HR-211
El. 818'-1"	P-26.5	1HR-201
El. 818'-1"	U-26.5	1HR-202
El. 818'-1"	Q-20.6	1HR-203

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YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.6.6 Yard fire hydrants 1FH122 and 1FH104 and associated hydrant hose houses shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With yard fire hydrants 1FH122 and/or 1FH104 and/or associated hydrant hose houses inoperable, route sufficient additional lengths of fire hose of equal or greater diameter located in adjacent OPERABLE hydrant hose house(s) to provide service to the unprotected area(s) within 1 hour. |26|

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

SURVEILLANCE REQUIREMENTS

4.7.6.6 Yard fire hydrants 1FH122 and 1FH104 and associated hydrant hose houses shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose houses to assure all required equipment is at the hose houses.

b. At least once per 6 months, during March, April or May and during September, October or November, by visually inspecting the yard fire hydrants and verifying that the hydrant barrels are dry and that the hydrants are not damaged.

c. At least once per 12 months by:

   1. Conducting hose hydrostatic tests at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

   2. Replacement of all degraded gaskets in couplings.

   3. Performing a flow check of each hydrant.

3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire rated 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 and piping 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 1 hour establish a continuous fire watch on at least one side of the affected assembly and/or sealing device or verify the OPERABILITY of fire detectors on at least one side of the inoperable fire assembly and/or sealing device and establish an hourly fire watch patrol. |26

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 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying the position of each closed fire door at least once per 24 hours.

b. Verifying that doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. Verifying the position of each locked closed fire door at least once per 7 days.

d. Verifying the OPERABILITY of the fire door supervision system by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

e. Inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months.

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or determine MCPR to be equal to or greater than the applicable MCPR limit without bypass within 1 hour or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.7.8 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 0.30 seconds.

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. Four separate and independent diesel generators*, each with:

   1. Separate engine mounted day fuel tanks containing a minimum of 325 gallons of fuel,

   2. A separate fuel storage system containing a minimum of 47,570 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or one diesel generator 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 within 1 hour and 4.8.1.1.2.a.4, for one diesel generator at a time, within 4 hours and at least once per 8 hours therafter; restore at least two offsite circuits and four diesel generators 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 one diesel generator 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 within 1 hour and 4.8.1.1.2.a.4, for one diesel generator at a time, within 3 hours 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 four diesel generators 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.

*Shared with Unit 2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With one diesel generator of the above required A.C. electrical power sources inoperable, in addition to ACTION a or b, above, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining 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.

d. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of four diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4, for one diesel generator at a time, within four hours 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.

e. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4, for one diesel generator at a time, within 2 hours, and at least once per 8 hours thereafter; restore at least three of the diesel generators 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 four diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

7. Simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 200 volts and 60 ± 0.5 Hz during this test.

   c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential and engine low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4700 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to 4000 kW. The generator voltage and frequency shall be 4160 ± 200 volts and 60 ± 0.5 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.d.4.b).*

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4700 kW.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

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*If Surveillance Requirement 4.8.1.1.2.d.4.b) is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Rather, the diesel generator may be operated at 4000 kW for 1 hour or until operating temperature has stabilized.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that with all diesel generator air start receivers pressurized to less than or equal to 240 psig and the compressors secured, the diesel generator starts at least 5 times from ambient conditions and accelerates to at least 600 rpm in less than or equal to 10 seconds for the first 2 starts and accelerates to at least 600 rpm in less than or equal to 19 seconds for the remaining 3 starts. 3

13. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted day tank of each diesel via the installed cross connection lines.

14. Verifying that each diesel generator loading sequence timer shown in Table 4.8.1.1.2-2 is OPERABLE with its setpoint within ± 10% of its design setpoint.

15. Verifying that the following diesel generator lockout features prevent diesel generator starting and/or operation only when required:

    a) Engine overspeed.
    b) Generator differential.
    c) Engine low lube oil pressure.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 600 rpm in less than or equal to 13 seconds.

f. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. 26

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one of the above required chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within 1 hour and at least once per 8 hours thereafter. If any Category A limit in Table 4.8.2.1-1 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required ± 24-volt, 125-volt and 250-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 26, 129, 258-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 22, 110 or 220 volts, as applicable, or battery overcharge with battery terminal voltage above 30, 150 or 300 volts, as applicable, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and

3. The average electrolyte temperature of 4, 10 or 20, as applicable, of connected cells for the 24, 125 and 250 volt batteries is above 60°F.

SURVEILLANCE REQUIREMENTS (Continued)

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 of each 125-volt and 250-volt battery is less than or equal to 150×10^{-6} ohm, and

4. The battery charger, for at least 4 hours, will supply at least:

a) For the ± 24-volt batteries, 25 amperes at a minimum of 25.7 volts.

b) For the 125-volt batteries, 100 amperes at a minimum of 127.8 volts.

c) For the 250-volt batteries, 300 amperes at a minimum of 255.6 volts.

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, which is verified to be greater than the actual emergency loads, while maintaining the battery terminal voltage greater than or equal to ± 21, 105 or 210 volts, as applicable.

a) For ± 24-volt battery banks 1D670, 1D670-1, 1D680 and 1D680-1, 9.37 amperes for the entire 4 hour test.

b) For 125-volt batteries:

1) Channel A battery 1D612:
325 amperes for 60 seconds
107 amperes for the remainder of the 4 hour test

2) Channel "B" battery 1D622:
318 amperes for 60 seconds
100 amperes for the remainder of the 4 hour test

3) Channel "C" battery 1D632:
340 amperes for 60 seconds
121 amperes for the remainder of the 4 hour test

4) Channel "D" battery 1D642:
323 amperes for 60 seconds
104 amperes for the remainder of the 4 hour test.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. 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.

b. With less than Division I and/or Division II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling or 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.

SURVEILLANCE REQUIREMENTS

4.8.3.2.1 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

4.8.3.2.2 The A.C. power distribution system swing bus automatic transfer switch shall be demonstrated OPERABLE at least once per 31 days by actuating the load test switch or by disconnecting the normal power source to the transfer switch and verifying that swing bus automatic transfer is accomplished.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 and all fuses tested pursuant to Specification 4.8.4.1.a.2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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. Declare the affected system or component inoperable, and

   3. 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.

   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 which have the inoperable device racked out or removed or, which have the alternate device tripped, racked out, or removed.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary containment penetration conductor overcurrent protective devices required above shall be demonstrated OPERABLE:

a. At least once per 18 months by:

   1. Selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. For the lower voltage circuit breakers the nominal trip setpoint and short circuit response times are listed in Table 3.8.4.1-1. Testing of these circuit breakers shall consist of injecting a current in excess of 120% of the breaker's nominal setpoint and measuring the response time.

The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

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.

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

CIRCUIT BREAKER LOCATION	TYPE*	FRAME RATING/UL#	TRIP SET POINT (Amperes)	RESPONSE TIME (Milli-seconds/ Cycles)	SYSTEMS OR EQUIPMENT POWERED
a. Type 2 Molded Case Circuit Breakers#					
1. 1B219022	HFB-M	150/30	270	NA	HVB311F031A RRP "A" DSCH VLV
2. 1B237043	HFB-M	150/30	270	NA	HVB311F023A Recirc. PP "A" Suction
3. 1B236052	HFB-M	150/30	215	NA	HVE111F009 RHR Pump Suction Shutoff
4. 1B236023	HFB-M	150/3	18	NA	HV12603 Containment Inst. Compressor Suct. Iso. Valve
5. 1B236011	HFB-M	150/30	250	NA	1V413A - Drywell Area Unit Cooler
6. 1B236033	HFB-M	150/30	220	NA	1V414A - Drywell Area Unit Cooler
7. 1B236021	HFB-M	150/30	175	NA	1V417A - Drywell Area Unit Cooler
8. 1B236032	HFB-M	150/30	180	NA	1V412A - Drywell Area Unit Cooler
9. 1B236042	HFB-M	150/30	150	NA	1V411A - Drywell Area Unit Cooler
10. 1B236043	HFB-M	150/30	160	NA	1V416A - Drywell Area Unit Cooler
11. 1B236082	HFB-M	150/30	150	NA	1V415A - Drywell Area Unit Cooler
12. 1B236102	HFB-M	150/3	8	NA	HVB211F001 - Reactor Head Vent Valve
13. 1B236053	HFB-M	150/5	45	NA	HVG331F001 - Reac. Wtr. Clean up inboard isolation
14. 1B237072	HFB-M	150/5	25	NA	HVB211F016 - Main Stm. Line Drain Inboard Isolation
15. 1B219023	HFB-M	150/10	80	NA	HVB311F032A - RRP "A" Dsch Byps Vlv

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

CIRCUIT BREAKER LOCATION		TYPE*	FRAME RATING/UL	TRIP SET POINT (Amperes)	RESPONSE TIME (Milli-seconds/ Cycles)	SYSTEMS OR EQUIPMENT POWERED	
Type 2 Molded Case Circuit Breakers# (Continued)							
16.	1B237073	HFB-M	150/10	60	NA	HVE111F022 - Reac Head Spray Shutoff Inboard	16 24
17.	1B237082	HFB-M	150/10	70	NA	HVE411F002 - HPCI Stm. Supply Inboard Iso.	
18.	1B246011	HFB-M	150/30	270	NA	HVB311F023B - Reactor Recirc Pump Suction	
19.	1B229022	HFB-M	150/30	270	NA	HVB311F031B - Reactor Recirc Pump Disch	
20.	1B246022	HFB-M	150/5	40	NA	HVE511F007 - RCIC Inbrd Steam Line Isolation	
21.	1B246051	HFB-M	150/30	170	NA	1V417B - Drywell Area Unit Clr Fan	16
22.	1B246061	HFB-M	150/30	170	NA	1V414B - Drywell Area Unit Clr Fan	16
23.	1B229023	HFB-M	150/10	70	NA	HVB311F032B - RRP "B" Dsch Byps Vlv	
24.	1B246072	HFB-M	150/30	170	NA	1V415B - Drywell Area Unit Clr Fan	16
25.	1B246081	HFB-M	150/30	170	NA	1V416B - Drywell Area Unit Clr. Fan	16
26.	1B246091	HFB-M	150/30	160	NA	1V411B - Drywell Unit Clr. Fan	16
27.	1B246102	HFB-M	150/30	215	NA	1V413B - Drywell Area Unit Clr. Fan	16
28.	1B246103	HFB-M	150/30	170	NA	1V412B - Drywell Area Unit Clr. Fan	16
29.	1B246112	HFB-M	150/3	8	NA	HVB211F002 - Reactor Head Vent Valve	

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

CIRCUIT BREAKER LOCATION		TYPE*	FRAME RATING/UL	TRIP SET POINT (Amperes)	RESPONSE TIME (Milli-seconds/ Cycles)	SYSTEMS OR EQUIPMENT POWERED	
Type 2 Molded Case Circuit Breakers# (Continued)							16
30.	1B246113	HFB-M	150/3	8	NA	HVB211F005 - Reactor Head Vent Valve	24
31.	1B246062	HFB-M	150/3	20	NA	HV11346 - RBCCW Containment Iso. Vlv.	
32.	1B246012	HFB-M	150/3	18	NA	HV11345 - RBCCW Containment Iso. Vlv.	
33.	1B253063	HFB-M	150/5	18	NA	1P402A - Drywell Floor Drain Sump "A" Pump "A"	16 24
34.	1B253053	HFB-M	150/5	27	NA	HVG331F102 - Line Suction Inside Control Valve	
35.	1B263043	HFB-M	150/3	10	NA	HVG331F100 - RWCU Loop "1A" Suction	
36.	1B263053	HFB-M	150/3	12	NA	HVG331F106 - RWCU Loop "B" Suction	
37.	1B263081	HFB-M	150/3	11	NA	HVG331F101 - RWCU Sys Vessel Drain Line Recirc.	16
38.	1B263071	HFB-M	150/5	20	NA	1P4023 - Drywell Floor Drain Sump "A" Pump "B"	16
39.	1B253043	HFB-M	150/5	20	NA	1P403A - Drywell Floor Drain Sump "B" Pump "A"	
40.	1B263072	HFB-M	150/5	20	NA	1P403B - Drywell Floor Drain Sump "B" Pump "B"	16
41.	1B253021	HFB-M	150/50	480	NA	HVB211F011A Feedwater Inlet Shutoff Valve	
42.	1B263023	HFB-M	150/50	480	NA	HVB211F011B - Feed water Inlet Shutoff Valve	

*HFB-M - Westinghouse Type HFB, magnetic only
#Each location number represents two breakers in series.

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

CIRCUIT BREAKER LOCATION	TYPE**	FRAME RATING/UL	RESPONSE TIME (Milli-seconds/ Cycles)	SYSTEMS OR EQUIPMENT POWERED	
b. Type 3 Molded Case Circuit Breakers					16
1. 1B236103	KB-TM	250/150	NA	1E440C - Containment Recomb Elect. Htr. Ass'y.	16
2. 1B246033	KB-TM	250/150	NA	1E440D - Containment Recomb Elect. Htr. Ass'y.	16
3. 1B226103	KB-TM	250/150	NA	1E440B - Containment Recomb Elect. Htr. Ass'y.	16
4. 1B216092	KB-TM	250/150	NA	1E440A - Containment Recomb. Elect. Htr. Ass'y.	16
5. 1B253041	KB-TM	150/15	NA	TBB15 Pri. Cont. MSIV Iso. Hoist	
6. 1B263021	KB-TM	150/15	NA	TBB16 Pri. Cont. MSIV Iso. Hoist	

* TM - Westinghouse Type KB, Thermal-magnetic

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8.4.2-1 shall be bypassed continuously by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With thermal overload protection for one or more of the above required valves not bypassed continuously by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously by an OPERABLE integral bypass device by verifying that the thermal overload protection is bypassed:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection shall be verified to be bypassed following activities during which the thermal overload protection was temporarily placed in force.

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
HV-01222A	RHRSW
HV-01222B	RHRSW
HV-01224A1	RHRSW
HV-01224B1	RHRSW
HV-01224A2	RHRSW
HV-01224B2	RHRSW
HV-01112A	ESW
HV-01112B	ESW
HV-01122A	ESW
HV-01122B	ESW
HV-01112C	ESW
HV-01112D	ESW
HV-01122C	ESW
HV-01122D	ESW
HV-01110A	ESW
HV-01110B	ESW
HV-01120A	ESW
HV-01120B	ESW
HV-01110C	ESW
HV-01110D	ESW
HV-01120C	ESW
HV-01120D	ESW
HV-11210A	RHRSW
HV-11210B	RHRSW
HV-11215A	RHRSW
HV-11215B	RHRSW
HV-15766	Cont. Isol.
HV-15768	Cont. Isol.
HV-12603	Cont. Isol.
HV-11345	Cont. Isol.
HV-11313	Cont. Isol.
HV-11346	Cont. Isol.
HV-11314	Cont. Isol.
HV-E11-1F009	RHR
HV-E11-1F040	RHR
HV-G33-1F001	RWCU
HV-E11-1F103A	RHR
HV-E11-1F075A	RHRSW
HV-E11-1F048A	RHR
HV-E11-1F006C	RHR

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
HV-E11-1F004C	RHR
HV-E11-1F015A	RHR
HV-E11-1F024A	RHR
HV-E21-1F015A	CS
HV-E41-1F002	HPCI
HV-B21-1F016	NSSS
HV-E11-1F022	RHR
HV-E11-1F010A	RHR
HV-E11-1F011A	RHR
HV-E11-1F004A	RHR
HV-E11-1F006A	RHR
HV-E11-1F027A	RHR
HV-E11-1F007A	RHR
HV-E11-1F104A	RHR
HV-E11-1F026A	RHR
HV-E11-1F028A	RHR
HV-E11-1F047A	RHR
HV-E11-1F073A	RHRSW
HV-E11-1F003A	RHR
HV-E11-1F017A	RHR
HV-E21-1F001A	CS
HV-E21-1F031A	CS
HV-E21-1F004A	CS
HV-E21-1F005A	CS
HV-E11-1F021A	RHR
HV-E11-1F016A	RHR
HV-15112	RHR
HV-E51-1F007	RCIC
HV-E51-1F084	RCIC
HV-E11-1F027B	RHR
HV-E11-1F048B	RHR
HV-E11-1F015B	RHR
HV-E11-1F006B	RHR
HV-E11-1F021B	RHR
HV-E11-1F010B	RHR
HV-E11-1F011B	RHR
HV-E11-1F004B	RHR
HV-E11-1F007B	RHR
HV-E11-1F104B	RHR
HV-E11-1F026B	RHR

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
HV-E11-1F028B	RHR
HV-E11-1F047B	RHR
HV-E11-1F016B	RHR
HV-E11-1F003B	RHR
HV-E11-1F017B	RHR
HV-E21-1F031B	CS
HV-E21-1F001B	CS
HV-E11-1F103B	RHR
HV-E11-1F075B	RHRSW
HV-E11-1F073B	RHRSW
HV-E11-1F006D	RHR
HV-E11-1F004D	RHR
HV-E11-1F024B	RHR
HV-E21-1F015B	CS
HV-E21-1F004B	CS
HV-E21-1F005B	CS
HV-E32-1F001K	MSIV
HV-E32-1F002K	MSIV
HV-E32-1F003K	MSIV
HV-E32-1F001P	MSIV
HV-E32-1F002P	MSIV
HV-E32-1F003P	MSIV
HV-E32-1F001B	MSIV
HV-E32-1F002B	MSIV
HV-E32-1F003B	MSIV
HV-E32-1F001F	MSIV
HV-E32-1F002F	MSIV
HV-E32-1F003F	MSIV
HV-E32-1F006	MSIV
HV-E32-1F007	MSIV
HV-E32-1F008	MSIV
HV-E32-1F009	MSIV
HV-E51-1F045	RCIC
HV-E51-1F012	RCIC
HV-E51-1F013	RCIC
HV-15012	RCIC
HV-E51-1F046	RCIC
HV-E51-1F008	RCIC
HV-E51-1F031	RCIC
HV-E51-1F010	RCIC

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
HV-E51-1F019	RCIC
HV-E51-1F060	RCIC
HV-E51-1F059	RCIC
HV-E51-1F022	RCIC
HV-E51-1F062	RCIC
HV-E41-1F012	HPCI
HV-E41-1F001	HPCI
HV-E41-1F011	HPCI
HV-E41-1F006	HPCI
HV-E41-1F079	HPCI
HV-E41-1F059	HPCI
HV-E41-1F004	HPCI
HV-E41-1F003	HPCI
HV-E41-1F042	HPCI
HV-E41-1F075	HPCI
HV-E41-1F008	HPCI
HV-E41-1F007	HPCI
HV-E41-1F066	HPCI
HV-G33-1F004	RWCU
HV-B21-1F019	NSSS
HV-E11-1F008	RHR
HV-E11-1F023	RHR
HV-E11-1F049	RHR
HV-B31-1F032A	RX RECIRC
HV-B31-1F032BB	RX RECIRC
HV-B31-1F031A	RX RECIRC
HV-B31-1F031B	RX RECIRC

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REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring assembly to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring assembly to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring assemblies shall be determined OPERABLE:

*a. By performance of a CHANNEL FUNCTIONAL TEST when in COLD SHUTDOWN if not performed within the previous 6 months.

b. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints:

		RPS Division A	RPS Division B
1.	Overvoltage	$\leq$ 128.3 VAC	$\leq$ 129.5 VAC
2.	Undervoltage	$\geq$ 110.7 VAC	$\geq$ 111.9 VAC
3.	Underfrequency	$\geq$ 57 Hz	$\geq$ 57 Hz

*Prior to startup after the first refueling shutdown, this Surveillance Requirement is deleted and replaced as follows:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Fuel grapple position.
5. Service platform hoist fuel-loaded.

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. 14

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

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. |14|

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. |14|

The reactor mode switch may be placed in the Run or Startup/Hot Standby 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. |14|

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.

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. Suspension of movement of the control rod and/or associated control rod drive mechanism shall not preclude completion of the movement of the component to a safe conservative position.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel, are removed from the core cell.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements. Suspension of movement of control rods and/or control rod drive mechanisms shall not preclude completion of the movement of the components to a safe conservative position.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

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

SURVEILLANCE REQUIREMENTS

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

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*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange.

ACTION:

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

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

TABLE NOTATION

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8. 14 26

[d]A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

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 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which 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 the radiological impact on finished drinking water supplies at the nearest downstream drinking water source. 26

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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 (see Figure 5.1.3-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ. (Inhalation pathways only.)

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limits.

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 of the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, 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 of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2.1.2-1.

TABLE 4.11.2.1.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
B. Reactor Building Vents, Turbine Building Vents, and SGTS	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g,h]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. All Release Types as listed in A and B.	Continuous[f]	W[c,d] Charcoal Sample	I-131	$1x10^{-12}$
	Continuous[f]	W[c,d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	$1x10^{-11}$
	Continuous[f]	Q Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$ (XE-133 equivalent)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement systems (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 is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples)

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverfied theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATION

b. If the iodine or particulate monitoring channel(s) is(are) inoperative, analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Particulate and/or charcoal samples shall be analyzed when an alarm is received indicating rate of activity buildup exceeds 3 times normal

d. 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. If the iodine or particulate monitoring channel(s) is (are) inoperative, 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 completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

e. (Deleted)

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-135m 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 which are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8. 14 26

h. Under the provisions of footnote g. above, only noble gases need to be considered. 20

EXPLOSIVE GAS MIXTURE

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.

APPLICABILITY: Whenever the main condenser air ejector (evacuation) system is in operation.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main-condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3.3.7.11-1 of Specification 3.3.7.11.

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The radioactivity rate of the noble gases Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, and Xe-138 measured at the motive steam jet condenser discharge shall be limited to less than or equal to 330 millicuries/second.

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

ACTION:

With the gross radioactivity rate of the specified noble gases at the motive steam jet condenser discharge exceeding 330 millicuries/second, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases at the motive steam jet condenser discharge shall be continuously monitored in accordance with Specification 3.3.7.11.

4.11.2.7.2 The gross radioactivity rate of the specified noble gases from the motive steam jet condenser discharge 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:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Condenser Offgas Pre-Treatment Radioactivity 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.

*When the main condenser air ejector is in operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

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 radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. 26

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 shall be determined in accordance with the methodology and parameters in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report per Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 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)

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 6.9.1.8, 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). 14 26

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 parameters in the ODCM.

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; 2) An outer ring of stations, one in each meteorological sector, in the 3 to 9 mile range from the site; 3) The balance 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 radio activity analysis following filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE (Continued)			
	1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction		
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 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 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.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. INGESTION			
a. Milk	a. Samples from milking animals in 3 locations within 5 km distant having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[i]. 1 sample from milking animals at a control location (15-30 km distant and in the least prevalent wind direction).	Semimonthly when animals are on pasture, monthly at other times.	Gamma isotopic and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
b. Fish and Invertebrates	a. 1 sample of two recreationally important species in vicinity of plant discharge area. 1 sample of same species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis on edible portions.
c. Food Products	a. 1 sample of each principal class of food products from any area which is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest.	Gamma isotopic analysis on edible portions.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
c. Food Products (cont'd)	Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic[e] and I-131 analysis.

TABLE 3.12.1-1 (Continued)

Table Notation

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. 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, identify the cause of the unavilability 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). 14 26 14 26

[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]Methodology to guarantee complete recovery of radioiodine shall be described in the ODCM.

[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.

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 tuborous and root food products.

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400[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.

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/m³)	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			15		
La-140	15					

Table 4.12.1-1 (Continued)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable 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. |14 |26|

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[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.

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 approriate (as counts per minute),

E is the counting efficiency (as counts per disintegration),

B 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.

It should be recognized that the LLD is defined as 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

TABLE NOTATION

routine conditions. Occassionally background fluctuations, unavoidably small sample sizes, the presence of interferring 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 purusant to Specification 6.9.1.7. 14 26

[d]LLD for drinking water samples.

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1. 3.2.1-2 and 3.2.1-3.

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.

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. Incorporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.06 or that $\geq$ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Rod Block Monitor (RBM) portion of the control rod block instrumentation contains multiplexing circuitry which interfaces with the reactor manual control system. The RBM is a redundant system which includes two channels of information which must agree before rod motion is permitted. Each of these redundant channels has a self-test feature which is implicitly tested during the performance of surveillance pursuant to this specification as well as the control rod operability specification (3/4.1.3.1).

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

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.

3/4.4.1 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.

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984. 24

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 pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 10 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

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.

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.

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.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

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

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam line rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

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

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

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

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-2, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

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 1971 Edition and Addenda through 1972.

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

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (°F)	LRT_{NDT} MAX. * (°F)	MIN. UPPER SHELF (LFT-LBS)	RT_{NDT} MAX. (°F)
Plate	SA-533 GR B CL.1	C2433-1	0.10	0.009	+18	24	N/A	42
Weld	N/A	L311A27A-1	0.03	0.016	-50	30	N/A	-20

NOTE:* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA-533 GR B DL.1	C1232-2	+20
Bottom Head Dome	"	C9942-2	+34
Bottom Head Torus	"	C9942-2	+34
Top Head Dome	"	C9220-2	+10
Top Head Torus	"	C9355-1	+10
Top Head Flange	SA-508, CL.2	N/A	+10
Vessel Flange	"	N/A	+10
Feedwater Nozzle	"	Q2Q49W	-16
Weld	Per GE Purch. Spec.	No CNVS Available	0 (Per Purch. Spec. Requirements
Closure Studs	SA-540 GR B24	82552	+70

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS) is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR 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 automatic depressurization system (ADS).

The CSS 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 CSS 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. Two subsystems, each with two pumps, 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.

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

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5000 gpm at reactor pressures between 1157 and 150 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CS 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 HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

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

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls six selected safety-relief valves although the safety analysis only takes credit for five valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI, CSS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is also required by Specification 3.6.2.1.

Repair work might require making the suppression chamber 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 chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber 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 water volume is based on NPSH, recirculation volume, vortex prevention plus a safety margin for conservatism.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 45.0 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.

Operating experience with the main steam line isolation valves and main steam line drain valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation and drain valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

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 45.0 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression in chamber internal pressure ensure that the containment peak pressure of 45.0 psig does not exceed the design pressure of 53 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 2.0 psig for initial positive containment pressure will limit the total pressure to 45.0 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. Until these valves have been demonstrated capable of closing during a LOCA or steam line break accident, they shall be blocked so as not to open more than 50°. Until these valves have been demonstrated capable of closing within the times assumed in the safety analysis, they shall not be open more than 90 hours in any consecutive 365 days.

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.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 53 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1055 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 53 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 45.0 psig which is below the design pressure of 53 psig. Maximum water volume of 133,540 ft^3 results in a downcomer submergence of 12 feet and the minimum volume of 122,410 ft^3 results in a submergence approximately 24 inches less. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 128°F immediately following blowdown which is below the 170°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified 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.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are five pairs of valves to provide redundancy so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet 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.

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.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1.3-1a and 5.1.3-1b.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined reinforced concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in the shape of a truncated cone on top of a water filled suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 239,600 cubic feet. The suppression chamber has an air region of 148,590 cubic feet and a water region of 142,160 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 53 psig.

b. Maximum internal temperature: drywell 340°F.
suppression chamber 220°F.

c. Maximum external pressure 5 psig.

d. Maximum floor differential pressure: 28 psid, downward.
5.5 psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, the Reactor Building recirculation fan room, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,650,000 cubic feet.

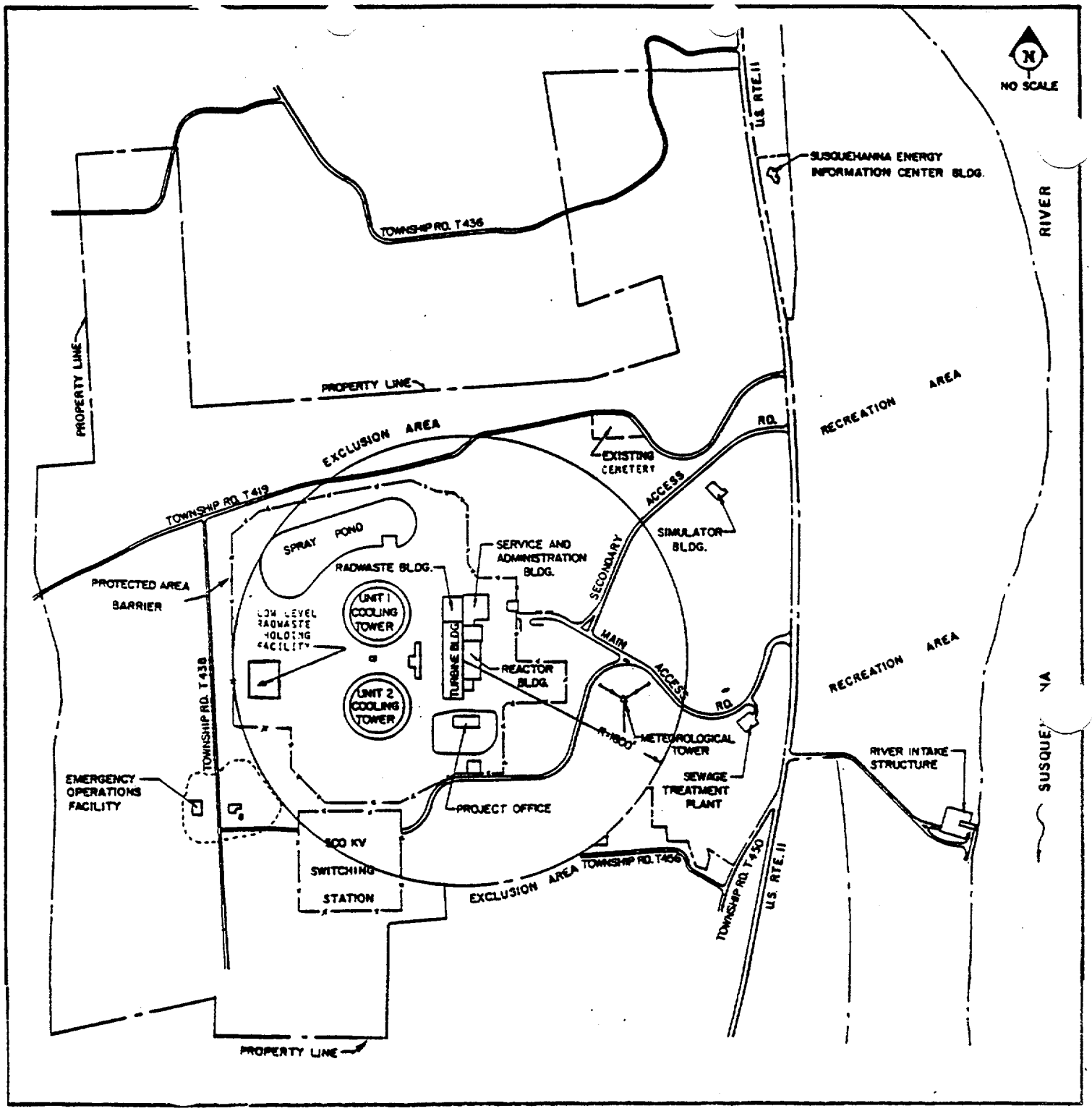

FIGURE 5.1.1-1

EXCLUSION AREA

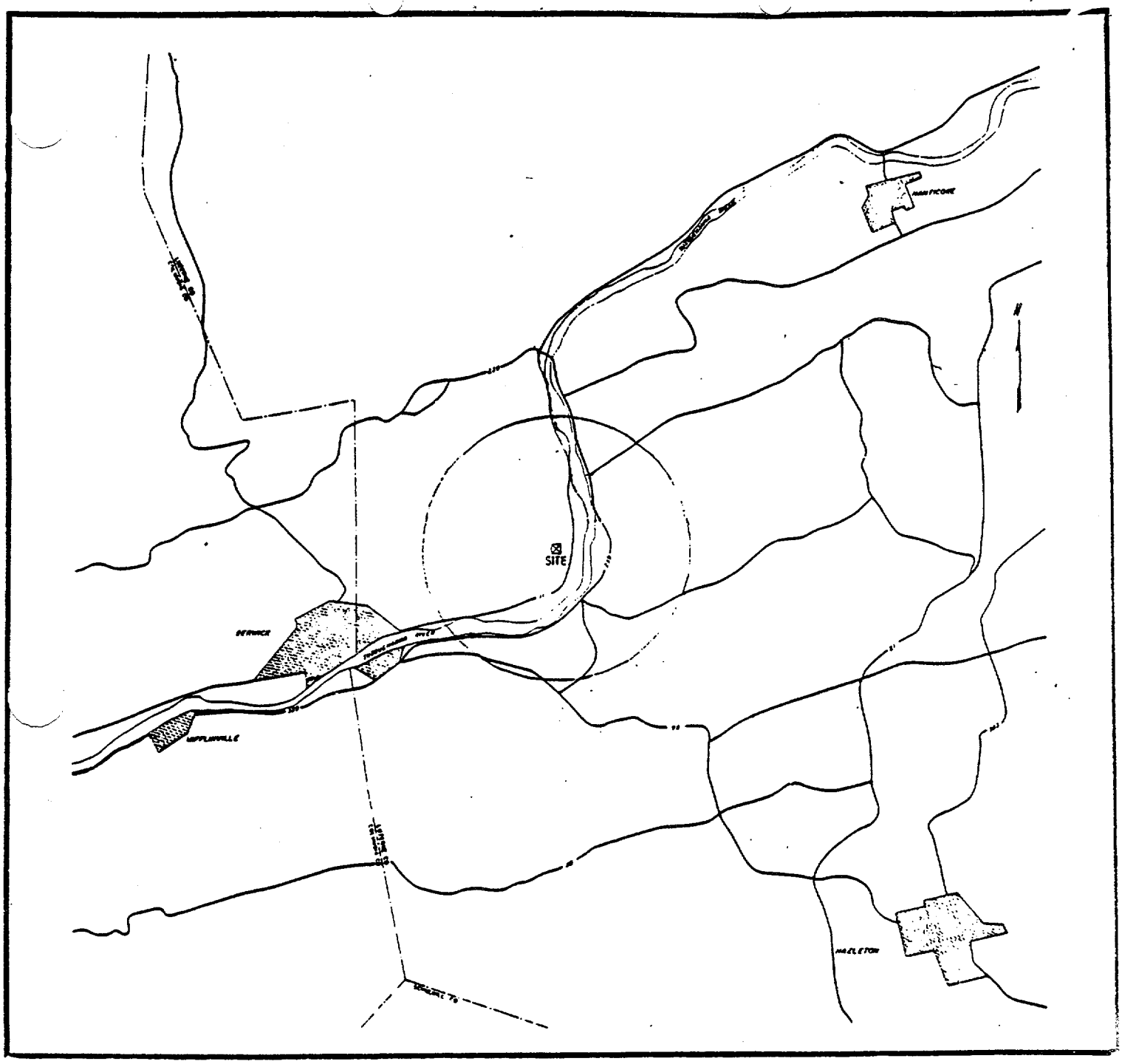

FIGURE 5.1.2-1

LOW POPULATION ZONE

(3-mile radius)

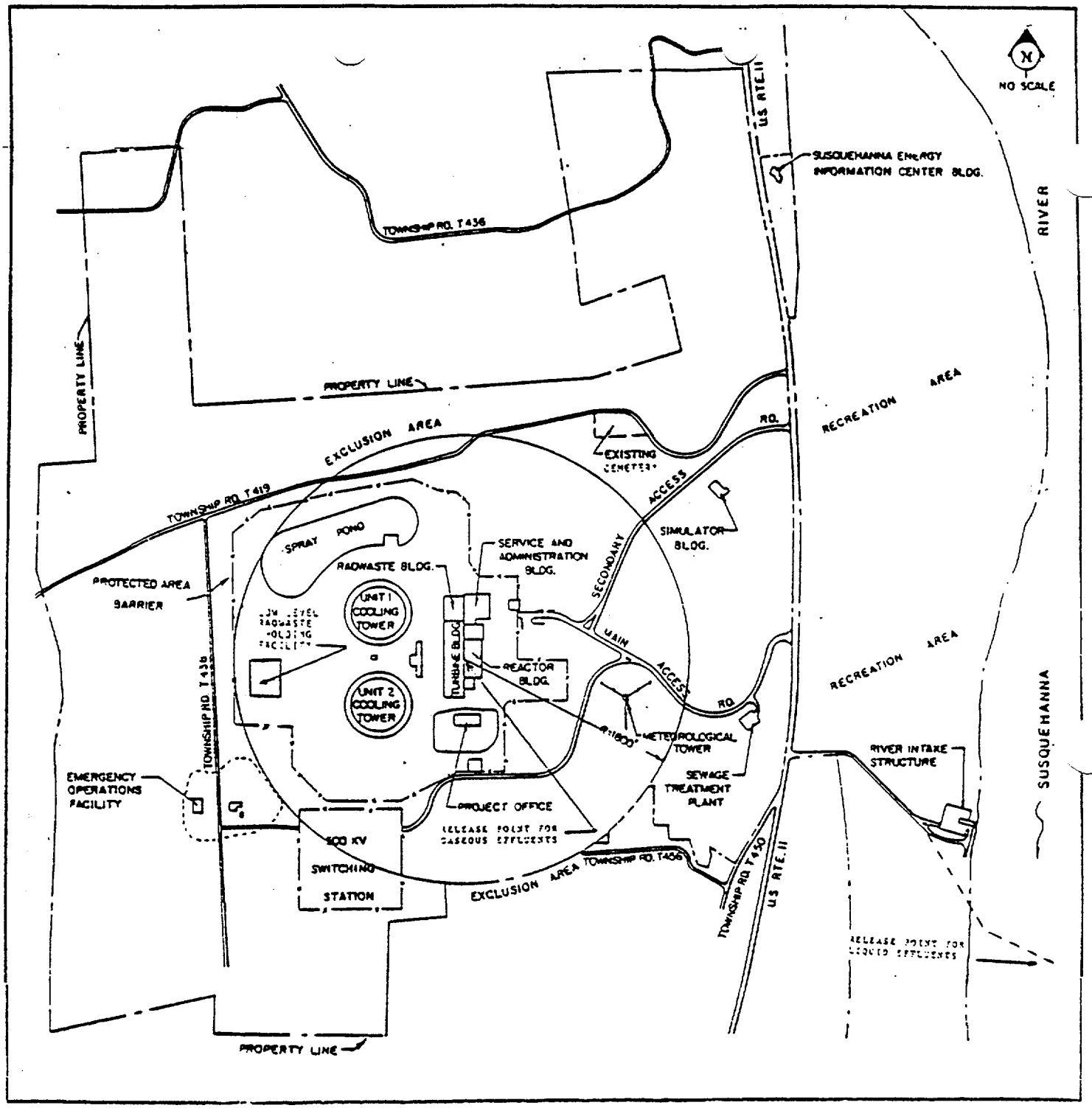

FIGURE 5.1.3-1a

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

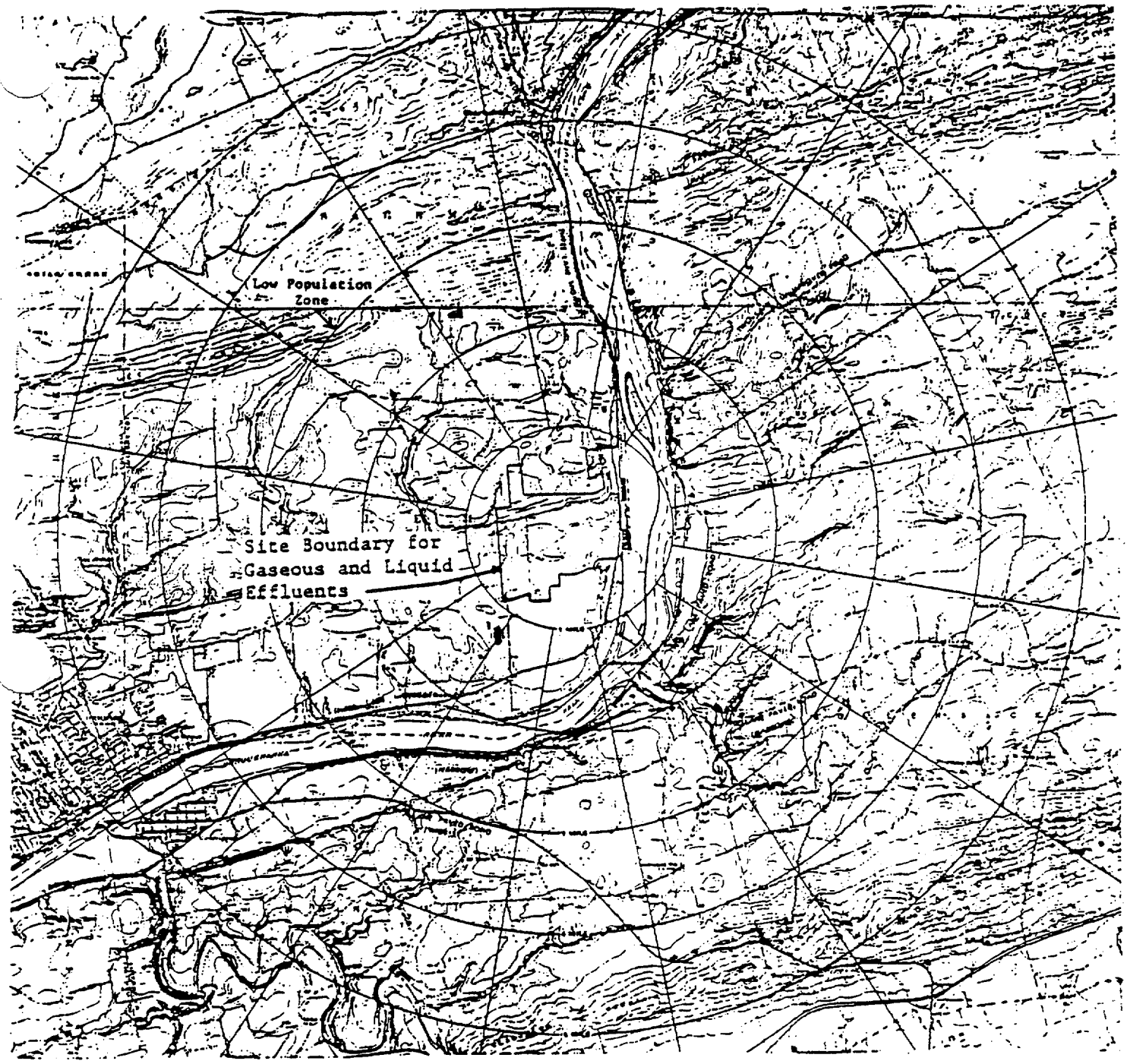

FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 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 3.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 528°F.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Superintendent of Plant - Susquehanna shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor or, during his absence from the Control Room, a designated individual, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Senior Vice President - Nuclear 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 Reactor Operator assigned to and qualified on that unit shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator qualified on this unit shall be in the Control Room. This individual may be qualified on both units and be serving in this capacity on both units.

c. A health physics technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include the Shift Supervisor and the 2 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.

*The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

1. A 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 Superintendent of Plant-Susquehanna 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 Superintendent of Plant-Susquehanna or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH COMMON CONTROL ROOM

With both units in CONDITION 1, 2 or 3

Position	Number of Individual Required
SS	1
SRO[a]	1
RO[b]	3
NLO	3
STA	1

With one unit in CONDITION 1, 2, or 3 and one unit in CONDITION 4 or 5 or Defueled

Position	Number of Individual Required
SS	1
SRO[a]	1
RO[b]	3
NLO	3
STA	1

With both units in CONDITION 4 or 5

Position	Number of Individual Required
SS	1
SRO	0
RO	2
NLO	3
STA	0

TABLE NOTATION

(a) - Individual filling this position shall be in the control room at all time except when relieved by an individual holding a Senior Reactor Operators License, who shall then provide the required SRO presence.

(b) - Individuals acting as relief operator shall hold a license for both units. Otherwise, provide a relief operator for each unit who holds a license for the unit assigned.

SS - Shift Supervisor with a Senior Reactor Operators License for each unit whose reactor contains fuel.

SRO - Individual with a Senior Reactor Operators License for each unit whose reactor contains fuel. Otherwise, provide an individual for each unit who holds a Senior Reactor Operators License for the unit assigned.

TABLE NOTATION (Continued)

RO - Individual with a Reactor Operators License or a Senior Reactor Operators License for unit assigned. One RO shall be assigned to each unit whose reactor contains fuel and one RO shall be assigned as relief operator for unit(s) in CONDITION 1, 2 or 3.

NLO - Non licensed operator properly qualified to support the unit to which assigned.

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 NUCLEAR SAFETY ASSESSMENT GROUP (NSAG)

FUNCTION

6.2.3.1 The NSAG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The NSAG shall be composed of at least five dedicated, full-time engineers with at least three located onsite, each with a bachelor's degree in engineering or related science and at least two years professional level experience in his field, at least one year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The NSAG 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.

AUTHORITY

6.2.3.4 The NSAG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Senior Vice President-Nuclear.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure 1 of the October 30, 1979 NRC letter to all operating nuclear power plants.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager - Nuclear Training, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section 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.

*Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Superintendent of Plant-Susquehanna on matters related to nuclear safety as described in Specification 6.5.1.6.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Superintendent of Plant-Susquehanna
Member:	Assistant Superintendent of Plant-Susquehanna
Member:	Supervisor of Operations
Member:	Technical Supervisor
Member:	Supervisor of Maintenance
Member:	I&C/Computer Supervisor
Member:	Reactor Engineering Supervisor or Unit Reactor Engineer
Member:	Health Physics Supervisor
Member:	Shift Supervisor or Unit Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of all administrative procedures and changes thereto.

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. Review of the safety evaluations for procedures and changes thereto completed under the provisions of 10 CFR 50.59.

f. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President-Nuclear Operations and to the Susquehanna Review Committee.

g. Review all REPORTABLE EVENTS. 26

h. Review of unit operations to detect potential nuclear safety hazards.

i. Performance of special reviews, investigations or analyses and reports thereon as requested by the Superintendent of Plant-Susquehanna or the Susquehanna Review Committee.

j. Review of the Security Plan and shall submit recommended changes to the Susquehanna Review Committee.

k. Review of the Emergency Plan and shall submit recommended changes to the Susquehanna Review Committee.

l. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice-President-Nuclear Operations and to the Chairman of the Susquehanna Review Committee.

m. Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend in writing to the Superintendent of Plant-Susquehanna approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (f) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President-Nuclear Operations and the Susquehanna Review Committee of disagreement between the PORC and the Superintendent of Plant- Susquehanna; however, the Superintendent of Plant-Susquehanna shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Vice President-Nuclear Operations and the Susquehanna Review Committee.

6.5.2 SUSQUEHANNA REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The SRC shall be composed of nine individuals who shall meet or exceed the requirements of ANSI 3.1-1981, Section 4.7. The Chairman, Vice Chairman and all members shall be appointed in writing by the Senior Vice President-Nuclear.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

MEETING FREQUENCY

6.5.2.5 The SRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The quorum of the SRC necessary for the performance of the SRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four SRC 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 SRC shall be responsible for the review of:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

REVIEW (Continued)

d. Proposed changes to Appendix A 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. 26

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.

i. Reports and meetings minutes of the PORC.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Appendix A 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 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months. 11

f. The Security Plan and implementing procedures at least once per 12 months. 11

g. Any other area of unit operation considered appropriate by the SRC or the Senior Vice President-Nuclear.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

AUDITS (Continued)

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 procedures 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

6.5.2.9 The SRC shall report to and advise the Senior Vice President-Nuclear on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the Senior Vice President-Nuclear within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President-Nuclear within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President-Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

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 Superintendent of Plant -Susquehanna, 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 Superintendent of Plant - Susquehanna shall approve all plant procedures, programs, and changes thereto.

b. Individuals responsible for reviews performed in accordance with item 6.5.3.1.a above shall be members of the plant staff previously designated by Superintendent of Plant - Susquehanna. 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 Superintendent of Plant - Susquehanna approval for implementation.

d. Written records of reviews performed in accordance with item 6.5.3.1.a 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 26

6.6.1 The following actions shall be taken for REPORTABLE EVENTS.

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and 26

b. Each REPORTABLE EVENT shall be reviewed by the PORC and submitted to the SRC and the Vice President-Nuclear Operations.

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-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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 SRC and the Vice President-Nuclear Operations 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. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. Offsite Dose Calculation Manual implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance of Regulatory Guide 4.15, February 1979.

6.8.2 Each procedure of 6.8.1(a) through (g) above, and changes thereto, shall be reviewed in accordance with Specifications 6.5.1.6 or 6.5.3, as appropriate, and approved by the Superintendent of Plant-Susquehanna prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

Each procedure of 6.8.1, above, and changes thereto, that is established to implement those portions of the radiological effluent and environmental monitoring programs and those portions of the ODCM that are the responsibility of the Nuclear Support Group shall be reviewed by the Environmental Group Supervisor-Nuclear and approved by the Manager-Nuclear Support.

PROCEDURES AND PROGRAMS (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed in accordance with Specification 6.5.1.6 or 6.5.3, as appropriate, and approved by the Superintendent of Plant-Susquehanna 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 core spray, high pressure coolant injection, reactor core isolation cooling, reactor water cleanup, standby gas treatment, scram discharge and containment air monitoring systems.

The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel.
2. Procedure for sampling and analysis,
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

26

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, unless otherwise noted.

STARTUP REPORTS

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 startup tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation, supplementary reports shall be submitted at least every three 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 manrem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

*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.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report. 11

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 PORC.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 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. 26

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate), with preoperational studies, operational controls and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 3.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

*A single submittal may be made for a multiple unit station.
**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality. 14 26

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction and atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, 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 assessment of radiation doses shall be performed in accordance with the methodology and parameters of the Offsite Dose Calculation Manual (ODCM).

*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.

**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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT* (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBERS OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each type of solid waste (as defined in 10 CFR PART 61) shipped offsite during the report period:

1. Container volume,
2. Total curie quantity (specify whether determined by measurement or estimate),
3. Principal radionuclides (specify whether determined by measurement or estimate),
4. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),
5. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
6. Solidification agent or absorbent (e.g., cement; urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS. 26

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.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

RECORD RETENTION (Continued)

h. Records of in-service 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 PORC and the SRC and records of reviews conducted in accordance with Specification 3.5.3.

l. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

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 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.

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 level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.12 HIGH RADIATION AREA (Continued)

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 unit Health Physics Supervision in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the unit Health Physicist. 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 one 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, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

*Measurement made at 18" from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

2. Shall become effective upon review and acceptance by the PORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the Manager-Nuclear Support.

2. Shall become effective upon review and acceptance by the Manager-Nuclear Support.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste systems, liquid, gaseous and solid:

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the (PORC). The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

2. Shall become effective upon review and acceptance by the PORC.

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

SAFETY EVALUATION

AMENDMENT NO. 29 TO NPF-14

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-387

Introduction

The licensee has proposed changes to the Technical Specifications of the operating license for Susquehanna Steam Electric Station, Unit 1 to reflect the administrative improvements already present in the Unit 2 Technical Specifications.

Evaluation

The changes in the Unit 1 Technical Specifications (TS) with regard to this amendment are administrative in nature. They provide clarity and correctness to the TS's. Changes, which are not obviously administrative relate to updating the Technical Specifications for correctness and to reflect two unit operation such as:

1) Page 3/4 2-7, in which action statement b is added, this addition corrects an omission in the Unit 1 Technical Specifications.

2) Page 3/4 8-22, in which section 3.8.4.1 was rewritten, the basic Technical Specification has been changed such that the action statements have been separated and clarified in order to relieve the need for operator interpretation of the required action for a specific case.

3) Page 3/4 8-22, action statement b, in which the change provides clarification that the removal of the inoperable device is equivalent to racking out the same device. Additionally, the change permits tripping, racking or removal of an alternative device to protect primary containment penetration conductors. No safety analysis is affected by this change since protection of the containment penetration conductor is maintained. This change is consistent with existing controls and remains within the bases for protection of the primary containment penetration conductors.

4) Page 6-5 thru 6-6 have been updated to reflect the present two-unit operation. Originally this Technical Specification was written for the operation of Unit 1 only. Since the licensing of Unit 2 it is necessary and correct to update these pages.

5) Page 3/4 6-6. This change provides operational flexibility yet achieves the same level of assurance that interlocks are tested. This change avoids the unnecessary testing of the interlocks when the Unit is down for short periods of time and entry is not required.

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The NRC staff finds these changes to be appropriate and acceptable, therefore recommending their approval.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: FEB 06 1985

FEB 06 1985

AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-14 - SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

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