August 30, 1988

Docket Nos. 50-387/388

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RDiggs, ARM/LFMB	Brent Clayton

Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: ADMINISTRATIVE CHANGES TO UPDATE THE TECHNICAL SPECIFICATIONS (TAC NOS. 64694/64695)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. NPF-14 and Amendment No. 50 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, (SSES) Units 1 and 2. These amendments are in response to your letter dated August 5, 1986.

These amendments revise the SSES Technical Specifications to correct errors, achieve consistency, change nomenclature, and delete previously dated requirements which have been completed. Some of the changes requested in your August 5, 1986 application were included in other amendments which were issued subsequent to your application. This Amendment contains the remaining requested changes and some editorial changes.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 82 to License No. NPF-14
2. Amendment No. 50 to License No. NPF-22
3. Safety Evaluation

cc w/enclosures:
See next page

PDI-2/LA	PDI-2/PM	PDI-2/D	OGC
MO'Brien	MThadani:mr	WButler	
8/18/88	8/18/88	8/30/88	4/22/88

m.m.

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

August 30, 1988

Docket Nos. 50-387/388

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
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cc w/enclosures:
See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

cc:
Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street N.W.
Washington, D.C. 20037

Bryan A. Snapp, Esq.
Assistant Corporate Counsel
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. E. A. Heckman
Licensing Group Supervisor
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. F. I. Young
Resident Inspector
P.O. Box 52
Shickshinny, Pennsylvania 18655

Mr. R. J. Benich
Services Project Manager
General Electric Company
1000 First Avenue
King of Prussia, Pennsylvania 19406

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Resources
Commonwealth of Pennsylvania
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Mr. Jesse C. Tilton, III
Allegheny Elec. Coorperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, Pennsylvania 17108-1266

Mr. W. H. Hirst, Manager
Joint Generation
Projects Department
Atlantic Electric
P.O. Box 1500
1199 Black Horse Pike
Pleasantville, New Jersey 08232

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

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. 82
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 August 5, 1986 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. 82 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 82

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 change. The overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
iii	iii
iv	iv*
v	v*
vi	vi
ix	ix*
x	x
xi	xi*
xii	xii
xiii	xiii
xiv	xiv*
xv	xv*
xvi	xvi
xxiii	xxiii*
xxiv	xxiv
3/4 2-1	3/4 2-1*
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12*
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20

3/4 3-97	3/4 3-97*
3/4 3-98	3/4 3-98
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2*
3/4 6-9	3/4 6-9*
3/4 6-10	3/4 6-10
3/4 6-13	3/4 6-13
3/4 6-14	3/4 6-14*
3/4 6-31	3/4 6-31
3/4 6-31a	3/4 6-31a*
3/4 6-35	3/4 6-35
3/4 6-36	3/4 6-36*
3/4 6-37	3/4 6-37*
3/4 6-38	3/4 6-38
3/4 6-39	3/4 6-39
3/4 7-1	3/4 7-1*
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3*
3/4 7-4	3/4 7-4
3/4 7-21	3/4 7-21*
3/4 7-22	3/4 7-22
3/4 8-7	3/4 8-7*
3/4 8-8	3/4 8-8
3/4 8-9	3/4 8-9
3/4 8-10	3/4 8-10*
3/4 8-33	3/4 8-33
3/4 9-1	3/4 9-1
3/4 9-2	3/4 9-2*
3/4 10-3	3/4 10-3*
3/4 10-4	3/4 10-4
3/4 10-5	3/4 10-5

3/4 11-19	3/4 11-19
3/4 11-20	3/4 11-20*
B 3/4 1-3	B 3/4 1-3*
B 3/4 1-4	B 3/4 1-4
B 3/4 1-5	-
-	-
B 3/4 10-1	B 3/4 10-1
-	-
5-1	5-1*
5-2	5-2
5-3	5-3*
5-4	5-4
6-15	6-15*
6-16	6-16

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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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 for GE fuel and AVERAGE BUNDLE EXPOSURE for ANF fuel 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:

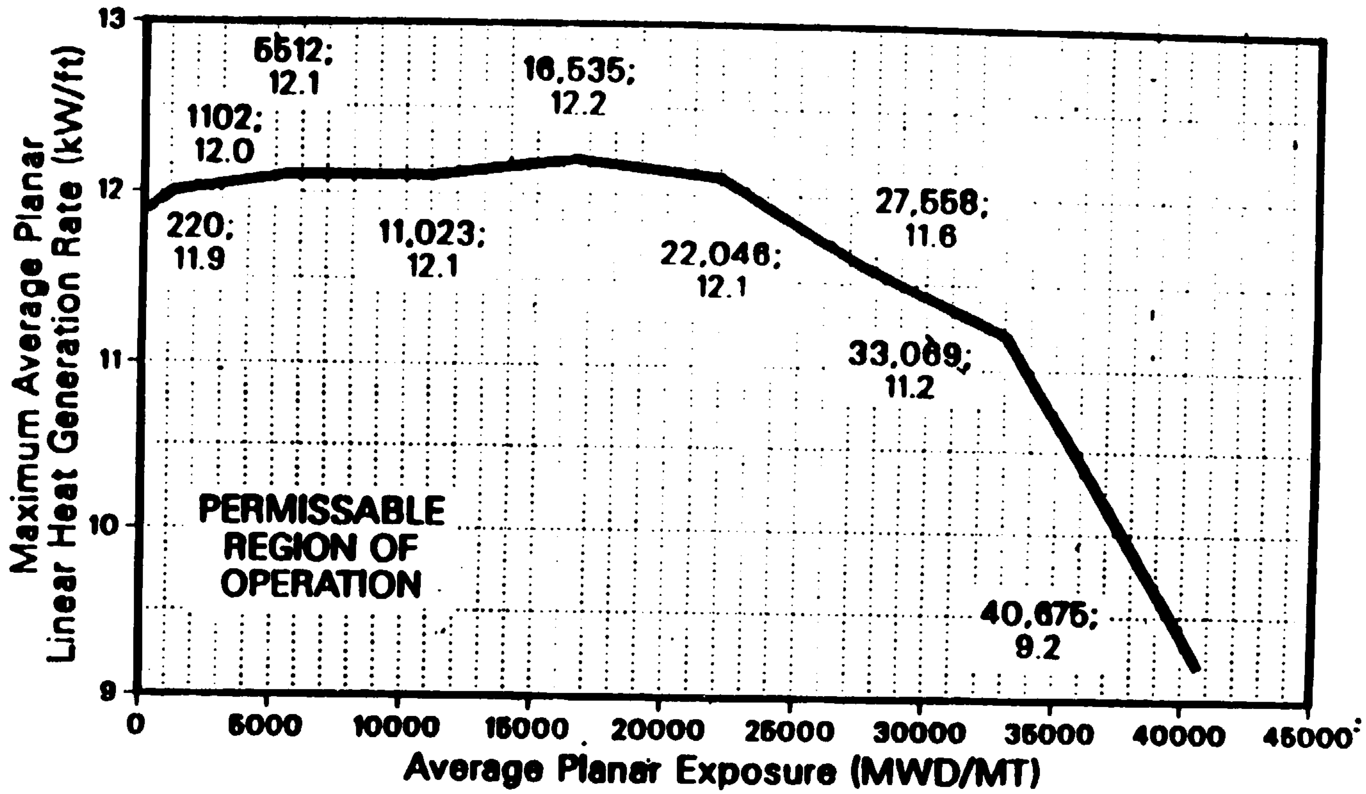
a. At least once per 24 hours,

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

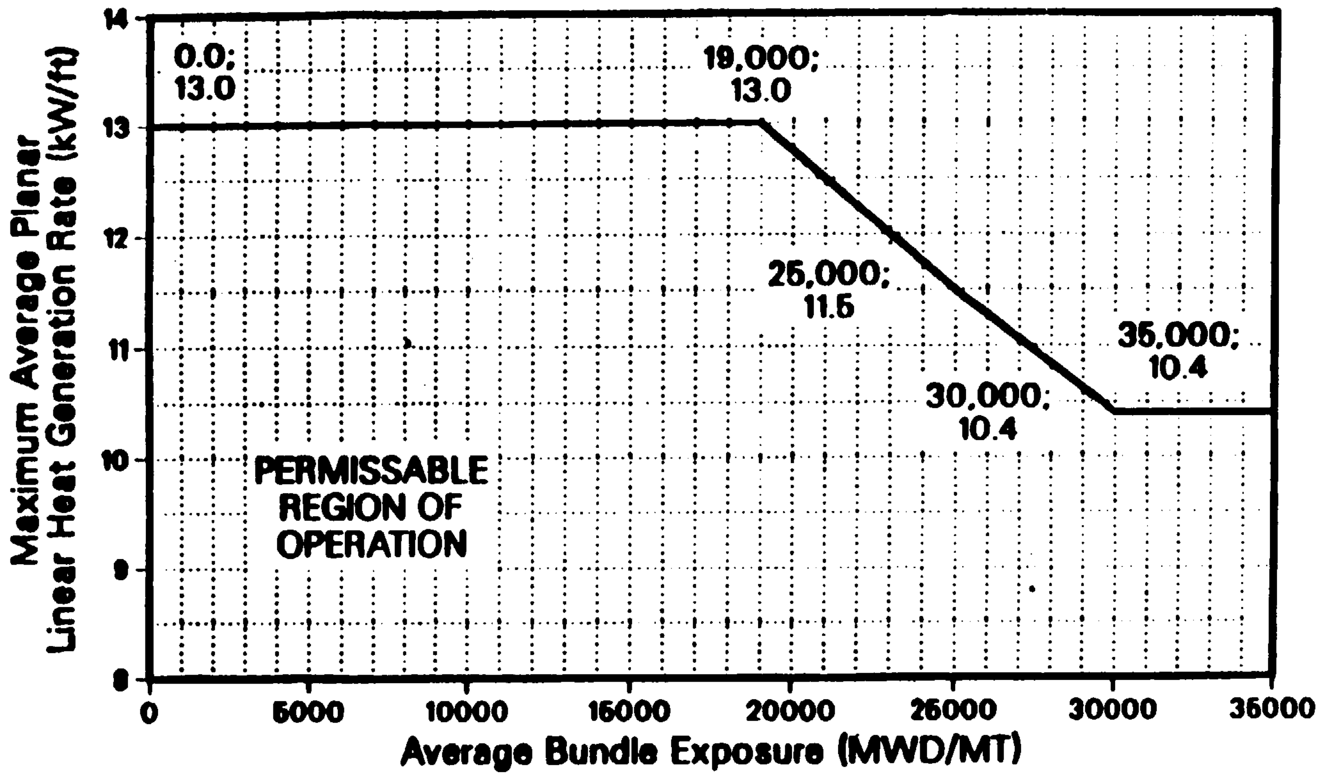
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

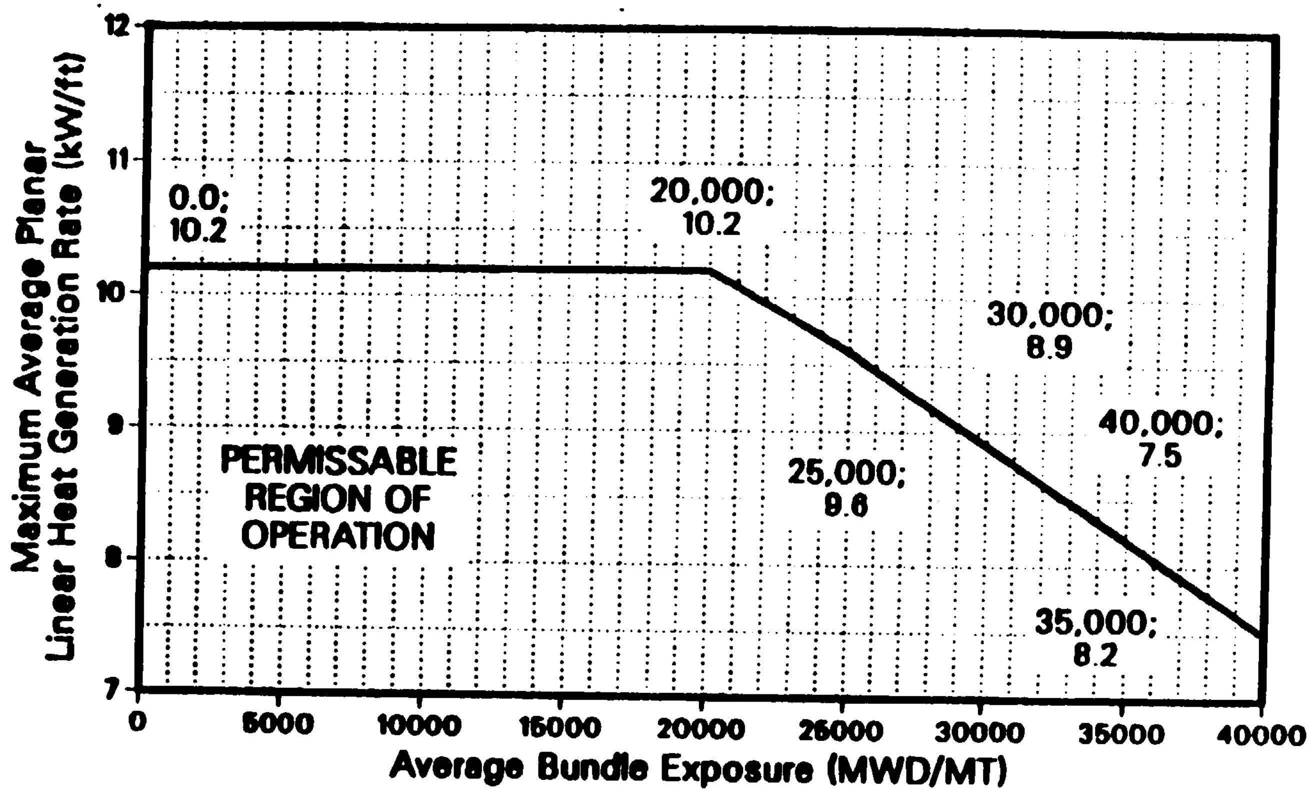
*See Specification 3.4.1.1.2.a for single loop operation requirements.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR233 (2.33% ENRICHED)
FIGURE 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.1-2



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.1-3

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.*

(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	
a. Level Transmitter	NA
b. Float Switch	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.

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
d. SGTS Exhaust Radiation-High	R	1	1, 2, 3,4***,5***	20
e. Main Steam Line Radiation-High	C	2	1, 2, 3	20
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	**	2	1, 2, 3 and *	25
b. Drywell Pressure - High	**	2	1, 2, 3	25
c. Refuel Floor High Exhaust Duct Radiation - High	**	2	*	25
d. Railroad Access Shaft Exhaust Duct Radiation - High	**	1	*	25
e. Refuel Floor Wall Exhaust Duct Radiation - High	**	2	*	25
f. Manual Initiation	NA	1	1, 2, 3 and *	24

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, Low, Level 1	X	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. Reactor Building Main Steam Line Tunnel Temperature - High	E	2	1, 2, 3	21
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	E	2	1, 2, 3	21
h. Manual Initiation	NA	1	1, 2, 3	24
i. Turbine Building Main Steam Line Tunnel Temperature-High	E	2	1, 2, 3	21
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	J	1	1, 2, 3	23
b. RWCU Area Temperature - High	W	3	1, 2, 3	23
c. RWCU Area Ventilation Δ Temp. - High	W	3	1, 2, 3	23
d. SLCS Initiation	I	2	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
g. Manual Initiation	NA	1	1, 2, 3	24

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	$\geq$ 13.0 inches*	$\geq$ 11.5 inches
2) Low Low, Level 2	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
3) Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches
b. Drywell Pressure - High	$\leq$ 1.72 psig	$\leq$ 1.88 psig
c. Manual Initiation	NA	NA
d. SGTS Exhaust Radiation - High	$\leq$23.0 mR/hr	$\leq$31.0 mR/Hr
e. Main Steam Line Radiation - High	$\leq$ 7.0 x full power background	$\leq$8.4 x full power background
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
b. Drywell Pressure - High	$\leq$ 1.72 psig	$\leq$ 1.88 psig
c. Refuel Floor High Exhaust Duct Radiation - High	$\leq$ 2.5 mR/hr.	$\leq$ 4.0 mR/hr.
d. Railroad Access Shaft Exhaust Duct Radiation - High	$\leq$ 2.5 mR/hr.	$\leq$ 4.0 mR/hr.
e. Refuel Floor Wall Exhaust Duct Radiation - High	$\leq$ 2.5 mR/hr.	$\leq$ 4.0 mR/hr.
f. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches
b. Main Steam Line Radiation - High	$\leq$ 7.0 X full power background	$\leq$ 8.4 X full power background
c. Main Steam Line Pressure - Low	$\geq$ 861 psig	$\geq$ 841 psig
d. Main Steam Line Flow - High	$\leq$ 107 psid	$\leq$ 110 psid

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. Reactor Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	≤ 99°F	≤ 108°F
h. Manual Initiation	NA	NA
i. Turbine Building Main Steam Line Tunnel Temperature-High	≤177°F	≤184°F
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 3.3.2-2 (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
j. Drywell Pressure - High	≤1.72 psig	≤1.88 psig
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 350 inches H_2O	≤ 367 inches H_2O
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##

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
h. HPCI Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤1.72 psig	≤1.88 psig
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY 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 Equipment Area Temperature - High	≤ 167°F	≤ 170.5°F
e. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
f. Manual Initiation	NA	NA
g. Drywell Pressure - High	≤1.72 psig	≤1.88 psig

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

#Lower setpoints for TSH-G33-N600 E, F and TSH-G33-N602 E, F.

##15 minute time delay.

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High	$\leq$ 54.0 inches	$\leq$ 55.5 inches

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel Water Level-High	D	M	R	1

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

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

1. Two OPERABLE CSS pumps, and

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

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of two subsystems with each subsystem comprised of:

1. Two OPERABLE LPCI pumps, and

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

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

1. One OPERABLE HPCI pump, and

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

d. The automatic depressurization system (ADS) with six OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*,**,#, and 3*,**,##.

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

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

#See Special Test Exception 3.10.5.

##One LPCI subsystem of the RHR system may be inoperable in that it is aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least one LPCI pump in each LPCI subsystem is OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

b. For the LPCI system:

1. With one LPCI pump in either or both LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump(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.

2. With no LPCI system cross-tie valve closed or with power not removed from the closed cross-tie valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

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

4. With both LPCI subsystems otherwise inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the CSS, the LPCI system, the ADS and the RCIC system are OPERABLE:

1. With the HPCI system inoperable, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 150 psig within the following 24 hours.

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

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -1.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the arithmetical average of the higher temperature at a minimum of 3 of the following areas and shall be determined to be within the limit at least once per 24 hours:

	Area	Elevation	Azimuth
a.	Top	797'8"	110°, 295°
b.	Middle	752'2"	90°, 270°
c.	Bottom	737'	150°, 300°
d.	Pedestal	711' or 720'	270°, 85°

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

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

c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators covering at least six locations 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 less than one suppression pool water temperature indicator at at least six different locations OPERABLE, restore at least one water level indicator and at least one water temperature indicator at at least six different locations to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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.

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.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT** INTEGRITY shall be maintained.
APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT** INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT** INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition*, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT** INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

1a. When the railroad bay door (No. 101) is closed; all Zone I and III hatches, removable walls, dampers, and doors connected to the railroad access bay are closed,## or

i) Only Zone I removable walls and/or doors are open to the railroad access shaft,## or

ii) Only Zone III hatches and/or dampers are open to the railroad access shaft.##

1b. When the railroad bay door (No. 101) is open; all Zone I and III hatches, removable walls, dampers, and doors connected to the railroad access bay are closed.

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

**Secondary Containment consists of Zone I, Zone II and Zone III or Zone I and Zone III when Zone II is isolated from Zone I and Zone III. During operational condition* when no operations with a potential for draining the reactor vessel are being performed, secondary containment may consist of Zone III as long as Zone I is isolated from Zone III.

##Personnel ingress and egress through doors within the secondary containment is not prohibited by this specification.

SURVEILLANCE REQUIREMENTS (Continued)

2a. At least one door in each access to the secondary containment zones is closed.

2b. At least one door in each access between secondary containment zones is closed.*

3. All secondary containment penetrations** not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

4. The truck bay hatch is closed.
5. The truck bay door (No. 102) is closed unless Zone II is isolated from Zones I and III.

c. At least once per 18 months:

1. For three zone operation with Zone II OPERABLE:

a. Verifying that one standby gas treatment subsystem will draw down the secondary containment (Zone I and Zone III) to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 15 seconds, and

b. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate of less than or equal to 2885 cfm from Zone I and Zone III, and

c. Verifying by calculation that one standby gas treatment subsystem will maintain greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate of less than or equal to 4000 cfm from Zone I, Zone II, and Zone III, or

2. For three zone operation:

a. Verifying that one standby gas treatment subsystem will draw down the secondary containment (Zone I, Zone II and Zone III) to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 92 seconds, and

*Personnel ingress and egress through doors within the secondary containment is not prohibited by this specification.
**Penetration between secondary containment zones, penetrations to no-zones, and penetrations to the outside atmosphere.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 10,100 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 10,100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 13 inches Water Gauge while operating the filter train at a flow rate of 10,100 cfm ± 10%.

   2. Verifying that the filter train starts and associated dampers open on each of the following test signals:

      a. Manual initiation from the control room, and

      b. Simulated automatic initiation signal.

   3. Verifying that the filter cooling bypass and outside air dampers open and the fan start on filter cooling initiation.

   4. Verifying that the temperature differential across each heating coil is $\geq$ 17°F when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 10,100 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a hydrogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 10,100 cfm ± 10%.

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

3.6.6.2 Drywell unit cooler fans 1V414 A&B, 1V416 A&B and recirculation fans 1V418 A&B shall be OPERABLE at low speed.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

b. With both fans in any pair inoperable at low speed, follow the requirements of Specification 3.0.3.

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each of the fans 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.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

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.

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*May not be a pump required for Unit 2.
**See Specification 3.9.11.1 and 3.9.11.2 for applicability.
***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.

PLANT SYSTEMS

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 at 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 otherwise 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. In OPERATIONAL CONDITION 4, 5 or *:
   1. With one pump in an emergency service water system loop inoperable, verify adequate cooling capability remains available for the diesel generators required to be operable by Specification 3.8.1.2 or declare the affected diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2.
   2. With two pumps in an emergency service water system loop inoperable or with the loop otherwise inoperable declare the associated safety related equipment inoperable (except diesel generators), and follow the applicable ACTION statements. Verify adequate cooling remains available for the diesel generators required to be operable by Specification 3.8.1.2 or declare the affected diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2.

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*When handling irradiated fuel in the secondary containment.

#When any diesel generator is removed from service in order to do work associated with tying in the additional diesel generator and its associated emergency service water pump is inoperable, Action a.1 shall read as follows:

a.1 With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status when its associated diesel generator is restored to OPERABLE status per Specification 3.8.1.1.

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. 29

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

c. At least once per 18 months by verifying that each automatic valve properly cycles to its proper position in its required time following receipt of an automatic pump start signal.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE.

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

ACTION:

a. With the groundwater level at any spray pond area observation well greater than or equal to 663' Mean Sea Level (MSL), 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 high groundwater level and the plans for restoring the level to within the limit.

b. With the spray pond otherwise inoperable:

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

   2. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

   3. In Operational Condition*, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE by verifying:

a. The average water temperature, which shall be the arithmetical average of the spray pond water temperature at the surface, mid and bottom levels, to be less than or equal to 88°F at least once per 24 hours.

b. The water level at the overflow weir is greater than or equal to 678'1" MSL USGS, at least once per 12 hours.

c. The groundwater level at observation wells 1, 3, 4, 5, 6, and 1113 to be less than 663' MSL at least once per 31 days.

*When handling irradiated fuel in the secondary containment.

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. South Cable Chase
f. Center Cable Chase
g. North Cable Chase
h. Room C-411 Soffit
i. Control Room Soffit, Unit 1
j. Control Room Soffit, Unit 2
k. Room C-412 Soffit

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 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 25% 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:

1U700	1U701	1U702	1U703	1U704	1U705
1U706	1U730	1U731	1U732		

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.

b. The provisions of Specification 3.0.3 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.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
2	At least once per 14 days
≥ 3	At least once per 7 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per diesel generator basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency.

TABLE 4.8.1.1.2-2
UNIT 1 AND UNIT 2
DIESEL GENERATOR LOADING TIMERS

DEVICE TAG NO.	SYSTEM	LOCATION	TIME SETTING
62A-20102	RHR Pump 1A	1A201	3 sec
62A-20202	RHR Pump 1B	1A202	3 sec
62A-20302	RHR Pump 1C	1A203	3 sec
62A-20402	RHR Pump 1D	1A204	3 sec
62A-20102	RHR Pump 2A	2A201	3 sec
62A-20202	RHR Pump 2B	2A202	3 sec
62A-20302	RHR Pump 2C	2A203	3 sec
62A-20402	RHR Pump 2D	2A204	3 sec
K116A	CS pp 1A	1C626	10.5 sec
K116B	CS pp 1B	1C627	10.5 sec
K125A	CS pp 1C	1C626	10.5 sec
K125B	CS pp 1D	1C627	10.5 sec
K116A	CS pp 2A	2C626	10.5 sec
K116B	CS pp 2B	2C627	10.5 sec
K125A	CS pp 2C	2C626	10.5 sec
K125B	CS pp 2D	2C627	10.5 sec
62AX2-20108	Emergency Service Water (ESW)	1A201	40 sec
62AX2-20208	Emergency Service Water (ESW)	1A202	40 sec
62AX2-20303	Emergency Service Water (ESW)	1A203	44 sec
62AX2-20403	Emergency Service Water (ESW)	1A204	48 sec
62X3-20304	Control Structure Chilled Water System	0C877A	60 sec
62X3-20404	Control Structure Chilled Water System	0C877B	60 sec
62X-20104	Emergency Switchgear Rm. Cooler A & RHR SW pp H&V Fan A	0C877A	60 sec

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two of the five separate and independent diesel generators each with:

   1. An engine mounted day fuel tank containing a minimum of 325 gallons of fuel.

   2. A fuel storage system containing a minimum of 47,570 gallons of fuel for diesels A, B, C, and D; and 60,480 gallons for diesel generator E.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 and 4.8.1.1.4, except for the requirement of 4.8.1.1.2.a.5.

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

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division I, consisting of:

1. Load group Channel "A" power source consisting of:
   a) 125 volt DC battery bank — 1D610, 2D610*
   b) Full capacity charger — 1D613, 2D613*

2. Load group Channel "C" power source consisting of:
   a) 125 volt DC battery bank — 1D630, 2D630*
   b) Full capacity charger — 1D633,2D633*

3. Load group "I" power source consisting of:
   a) 250 volt DC battery — 1D650
   b) Half-capacity chargers — 1D653A, 1D653B

4. Load group "I" power source consisting of:
   a) ± 24 volt DC battery bank — 1D670
   b) Two half-capacity chargers — 1D673, 1D674

b. Division II, consisting of:

1. Load group Channel "B" power source consisting of:
   a) 125 volt DC battery bank — 1D620, 2D620*
   b) Full capacity charger — 1D623, 2D623*

2. Load group Channel "D" power source consisting of:
   a) 125 volt DC battery bank — 1D640, 2D640*
   b) Full capacity charger — 1D643, 2D643*

3. Load group "II" power source consisting of:
   a) 250 volt DC battery bank — 1D660
   b) Full capacity charger — 1D663

*Not required to be OPERABLE when the requirements of ACTION b have been satisfied.

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 each time the unit is in COLD SHUTDOWN for a period of more than 24 hours, unless 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

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

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

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

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

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

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

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.

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

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1.1, 3.4.1.1.2, and 3.4.1.3 may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

3/4.10.5 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.5 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

LIMITING CONDITION FOR OPERATION

3.11.2.8 VENTING or PURGING of the Mark II containment drywell shall be through the Standby Gas Treatment System.

APPLICABILITY: Whenever the drywell is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIRMENTS

4.11.2.8.1 The containment drywell shall be determined to be aligned for VENTING or PURGING through the Standby Gas Treatment System within 4 hours prior to start of and at least once per 12 hours during VENTING or PURGING of the drywell.

4.11.2.8.2 Prior to use of the purge system through the standby gas treatment system assure that:

a. Both standby gas treatment system trains are OPERABLE whenever the purge system is in use, and

b. Whenever the purge system is in use during OPERATIONAL CONDITION 1 or 2 or 3, only one of the standby gas treatment system trains may be used.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADWASTE SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM, for the processing and packaging of radioactive wastes to ensure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal regulations governing the disposal of the waste.

APPLICABILITY: At all times.

ACTION:

a. With the requirements of 10 CFR Part 20, and/or 10 CFR Part 71, not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient, effective delayed neutron fraction, and maximum four-bundle local peaking factor are compared with the inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle-specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in XN-NF-80-19 Volume 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement for the sodium penetrate solution is necessary to ensure that the sodium penetaborate remains in solution.

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

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

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 Unit 1 and Unit 2 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 5,755,600 cubic feet.

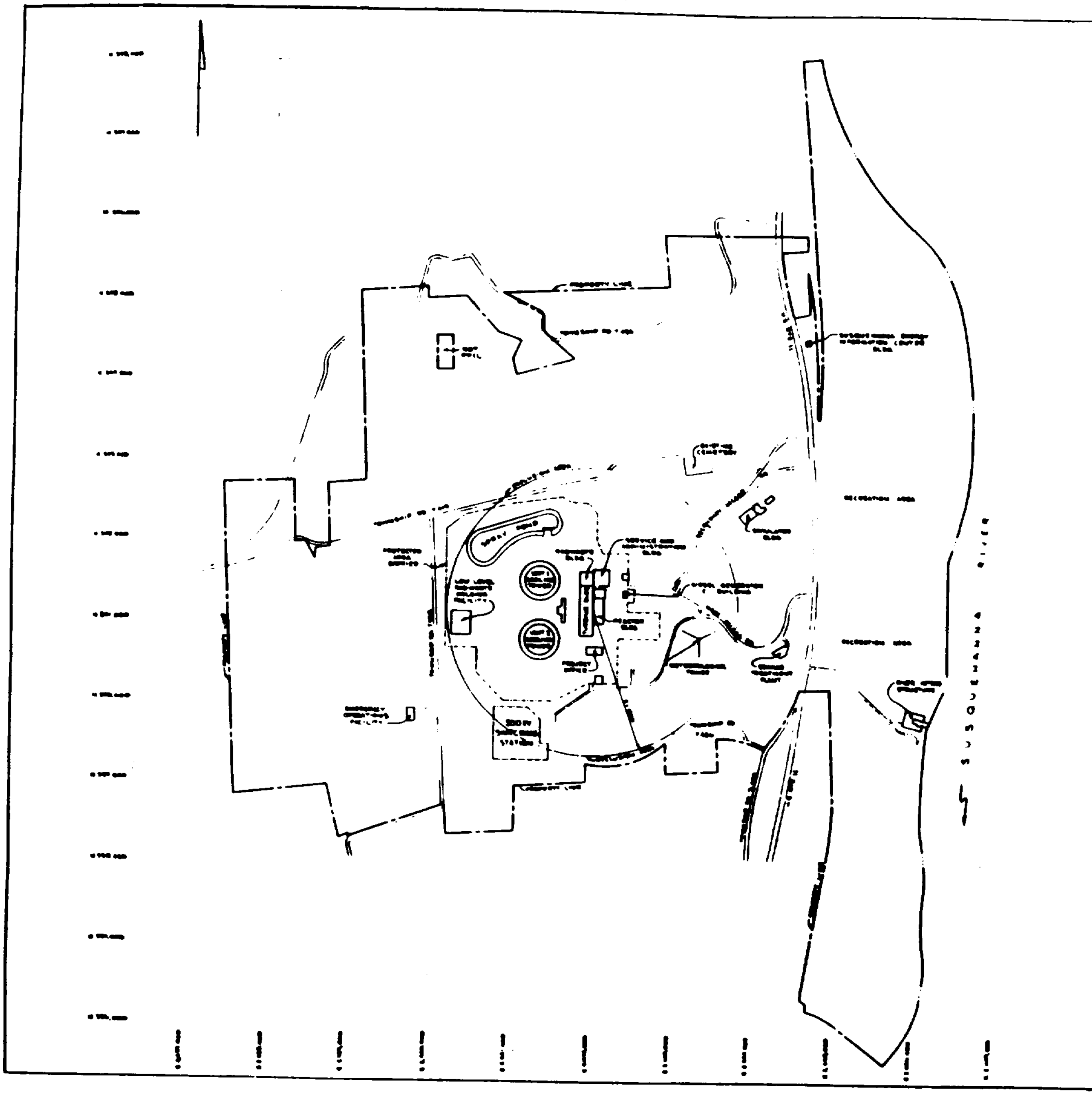

FIGURE 5.1.1-1

EXCLUSION AREA

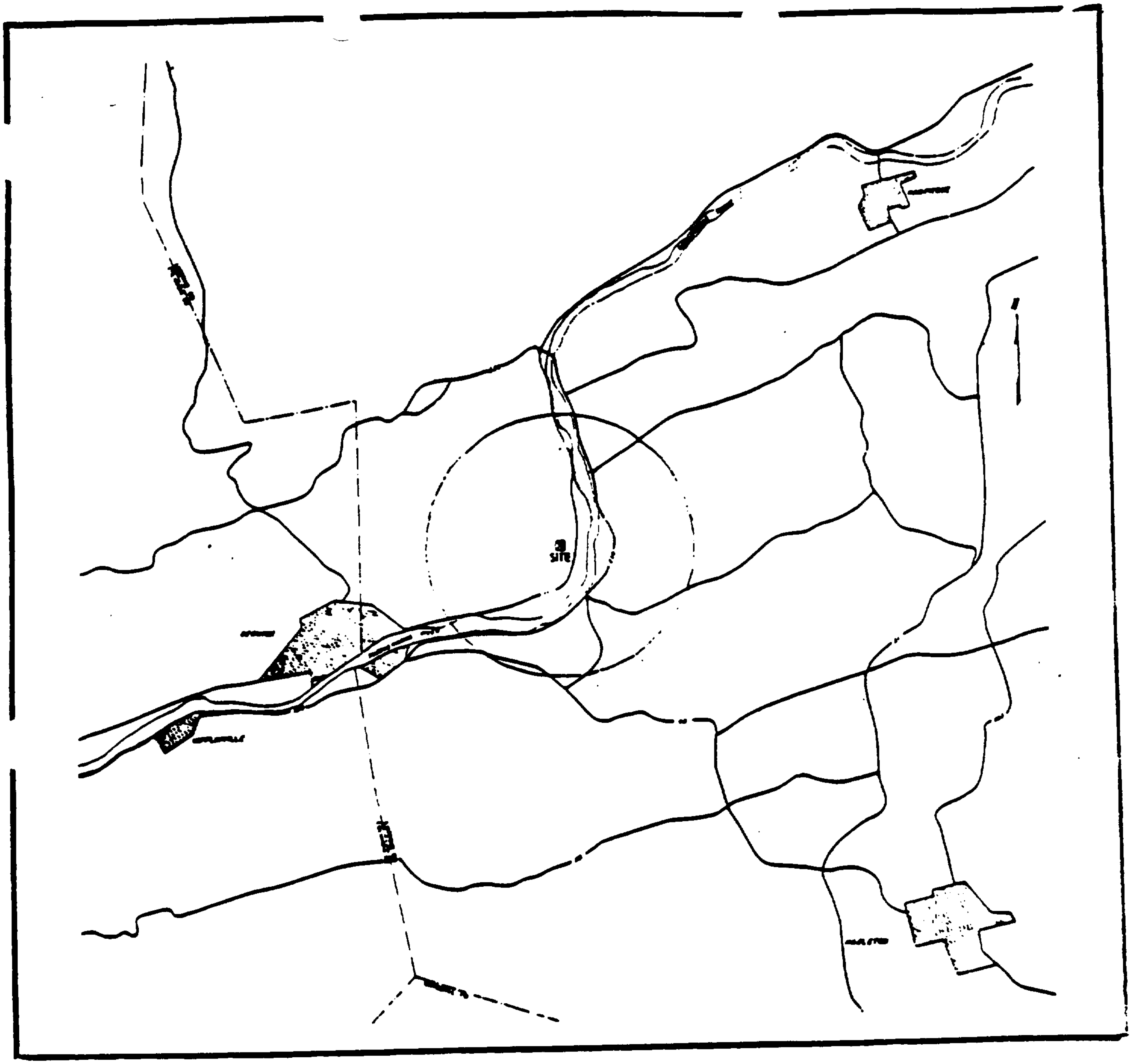

FIGURE 5.1.2-1

LOW POPULATION ZONE
(3-mile radius)

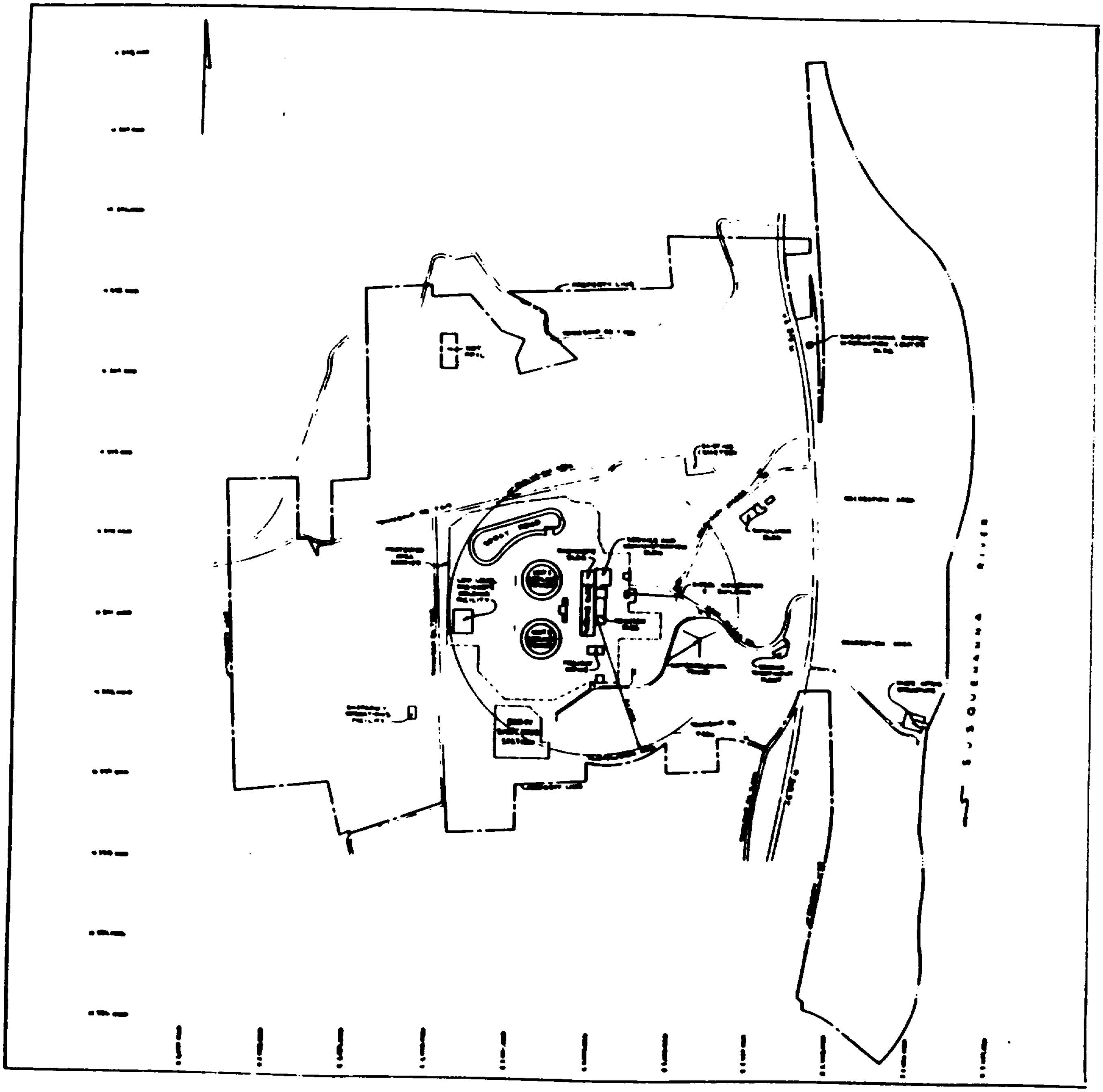

FIGURE 5.1.3-1a

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

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, residual heat removal, post accident sampling 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.

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. NPF-22

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 August 5, 1986 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-22 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. 50 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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 change. The overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
v	v*
vi	vi
ix	ix*
x	x
xiii	xiii
xiv	xiv*
xv	xv*
xvi	xvi
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12*
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54*
3/4 3-71	3/4 3-71
3/4 3-72	3/4 3-72
3/4 3-77	3/4 3-77*
3/4 3-78	3/4 3-78
3/4 3-79	3/4 3-79
3/4 3-80	3/4 3-80*
3/4 4-1b	3/4 4-1b*
3/4 4-1c	3/4 4-1c

3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4*
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2*
3/4 6-9	3/4 6-9*
3/4 6-10	3/4 6-10
3/4 6-31	3/4 6-31
3/4 6-32	3/4 6-32*
3/4 6-41	3/4 6-41
-	-
3/4 7-1	3/4 7-1*
3/4 7-2	3/4 7-2
3/4 8-9	3/4 8-9*
3/4 8-10	3/4 8-10
3/4 8-15	3/4 8-15*
3/4 8-16	3/4 8-16
3/4 8-35	3/4 8-35
-	-
3/4 9-1	3/4 9-1
3/4 9-2	3/4 9-2*
3/4 10-3	3/4 10-3*
3/4 10-4	3/4 10-4
3/4 10-5	3/4 10-5
3/4 10-6	-
B 3/4 10-1	B 3/4 10-1
-	-
5-1	5-1*
5-2	5-2
5-3	5-3*
5-4	5-4
6-3	6-3*
6-4	6-4

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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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.*

(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	$\leq$ 0.09**
c. Fixed Neutron Flux - Upscale	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switch	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 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.

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	2	1, 2, 3	20
c. Manual Initiation	NA	1	1, 2, 3	24
d. SGTS Exhaust Radiation - High	R	1	1, 2, 3, 4***, 5***	20
e. Main Steam Line Radiation - High	C	2	1, 2, 3	20
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	**	2	1, 2, 3 and *	25
b. Drywell Pressure - High	**	2	1, 2, 3	25
c. Refuel Floor High Exhaust Duct Radiation - High	**	2	*	25
d. Railroad Access Shaft Exhaust Duct Radiation - High	**	1	*	25
e. Refuel Floor Wall Exhaust Duct Radiation - High	**	2	*	25
f. Manual Initiation	NA	1	1, 2, 3 and *	24

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 Low, Level 1	X	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. Reactor Building Main Steam Line Tunnel Temperature - High	E	2	1, 2, 3	21
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	E	2	1, 2, 3	21
h. Manual Initiation	NA	1	1, 2, 3	24
i. Turbine Building Main Steam Line Tunnel Temperature - High	E	2	1, 2, 3	21
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	J	1	1, 2, 3	23
b. RWCU Area Temperature - High	W	3	1, 2, 3	23
c. RWCU Area Ventilation Δ Temperature - High	W	3	1, 2, 3	23
d. SLCS Initiation	I	2	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
g. Manual Initiation	NA	1	1, 2, 3	24

effective upon startup following the unit's first

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
d. SGTS Exhaust Radiation - High	≤ 23.0 mR/hr	≤ 31.0 mR/hr
e. Main Steam Line Radiation - High	≤ 7.0 X full power background	≤ 8.4 X full power background
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 mR/hr	≤ 4.0 mR/hr
f. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Main Steam Line Radiation - High	≤ 7.0 X full power background	≤ 8.4 X full power background
c. Main Steam Line Pressure - Low	≥ 861 psig	≥ 841 psig
d. Main Steam Line Flow - High	≤ 107 psid	≤ 110 psid

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. Reactor Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	≤ 99°F	≤ 108°F
h. Manual Initiation	NA	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
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	≤ 153" H_2O	≤ 165" 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 3.3.2-2 (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
j.	Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
6.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Flow - High	≤ 275 inches H_2O	≤ 292 inches H_2O
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
j.	Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	$\geq$ 13.0 inches*	$\geq$ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 98 psig	$\leq$ 108 psig
c. RHR Equipment Area Δ Temperature - High	$\leq$ 89°F	$\leq$ 90.5°F
d. RHR Equipment Area Temperature - High	$\leq$ 167°F	$\leq$ 170.5°F
e. RHR Flow - High	$\leq$ 25,000 gpm	$\leq$ 26,000 gpm
f. Manual Initiation	NA	NA
g. Drywell Pressure - High	$\leq$ 1.72 psig	$\leq$ 1.88 psig

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

#Lower setpoints for TSH-G33-N600 E, F and TSH-G33-N602 E, F.

##15 minute time delay.

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 $\geq$ 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.

*** Not required when eight or fewer fuel assemblies (adjacent to the SRMs) are in the core.

(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 $\geq$ 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##	≤ 0.66 W + 42%	≤ 0.66 W + 45%
b. Inoperative	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale##	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 2×10^5 cps	≤ 4×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps**	≥ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.
**Provided signal-to-noise ratio is ≥ 2. Otherwise, 3 cps as trip setpoint and 2.8 cps for allowable value.
##See Specification 3.4.1.1.2.a for single loop operation requirements.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Steam Dome Pressure	2	1	80	1, 2
2. Reactor Vessel Water Level	2	1	80	1, 2
3. Suppression Chamber Water Level	2	1	80	1, 2
4. Suppression Chamber Water Temperature	8, 6 locations	6, 1/location	80	1, 2
5. Suppression Chamber Air Temperature	2	1	80	1, 2
6. Primary Containment Pressure	2/range	1/range	80	1, 2
7. Drywell Temperature	2	1	80	1, 2
8. Drywell Gaseous Analyzer				
a. Oxygen	2	1	80	1,# 2#
b. Hydrogen	2	1	82	1,# 2#
9. Safety/Relief Valve Position Indicators	1/valve*	1/valve*	80	1, 2
10. Containment High Radiation	2	1	81	1, 2
11. Noble gas monitors**				
a. Reactor Bldg. Vent	1	1	81	1, 2 and***
b. SGTS Vent	1	1	81	1, 2 and***
c. Turbine Bldg. Vent	1	1	81	1, 2
12. Primary Containment Isolation Valve Position	1/valve	1/valve	80	1, 2
13. Neutron Flux	2	1	80	1, 2

*Acoustic monitor.
**Mid-range and high-range channels.
***When moving irradiated fuel in the secondary containment.
#See Special Test Exception 3.10.1.

TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION STATEMENTS

ACTION 80 -

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

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

ACTION 81 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore at least one channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.9-1:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside an inaccessible zone, then inspect the area surrounding the inaccessible zone at least once per hour.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
FIRE ZONE	ROOM OR AREA	ELEV.	HEAT TOTAL	HEAT MIN.	IONIZATION TOTAL	IONIZATION MIN.	PHOTO-ELECTRIC TOTAL	PHOTO-ELECTRIC MIN.
a. Control Building								
0-22A	Filter Area	687'-8"	NA	NA	11	6	NA	NA
0-24D	Lower Relay Room	698'-1"	4	2	4	2	NA	NA
0-24G	Lower Relay Room	698'-1"	4	2	4	2	NA	NA
0-24G	PGCC	698'-1"	54	27	30	15	NA	NA
0-25A	Lower Cable Spreading Rm.	714'-0"	20	10	6	3	NA	NA
0-25B	South Cable Chase	714'-0"	1	1	NA	NA	NA	NA
0-25C	Center Cable Chase	714'-0"	1	1	NA	NA	NA	NA
0-25D	North Cable Chase	714'-0"	1	1	NA	NA	NA	NA
0-25E	Lower Cable Spreading Rm.	714'-0"	26	13	6	3	NA	NA
0-26B	South Cable Chase	729'-1"	NA	NA	1	1	NA	NA
0-26C	Center Cable Chase	729'-1"	NA	NA	1	1	NA	NA
0-26D	North Cable Chase	729'-1"	NA	NA	1	1	NA	NA
0-26F	Vestibule	729'-1"	NA	NA	1	1	NA	NA
0-26G	Shift Office	729'-1"	NA	NA	1	1	NA	NA
0-26H	Control Rm. (Under Flr. Unit 1)*	729'-1"	NA	NA	18	9	NA	NA
0-26H	Control Room (Under Flr. Unit 2)*	729'-1"	NA	NA	15	8	NA	NA
0-26H	Control Room	729'-1"	NA	NA	10	5	NA	NA
0-26H	Control Rm. (Above Clg)*	729'-1"	NA	NA	9	5	NA	NA
0-26I	Operational Support Center	729'-1"	NA	NA	1	1	NA	NA
0-26J	Vestibule	729'-1"	NA	NA	1	1	NA	NA
0-26M	Soffit	729'-1"	NA	NA	4	2	NA	NA
0-26N	Control Room Soffit	729'-1"	NA	NA	2	1	NA	NA
0-26P	Control Room Soffit	729'-1"	NA	NA	2	1	NA	NA
0-26R	Soffit	729'-1"	NA	NA	4	2	NA	NA
0-26S	South Cable Chase	729'-1"	1	1	NA	NA	NA	NA

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
FIRE ZONE	ROOM OR AREA	ELEV.	HEAT TOTAL	HEAT MIN.	IONIZATION TOTAL	IONIZATION MIN.	PHOTO-ELECTRIC TOTAL	PHOTO-ELECTRIC MIN.
a.	Control Building (Continued)							
0-26T	Center Cable Chase	729'-1"	1	1	NA	NA	NA	NA
0-26V	North Cable Chase	729'-1"	1	1	NA	NA	NA	NA
0-27A	Upper Relay Room	754'-1"	2	1	2	1	NA	NA
0-27A	PGCC	754'-1"	55	28	30	15	NA	NA
0-27B	Upper Cable Spreading Rm.	753'-0"	24	12	5	3	NA	NA
0-27C	Upper Cable Spreading Rm.	753'-0"	25	13	8	4	NA	NA
0-27E	Upper Relay Room	754'-1"	4	2	2	1	NA	NA
0-27F	South Cable Chase	754'-1"	1	1	NA	NA	NA	NA
0-27G	Center Cable Chase	754'-1"	1	1	NA	NA	NA	NA
0-27H	North Cable Chase	754'-1"	1	1	NA	NA	NA	NA
0-28A	Equipment Room	771'-0"	NA	NA	4	2	NA	NA
0-28B	Equipment Room	771'-0"	NA	NA	4	2	NA	NA
0-28C	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28D	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28E	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28F	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28G	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28H	Repair Shop	771'-0"	NA	NA	2	1	NA	NA
0-28I	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28J	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28K	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28L	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28M	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28N	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-28P	South Cable Chase	771'-0"	1	1	NA	NA	NA	NA
0-28Q	Center Cable Chase	771'-0"	1	1	NA	NA	NA	NA
0-28R	North Cable Chase	771'-0"	1	1	NA	NA	NA	NA

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION			INSTRUMENTS OPERABLE					
FIRE ZONE	ROOM OR AREA	ELEV.	HEAT TOTAL	HEAT MIN.	IONIZATION TOTAL	IONIZATION MIN.	PHOTO-ELECTRIC TOTAL	PHOTO-ELECTRIC MIN.
a.	Control Building (Continued)							
0-28T	Battery Room	771'-0"	NA	NA	1	1	NA	NA
0-29B	H&V Equipment Room	783'-0"	NA	NA	10	5	NA	NA
0-30A	HVAC Equipment Room	806'-0"	NA	NA	20	10	NA	NA
b.	Reactor Building							
2-1B	Core Spray Pump Room	645'-0"	NA	NA	6	3	NA	NA
2-1A	Core Spray Pump Room	645'-0"	NA	NA	8	4	NA	NA
2-1E	RHR Pump Room	645'-0"	NA	NA	NA	NA	13	7
2-1F	RHR Pump Room	645'-0"	NA	NA	NA	NA	15	8
2-1D	RCIC Pump Room	645'-0"	2	1	NA	NA	5	3
2-1C	HPCI Pump Room	645'-0"	2	1	NA	NA	7	4
2-1G	Sump Room	645'-0"	NA	NA	2	1	NA	NA
2-2B	Core Spray Pump Room	670'-0"	NA	NA	11	6	NA	NA
2-4C	Switchgear Room	719'-0"	NA	NA	2	1	NA	NA
2-4D	Switchgear Room	719'-0"	NA	NA	2	1	NA	NA
2-4A	Containment Access Area	719'-0"	NA	NA	26	13	3	2
2-5F	Load Center Room	749'-1"	NA	NA	2	1	NA	NA
2-5G	Load Center Room	749'-1"	NA	NA	2	1	NA	NA
2-2A	Access Area and Remote Shutdown Panel Room	670'-0"	NA	NA	6	3	NA	NA
2-3A	Access Area	683'-0"	NA	NA	4	2	NA	NA
2-3B	Access Area	683'-0"	NA	NA	14	7	NA	NA
2-3C	Access Area	683'-0"	NA	NA	NA	NA	13	7
2-4B	Pipe Penetration Room	719'-1"	NA	NA	1	1	NA	NA
2-4G	Main Steam Piping	719'-1"	NA	NA	NA	NA	4	2
2-5A	Fuel Pool Pumps and Heat Exchangers	749'-1"	NA	NA	21	11	7	4
2-5B	Valve Access Area	761'-10"	NA	NA	NA	NA	2	1
2-5C	RWCU Backwash Tank	749'-1"	NA	NA	1	1	2	1

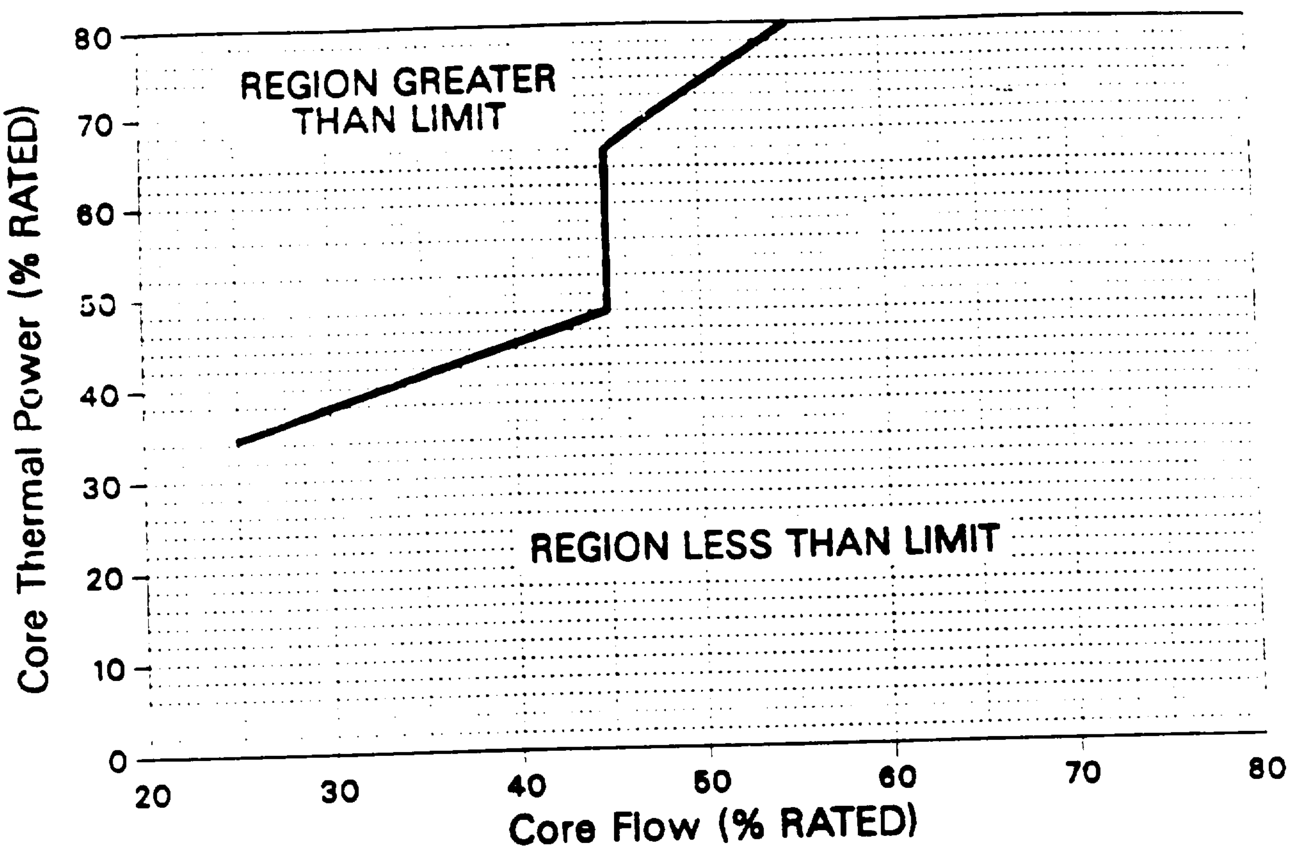

Figure 3.4.1.1.1-1
THERMAL POWER/CORE FLOW LIMITATIONS

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 55%	≤ 0.58W + 58%.

3. Specification 3.2.1: The MAPLHGR limits shall be the limits specified in Figures 3.2.1-1 and 3.2.1-2 multiplied by 0.81 and Figure 3.2.1-3 multiplied by 1.0.

4. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the largest of the following values:

   a. 1.37,

   b. the MCPR determined from Figure 3.2.3-1 plus 0.01, and

   c. the MCPR determined from Figure 3.2.3-2 plus 0.01.

6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.66W + 37%	≤ 0.66W + 40%
b. APRM-Flow Biased	≤ 0.58W + 46%	≤ 0.58W + 49%

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.2-1.

c. Total core flow shall be greater than or equal to 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

ACTION:

a. With no reactor coolant system recirculation loops in operation, take the ACTION required by Specification 3.4.1.1.1.

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed mismatch shall be maintained within:

a. 5% of each other with core flow greater than or equal to 75% of rated core flow.

b. 10% of each other with core flow less than 75% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* when both recirculation loops are in operation.

ACTION:

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

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

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.1.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

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

1. Two OPERABLE CSS pumps, and

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

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of two subsystems with each subsystem comprised of:

1. Two OPERABLE LPCI pumps, and

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

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

1. One OPERABLE HPCI pump, and

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

d. The automatic depressurization system (ADS) with six OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*,**,#, and 3*,**,##.

---

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

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

#See Special Test Exception 3.10.5.

##One LPCI subsystem of the RHR system may be inoperable in that it is aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least one LPCI pump in each LPCI subsystem is OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

b. For the LPCI system:

1. With one LPCI pump in either or both LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump(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.

2. With no LPCI system cross-tie valve closed or with power not removed from the closed cross-tie valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

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

4. With both LPCI subsystems otherwise inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the CSS, the LPCI system, the ADS and the RCIC system are OPERABLE:

1. With the HPCI system inoperable, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 150 psig within the following 24 hours.

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

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -1.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the arithmetical average of the higher temperature at a minimum of three of the following areas and shall be determined to be within the limit at least once per 24 hours:

	Area	Elevation	Azimuth
a.	Top	797'8"	105°, 285°
b.	Middle	752'2"	80°, 280°
c.	Bottom	725' or 711'	40°, 260°
d.	Pedestal	711' or 720'	80°, 270°

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT** INTEGRITY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT** INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT** INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT** INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

1a. When the railroad bay door (No. 101) is closed; all Zone I and III hatches, removable walls, dampers, and doors connected to the railroad access bay are closed,## or

i) Only Zone I removable walls and/or doors are open to the railroad access shaft,## or

ii) Only Zone III hatches and/or dampers are open to the railroad access shaft.##

1b. When the railroad bay door (No. 101) is open; all Zone I and III hatches, removable walls, dampers, and doors connected to the railroad access bay are closed.

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

**Secondary Containment consists of Zone I, Zone II and Zone III or Zone II and Zone III when Zone I is isolated from Zone II and Zone III. During operational condition* when no operations with a potential for draining the reactor vessel are being performed, secondary containment may consist of Zone III as long as Zone II is isolated from Zone III.

##Personnel ingress and egress through doors within the secondary containment is not prohibited by this specification.

SURVEILLANCE REQUIREMENTS (Continued)

2a. At least one door in each access to the secondary containment zones is closed.

2b. At least one door in each access between secondary containment zones is closed.*

3. All secondary containment penetrations** not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

4. The truck bay hatch is closed.
5. The truck bay door (No. 102) is closed unless Zone II is isolated from Zones I and III.

c. At least once per 18 months:

1. For three zone operation with Zone I OPERABLE:

a. Verifying that one standby gas treatment subsystem will draw down the secondary containment (Zone II and Zone III) to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 15 seconds, and

b. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate of less than or equal to 2960 cfm from Zone II and Zone III, and

c. Verifying by calculation that one standby gas treatment subsystem will maintain greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate of less than or equal to 4000 cfm from Zone I, Zone II, and Zone III, or

2. For three zone operation:

a. Verifying that one standby gas treatment subsystem will draw down the secondary containment (Zone I, Zone II and Zone III) to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 92 seconds, and

*Personnel ingress and egress through doors within the secondary containment is not prohibited by this specification.
**Penetration between secondary containment zones, penetrations to no-zones, and penetrations to the outside atmosphere.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

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.

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*May not be a pump required for Unit 1.

**See Specifications 3.9.11.1 and 3.9.11.2 for applicability.

***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 at 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 otherwise 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. In OPERATIONAL CONDITION 4, 5 or *:

1. With one pump in an emergency service water system loop inoperable, verify adequate cooling capability remains available for the diesel generators required to be operable by Specification 3.8.1.2 or declare the affected diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2.

2. With two pumps in an emergency service water system loop inoperable or with the loop otherwise inoperable declare the associated safety related equipment inoperable (except diesel generators), and follow the applicable ACTION statements. Verify adequate cooling remains available for the diesel generators required to be operable by Specification 3.8.1.2 or declare the affected diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2.

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*When handling irradiated fuel in the secondary containment.

#When any diesel generator is removed from service in order to do work associated with tying in the additional diesel generator and its associated emergency service water pump is inoperable, Action a.1 shall read as follows:

a.1 With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status when its associated diesel generator is restored to OPERABLE status per Specification 3.8.1.1.

TABLE 4.8.1.1.2-2 (Continued)

UNIT 1 AND UNIT 2

DIESEL GENERATOR LOADING TIMERS

DEVICE TAG NO.	SYSTEM	LOCATION	TIME SETTING
62X-20204	Emergency Switchgear Rm Cooler B & RHR SW pp H&V Fan B	0C877B	60 sec
262X-20104	Emergency Switchgear Rm Cooler A	0C877A	120 sec
262X-20204	Emergency Switchgear Rm Cooler B	0C877B	120 sec
*62X-516	DG Rm Exh Fan A	0B516	2 min
*62X-526	DG Rm Exh Fan B	0B526	2 min
*62X-536	DG Rm Exh Fan C	0B536	2 min
*62X-546	DG Rm Exh Fan D	0B546	2 min
*CRX-5652A	DG Room Supply Fans E1 and E2	0B565	2 min
*62X-5653A	DG Room Exhaust Fan E3	0B565	1 min
*62X-5652A	DG Room Exhaust Fan E4	0B565	1 min
62X1-20304	Control Structure Chilled Water System	0C877A	3 min
62X1-20404	Control Structure Chilled Water System	0C877B	3 min
62X2-20310	Control Structure Chilled Water System	0C876A	3 min
62X2-20410	Control Structure Chilled Water System	0C876B	3 min
62X2-20304	Control Structure Chilled Water System	0C877A	3.5 min
62X2-20404	Control Structure Chilled Water System	0C877B	3.5 min
62X-K11AB	Emergency Switchgear Rm Cooling Compressor A	2CB250A	260 sec
62X-K11BB	Emergency Switchgear Rm Cooling Compressor B	2CB250B	260 sec

*When associated diesel generator is declared OPERABLE.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two of the five separate independent diesel generators each with:

   1. An engine mounted day fuel tank containing a minimum of 325 gallons of fuel.

   2. A fuel storage system containing a minimum of 47,570 gallons of fuel for diesel generators A, B, C and D; and 60,480 gallons of fuel for diesel generator E.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 and 4.8.1.1.4, except for the requirement of 4.8.1.1.2.a.5.

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

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195[b] Average of all connected cells > 1.205[b]	Not more than 0.020 below the average of all connected cells Average of all connected cells ≥ 1.195[b]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 0.01, 0.1 and 0.25 amperes for the ±24, 125 and 250 volt batteries respectively, when on float charge.

(c) May be corrected for average electrolyte temperature.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division I and diesel generator E or Division II and diesel generator E of the D.C. electrical power sources shall be OPERABLE with:

a. Division I consisting of:

1. Load group Channel "A" power source, consisting of:
   a) 125-volt D.C. battery bank — 1D610**, 2D610
   b) Full capacity charger — 1D613**, 2D613
2. Load group Channel "C" power source, consisting of:
   a) 125-volt D.C. battery bank — 1D630**, 2D630
   b) Full capacity charger — 1D633**, 2D633
3. Load group "I" power source, consisting of:
   a) 250-volt D.C. battery bank — 2D650
   b) Half-capacity chargers — 2D653A, 2D653B
4. Load group "I" power source, consisting of:
   a) ± 24-volt D.C. battery bank — 2D670
   b) Two half-capacity chargers — 2D673, 2D674

b. Division II consisting of:

1. Load group Channel "B" power source, consisting of:
   a) 125-volt D.C. battery bank — 1D620**, 2D620
   b) Full capacity charger — 1D623**, 2D623
2. Load group Channel "D" power source, consisting of:
   a) 125-volt D.C. battery bank — 1D640**, 2D640
   b) Full capacity charger — 1D643**, 2D643
3. Load group "II" power source, consisting of:
   a) 250-volt D.C. battery bank — 2D660
   b) Full capacity charger — 2D663
4. Load group "II" power source, consisting of:
   a) ± 24-volt D.C. battery bank — 2D680
   b) Two half-capacity chargers — 2D683, 2D684

c. Diesel Generator E

1. Load group power source, consisting of:
   a) 125 volt DC battery bank — 0D595
   b) Full capacity charger — 0D596

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

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

**Not required to be OPERABLE when the requirements of ACTION b have been satisfied.

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 each time the unit is in COLD SHUTDOWN for a period of more than 24 hours, unless 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	$\leq$ 129.1 VAC	$\leq$ 130.3 VAC
2. Undervoltage	$\geq$ 112.0 VAC	$\geq$ 112.5 VAC
3. Underfrequency	$\geq$ 57 Hz	$\geq$ 57 Hz

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: OPERATIONAL CONDITION 5*.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

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* See Special Test Exceptions 3.10.1 and 3.10.3.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks## shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch refuel position interlocks## that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1.1, 3.4.1.1.2, and 3.4.1.3 may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.5 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

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 Unit 1 and Unit 2 Reactor Buildings, 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 5,755,600 cubic feet.

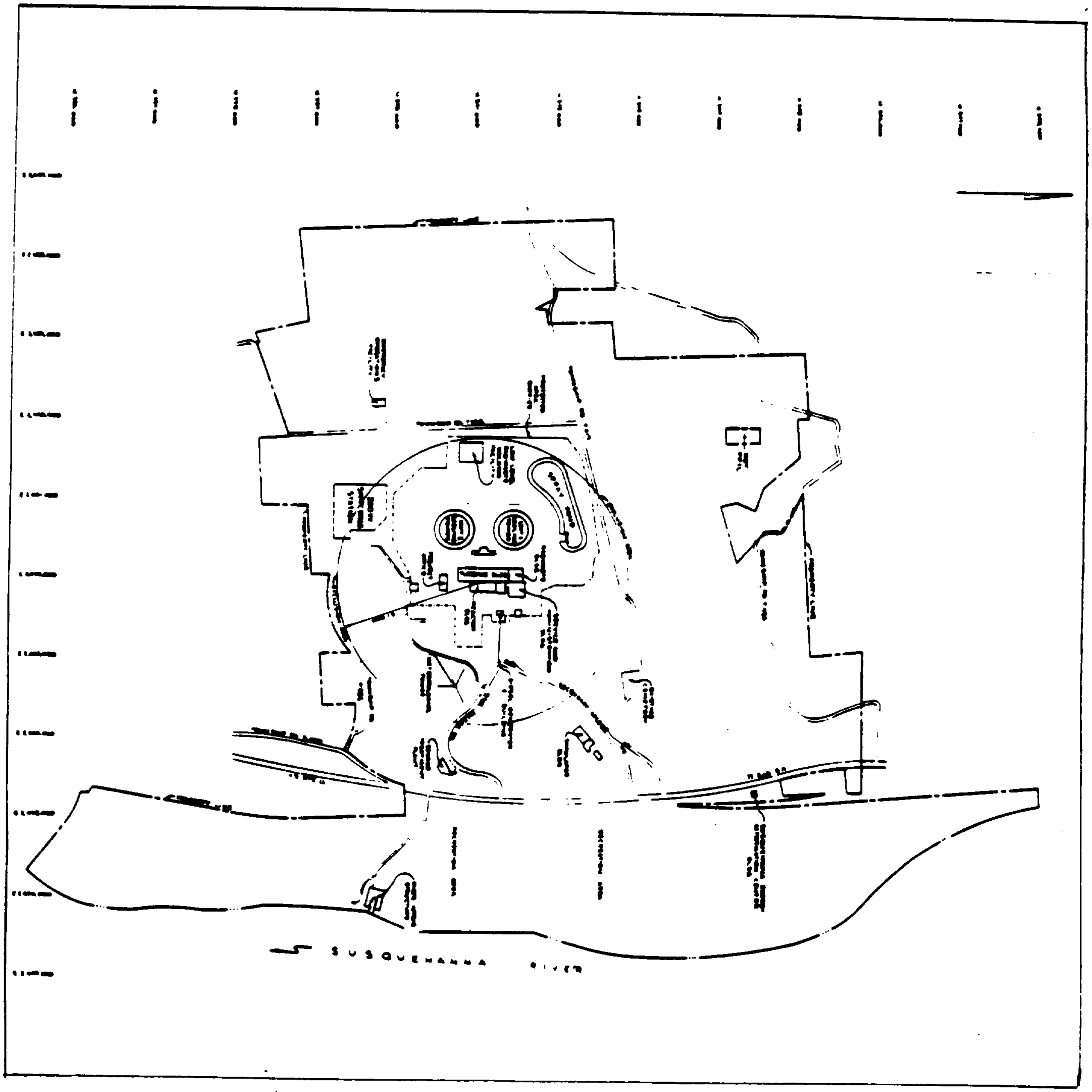

FIGURE 5.1.1-1

EXCLUSION AREA

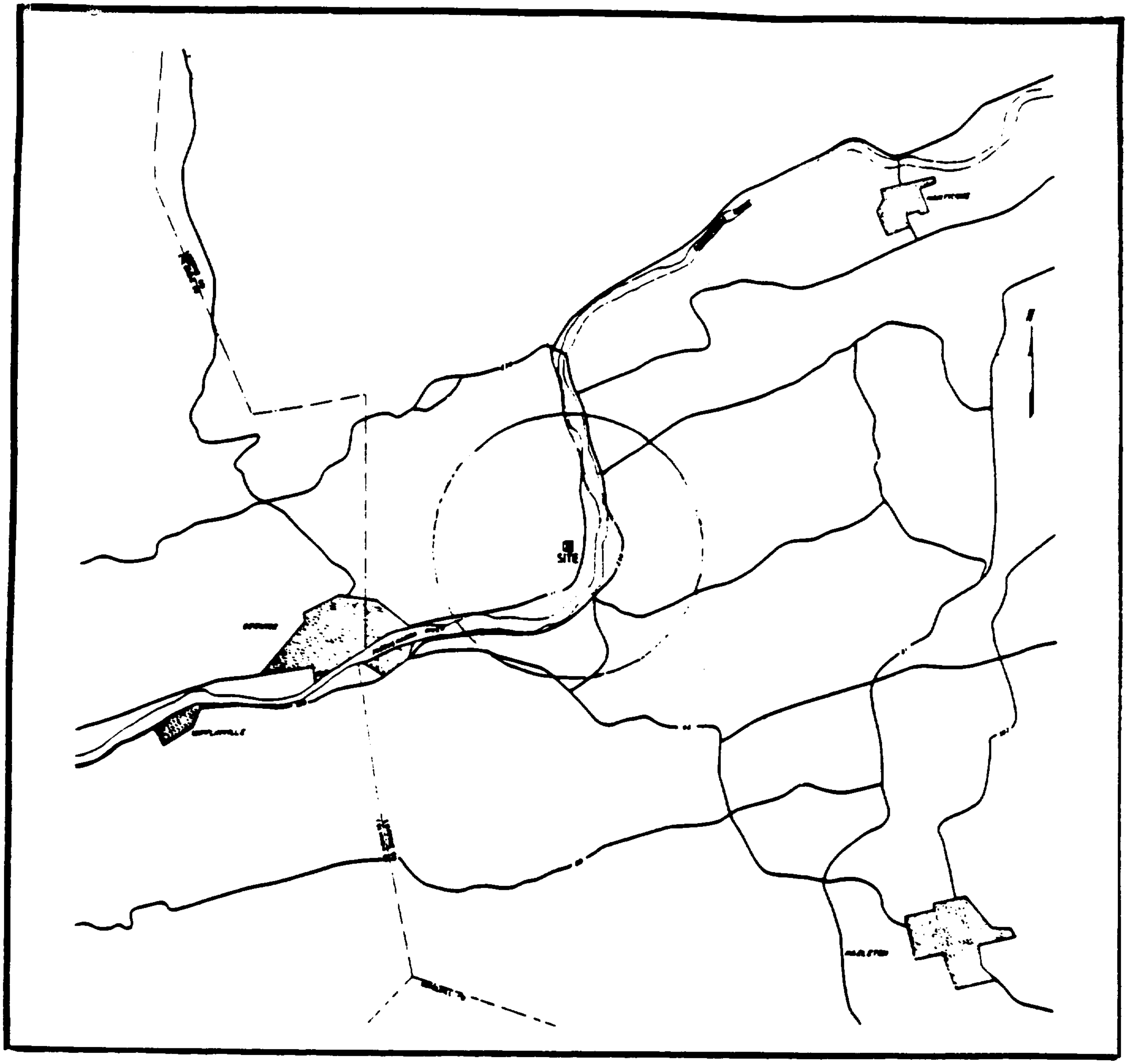

FIGURE 5.1.2-1

LOW POPULATION ZONE

(3-mile radius)

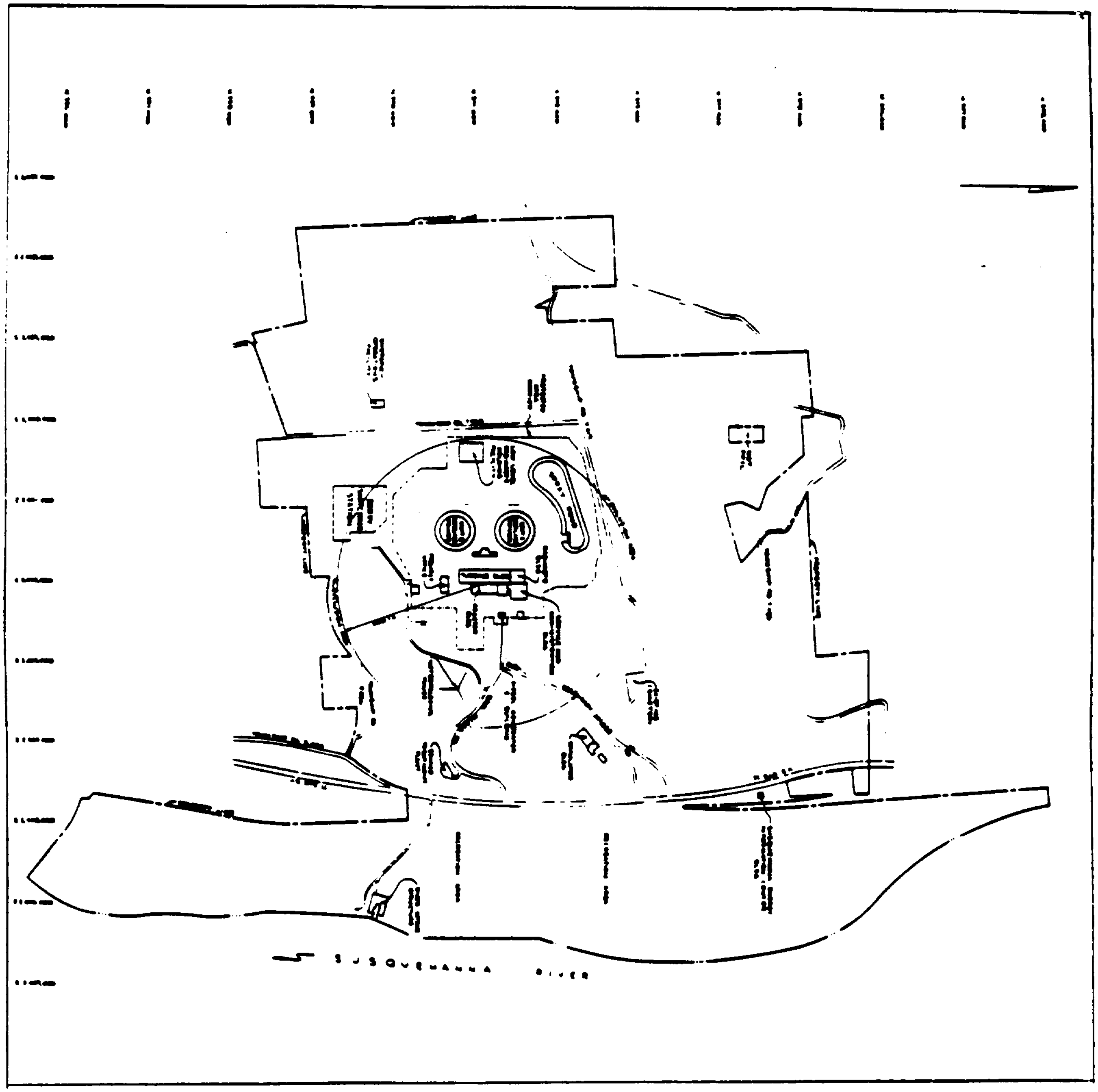

FIGURE 5.1.3-1a

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

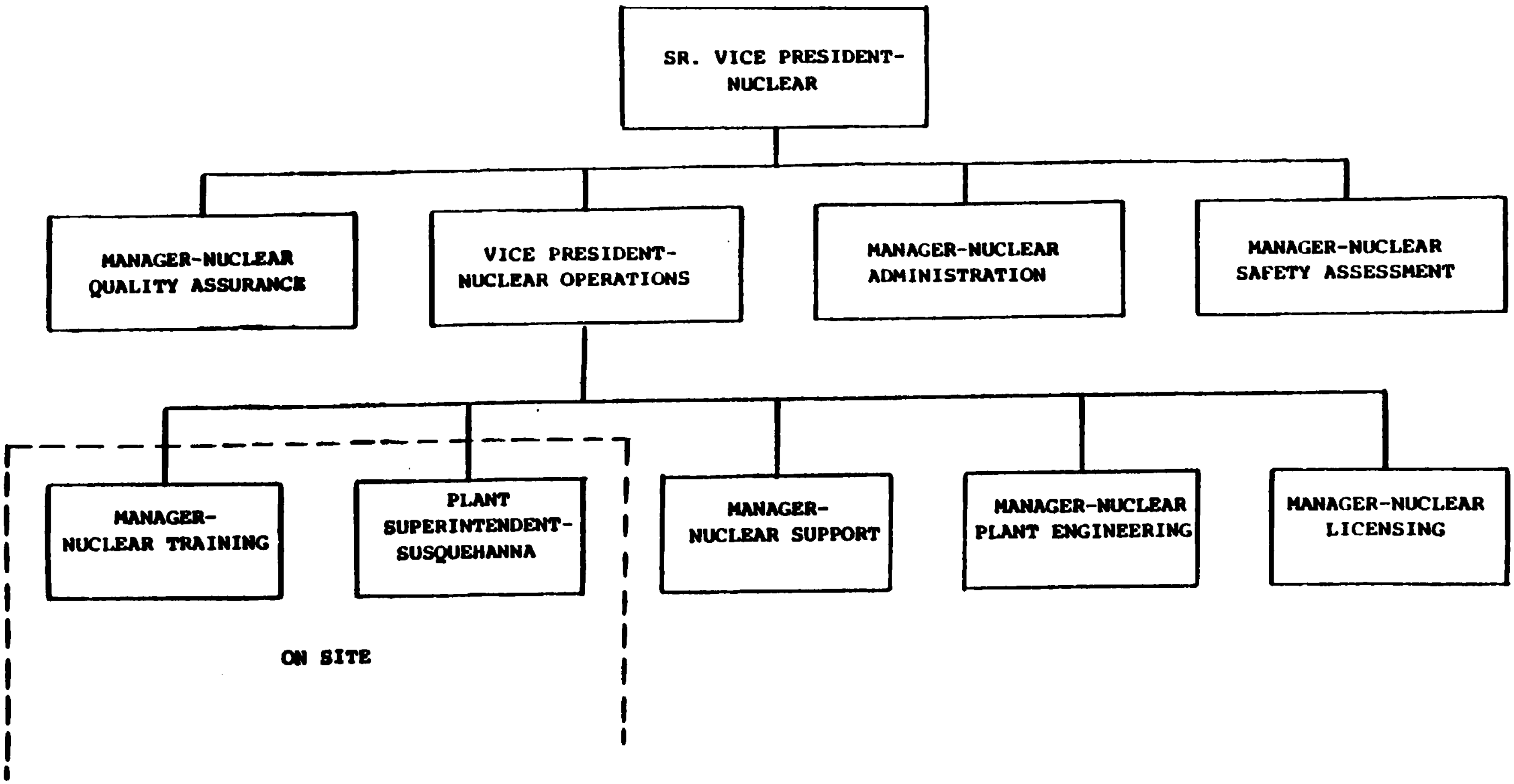

Figure 6.2.1-1

OFFSITE ORGANIZATION

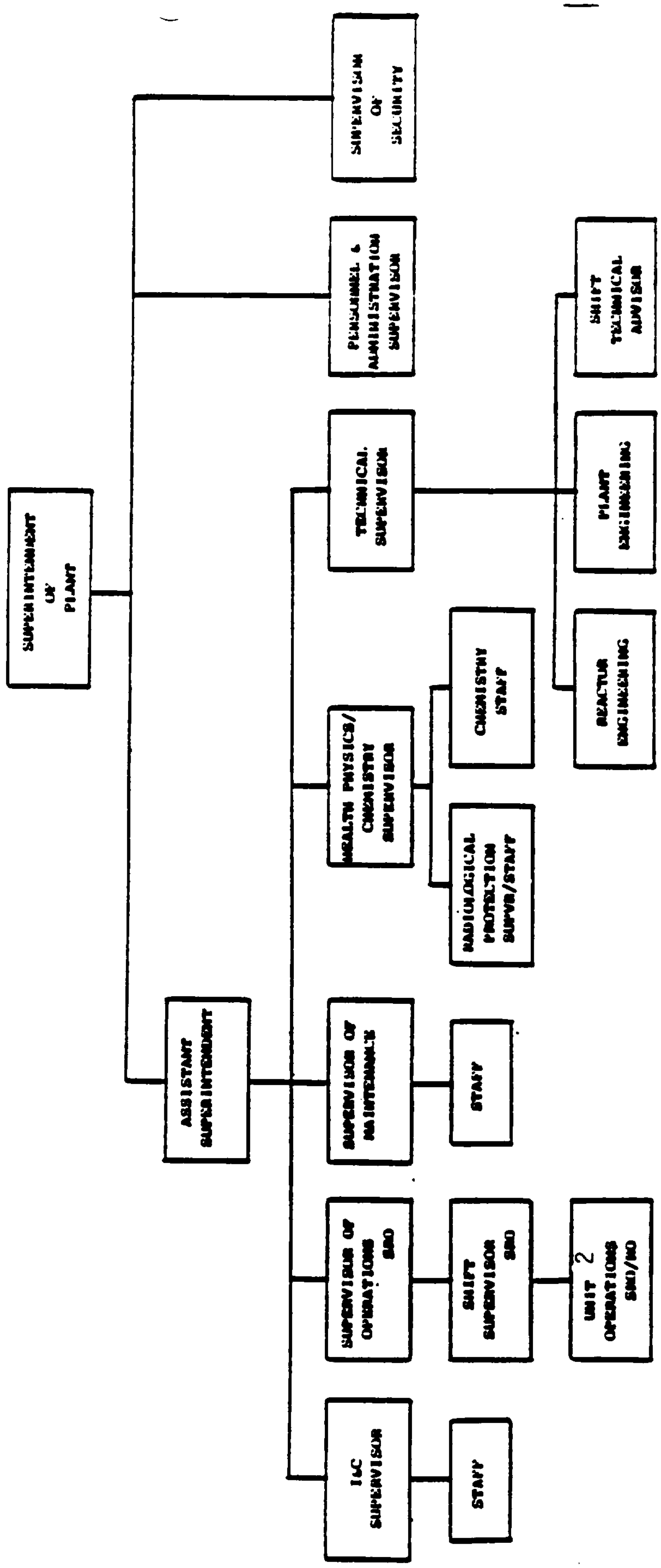

FIGURE 6.2.2-1
UNIT ORGANIZATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. NPF-14 AND

AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NOS. 50-387 AND 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated August 5, 1986, Pennsylvania Power & Light Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-14 and NPF-22 for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The proposed amendments would revise the SSES Unit 1 and Unit 2 Technical Specifications to correct errors, delete dated requirements which have been completed and the effective dates which have expired, achieve consistency in the Technical Specifications, and to change nomenclature. Some of the changes requested in the August 5, 1986 letter have been included in other amendments issued in the intervening period. Therefore, this amendment addresses only the remaining requested changes and some editorial changes.

2.0 EVALUATION

The licensee has proposed 49 changes to the Unit 1 Technical Specifications and 33 changes to the Unit 2 Technical Specifications. All proposed changes can be grouped as follows:

Unit 1

- 17 corrections of misspelled words and other typographical errors
- 29 changes to achieve clarity and consistency in the Technical Specifications
- 2 changes to reflect changes in nomenclature
- 1 deletion of a dated requirement which has been completed

Unit 2

- 12 corrections of typographical errors
- 14 changes to achieve clarity and consistency in the Technical Specifications

- 5 deletions of dated requirements which have been completed
- Two changes updating site drawings to show new construction

As stated above, some of the requested changes were approved in other amendments subsequent to the August 5, 1986 application. The remaining requested changes and some editorial changes have been reviewed here.

Our reviews of the remaining proposed Technical Specification changes indicate that the changes do not involve any issues affecting SSES Unit 1 and Unit 2 safety. The proposed changes are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the requirements with respect to 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 these amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 18983) on May 20, 1987 and consulted with the State of Pennsylvania. No public comments were received, and the State of Pennsylvania did not have any comments.

The staff has 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 these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: Mohan C. Thadani

Dated: August 30, 1988