



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20585-0001

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
License No. NPF-39

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Philadelphia Electric Company (the licensee) dated December 9, 1993, as supplemented by letters of July 5, September 9, October 19, November 15, and December 2, 1994, January 6, and January 23, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;
 - 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, Facility Operating License No. NPF-39 paragraph 2.C.(1) is hereby amended to read as follows:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

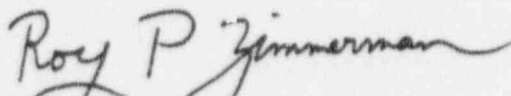
3. Further, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 106, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. This license amendment is effective as of its date of issuance and is to be implemented prior to startup in Cycle 7.

FOR THE NUCLEAR REGULATORY COMMISSION



Roy P. Zimmerman, Acting Director
Office of Nuclear Reactor Regulation

Attachments:

1. Page 3 of License*
2. Changes to the Technical Specifications

Date of Issuance: January 24, 1996

*Page 3 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Facility Operating License (FOL) and the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

	<u>Remove</u>	<u>Insert</u>
FOL	3	3
Appendix A	1-6	1-6
	2-4	2-4
	3/4 1-20	3/4 1-20
	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 3-44	3/4 3-44
	3/4 3-60	3/4 3-60
	3/4 4-1	3/4 4-1
	3/4 4-2	3/4 4-2
	3/4 4-3	3/4 4-3
	3/4 4-7	3/4 4-7
	3/4 4-18	3/4 4-18
	3/4 4-20	3/4 4-20
	3/4 4-22	3/4 4-22
	3/4 5-4	3/4 5-4
	3/4 6-46	3/4 6-46
	3/4 6-54	3/4 6-54
	3/4 7-9	3/4 7-9
	B 3/4 4-5	B 3/4 4-5
	B 3/4 4-7	B 3/4 4-7
	B 3/4 4-8	B 3/4 4-8
	B 3/4 5-1	B 3/4 5-1
	B 3/4 6-1	B 3/4 6-1
	B 3/4 6-2	B 3/4 6-2
	B 3/4 6-3	B 3/4 6-3
	B 3/4 6-5	B 3/4 6-5
	5-8	5-8

- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
 - (4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
 - (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.
- C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.
 - (2) Technical Specifications

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

DEFINITIONS

PURGE - PURGING

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

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

- a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or
 2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.
- b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.
- c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.
- d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.
- e. At least one door in each access to the reactor enclosure secondary containment is closed.
- f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.
- g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

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

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.35 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Neutron Flux-Upscale		
1) During two recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 62%, with a maximum of	≤ 0.66 W+ 64%, with a maximum of
b) High Flow Clamped	≤ 115% of RATED THERMAL POWER	≤ 117% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 57%,	≤ 0.66 W+ 59%,
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
b. Float Switch	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

* See Bases Figure B 3/4.3-1.

** Equivalent to 25.45 gallons/scram discharge volume.

- b. At least once per 31 days by:
1. Verifying the continuity of the explosive charge.
 2. Determining by chemical analysis and calculation* that the available weight of sodium pentaborate is greater than or equal to 3754 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\% \text{ wt.}} \times \frac{E}{29 \text{ atom \%}} \times \frac{Q}{86 \text{ gpm}} \geq 1$$
 where
 - C = Sodium pentaborate solution (% by weight)
 - Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
 - E = Boron 10 enrichment (atom % Boron 10)
 3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.
- d. At least once per 24 months during shutdown by:
1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.
 2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**
- e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
<u>1. MAIN STEAM LINE ISOLATION</u>		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ 756 psig	≥ 736 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥ 10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
<u>2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION</u>		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
3. <u>REACTOR WATER CLEANUP SYSTEM ISOLATION</u>		
a. RWCS Δ Flow - High	≤ 54.9 gpm	≤ 65.2 gpm
b. RWCS Area Temperature - High	$\leq 142^{\circ}\text{F}$ or 132°F^{**}	$\leq 147^{\circ}\text{F}$ or 137°F^{**}
c. RWCS Area Ventilation Δ Temperature - High	$\leq 32^{\circ}\text{F}$	$\leq 40^{\circ}\text{F}$
d. SLCS Initiation	N.A.	N.A.
e. Reactor Vessel Water Level - Low, Low, - Level 2	≥ -38 inches *	≥ -45 inches
f. Manual Initiation	N.A.	N.A.
4. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u>		
a. HPCI Steam Line Δ Pressure - High	$\leq 974^{\circ}$ H ₂ O	$\leq 984^{\circ}$ H ₂ O
b. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	225°F	$\geq 218^{\circ}\text{F}$, $\leq 247^{\circ}\text{F}$
e. HPCI Equipment Room Δ Temperature - High	$\leq 126^{\circ}\text{F}$	$\leq 130.5^{\circ}\text{F}$
f. HPCI Pipe Routing Area Temperature - High	175°F	$\geq 165^{\circ}\text{F}$, $\leq 200^{\circ}\text{F}$
g. Manual Initiation	N.A.	N.A.
h. HPCI Steam Line Δ Pressure - Timer	$3 \leq \tau \leq 12.5$ seconds	$2.5 \leq \tau \leq 13$ seconds

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>		
a. RCIC Steam Line Δ Pressure - High	$\leq 373^{\circ} \text{H}_2\text{O}$	$\leq 381^{\circ} \text{H}_2\text{O}$
b. RCIC Steam Supply Pressure - Low	$\geq 64.5 \text{ psig}$	$\geq 56.5 \text{ psig}$
c. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq 10.0 \text{ psig}$	$\leq 20.0 \text{ psig}$
d. RCIC Equipment Room Temperature - High	205°F	$\geq 198^{\circ}\text{F}, \leq 227^{\circ}\text{F}$
e. RCIC Equipment Room Δ Temperature - High	$\leq 109^{\circ}\text{F}$	$\leq 113.5^{\circ}\text{F}$
f. RCIC Pipe Routing Area Temperature - High	175°F	$\geq 165^{\circ}\text{F}, \leq 200^{\circ}\text{F}$
g. Manual Initiation	N.A.	N.A.
h. RCIC Steam Line Δ Pressure Timer	$3 \leq \tau \leq 12.5 \text{ seconds}$	$2.5 \leq \tau \leq 13 \text{ seconds}$

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TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
2. Reactor Vessel Pressure - High	≤ 1149 psig	≤ 1156 psig

* See Bases Figure B3/4 3-1.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. <u>ROD BLOCK MONITOR</u>		
a. Upscale ^(a)		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint ^(b)		
1) Low Power Setpoint (LPSP)	23% RATED THERMAL POWER	26% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	58% RATED THERMAL POWER	61% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	78% RATED THERMAL POWER	81% RATED THERMAL POWER
2. <u>APRM</u>		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation	≤ 0.66 W + 55%	≤ 0.66 W + 59%
2) During single recirculation loop operation	≤ 0.66 W + 50%	≤ 0.66 W + 54%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% 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. <u>SOURCE RANGE MONITORS</u>		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10 ⁵ cps	≤ 1.6 x 10 ⁵ cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
4. <u>INTERMEDIATE RANGE MONITORS</u>		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. <u>SCRAM DISCHARGE VOLUME</u>		
a. Water Level-High	≤ 257' 5 9/16" elevation***	≤ 257' 7 9/16" elevation
a. Float Switch		

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

- a. Total core flow greater than or equal to 45% of rated core flow, or
- b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

- a. With one reactor coolant system recirculation loop not in operation:
 1. Within 4 hours:
 - a. Place the recirculation flow control system in the Local Manual mode, and
 - b. Reduce THERMAL POWER to $\leq 76.2\%$ of RATED THERMAL POWER, and,
 - c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
 - d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is $\leq 30\%$ of RATED THERMAL POWER or the recirculation loop flow in the operating loop is $\leq 50\%$ of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to the setpoints as noted in the CORE OPERATING LIMITS REPORT, as a percentage of rated core flow, at least once per 24 months.

4.4.1.1.3 Establish a baseline APRM and LPRM** neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

- a. Reactor THERMAL POWER is $\leq 76.2\%$ of RATED THERMAL POWER,
- b. The recirculation flow control system is in the Local Manual mode, and
- c. The speed of the operating recirculation pump is $\leq 90\%$ of rated pump speed.
- d. Core flow is greater than 39% when THERMAL POWER is within the restricted zone of Figure 3.4.1.1-1.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $\leq 30\%$ of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq 50\%$ of rated loop flow.

- a. $\leq 145^{\circ}\text{F}$ between reactor vessel steam space coolant and bottom head drain line coolant,
- b. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and
- c. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

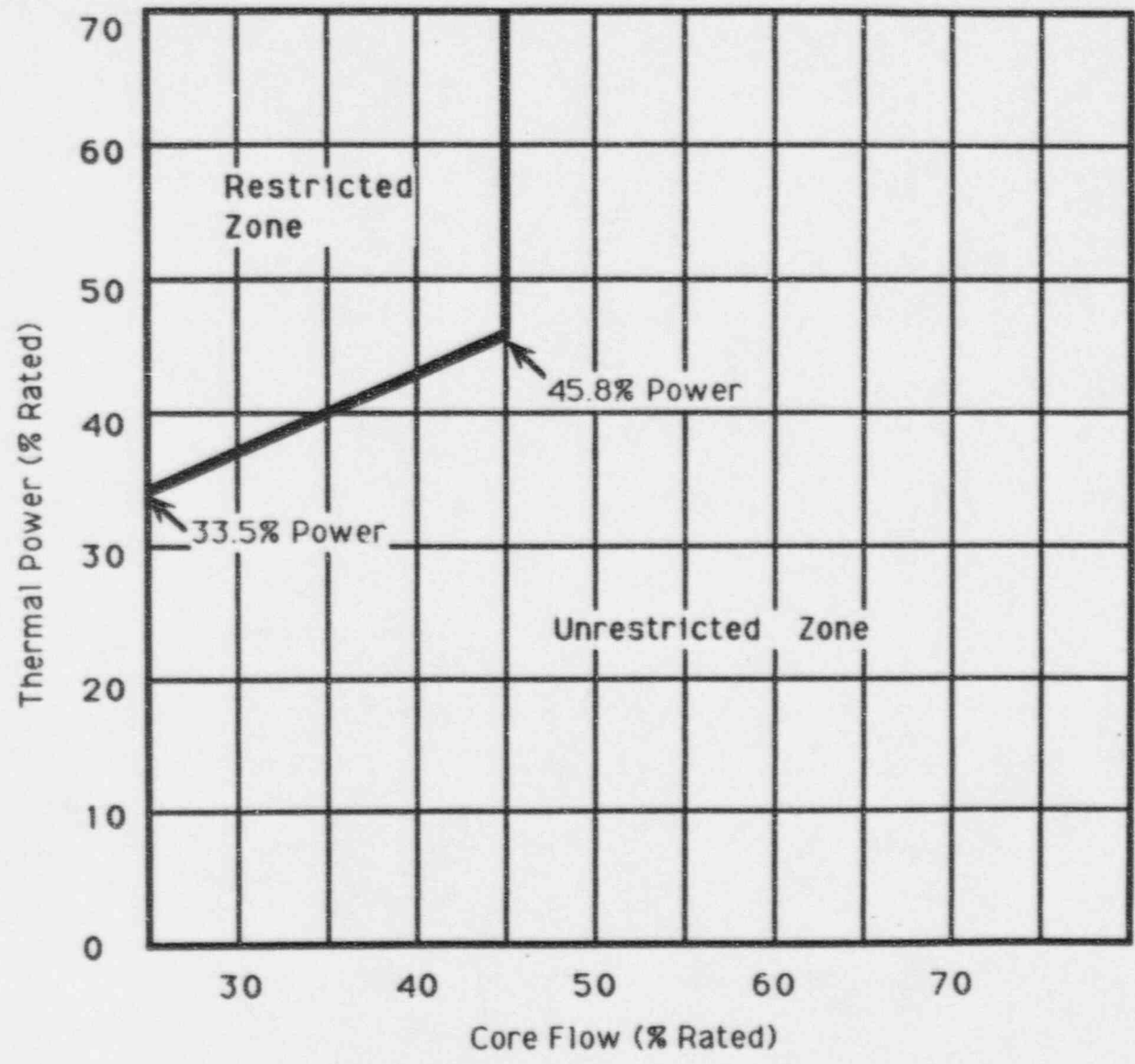


Figure 3.4.1.1-1 THERMAL POWER VERSUS CORE FLOW

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:##

- 4 safety/relief valves @ 1170 psig $\pm 1\%$
- 5 safety/relief valves @ 1180 psig $\pm 1\%$
- 5 safety/relief valves @ 1190 psig $\pm 1\%$

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.
- c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

- a. CHANNEL FUNCTIONAL TEST at least once per 92 days, and a
- b. CHANNEL CALIBRATION at least once per 24 months**.

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months.

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

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

Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A and A' for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cool-down following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

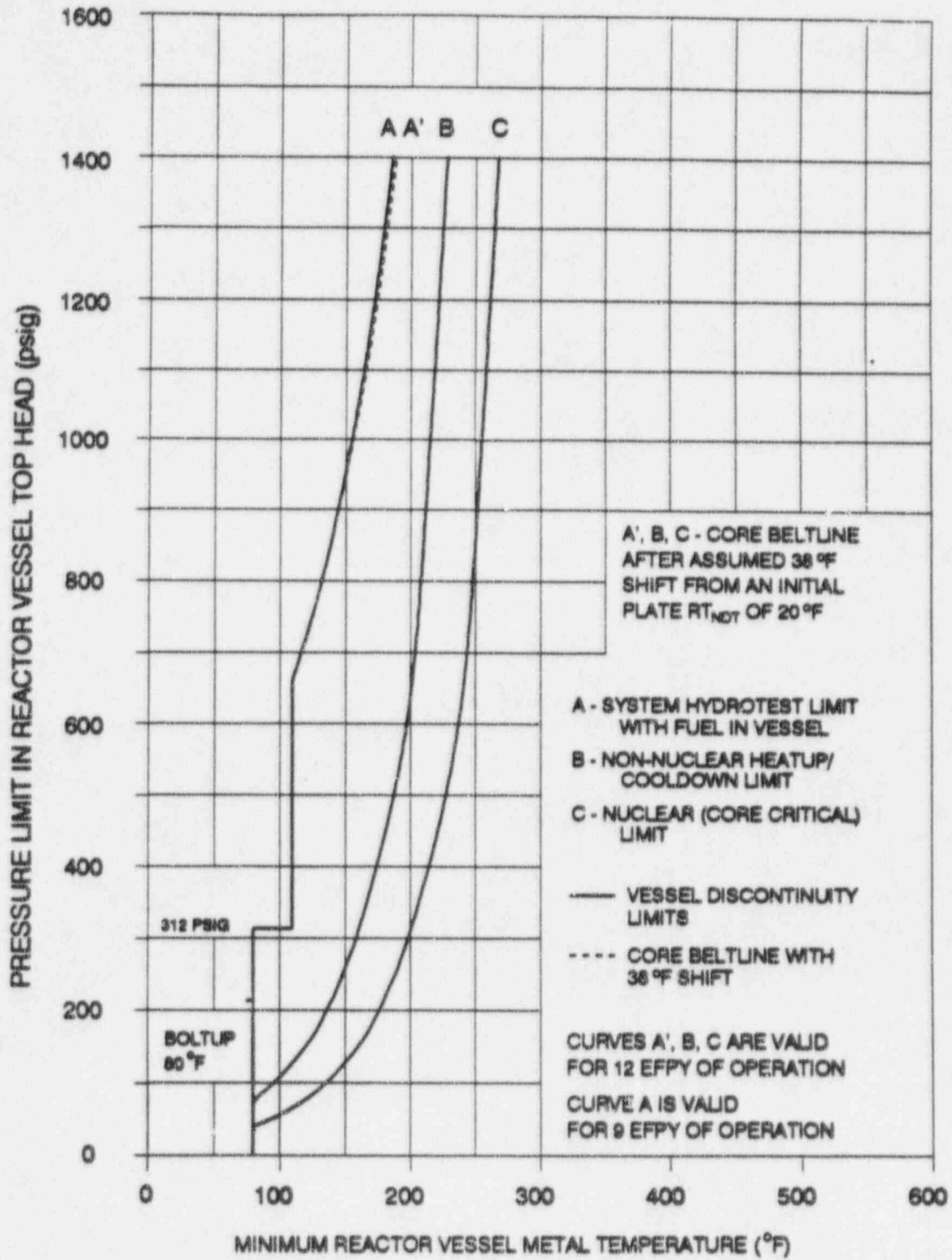
APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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



MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE
FIGURE 3.4.6.1-1

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1053 psig. |

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1053 psig, reduce the pressure to less than 1053 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours. |

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1053 psig at least once per 12 hours. |

*Not applicable during anticipated transients.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

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

- a. At least once per 31 days:
 1. For the CSS, the LPCI system, and the HPCI system:
 - a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 - b) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.
 2. For the LPCI system, verifying that both LPCI system subsystem cross-tie valves (HV-51-182 A, B) are closed with power removed from the valve operators.
 3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.
 4. For the CSS and LPCI system, performance of a CHANNEL FUNCTIONAL TEST of the injection header ΔP instrumentation.
- b. Verifying that, when tested pursuant to Specification 4.0.5:
 1. Each CSS pump in each subsystem develops a flow of at least 3175 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 105 psid plus head and line losses.
 2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psid plus head and line losses.
 3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure which corresponds to a reactor vessel pressure of 1040 psig plus head and line losses when steam is being supplied to the turbine at 1040, +13, -120 psig.**
- c. At least once per 24 months:
 1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72 hours.

3/4.6.5 SECONDARY CONTAINMENT

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY, restore REACTOR ENCLOSURE 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.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

- a. Verifying at least once per 24 hours that the pressure within the reactor enclosure secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.
- b. Verifying at least once per 31 days that:
 1. All reactor enclosure secondary containment equipment hatches and blowout panels are closed and sealed.
 2. At least one door in each access to the reactor enclosure secondary containment is closed.
 3. All reactor enclosure secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.
- c. At least once per 24 months:
 1. Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 126 seconds with the reactor enclosure recirc system in operation and
 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 1250 cfm with wind speeds of ≤ 7.0 mph as measured on the wind instrument on Tower 1, elevation 30' or, if that instrument is unavailable, Tower 2, elevation 159'.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the fan starts and isolation valves necessary to draw a suction from the refueling area or the reactor enclosure recirculation discharge 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 temperature differential across each heater is $\geq 15^{\circ}\text{F}$ when tested in accordance with ANSI N510-1980.
- e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 3000 cfm \pm 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 leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 3000 cfm \pm 10%.
- g. After any major system alteration:
 1. Verify that when the SGTS fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III).
 2. Verify that one standby gas treatment subsystem will drawdown reactor enclosure Zone I secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 126 seconds with the reactor enclosure recirculation system in operation and the adjacent reactor enclosure and refueling area zones are in their isolation modes.

PLANT SYSTEMS

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

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

ACTION:

- a. With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.
- b. In the event the RCIC system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
 1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
 3. Verifying that the pump flow controller is in the correct position.
- b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1040 + 13, - 120 psig.*

* The provisions of Specification 4.0.4 are not applicable, provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam pressure to less than 150 psig within the following 72 hours.

BASESPRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT} . The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curve A includes a shift in RT_{NDT} for conditions at 9 EFPY. The A', B and C limit curves are predicted to be bounding for all areas of the RPV until 12 EFPY when the beltline material's RT_{NDT} will shift due to neutron fluence and the beltline curves will intersect the non-beltline discontinuity curves.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and Charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires, Charpy specimens and vessel inside radius are essentially identical, the irradiated Charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and Charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

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

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

BASES TABLE B 3/4.4.6-1
REACTOR VESSEL TOUGHNESS*

<u>BELTLINE COMPONENT</u>	<u>WELD SEAM I.D. OR MAT'L TYPE</u>	<u>HEAT/SLAB OR HEAT/LOT</u>	<u>CU (%)</u>	<u>Ni (%)</u>	<u>STARTING RTNDT (°F)</u>	<u>ΔRTNDT ** (°F)</u>	<u>MIN. UPPER SHELF (LFT-LBS)</u>	<u>ART (°F)</u>
Plate	SA-533 Gr. B, CL. 1	C 7677-1	.11	.5	+20	+69	NA	+89
Weld	AB (Field Weld)	640892/ J424B27AE	.09	1.0	-60	+114	NA	+54

NOTES: * Based on 110% of original rated power.

** These values are given only for the benefit of calculating the end-of-life (EOL/32 EFPY) RTNDT

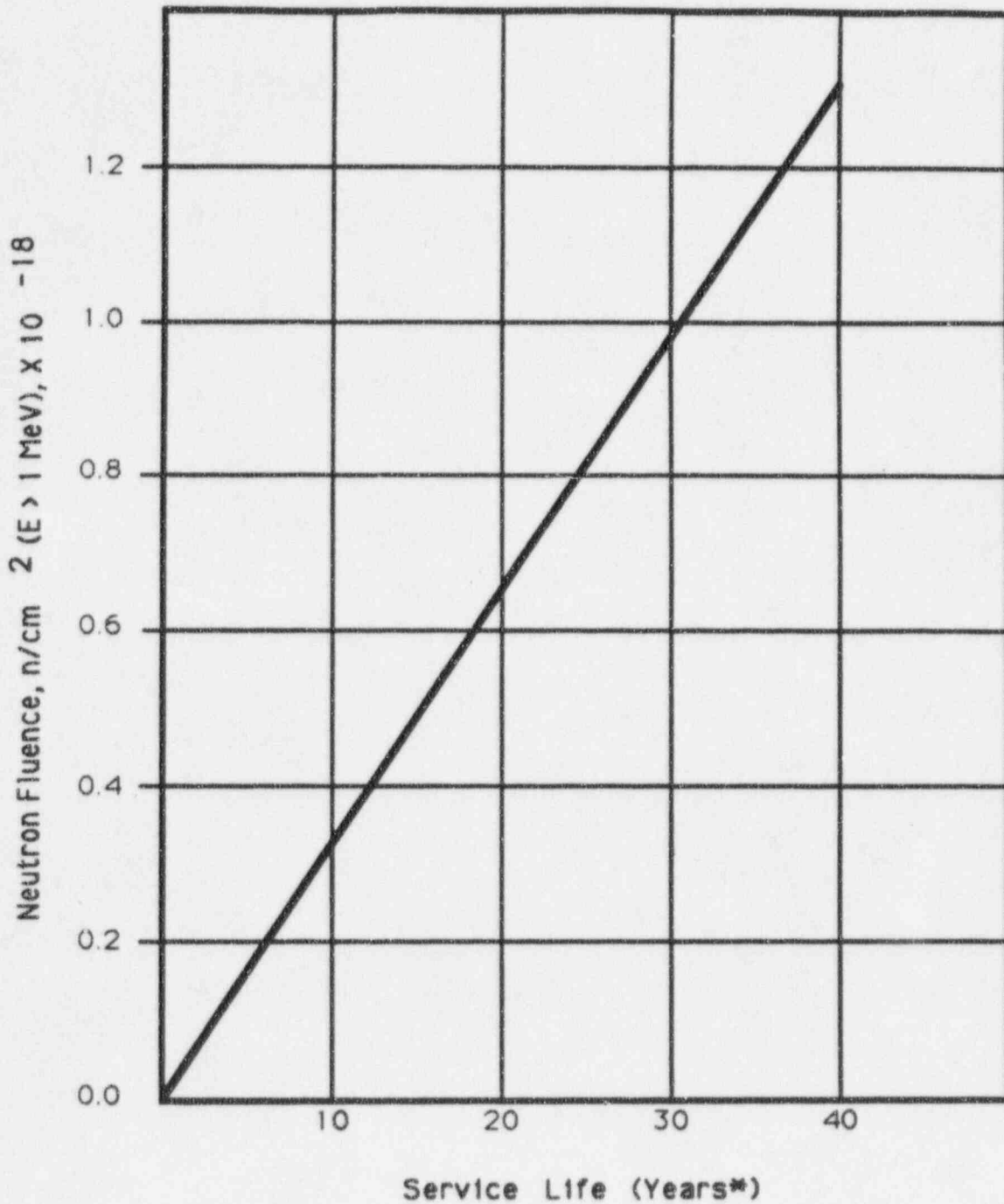
<u>NON-BELTLINE COMPONENT</u>	<u>MT'L TYPE OR WELD SEAM I.D.</u>	<u>HEAT/SLAB OR HEAT/LOT</u>	<u>HIGHEST STARTING RTNDT (°F)</u>
Shell Ring	SA 533, Gr. B, CL. 1	C7711-1	+20
Bottom Head Dome	"	C7973-1	+12
Bottom Head Torus	"	C7973-1	+12
Top Head Dome	"	A6834-1	+10
Top Head Torus	"	B1993-1	+10
Top Head Flange	SA-508, CL. 2	123B195-289	0
Vessel Flange	"	2V1924-302	-30
Feedwater Nozzle	"	Q2Q22W-412	-10
Weld	Non-Beltline	All	0
LPCI Nozzle***	SA-508, CL. 2	Q2Q25W	-6
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. Exp. at +10°F

Note: *** The design of the LPCI nozzles results in their experiencing an EOL fluence in excess of 10^{17} N/Cm² which predicts an EOL (32 EFPY) RTNDT of +42°F.

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BASES FIGURE B 3/4.4.6-1

FAST NEUTRON FLUENCE (E>1 MeV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

* At 90% of Rated Thermal Power and 90% availability

BASES3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

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

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

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

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

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

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5600 gpm at reactor pressures between 1182 and 200 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value calculated in the safety analyses for the peak accident pressure of ≤ 44 psig, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75 L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for leak testing of the main steam isolation valves, the airlock and TIP shear valves.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCK

The limitations on closure and leak rate for the primary containment air lock are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air lock will be in a closed and secured position during reactor operation. Only one closed door in the air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

BASES

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5.0 psid. The limit of - 1.0 to + 2.0 psig for initial containment pressure will limit the total pressure to ≤ 44 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting and pressure control. The 90 hours per 365 day limit on purge valve operation is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of SGTS. This limit is not imposed, however, on the subject valves when pressure control is being performed through the 2-inch bypass line, since a pressure surge through this line does not threaten the OPERABILITY of SGTS.

BASES

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 55 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from rated conditions. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the suppression chamber air space must not exceed 55 psig. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, suppression pool pressure during the design basis accident is below the design pressure. Maximum water volume of 134,600 ft³ results in a downcomer submergence of 12'3" and the minimum volume of 122,120 ft³ results in a submergence approximately 2'3" less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown through safety/relief valves assuming an initial suppression chamber water temperature of 95°F results in a bulk water temperature of approximately 140°F immediately following blowdown which is below the 190°F bulk temperature limit used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 140 seconds, these surveillance requirements specify a draw down time of 126 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS fans are sized for three zones and therefore, when aligned to a single zone or two zones, will have excess capacity to more quickly drawdown the affected zones. There is no maximum flow limit to individual zones or pairs of zones and the air balance and drawdown time are verified when all three zones are connected to the SGTS.

The three zone air balance verification and drawdown test will be done after any major system alteration, which is any modification which will have an effect on the SGTS flowrate such that the ability of the SGTS to drawdown the reactor enclosure to greater than or equal to 0.25 inch of vacuum water gage in less than or equal to 126 seconds could be affected.

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE (Continued)

- b. For a pressure of:
 - 1. 1250 psig on the suction side of the recirculation pump.
 - 2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
 - 3. 1500 psig from the discharge shutoff valve to the jet pumps.
- c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 552°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.
- b. A nominal center-to-center distance between fuel assemblies placed in the storage racks of greater than or equal to 6.244 inches.

5.5.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346'0".

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4117 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE (Continued)

- b. For a pressure of:
 - 1. 1250 psig on the suction side of the recirculation pump.
 - 2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
 - 3. 1500 psig from the discharge shutoff valve to the jet pumps.
- c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 552°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.
- b. A nominal center-to-center distance between fuel assemblies placed in the storage racks of greater than or equal to 6.244 inches.

5.5.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346'0".

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4117 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.