January 24, 1996

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: REVISED MAXIMUM AUTHORIZED THERMAL POWER LIMIT, LIMERICK GENERATING STATION, UNIT NO. 1 (TAC NO. M88392)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Facility Operating License and Technical Specifications in response to your application dated December 9, 1993, as supplemented in letters of July 5, September 9, October 19, November 15, and December 2, 1994, January 6, and January 23, 1995.

This amendment raises the authorized maximum power level from 3293 MWt to a new limit of 3458 MWt. The amendment also approves changes to the Technical Specifications to implement uprated power operation.

The staff issued the power uprate amendment for Unit 2 on February 16, 1995.

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/
Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-352

Enclosures: 1. Amendment No. 106 to License No. NPF-39
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Docket No. 50-352

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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) <u>Maximum Power Level</u>

   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:

   <u>Technical Specifications</u>

   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

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.

	Remove	Insert
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.

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

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. 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.

SURVEILLANCE REQUIREMENTS (...ntinued)

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

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
1.	MAIN STEAM LINE ISOLATION		
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.
2.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
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.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION		
a.	RWCS Δ Flow - High	≤ 54.9 gpm	≤ 65.2 gpm
b.	RWCS Area Temperature - High	≤ 142°F or 132°F**	≤ 147°F or 137°F**
c.	RWCS Area Ventilation Δ Temperature - High	≤ 32°F	≤ 40°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.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Δ Pressure - High	≤ 974" H_2O	≤ 984" H_2O
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	≥ 218°F, ≤ 247°F
e.	HPCI Equipment Room Δ Temperature - High	≤ 126°F	≤ 130.5°F
f.	HPCI Pipe Routing Area Temperature - High	175°F	≥ 165°F, ≤ 200°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

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure - High	≤ 373" H_2O	≤ 381" H_2O
b. RCIC Steam Supply Pressure - Low	≥ 64.5 psig	≥ 56.5 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig
d. RCIC Equipment Room Temperature - High	205°F	≥ 198°F, ≤ 227°F
e. RCIC Equipment Room Δ Temperature - High	≤ 109°F	≤ 113.5°F
f. RCIC Pipe Routing Area Temperature - High	175°F	≥ 165°F, ≤ 200°F
g. Manual Initiation	N.A.	N.A.
h. RCIC Steam Line Δ Pressure Timer	$3 \leq \tau \leq 12.5$ seconds	$2.5 \leq \tau \leq 13$ seconds

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -38 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

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
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. APRM		
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. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
4. INTERMEDIATE RANGE MONITORS		
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. SCRAM DISCHARGE VOLUME		
a. Water Level-High		
a. Float Switch	≤ 257' 5 9/16" elevation***	≤ 257' 7 9/16" elevation

3/4.4.1 RECIRCULATIC 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 ≤ 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 ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

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 ≤ 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 ≤ 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 ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°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.

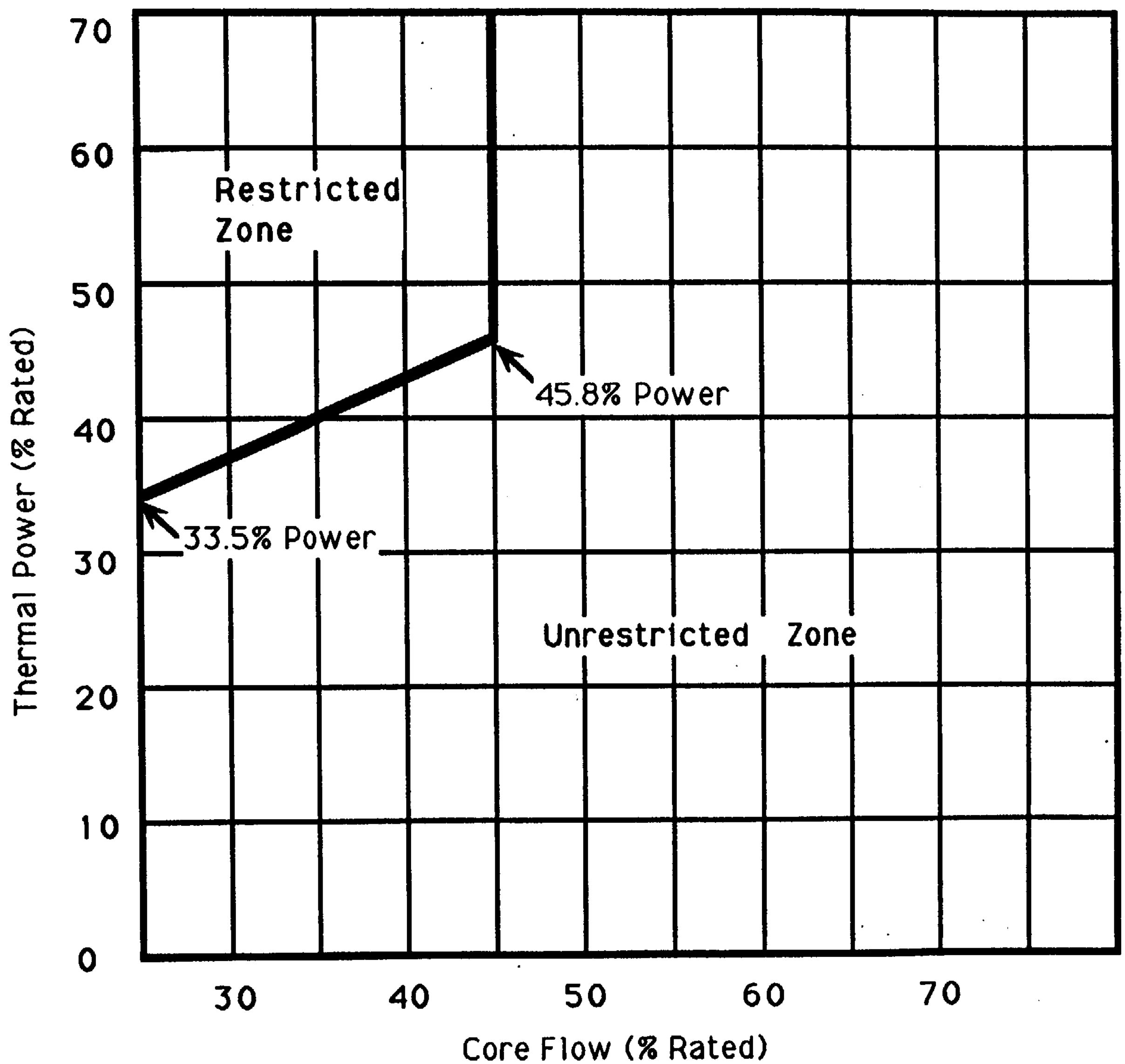

Figure 3.4.1.1-1 THERMAL POWER VERSUS CORE FLOW

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 ±1%
5	safety/relief valves @ 1180 psig ±1%
5	safety/relief valves @ 1190 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open 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.

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* 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.

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, cooldown 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.

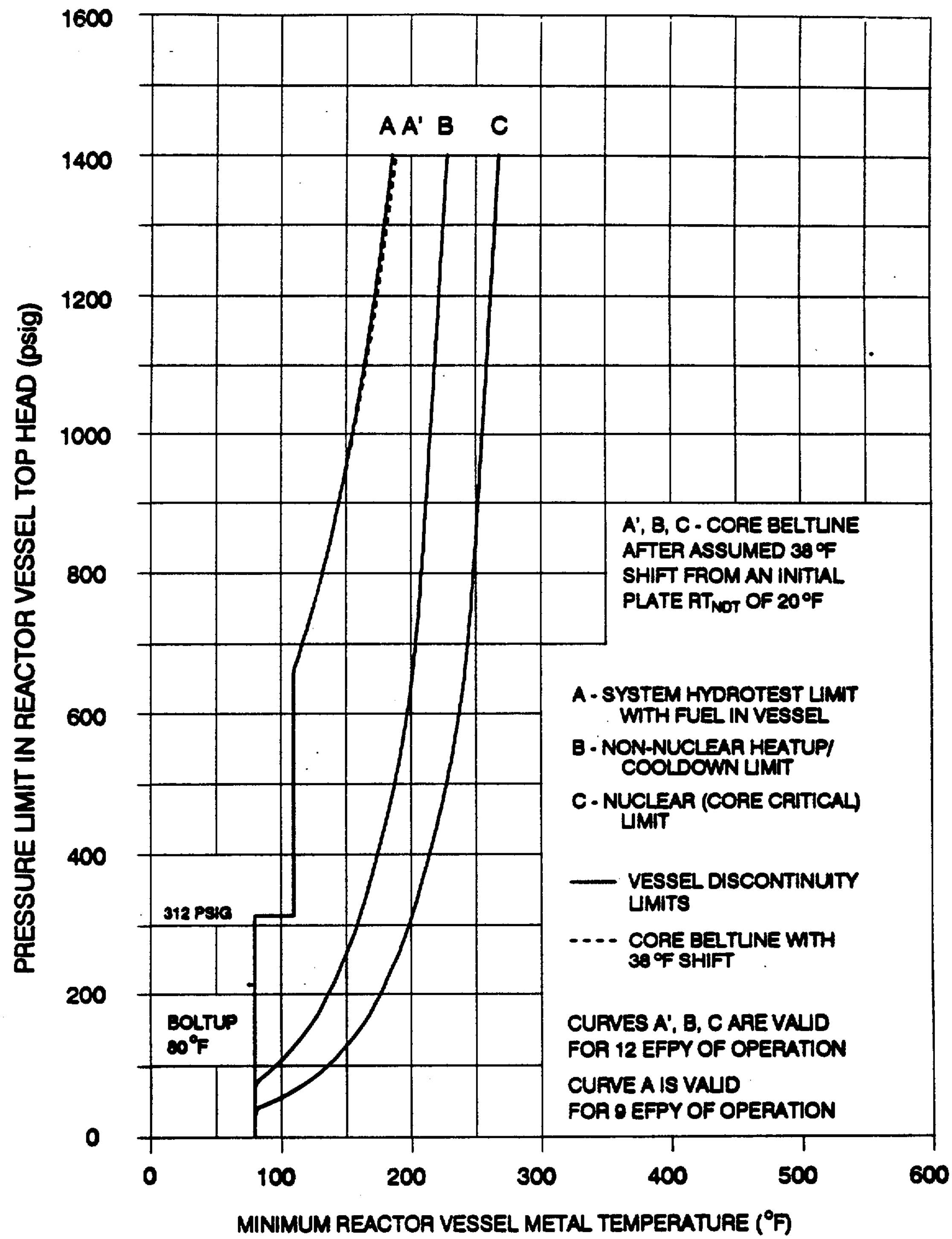
APPLICABILITY: At all times.

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.

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 $\geq$ 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 $\geq$ 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'.

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 ≥ 15°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 inplace 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 ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace 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 ± 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.

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.*

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

PRESSURE/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.

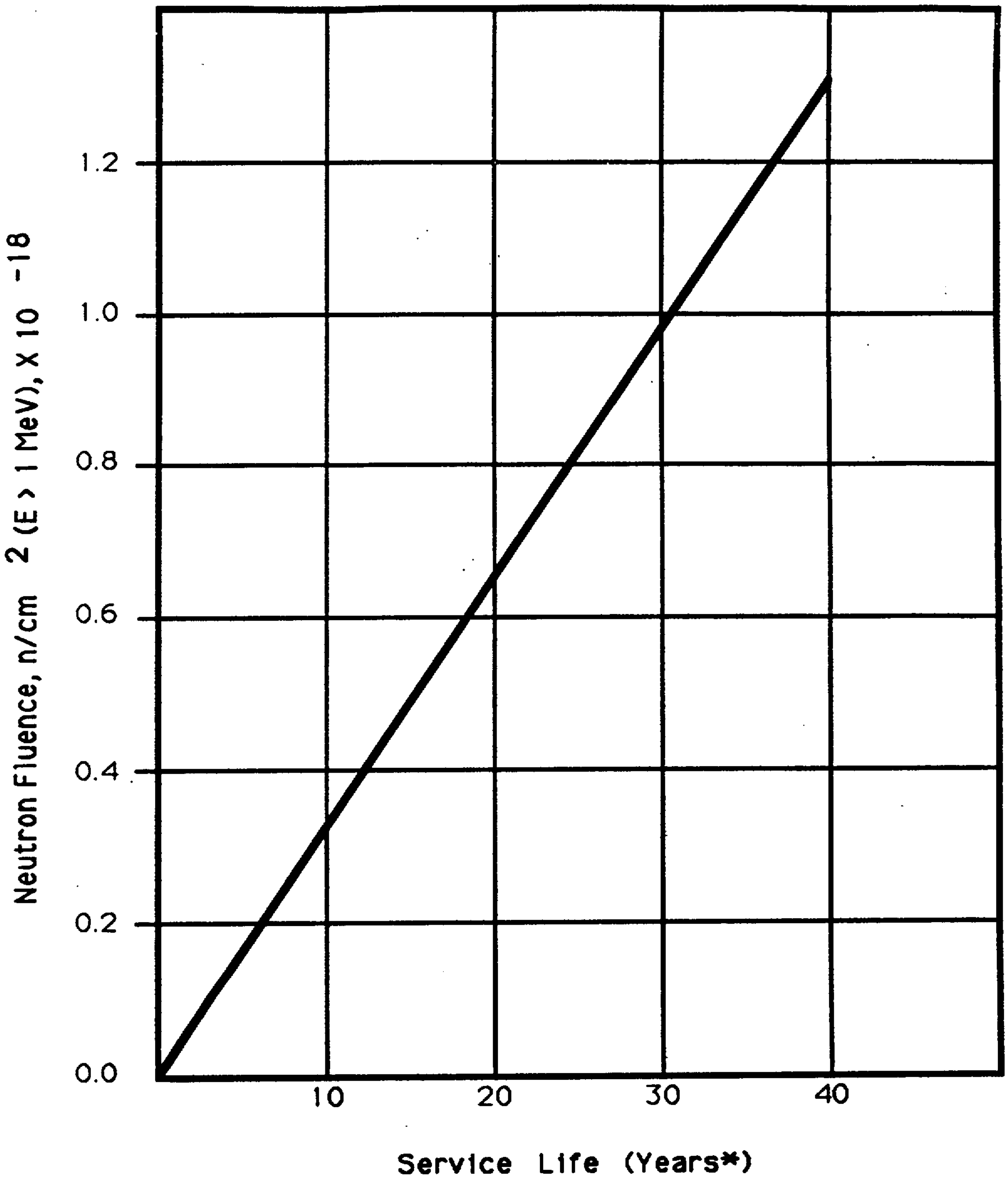
BASES TABLE B 3/4.4.6-1
REACTOR VESSEL TOUGHNESS*

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU (%)	Ni (%)	STARTING RT_{NDT} (°F)	ΔRT_{NDT} **(°F)	MIN.UPPER SHELF (LFT-LBS)	ART (°F)
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) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA 533, Gr. B, 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/Cm^2 which predicts an EOL (32 EFPY) RT_{NDT} of +42°F.



BASES FIGURE B 3/4.4.6-1

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

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

3/4.5.1 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 HCPI 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.

3/4.6.1 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.

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.

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 ft^3 results in a downcomer submergence of 12'3" and the minimum volume of 122,120 ft^3 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 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-39

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1 INTRODUCTION

In a letter of December 9, 1993 (Reference 1), as supplemented by letters of July 5, September 9, October 19, November 15, and December 2, 1994, January 6, and January 23, 1995, the Philadelphia Electric Company (PECO, the licensee) submitted a request for changes to the Operating License for Limerick Generating Station, Units 1 and 2, and for changes to Appendix A (Technical Specifications [TS]) to the Operating License. The licensee submitted NEDC-32225P, "Power Rerate Safety Analysis Report For Limerick Generating Station Units 1 and 2," Class III, September 1993 (Reference 2) as attachment 3 to Reference 1. The proposed amendment would increase the licensed thermal power level of the reactor from the current limit of 3293 megawatts thermal (MWt) to 3458 MWt. This request is in accordance with the generic BWR power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter of September 30, 1991. In letters of July 5, September 9, October 19, November 15, and December 2, 1994, January 6, and January 23, 1995, the licensee submitted clarifying information that did not change the initial proposed no significant hazards determination, which was published in the *Federal Register* on February 16, 1994 (59 FR 7695). The staff issued the power uprate amendment for Unit 2 on February 16, 1995.

2 DISCUSSION

On December 28, 1990, GE submitted GE Licensing Topical Report (LTR) NEDC-31897P-1, in which it proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent (Reference 3). The report contained a proposed outline for individual license amendment submittals and discussed the scope and depth of reviews needed and the methodologies used in these reviews. In a letter of September 30, 1991, the NRC approved the program proposed in the report, on the condition that individual power uprate amendment requests meet certain requirements in the document (Reference 4).

The generic BWR power uprate program gives each licensee a consistent means to recover additional generating capacity beyond its current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level for most licensees was based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the design power level is often referred to as *stretch power*. The design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS). Therefore, increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment and does not significantly affect the reliability of this equipment.

The licensee's amendment request to increase the current licensed power level of 3293 MWt to a new limit of 3458 MWt represents an approximate 5-percent increase in thermal power with a corresponding 5.9-percent increase in rated steam flow (an increase in vessel steam flow from 14.16 to 14.99 Mlb/h). The licensee will increase power to the higher level by (1) increasing the core thermal power to increase steam flow, (2) increasing the feedwater system flow by a corresponding amount, (3) increasing reactor pressure to ensure adequate turbine control margin, (4) not increasing the current maximum core flow, and (5) operating the reactor primarily along extensions of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in Reference 3. The operating pressure will be increased approximately 40 psi to ensure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

3 EVALUATION

The staff reviewed the request for a Limerick power uprate amendment using applicable rules, regulatory guides, sections of the Standard Review Plan, and NRC staff positions. The staff also evaluated the Limerick submittal (Reference 2) for compliance with the generic BWR power uprate program as defined in Reference 3. Detailed discussions of individual review topics follow.

3.1 Reactor Core and Fuel Performance

The staff evaluated the power uprate for its effect on areas related to reactor thermo-hydraulic and neutronic performance such as the power/flow operating map, core stability, reactivity control, fuel design, control rod drives, and scram performance. The staff also considered the effect of power uprate on reactor transients, anticipated transients without scram (ATWS), ECCS performance, and peak cladding temperature for design basis accident break spectra.

3.1.1 Fuel Design and Operation

The licensee stated that no new fuel designs would be needed to increase power, which is consistent with the information submitted by GE in LTR NEDC-31984P (Reference 5). The plant will continue to meet fuel operating limits such as the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) for future reloads. The methods for calculating MAPLHGR and OLMCPR limits will not be changed by power uprate, although actual thermal limits may vary between cycles. Cycle-specific thermal limits will be included in the plant core operating limits report (COLR).

3.1.2 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. Changes to the power/flow operating map are consistent with previously approved generic descriptions in Sections 5.2 and C.2.3 of Reference 3. The maximum thermal operating power and maximum core flow correspond to the uprated power and the previously analyzed core flow range. Uprated power has been rescaled so that it is equal to 100-percent rated.

3.1.3 Stability

The licensee evaluated the effect of power uprate on core stability issues according to the generic guidelines for power uprate (Reference 5). To determine the effect on core stability, the licensee reviewed recommendations from GE Service Information Letter SIL-380, Revision 1, NRC Bulletin 88-07, Supplement 1 (Reference 6), and current Boiling Water Reactor [BWR] Owners Group (BWROG) efforts including interim corrective actions (ICAs) recommended by GE and the BWROG.

The licensee adjusted the percent power on the revised power/flow map such that the ICA region boundaries have the same actual power (MWt); thus, LGS Units 1 and 2 will have the same level of protection against thermal-hydraulic instability. Furthermore, the analysis shows that the power increase will not affect the application of any of the BWROG stability long-term solution options at LGS Units 1 and 2.

The staff concluded that the licensee addressed thermal hydraulic stability in an acceptable manner.

3.1.4 Control Rod Drives and Scram Performance

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The licensee evaluated the CRD system at the uprated steam flow and dome pressure.

The increase in dome pressure at uprated power will increase the bottom head pressure a corresponding amount. Although the increased pressure will slow rod insertion initially, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Increased reactor pressure has little effect on scram time, and CRD performance during power uprate will meet current TS requirements. The licensee will monitor scram performance by following various TS surveillance requirements to preserve the original licensing basis for the scram system.

Power uprate conditions reduce the operating margin between available and required drive water differential pressure. For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The licensee analyzed plant CRD pump and system data and found that the Unit 1 and Unit 2 CRD pumps must be modified to give adequate head and flow for CRD positioning at uprated conditions. In a letter of September 9, 1994 (Reference 7), the licensee stated that the modification will replace the existing CRD pumps, motors and gear boxes with new, higher capacity, direct-drive pumps. This is the only CRD system modification determined to be necessary by the licensee.

The licensee did an uprate analysis and determined that the existing CRD pumps would not have sufficient capacity under uprated conditions for normal control rod positioning operations. The pumps lacked sufficient capacity to compensate for the high line losses between the CRD pump discharge and the CRD flow control station. Use of the existing CRD pumps at uprated conditions and a 250 psid between the CRD system and the reactor would have resulted in cooling water flows of approximately 35 gpm. This would have led to an increased number of control rod drives running hot. The licensee discussed the issue with the manufacturer of the existing pumps, reviewed the pump operating history and decided to replace the existing pumps with new pumps.

Based on the above, the staff concludes that the CRD system, as modified, will continue to perform all its safety-related functions at uprated power, and will function adequately during insert and withdraw modes and is, therefore, acceptable.

3.2 Reactor Coolant System and Connected Systems

The staff reviewed the mechanical engineering portions of the LGS power uprate amendment request to determine the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components. The staff's review is discussed below.

3.2.1 Nuclear Steam Pressure Relief

The plant safety/relief valves (SRV) and reactor scram function relieve pressure from the nuclear system to prevent over pressurization of the nuclear system during abnormal operational transients. The only change in the nuclear system pressure relief for power uprate is to increase the SRV setpoints to accommodate the increased uprate dome pressure. The maximum operating dome pressure was selected to enable the turbine control valves (TCVs) to operate effectively at the higher steam flow condition corresponding to rerated power. The SRV setpoints will be increased to compensate for the increased rerate dome pressure. An appropriate increase in the SRV setpoints ensures that adequate differences between operating pressure and setpoints (simmer margins) are maintained, and that the increase in dome pressure does not increase the number of unnecessary SRV actuations. The analysis described in Section 3.2.2 indicates that the nuclear boiler pressure relief system has the capability to accommodate the power uprate.

3.2.2 Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor pressure coolant boundary remains at 1250 psig. The American Society of Mechanical Engineers (ASME) code allows a peak pressure of 1375 psig (110 percent of design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steam isolation valve (MSIV) closure with a failure of valve position scram. The licensee's rerate analysis assumes that the event initiates at a pressure of 1053 psig, which is higher than the normal rerated dome pressure. The analysis also assumes that three SRVs are out of service. The analysis resulted in a conservatively high peak RPV bottom pressure of 1342 psig which remains below the 1375-psig ASME limit. The staff finds this acceptable.

3.2.3 Reactor Vessel and Internals

The licensee evaluated the reactor vessel and internal components considering load combinations that include reactor internal pressure difference (RIPD), seismic, other loads such as loss-of-coolant-accident (LOCA), safety relief valve (SRV), annulus pressurization (AP), and fuel lift loads. The seismic loads will not change for the power uprate conditions. In Section 4.1.2.1 of Reference 2, the licensee evaluated LOCA loads such as pool swell, condensation oscillation (CO), and chugging for the LGS power uprate and found that the original LOCA analyses remain unchanged because the containment condition with the power uprate are within the range of test conditions used to define the LOCA dynamic loads. In Section 4.1.2.2 of Reference 2, the licensee evaluated SRV containment dynamic loads that affect the reactor vessel and piping systems. The licensee stated that the increase of SRV loads resulting from the change of SRV setpoints are within the range of the original LGS SRV load definitions that were based on a reference reactor pressure of 1276 psig. Therefore, the original SRV loads remain bounding for the power uprate condition. The licensee reviewed the original analyses for the AP loads and found that the mass and energy release rates used for calculation of the original analyzed loads bound the uprated power conditions. Based on the above review, the staff concurs with the licensee's evaluation that the LOCA, SRV, and AP design basis loads remain bounding for the LGS power uprate.

The licensee determined that the fuel lift load for the top guide will increase by less than 10 percent and that all previous fuel lift loads on other reactor internal components are bounding for the power uprate. The calculated RIPDs for the uprated power conditions were summarized in Tables 3-1, 3-2 and 3-3 of Reference 2 for normal, upset and faulted conditions respectively.

Considering the increase in fuel lift loads and the RIPDs, the licensee evaluated the stresses and fatigue usage factors for reactor vessel components in accordance with the requirements of the 1968 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division I, 1968 Edition with Addenda through Winter 1969 (Reference 8), Subsection NB, to assure compliance with the LGS original Code of Record. The maximum stresses at the critical locations were summarized in Table 1 of Reference 9. The fatigue usage factors of limiting components calculated for the uprated power level were listed in Table 3-4 of Reference 2. The fatigue usage factors were also found to be acceptable except that the increased cumulative usage in the feedwater nozzles reduced their fatigue life from the original 32 years to 23 years. In response to NRC's request for additional information (Reference 9), the licensee committed to revise the LGS refurbishment interval to reflect the new feedwater nozzle fatigue life of 23 years. No new assumptions were used in the analysis for the power uprate condition.

Based on its review of the information provided by the licensee, the staff concludes that the maximum stresses and fatigue usage factors submitted by the licensee are within the Code-allowable limits and are therefore acceptable.

3.2.4 Control Rod Drive Mechanism

The licensee evaluated the adequacy of the LGS control rod drive mechanism (CRDM) in accordance with the ASME Boiler and Pressure Vessel Code Section III, Division I, 1968 Edition with Addenda through Winter 1969 (Reference 8). The licensee found the limiting component of the CRDM to be the indicator tube. The maximum calculated stress was based on a maximum CRD internal water pressure of 1750 psig, and this basis is not affected by the operation at uprated conditions. The licensee calculated a maximum fatigue usage factor of 0.15 for the CRD main flange considering 40 years of plant operation. The increase in the reactor dome pressure, operating temperature and steam flow rate as a result of the power uprate are bounded by the conditions assumed in the General Electric generic guidelines for the power uprate (Reference 3). The CRDM was originally evaluated for a normal maximum reactor dome pressure of 1060 psig which is higher than the power uprate dome pressure of 1045 psig. The licensee also stated that the CRDM has been tested at simulated reactor pressure up to 1250 psig, which bounds the high-pressure scram setpoint of 1111 psig for the power uprate.

Based on its review of the licensee's information, the staff concludes that the CRDM will continue to meet its design basis and performance requirements at uprated power conditions, and is therefore, acceptable.

3.2.5 Reactor Recirculation System

The licensee will increase power to the uprated level by operating along higher rod lines with no increase in the licensed maximum core flow of 105 percent of rated power. The core reload analyses are performed with the most conservative allowable core flow. The thermal-hydraulic performance of the reactor recirculation system (RRS) at the uprated power condition indicated that the core flow can be maintained at 105 percent.

In response to a staff question regarding performing evaluation and testing of recirculation system vibration, the licensee stated in their September 9, 1994 letter (Reference 7) that a detailed vibration analysis was performed for the RRS piping for rerate conditions, and that rerate resulted in a negligible effect. The licensee further stated in the letter that recently a phenomenon described as "containment noise" has occurred at another U.S. BWR. This phenomenon has been related to operation at increased core flow (ICF) above 100-percent rated flow. The increased recirculation pump speed associated with ICF is theorized to be a contributing factor to this phenomenon. According to the licensee, this phenomenon is not believed to be related to power rerate since only a very small increase in recirculation pump speed is required to maintain a given core flow at rerate conditions. An investigation at this other U.S. BWR to determine the source of this phenomenon is underway, and the licensee is following this investigation closely.

LGS Units 1 and 2 have operated with ICF up to 105-percent rated flow for many cycles, and no incidents of the "Containment Noise" phenomenon have been reported. Analyses have been performed for LGS justifying ICF operation up to 110-percent rated flow including rerate conditions. The licensee stated that it planned to implement 110-percent ICF near the end of the LGS Unit 2 Cycle 3 Operating Cycle (September 1994), prior to the implementation of power rerate the following cycle.

The licensee is developing a monitoring program for the implementation of 110-percent ICF. This program is likely to include baseline measurements and trending of noise and vibration levels in key areas of the reactor building and monitoring the vibration instrumentation currently installed on the recirculation pump motors and shafts. Implementation of 110-percent ICF will consist of a gradual progression from 105-percent core flow up to 110-percent core flow over a period of about 1 week. This approach will allow the licensee to closely monitor the effects of the increasing recirculation pump speed and respond appropriately if the "Containment Noise" phenomenon is encountered.

Design pressures for the RRS components (including the suction and discharge valves, recirculation pumps, and piping) are based on the design pressure for the reactor pressure vessel because the recirculation piping loops are part of the reactor coolant pressure boundary (RCPB). Raising the steam dome pressure to operate at the uprated power will increase the RRS pump suction pressure and the RRS pump discharge pressure. These increases are within the system design pressures. Thus, the design pressure margin for the RRS suction and discharge lines will support operation at the uprated power.

Design temperatures for the RRS components (including the suction and discharge valves, recirculation pumps, and piping) are based on the design temperature for the reactor vessel. Operation at the uprated power condition will increase the RRS pump suction and discharge temperatures by less than 5 °F. This increase is within the RRS design temperature. Therefore, the RRS has sufficient design temperature margin for operation at the uprated power condition.

The RRS thermal-hydraulic performance results show that operations at the uprated power condition will require small (approximately 0.6-percent) increases in the RRS pump speed, pump drive flow, pump motor horsepower, and motor generator (MG)-set generator output power. The RRS pump, pump motor, and MG-Set include sufficient design capacity margins to accommodate the required increases and to support operation at the uprated power.

The interlocks and pump runbacks affected by power uprate are discussed below.

1. Originally, when the feedwater flow was less than a minimum value (typically 20 percent of rated), the RRS pump speed would decrease (*run back*) to its minimum value to prevent cavitation, which might occur if the feedwater subcooling becomes low enough to sufficiently reduce the net positive suction head (NPSH) available to the pump.

   The licensee evaluated whether or not increasing the feedwater flow by 5.8 percent as needed for the power uprate would affect the cavitation setpoint. The licensee found no change needed in the setpoint because the setpoint is expressed in terms of *absolute feedwater flow*. Therefore, as feedwater flow increases, the cavitation setpoint (expressed in percentage) will be slightly lower than the original setpoint.

2. If a feedwater or condensate pump trips while the reactor is operating at high power and the reactor water level is at or below level 4, the RRS pump speed is automatically decreased to an intermediate speed. The purpose of the runback is to avoid unnecessary scrams by reducing the RRS drive flow to a rate more compatible with the reduced feedwater flow, thus avoiding unnecessary scrams. The RRS pump speed runback setpoint is 42 percent of rated pump speed, which corresponds to a core flow greater than the maximum core flow in the power/flow map stability exclusion region.

Based on the information discussed above in this section, the staff concludes that the existing RRS design has sufficient margin to accommodate operation at the uprated power condition, and is, therefore, acceptable.

3.2.6 Reactor Coolant Piping

The licensee evaluated the effects of the power uprate, including higher flow rate, temperature and pressure for thermal expansion, the effects of fluid transients and vibration on the reactor coolant pressure boundary (RCPB) and the balance-of-plant (BOP) piping systems, including inline components such as equipment nozzles, valves and flange connections, and pipe supports. The licensee performed this evaluation to ensure compliance with requirements of the code of record for various systems and components as specified in the LGS UFSAR. The licensee evaluated piping systems affected by the power uprate and by the methodology listed in Section 3.12 of Reference 2.

The RCPB piping systems evaluated include main steam and associated extraction and drain system, reactor recirculation line, reactor water clean-up (RWCU), reactor core isolation cooling (RCIC), condensate and feedwater system, high pressure coolant injection (HPCI), residual heat removal (RHR) and instrumentation sensing lines. The licensee evaluated the RCPB piping systems by comparing the maximum percentage increase in stress for the power uprate (caused by increased pressure, temperature, and fluid transient loads) with the design margins available in the original design basis analyses, and doing stress analyses for the power uprate in accordance with requirements of the Code and the Code Addenda of Record. The licensee concluded that the Code requirements remain satisfied for the evaluated piping systems and that power uprate will not have an adverse effect on the reactor coolant piping system design.

The licensee verified the adequacy of BOP systems from the uprated reactor and BOP heat balances. These systems include lines affected by the power uprate, of which the most limiting systems determined by the licensee are the main steam relief valve discharge, main steam (outside drywell) and feedwater systems (Reference 9). The licensee evaluated the maximum stress levels and fatigue usage factors for BOP piping based on the bounding percentage increases given in Table 3-5 of Reference 2 and concluded that in a majority of the BOP systems, there are sufficient margins between the original design stresses and the Code limits to accommodate the stress increase due to the power uprate. The licensee evaluated those systems whose design temperature and pressure did not envelop the uprate power conditions and concluded that the actual calculated pipe stresses and support loads remained within the Code-allowable limits.

The licensee evaluated pipe supports including anchorage, equipment nozzles, and penetrations by comparing the increased piping interface loads on the system components with the margin in the original design basis calculation. The increased interface loads would result from thermal expansion from the power uprate. The licensee found sufficient margin between the original design stresses and the Code limits to accommodate the stress increase for all service levels at the uprated power. The licensee also evaluated the effect of power uprate conditions on thermal and vibration displacement limits for struts, springs and pipe snubbers, and found it acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and found no new pipe break locations.

The licensee's submittal shows that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant piping and supports in the power uprate conditions, and is therefore acceptable.

3.2.7 Main Steam Isolation Valves

The licensee evaluated the MSIVs and found them consistent with the bases and conclusions of the generic evaluation. However, the bases for the LGS evaluation vary slightly from the generic evaluation. The differences are that the reactor operating pressure for LGS is 1060 psia instead of 1045 psia, and the operating temperature for LGS is 551.7°F instead of 545.7°F. The licensee indicates that these differences were assessed to be insignificant.

MSIV performance will be monitored according to surveillance requirements in the technical specification to ensure original licensing basis for MSIVs are preserved. Maintenance of MSIV performance to existing licensing basis standards is acceptable to the staff.

3.2.8 Reactor Core Isolation Cooling System

The RCIC system provides core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than maximum allowable for initiation of a low pressure core cooling system. The RCIC system was evaluated and found consistent with the generic evaluation. The 40 psi increase in nominal reactor operating pressure (for rerate) will require an increase in turbine speed in order to maintain system design margin. This reduces the available overspeed trip margin. But as discussed by the licensee in the November 15, 1994 letter (Reference 10), which is summarized in the paragraphs below, the licensee expects that it will not impact system operability and reliability. Nevertheless, in order to reduce the possibility of turbine overspeed trips, the licensee plans to install the RCIC bypass start modification (or equivalent modifications) described in GE SIL No. 377 as part of the power rerate program.

The modifications to the RCIC turbine assembly, as described in GE SIL No. 377, are for improvement of the turbine startup transient response. This modification is in use on a number of turbine assemblies and has performed very satisfactorily and reliably in the industry. The implementation of this modification effectively limits the initial response of the turbine speed on startup at high reactor pressures. This reduces the probability of turbine overspeed trips, as well as reducing cyclic pressure forces and loads on certain components, thus improving overall system reliability. Consequently, this modification results in the higher power rerate reactor steam pressure having an insignificant impact on the turbine startup transient response.

For the RCIC system, the increase in reactor operating pressure with power rerate required an increase in the pump total dynamic head (TDH) to maintain the same injection rate to the reactor. This was accomplished through an increase in the RCIC pump speed. Vendor pump test data for the LGS RCIC pumps and the affinity laws for centrifugal pumps were used to determine the new speed and horsepower requirements for the pump. Operation of the pump at the higher speed does not result in the pump or system components exceeding their specified design pressures. Turbine performance curves indicate that the unit has more than adequate speed and horsepower capability to drive the pump at its new operating point.

The change to the RCIC turbine control system to increase its maximum rated speed from 4500 RPM to 4575 RPM will have no adverse effect on system reliability. This maximum rated speed forms part of the turbine controller calibration and will only be limiting when the system is operating at its maximum design injection pressure as established by the safety relief valve setpoint and upper tolerance limits. The majority of RCIC system operation occurs at reactor pressures equal to or less than the reactor normal operating pressure. Furthermore, a flow test of RCIC system injection capabilities will be included in the testing program. This flow test will involve RCIC returning flow to the condensate storage tank. This testing is in conformance with the Power Rerate Startup Testing Program recommended by General Electric.

Based on the staff's review of the licensee's information, the staff concludes the RCIC system is acceptable for operation at uprated power conditions.

3.2.9 Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove decay heat from the primary coolant system after reactor shutdown for both normal and postaccident conditions. The RHR system is also designed to operate in the low-pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The LPCI mode is discussed in Section 3.3.2.2 of this safety evaluation.

(a) Shutdown Cooling Mode

The licensee evaluated the shutdown cooling mode of the RHR system. The operational objective of the system during normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At 110-percent of original rated thermal power, the decay heat is increased, which slightly increases (to 22.7 hours) the time required to reach the (125°F) shutdown temperature. The staff agrees with the licensee's conclusion that this has no effect on plant safety.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," provides guidance for demonstrating cold shutdown (200°F reactor fluid temperature) capability within 36 hours. The UFSAR Section 5.4.7 indicates that cold shutdown can be reached in a much shorter time even considering the availability of only one RHR heat exchanger. For power rerate, an alternate shutdown cooling analysis was performed by the licensee, based on the criteria in Regulatory Guide 1.139 and 110-percent of original rated thermal power. The results of this analysis show that with power rerate the reactor can still be cooled to 200°F in less than the 36 hours. The staff finds this acceptable.

(b) Suppression Pool Cooling Mode

The functional design basis as stated in the UFSAR for the suppression pool cooling mode is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate since the licensee's analysis confirms that the pool temperature will stay below its design limit. Section 3.3.1 provides further discussion on suppression pool temperature response.

(c) Containment Spray Cooling Mode

In the containment spray cooling mode, the RHR system supplies water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during postaccident conditions. Power uprate will increase the containment spray temperature by only a few degrees. This increase will have a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these parameters reach peak values before containment spray begins. The temperature increase has been evaluated by the licensee, and determined not to affect the function and operation of the containment spray mode.

3.2.10 Reactor Water Cleanup System

The operating temperature and pressure of the RWCU system will increase slightly as a result of power uprate. The licensee evaluated the effect of these increases and concluded that uprate will not adversely affect RWCU system integrity. Although increased feedwater flow to the reactor may slightly diminish the cleanup effectiveness of the RWCU system, the power uprate will not require a change in TS limits for reactor water chemistry. Therefore, the power uprate will not significantly affect the operation or coolant boundary integrity of the RWCU system.

3.3 Engineered Safety Features

The staff reviewed the effect of power uprate on containment system performance, the standby gas treatment system (as affected by increased iodine loading), post-LOCA combustible gas control, the control room atmosphere control system, and the emergency cooling water system. The staff did this review to verify that the uprate would not impair the ability of these systems to do their safety functions to respond to or mitigate the effects of design-basis accidents. The staff also considered the effects on high-energy line breaks, fire protection, and station blackout.

3.3.1 Containment System Performance

The Limerick final safety analysis report provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with power uprate changes some of the conditions for the containment analyses. Section 5.10.2 of Topical Report Reference 3 specifies that the power uprate applicant must show acceptability of the uprated power level for: (1) containment pressures and temperatures, (2) LOCA containment dynamic loads, and (3) safety-relief valve dynamic loads. Appendix G of Reference 3 prescribes the approach to be used by power uprate applicants for performing required plant-specific analyses. The licensee did the necessary analyses and discussed the results in the application.

Appendix G of Reference 3 states that the applicant will analyze short-term containment responses using the staff-approved M3CPT code. M3CPT is used to analyze the period from when the break begins to when pool cooling begins. M3CPT generates data on the response of containment pressure and temperature (Section 3.3.1.1), for dynamic loads analyses (Section 3.3.1.2), and for equipment qualification analyses (Section 3.8.2).

Appendix G of Reference 3 states that the applicant will do long-term containment heatup (suppression pool temperature) analyses for the limiting safety analysis report events to show pool temperatures will be within the limits for:

- containment design temperature
- local pool temperature
- net positive suction head
- pump seals, piping design temperature, and other limits.

These analyses will use the SHEX code and ANS 5.1-1979 decay heat assumptions consistent with the staff's letter to Mr. Gary L. Sozzi (Reference 11). SHEX, which is partially based on M3CPT, is a long-term code to analyze the period from when the break begins until after peak pool heatup.

3.3.1.1 Containment Pressure and Temperature Response

The UFSAR documents short-term and long-term containment analyses of the response of containment pressure and temperature after a large break inside the drywell. The short-term analysis is primarily to determine the peak drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment after a design basis accident (DBA) LOCA. The long-term analysis is primarily to determine the peak pool temperature response.

3.3.1.1.1 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

The licensee evaluated the long-term bulk response of the suppression pool temperature for the DBA LOCA at 3694 MWt (102 percent of 110 percent of original rated power) using the SHEX code and ANS 5.1 decay heat assumptions prescribed by Reference 3. All other key input parameters for power uprate analyses were essentially the same as those for the original analyses. For the power uprate, the DBA-LOCA peak suppression pool temperature was calculated to be 205°F. The peak suppression pool temperature is well within the 220°F suppression pool structural design temperature and does not exceed the low pressure ECCS pump limit of 212°F.

The licensee indicated that the highest pool temperature response from a non-LOCA event results from an alternate shutdown cooling event analyzed at 3694 MWt. The event assumes reactor isolation with only one RHR heat exchanger available to accommodate SRV discharge to the suppression pool and results in a maximum pool temperature of 212°F, which does not exceed the above pool design or ECCS temperature limits.

The staff reviewed the results of these analyses and concludes that the bulk suppression pool temperature response remains acceptable after power rerate.

(2) Local Pool Temperature with SRV Discharge

The local pool temperature limit for SRV discharge is specified in NUREG-0783 (Reference 12), because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. The licensee indicated that since the Limerick Station Units 1 and 2 have quenchers, no evaluation of this limit is considered necessary. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limits on Local Suppression Pool Temperature for SRV Discharge with Quenchers." However, the local pool temperature has been evaluated at rerated power, and was found to be acceptable.

Based on the above, the staff concludes that the local pool temperature limit will remain acceptable after power rerate.

3.3.1.1.2 Containment Gas Temperature Response

The licensee indicated that the containment drywell design temperature of 340°F was determined based on a bounding analysis of the blowdown of steam to the drywell during a LOCA. The changes in the reactor vessel conditions with power rerate will increase the calculated long-term peak drywell gas temperature during a small break LOCA by a maximum of a few degrees but will not exceed the drywell design value of 340°F. For larger steam line breaks, the superheat temperature is nearly the same as for small breaks, but the duration of the high temperature condition is less for large break. Therefore, the drywell gas temperature response after power rerate will remain below the containment design temperature of 340°F.

The licensee indicated that the wetwell gas space peak temperature response is calculated assuming thermal equilibrium between the pool and wetwell gas space. The reanalysis has shown that the maximum bulk pool temperature can reach 205°F due to power rerate. Therefore, the maximum wetwell gas space temperature of 205°F due to power rerate will remain below the wetwell design temperature of 220°F.

Based on its review, the staff concludes that the containment drywell and wetwell gas temperature response will remain acceptable after power rerate.

3.3.1.1.3 Short Term Containment Pressure Response

The licensee indicated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double ended guillotine break of a recirculation suction line to demonstrate that power uprate operation will not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occur. These analyses were performed at 102-percent of 110-percent of the original rated power, using the GE M3CPT computer code. The reanalysis predicted a maximum containment pressure of 42.1 psig which remains below the containment design pressure of 55 psig.

Technical specifications definitions, limiting conditions for operation, surveillance requirements and bases relating to the current 44.0 psig value of P_a will not be revised as it remains higher than the maximum containment pressure of 42.1 psig calculated for the power rerate.

Based on its review, the staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after power rerate.

3.3.1.1.4 Steam Bypass Case

The licensee indicated that the steam bypass of the suppression pool due to a leakage between the drywell and the wetwell airspace during a LOCA event was analyzed to ensure that there is sufficient time for manual actuation of the containment spray to prevent the containment pressure from exceeding the design limit. The evaluation performed at rerated conditions shows that the operator will have over 2 hours and 30 minutes following the time drywell pressure reaches 30 psig, alerting the existence of significant steam leakage due to a small-line break in the drywell, for manual actuation of the containment sprays. The allowed time required for operator action is 30 minutes. The evaluation shows that the power rerate has negligible impact on the suppression pool steam bypass effects.

Based on the above, the staff concludes that the steam bypass response will remain acceptable after power rerate.

3.3.1.2 Containment Dynamic Loads

(1) LOCA Containment Dynamic Loads

Reference 3 specifies that the power uprate applicant must determine if the containment pressure, temperature, and vent flow conditions calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate and thus do not require further analysis.

The licensee stated that the containment response is negligibly affected by power uprate, the loads being bounded by the test conditions used to define the original loads. The short-term analyses demonstrated that the uprate would not significantly affect parameters important for LOCA containment dynamic loads (e.g., drywell and wetwell pressure, vent flow rate, and suppression pool temperature).

Based on its review of the licensee's information, the staff concludes that LOCA containment dynamic loads will remain acceptable after power uprate.

(2) SRV Containment Dynamic Loads

The licensee stated that SRV containment dynamic loads include discharge line loads, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration and suppression pool configuration. The SRV setpoint would be the only one of these affected by power uprate. Reference 3 states that if the SRV setpoints are increased, the power uprate applicant will attempt to show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

The licensee indicated that the SRV analytical limits for setpoints show a 3.5-percent increase in the analytical values of the SRV opening pressure with power rerate. The increase in SRV setpoint pressure will result in a corresponding 3.5-percent increase in flow rate and hydrodynamic loads. The increased SRV loads resulting from this increase in the SRV setpoint pressure and flow rate were compared with the original flow rate used to define the quencher hydrodynamic loads. The comparison shows there is sufficient conservatism in the original containment dynamic loads definition to accommodate the increase SRV loads. The results of the reanalysis indicate that the loads remain below their original design values. The staff finds the licensee's conclusions acceptable.

(3) Subcompartment Pressurization

The licensee stated that the design loads on the sacrificial shield wall due to a postulated pipe break in the annulus between this wall and the reactor vessel are acceptable for the higher reactor pressure at uprated conditions. The shield wall design remains adequate because the original analyzed loads were based on mass and energy releases which bounded the rerated conditions. It is also noted that the Reference 3 methodology does not require subcompartment reanalysis.

Based on the above, the staff concludes that the subcompartment pressurization effects will remain acceptable after power uprate.

3.3.1.3 Containment Isolation

Reference 3 methodology does not address a need for reanalysis of the isolation system. The isolation system is not affected by power uprate. The licensee evaluated the capability of the actuation devices to perform with the higher pressure and flow and determined them to be acceptable. The licensee stated that all motor-operated valves (MOVs) used as containment valves will comply to the licensee's commitments regarding Generic Letter 89-10 at uprated conditions. The staff agrees with the licensee that the operation of the plant at the uprated power level will not affect the containment isolation system.

3.3.1.4 Post-LOCA Combustible Gas Control

The licensee stated that the containment atmospheric control system and the hydrogen recombiner subsystem are provided to maintain the containment atmosphere as a non-combustible mixture after DBA LOCA. The combustibility of the post-LOCA containment atmosphere is controlled by the concentration of oxygen. A result of power rerate is that the production of oxygen by radiolysis after a LOCA will increase proportionally with the power level. The licensee stated that sufficient capacity exists in the combustible gas control system to accommodate the increased oxygen production. Also, recombiner operation is controlled procedurally based on gas concentration in the containment.

Based on its review, the staff concludes that the post-LOCA combustible gas control will remain acceptable at rerated power.

3.3.2 Emergency Core Cooling Systems

The following sections address the manner in which the functional capability of each ECCS will be affected by the power uprate and the increase in RPV dome pressure. Section 3.3.3 is an evaluation of ECCS performance.

Power rerate does not increase the calculated peak suppression pool temperature when compared to the current UFSAR analysis. An increase in calculated peak suppression pool temperature could decrease the NPSH available to the ECCS pumps. Assuming a LOCA occurs during operation at the rerated power, the calculated suppression pool temperature will remain below the value (212°F) used in the current NPSH analysis. Therefore, power rerate will not affect ECCS pump NPSH requirements.

3.3.2.1 High-Pressure Coolant Injection System

The HPCI system and hardware capabilities have been evaluated by the licensee for power rerate conditions, and HPCI was found to be consistent with the bases and conclusions of the generic evaluation.

In response to a staff question, the licensee stated that it modified the LGS HPCI system in accordance with GE SIL 480 (Reference 7 and Reference 10). The licensee modified the HPCI turbine assembly to improve the turbine startup response. Various licensees have installed this modification and found it reliable for their turbine assemblies. The modification limits the initial speed of the turbine on startup from high reactor pressures to reduce the probability of turbine overspeed trips and reduce cyclic pressure forces and loads on certain components, thus improving overall system reliability. Consequently, this modification results in the higher power rerate reactor steam pressure having an insignificant impact on the turbine startup transient response.

For the HPCI system, the increase in reactor operating pressure with power rerate was found to be less than the calculated existing system operating margin. The introduction of the HPCI flow split modification (core spray and feedwater systems) during the plant design phase resulted in a reduction in the system flow losses and in the required system injection pressure. No changes were made to the equipment specifications or their capabilities as the result of this system modification. Furthermore, a flow test of HPCI system injection capabilities will be included in the testing program. This flow test will involve HPCI returning flow to the condensate storage tank. This testing conforms with the Power Rerate Startup Testing Program recommended by General Electric.

Based on the staff's review of the licensee's information, the staff concludes the HPCI system is acceptable for operation at uprated power conditions.

3.3.2.2 Residual Heat Removal System (Low-Pressure Coolant Injection)

Section 3.3.3 addresses the adequacy of the LPCI mode of the RHR system to provide core cooling during a LOCA. The hardware capability of the equipment in the system is bounded by the generic evaluation (Reference 3).

3.3.2.3 Low Pressure Core Spray System

Section 3.3.3 addresses the adequacy of the low-pressure core spray (CS) system to provide core cooling during a LOCA. The hardware capability of the equipment in the CS system is bounded by the generic evaluation (Reference 3).

3.3.2.4 Automatic Depressurization System

The automatic depressurization system (ADS) uses safety/relief valves to reduce reactor pressure following a small break LOCA with high-pressure ECCS failure. This function allows LPCI and CS to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. ECCS design requires a minimum flow capacity for the SRVs, and that ADS initiates (after a time delay) on low water level plus high drywell pressure or low water level alone. ADS capacity at uprated power levels was evaluated by the licensee using the methodologies described in Section 3.3.3. The ability to give the required flow capacity and initiate ADS on appropriate signals is still achieved under operation at uprated conditions. Performance of the ECCS, including ADS, at uprated power levels is discussed in Section 3.3.3.

3.3.3 ECCS Performance Evaluation

The ECCS are designed to protect against a hypothetical LOCA caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

The SAFER/GESTR (S/G) LOCA analysis for LGS was performed by the licensee in accordance with NRC requirements and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal peak cladding temperature (PCT) and the Appendix K PCT as a function of break size. The LGS specific analysis was performed with conservatively high Peak Linear Heat Generation Rate (PLHGR) and conservatively low minimum critical power ratio (MCPR). In addition, many of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is below 1000 °F. The statistical Upper Bound PCT is below 1430 °F. The Licensing Basis PCT for Limerick 1/2 is 1625 °F, which is well below the PCT limit of 2200 °F.

An analysis for the maximum extended load line limit (MELLL) region was performed by the licensee. The higher rod line in the MELLL region permits reactor operation at rated power for core flows below rated (down to 75-percent core flow at current rated power). For low core flow operation, boiling transition at the limiting fuel node (the high power node) can occur sooner than observed at rated core flow conditions. This phenomenon is referred to as early boiling transition (EBT). If EBT occurs for the high power node as a result of the reduced initial core flow, the resultant PCT can exceed the rated core flow condition results. Low core flow effects were generically addressed, which was approved by the staff. The LOCA analysis for low core flow conditions was re-evaluated for Limerick 1/2 with SAFER/GESTR. In general, a LOCA analysis for operation at high power/low core flow requires performing LAMB/SCAT calculations for the DBA recirculation suction line break to determine if EBT occurs at the highest power node of the hot bundle prior to jet pump uncovery. Given the LAMB/SCAT results, a SAFER calculation is performed to evaluate PCT. LAMB and SCAT calculations were performed at 3694 MWt (Appendix K) and 75-percent core flow for BP/P8x8R fuel to determine if EBT of the high power node will occur. The BP/P8x8R fuel type was expected to result in EBT because of the high initial fuel stored energy. The results demonstrated that EBT does not occur at 75-percent initial core flow for the high power node, although the results are more severe than for the rated flow case. SAFER calculations were then performed at 75-percent initial core flow with BP/P8x8R and GE11 fuel. For GE11 fuel, EBT of the high power node was conservatively assumed to occur. The results of the 75-percent core flow calculations demonstrate that the increase in PCT would be less than 20°F when compared to the rated flow case with Appendix K assumptions. The results of this bounding evaluation show that the potential increase in PCT for a design basis LOCA at the MELLL condition (102-percent power/75-percent flow) is small relative to the PCT margin currently available with respect to the 2200°F criteria. As such, there is no required low flow MAPLHGR multiplier for ECCS considerations.

The ECCS performance for LGS under Single Loop Operation (SLO) was also evaluated by the licensee using S/G - LOCA calculations for the DBA. This analysis assumes that there is essentially no period of recirculation pump coastdown, and thus, dryout is assumed to occur simultaneously at all axial locations of the hot bundle less than one second after initiation of the event. The most bounding BP/P8x8R fuel type was analyzed. These assumptions are very conservative and provide bounding results for the DBA under SLO. In addition, the core power was assumed to be at 3694 MWt power level. SLO will affect the DBA results more than the smaller breaks, since with breaks smaller than the DBA there is a longer period of nucleate and/or film boiling prior to fuel uncovery to remove the fuel stored energy. With a MAPLHGR multiplier of 0.90, the SLO DBA Appendix K PCT is 1587°F for BP/P8x8R fuel, which is lower than the two-loop DBA Appendix K PCT result. Since maximum core power during SLO will typically be $\leq$ 80-percent and core flow will be $\leq$ 60-percent, the 0.90 MAPLHGR multiplier is very conservative. Therefore, the actual PCT for SLO will always be lower than that for two-loop operation.

Therefore, Limerick Units 1 and 2 meet the NRC SAFER/GESTR-LOCA licensing analysis requirements.

3.3.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to achieve and maintain a slightly negative pressure (-0.25 inch water gauge with respect to the outside atmosphere) in the secondary containment (SC) within a prescribed time following a LOCA to prevent unfiltered release of radioactive material from the SC to the environment.

The licensee stated that, as a result of the plant operations at the proposed uprated power level, heat loads from piping in the reactor building will increase slightly. This increase in piping heat loads, in turn, will cause a slight increase in the pressure drawdown time (by approximately 5 seconds) in order for the SGTS to achieve the above cited negative pressure in the SC. However, the current radiological release is conservatively determined based on an SC pressure drawdown time of 5 minutes and 10 seconds which is well above the calculated drawdown time of 2 minutes and 15 seconds. Therefore, an increase of 5 seconds in the SC pressure drawdown will have no impact on the radiological release analysis. The licensee also stated that the total post-LOCA iodine loading on the filters will increase slightly, but it will remain well below the original design capacity of the filters.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the ability of the SGTS to meet its design objectives.

3.3.5 Other ESF Systems

3.3.5.1 Emergency Cooling Water Systems

Safety-related and nonsafety-related water systems are addressed in Section 3.5.2.

3.3.5.2 Emergency Core Cooling Auxiliary Systems

Power dependent heating, ventilation and air conditioning (HVAC) systems and other auxiliary systems are addressed in Section 3.5.

3.3.5.3 Main Control Room Atmosphere Control System

The control room atmosphere control system (CRACS) containing an emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room following a LOCA. The licensee stated that since plant operation at the proposed uprated power level does not change the design and operational aspects of the control room emergency filtration system, there will not be an increase in unfiltered inleakage of contaminated outside air into the control room following a LOCA.

The staff recognizes that following a LOCA, iodine loading in the filters will increase marginally due to plant operations at the proposed uprated power level, however, the staff agrees with the licensee that it will remain well below the original design capacity of the filters.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have little or no effect on the CRACS, and is therefore, acceptable.

3.4 Instrumentation and Control

Many of the TS changes proposed in the licensee's application (Reference 1) involve changes to the Reactor Protection System trip and interlock setpoints. These changes are intended to maintain the same margin between the new operating conditions and the new trip points as existed before the proposed power uprate.

The conservative design calculations for the initial licensing of LGS resulted in setpoints which provided excess reactor coolant flow capacity and corresponding margins in the power conversion system. For LGS, these margins (e.g. 5-percent rated steam flow) result in the capability to increase the core operating power level by approximately 5 percent. This safety evaluation is limited to setpoint changes for the identified instrumentation and is predicated on the assumption that the analytical limits used by the licensee are based on application of approved design codes.

The licensee proposed the following setpoint changes:

1. APRM Flow Biased Neutron Flux - Upscale
   a. During two recirculation loop operations.
      Change trip from 0.66W + 66% to 0.66W + 62%.
      Change Allowable Value from 0.66W + 68% to 0.66W + 64%.

   b. During single recirculation loop operation.
      Change trip from 0.66W + 61% to 0.66W + 57%.
      Change Allowable Value from 0.66W + 63% to 0.66W + 59%.

2. Reactor Vessel Steam Dome Pressure High
   Change trip from 1037 psig to 1096 psig.
   Change Allowable Value from 1057 psig to 1103 psig.

3. Main Steam High Flow
   Change trip from 108.7 psid to 122.1 psid.
   Change Allowable Value from 111.7 psid to 123.0 psid.

4. APRM Rod Block - Flow Biased Neutron Flux Upscale
   (a) During two recirculation loop operations.
       Change trip from 0.66W + 59% to 0.66W + 55.0%.
       Change Allowable Value from 0.66W + 63% to 0.66W + 59%.

   (b) During single recirculation loop operation.
       Change trip from 0.66W + 54% to 0.66W + 50%.
       Change Allowable Value from 0.66W + 58% to 0.66W + 54%.

5. Turbine Stop Valve and Turbine Control Valve Fast Closure Scram Bypass
   The turbine first stage pressure setpoint was changed to reflect the expected pressure at the new 30% power point.

6. Reactor Water Cleanup System Area Temperature
   Change isolation setpoint from 135°F for pump room to 142°F and 122°F for heat exchanger room to 132°F.
   Change Allowable Value from 145°F for pump room to 147°F and from 130°F for heat exchanger room to 137°F.

7. High Pressure Coolant Injection Steamline ΔP - High
   Change isolation setpoint from 343" H_2O to 974" H_2O.
   Change Allowable Value from 358" H_2O to 984" H_2O.

8. Reactor Core Isolation Cooling Steamline - ΔP - High
   Change isolation setpoint from 213" H_2O to 373" H_2O.
   Change Allowable Value from 223" H_2O to 381" H_2O.

9. ATWS Recirculation Pump Trip Reactor Vessel Pressure - High
   Change trip setpoint from 1093 psig to 1149 psig.
   Change Allowable Value from 1108 psig to 1156 psig.

The licensee's application (Reference 1) did not describe the methodology used for instrument setpoint calculations. Therefore, in a letter of June 4, 1994, (Reference 13), the staff requested additional information regarding instrument setpoint methodology. The licensee, in a letter of July 5, 1994, (Reference 14) confirmed that GE Licensing Topical Report NEDC-31336P (Reference 15) was used for instrument setpoint calculations except for turbine valves and pressure regulator setpoints. The staff previously reviewed this Topical Report and accepted it with minor exceptions. The staff is reviewing the exceptions and will resolve them generically. They do not affect the staff's evaluation of the proposed LGS changes.

For the pressure regulator, the setpoint is controlled manually by the operator to maintain turbine inlet pressure within the required operating range. This is consistent with the current licensing basis for this system.

The proposed setpoint changes are intended to maintain the existing margins between operating conditions and the reactor trip setpoints. Thus, margins to the new safety limits will remain the same as the current margins. However, the staff was concerned that the trip setpoint and allowable values for some instruments were too close and requested that the licensee send the calculations for the reactor vessel high pressure scram and main steamline high flow for staff review. On July 27, 1994, the staff reviewed the calculations and determined that they were done in accordance with Topical Report NEDC-31336. During a conference call on July 27, 1994, the licensee stated that its procedure required the instrument technician to set the instrument setpoint at a number which will increase the band between the setpoint and allowable value. These new setpoints also do not significantly increase the likelihood of a false trip nor failure to trip upon demand. Therefore, the existing licensing basis is not affected.

The staff concludes that the licensee's instrument setpoint methodology and the resulting setpoint changes incorporated in the TSs for power uprate are consistent with the LGS licensing basis and are, therefore, acceptable.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling

The licensee stated that spent fuel pool heat loads and radiological consequences were evaluated for plant operations at the uprated power level. The results of the evaluation indicate that the original analyses associated with decay heat rate, time-to-boil, evaporation from boiling and the associated consequences, are still valid due to conservatism used in the original analyses.

In a letter of January 14, 1994, (Reference 16) the licensee proposed TS changes to support implementation of a modification to install new high density spent fuel storage racks in each of the spent fuel pools at LGS. In a letter of November 29, 1994 (Reference 17), the staff issued license amendments 82 and 43 for licenses NPF-39 and NPF-85, respectively, revising the LGS TS to permit installation of new high density spent fuel storage racks in each of the spent fuel pools at LGS. The staff reviewed the decay heat removal capability of the spent fuel pool cooling systems within the context of the existing licensing basis and concluded that it was adequate to accommodate the additional decay heat loads resulting from the new high density spent fuel storage in the pools.

Based on its review of licensee's rationale and the review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the spent fuel pool cooling system at LGS and is therefore, acceptable.

An issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992 (Reference 18). The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the licensee under separate cover.

3.5.2.1 Water Systems

The licensee evaluated the effect of power uprate on the various plant water systems including the safety-related and nonsafety-related service water systems, closed loop cooling water system, circulating water system, and the plant ultimate heat sink. The licensee's evaluation considered increased heat loads, temperatures, pressures, and flow rates.

3.5.2.1.1 Safety Related Loads

The safety-related heat loads are rejected to one of the two safety-related service water systems. These systems include the emergency service water system (ESWS) and the RHR service water system (RHRSWS). All heat removed from these systems is rejected to the ultimate heat sink (UHS). The staff's evaluation of the effects of uprated power level operation on each of these systems is provided below.

The ESWS is designed to provide cooling water to emergency core cooling system components and other essential equipment during a loss of off-site power event and/or a LOCA. The licensee, having performed evaluations, stated that heat loads for this system are only slightly impacted by plant operations at the proposed uprated power level. The ESWS return water temperature is expected to increase by less than 0.5 ^{0}F. Additionally, the existing design heat loads for this system are higher than the anticipated equipment heat loads resulting from plant operations at the proposed uprated power level.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the ESWS.

The RHRSWS provides safety-related cooling water to the RHR system under normal or post-accident conditions. The system pumps water from the ultimate heat sink (spray pond) through the RHR heat exchangers and returns it to the pond via a spray network. Heat loads on the RHRSW system will increase proportionally to the increase in reactor operating power level. The licensee, having performed evaluations, stated that the existing design heat loads for this system are higher than the anticipated equipment heat loads resulting from the proposed uprated power operations.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the RHRSWS.

3.5.2.1.2 Nonsafety-Related Loads

The effects of the power uprate on nonsafety-related loads is mainly felt in the increase in heat losses needed to be rejected from the main generator via the stator water coolers, hydrogen coolers, and exciter coolers, as well as increased bus cooler heat loads. Additional small increases in heat loads are felt in the closed cooling water systems and other auxiliary heat loads.

The service water system (SWS) is designed to continuously supply cooling water to various non-safety related components and heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation, and has no safety-related function. The licensee, having performed evaluations, stated that the SWS as designed will supply sufficient water to remove the additional heat loads resulting from plant operations at the proposed uprated power level.

Since plant operations at the proposed uprated power level do not change the design aspects and operations of the SWS, and the SWS does not perform any safety-related function, the staff has not reviewed the impact of the proposed uprated power level operations on the SWS design and performance.

3.5.2.2 Main Condenser/Circulating Water/Normal Heat Sink

The main condenser and circulating water system are designed to condense steam in the condenser and reject heat to the circulating water system. This maintains an adequately low condenser pressure required for efficient turbine performance.

The licensee stated in its power uprate submittal that the performance of the main condenser was evaluated for power uprate based on a design over a range of circulating water inlet temperatures. The licensee stated that its review confirmed that the condenser and circulating water system are adequate for uprated conditions.

Since the main condenser and circulating water system do not perform any safety function, the staff has not reviewed the effect of the uprated power level operation on the designs and performances of these systems.

3.5.2.3 Reactor Enclosure Cooling Water System

The reactor enclosure cooling water system (RECWS) is designed to remove heat from various auxiliary plant equipment in the reactor and radwaste buildings during normal and loss-of-offsite power (LOOP) conditions, and has no safety-related function. The licensee performed evaluations and stated that the increase in heat loads to this system due to uprated power operations has an insignificant impact on the RECWS.

Since plant operations at the proposed uprated power level do not change the design aspects and operations of the RECWS, and the RECWS does not perform any safety-related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the RECWS design and performance.

3.5.2.4 Turbine Enclosure Cooling Water System

The turbine enclosure cooling water system (TECWS) is designed to remove heat from the miscellaneous turbine plant equipment during normal plant and loss-of-offsite power operation, and has no safety-related function. The licensee, having performed evaluations, stated that the increase in heat loads from the equipment due to the proposed uprated power operations is insignificant and that the TECWS design cooling capacity will not be exceeded.

Since the proposed uprated power operation does not change the design aspects and operations of the TECWS, and the TECWS does not perform any safety-related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the TECWS design and performance.

3.5.2.5 Drywell Chilled Water System

The drywell chilled water system (DCWS) supplies chilled water to various reactor building and drywell HVAC and equipment loads during normal plant operation, and does not perform any safety-related function. The licensee, having performed evaluations, stated that due to the slightly additional heat loads, the temperature of the drywell chilled water returning to the chiller is expected to increase by less then 0.5 ^{0}F. This temperature increase is insignificant to the DCWS as designed.

Since the proposed uprated power level operations do not change the design aspects and operations of the DCWS, and the DCWS does not perform any safety-related function, the staff has not reviewed the impact of the proposed uprated power level operations on the DCWS design and performance.

3.5.2.6 Control Structure Chilled Water System

The control structure chilled water system (CSCWS) is designed to provide chilled water to the safety-related unit coolers including the control room, auxiliary equipment room, emergency switchgear and battery rooms, and SGTS rooms. The licensee stated that the SGTS access areas unit cooler heat loads will increase slightly because of the higher equipment and ductwork temperatures. The remaining heat loads are not power dependent and will not be impacted by plant operations at the proposed uprated power level. Thus, the control structure chilled water return temperature is expected to increase by less than 0.5 ^{0}F which is insignificant to the CSCWS as designed.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on CSCWS.

3.5.2.7 Ultimate Heat Sink

The UHS is designed to remove heat from the cooling water for the ESWS and the RHRSWS. As discussed in the above section 3.5.2.1.1, the anticipated ESWS and RHRSWS equipment heat loads resulting from plant operations at the proposed uprated power level are less than the existing design heat loads for these systems. Thus, the UHS heat removal capability will not be affected by plant operations at the proposed uprated power level.

Based on the above discussion, the staff concludes that the UHS design is acceptable for plant operations at the proposed uprated power level and no modification to the UHS system is required.

3.5.3 Standby Liquid Control System

The ability of the standby liquid control systems (SLCS) to achieve and maintain safe shutdown is not directly affected by core thermal power; rather, it is a function of amount of excess reactivity present in the core; and as such, is dependent upon fuel-loading techniques and uranium enrichment. The SLCS is designed to inject at a maximum pressure equal to that of the lowest safety/relief valve setpoint. The SLCS pumps are positive displacement pumps, and the small (approximately 40 psig) increase in the lowest safety/relief valve setting as a result of uprate will not impair the performance of the pumps. The staff concludes that the ability of the SLCS system to inject to the reactor will not be impaired by uprate.

However, in the future, the licensee may wish to increase fuel enrichments in order to meet fuel energy requirements for longer fuel cycles. The increased excess reactivity associated with this increase in fuel enrichment will affect the reactivity requirements of the SLCS. The SLCS requirements for future operating cycles will be evaluated by the licensee on a cycle-specific basis.

3.5.4 Power Dependent Heating, Ventilation and Air-Conditioning

The licensee stated that the HVAC systems affected by power uprate include the turbine enclosure, reactor enclosure, steam tunnel and drywell HVAC systems.

The increase in the heat loads on the HVAC system stem from increases in area temperatures resulting from the increase in steam cycle process temperatures which rise from the power uprate. The licensee stated in its power uprate submittal that all steam cycle process temperatures including main steam, feedwater, condensate, extraction steam, and heater drains experience less than an 8 °F increase, while the majority of the cooling water systems experience a maximum temperature increase of approximately 2 °F.

Area temperatures that result from the increase in process temperatures are not expected to exceed a rise of more than 3 °F, with the exception of the regenerative heat exchanger area which will experience an increase in area temperature of approximately 4 °F.

The licensee stated that the area design temperatures for all plant operating modes envelop the temperatures resulting from the anticipated increase in heat loads due to the plant operations at the proposed uprated power level. Thus, the existing design of the HVAC systems for the above cited areas is acceptable for plant operations at the uprated power level. The licensee stated in its submittal that area heat gains due to increase in electrical loads are negligible.

The staff agrees with the licensee that these operational increases are minor and that the designs of the HVAC systems are acceptable for operation at the uprated power level.

3.5.5 Fire Protection

In its power uprate submittal, the licensee stated that operation of the plant at the uprated power level does not affect the fire suppression or detection systems and would cause no changes to the physical plant configuration or combustible load. The staff recognizes that operation at an uprated power level requires a small increase in the reactor vessel pressure during full power operation, which would increase the heat load in the HPCI, RCIC, RHR, and core spray pump rooms during a postulated fire event. The licensee analyzed the temperature response for these rooms, as revised for the power uprate, and found that the required equipment would be operational for the event. The staff agrees that the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change and are acceptable for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected.

The staff agrees that the power upgrade will not affect the fire suppression and detection systems and their associated components.

3.5.6 Power Conversion Systems

The steam and power conversion systems and associated components (e.g. the turbine/generator, condenser vacuum pump and steam jet air ejectors, turbine steam bypass valves, feedwater and condensate systems, etc.) were originally designed to use 105 percent of the rated power available from the nuclear steam supply system. The licensee did evaluations and stated that the existing systems and equipment are acceptable for plant operations at the proposed uprated power level.

The objective of the pressure control system is to give a fast and stable response to pressure and steam flow disturbances to ensure that the reactor pressure is controlled within its allowed high and low limits. In order to ensure that the system objective is met, adequate turbine control valve range must be available at uprated conditions. The licensee stated that this system will have sufficient control pressure range during system disturbances with power uprate.

Based on its review of the licensee's information, the staff agrees that the power conversion systems are acceptable for operation at the uprated power level.

3.6 Radwaste Systems and Radiation Sources

The licensee evaluated the proposed power increase to show that the applicable regulatory acceptance criteria continue to be satisfied. The licensee considered the effect of the higher power level on source terms, onsite and offsite doses, and control room habitability during both normal operation and accident conditions.

3.6.1 Liquid Waste Management

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge. The single largest source of liquid waste is from the backwash of the condensate demineralizers. Operation at uprated power levels results in an increased flow rate through the condensate demineralizers and deepbeds. The rate of loading on the demineralizers will increase resulting in the average time between backwash precoats decreasing slightly. This reduction does not affect plant safety. Similarly, the RWCU filter/demineralizer will require slightly more frequent backwashes due to slightly higher levels of activation and fission products. The activated corrosion products in liquid wastes are expected to increase proportionally to the square of the power increase. The licensee concludes that the requirements of 10 CFR Part 20 and Appendix I to 10 CFR Part 50 will be met. Therefore, power uprate does not have an adverse effect on the processing of liquid waste.

Based on its review of available plant data and experience with previous power uprates, the staff agrees with the licensee's conclusion that the operation at uprated power levels will have no significant adverse effect on liquid effluents and is therefore, acceptable.

3.6.2 Gaseous Waste Management

The gaseous waste management systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system, SGTS, and various building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans are used to control airborne radioactive gases. The systems are designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

In its power uprate submittal, the licensee stated that the greatest contributors of radioactive gases are the noncondensible radioactive gases from the main condenser, which contain activation gases (principally N-16, O-19, and N-13) and radioactive noble gas parents. The steam jet air ejectors continually remove these noncondensible radioactive gases as well as nonradioactive air that leaks into the condenser. The steam jet air ejectors discharge these gases into the offgas system. The flow of these gases into the offgas system is included with the flow of H_2 and O_2 from the recombiners, which will increase linearly with core power. The licensee stated that the operational increases in gases are not significant when compared to the current total system flow. The power increase will not increase pressure losses, hold up times, heat of combustion, and peak pressures caused by H_2-O_2 gas detonation, and therefore, will not affect the offgas system design.

The power increase will not increase the contribution of gases from the building ventilation systems to the gaseous waste management system for the following reasons:

a. The amount of fission products released into the reactor coolant depends on the number and nature of the fuel rod defects and not on reactor power, and

b. The concentration of coolant activation products will not change since the linear increase in the production of these products will be offset by the linear increase in steaming rate.

On the basis of its review of available plant data and previous experience with other power uprates, the staff agrees with the licensee that there will not be a significant adverse effect on airborne effluents as a result of the power rerate.

3.6.3 Radiation Sources in the Core and Coolant

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The reactor coolant contains corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Fission products in the reactor coolant are separated into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover from the reactor water. The licensee has evaluated the effects of the power uprate on coolant activation products, activated corrosion products, and fission products which show that they are expected to be approximately equal to current measured data which is within the design basis of the plant.

On the basis of its review of the licensee's submittal, the staff concludes that the radiation sources in the core or reactor coolant will continue to meet its design-basis and performance requirements at uprated power conditions.

3.6.4 Radiation Levels

The licensee evaluated the effects of power uprate on radiation levels in the LGS facility during normal and abnormal operation as well as from postulated accident conditions. The licensee has concluded that radiation levels from both normal and accident conditions may increase slightly. However, any such increases would be slight and would be bounded by conservatism in the original design and analysis. Individual worker exposures will be maintained within acceptable limits by the existing ALARA program, which controls access to radiation areas. Procedural controls compensate for slightly increased radiation levels. The staff finds the licensee's conclusions acceptable.

The off-site doses associated with normal operation are not significantly affected by operation at the uprated power level, and are expected to remain below the limits of 10 CFR Part 20 and Appendix I of 10 CFR Part 50.

The main control room (MCR) habitability was evaluated. Post-accident MCR and technical support center (TSC) doses were confirmed by the licensee to be within the limits of General Design Criterion (GDC) 19 of 10 CFR Part 50, Appendix A.

On the basis of its review, the staff agrees with the licensee that no significant adverse effect on radiation levels will result on-site or off-site from the planned power uprate.

3.7 Reactor Safety Performance Evaluations

The staff reviewed information requested in Regulatory Guide 1.70, Chapter 15, for power uprate.

3.7.1 Reactor Transients

The UFSAR evaluates the effects of a wide range of potential plant transients. Disturbances of the plant caused by a malfunction or a single failure of equipment or the operator are investigated according to the type of initiating event (Regulatory Guide 1.70, Chapter 15). The generic guidelines for BWR power uprate (rerate) identifies the limiting event(s) to be considered in each category of events. The generic guideline also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied.

The following sections address each event and provide a summary of the resulting transient safety evaluations. The results given here are for a representative core (based on LGS Unit 1 Cycle 5), and show the overall capability of the design to meet all transient safety criteria for rerated operation.Table E-1 of Reference 3 provides the specific events to be analyzed for power rerate, the power level to be assumed, and the computer models to be used. The power rerate analysis used the GEMINI transient analysis methods listed there.

The reactor operating conditions that apply most directly to the transient analysis are summarized in Table 9-1 of Reference 2. They are compared to the conditions used for the UFSAR and the most recent reload fuel cycle (Unit 1 Cycle 5) analyses. The Cycle 5 core was used as the representative fuel cycle for power rerate. Most of the transient events are analyzed at the full rerated power and maximum allowed core flow operating point on the power/flow map. Direct or statistical allowance for 2-percent power uncertainty is included in the analysis. The Safety Limit MCPR (SLMCPR) was used to calculate the MCPR Operating Limits provided for the analyzed events. For all pertinent events, no SRV is assumed to be out-of-service. The overpressure protection analysis is based on 3 SRVs out-of-service. The effect of power rerate on the SLMCPR is generically evaluated in Reference 5.

The limiting events for each limiting transient category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The results from these analyses developed the new licensing basis for transient analyses at rerated power. No changes to the basic characteristics of any of the limiting events are caused by power rerate. The limiting events which establish the largest delta CPR and the MCPR operating limits are Turbine/Generator Trips and Feedwater Controller Failure.

The consequences of a Loss of Feedwater Flow (LOFW) event are discussed in Section 3.1 of Reference 5. During an LOFW event and assuming an additional single failure (loss of RCIC or HPCI), reactor water level is automatically maintained above the top of the active fuel (TAF) by the RCIC (or the HPCI) system without any operator action required. If both of these high pressure systems fail, ADS will automatically initiate on low water level and the low pressure ECCS will automatically maintain water without any operator action required. Because of the extra decay heat from power rerate, slightly more time will be required for the automatic systems to restore water level.

Operator action is only needed for long-term plant shutdown once water level is restored (control water level, reduce pressure and initiation of RHR shutdown cooling). These sequences of events do not require any new operator actions or shorter operator response time. Therefore, the operator actions for an LOFW transient do not significantly change for power rerate.

3.7.2 Design Basis Accident

The increase in LOCA radiological consequences due to power rerate was analyzed. The resultant offsite doses were found to be within guidelines of 10 CFR Part 100. The events evaluated for rerate were the LOCA, the main steam line break accident (MSLBA), the fuel handling accident (FHA) and the control rod drop accident (CRDA). The whole body and thyroid dose were calculated for the exclusion area boundary (EAB), low population zone (LPZ) and, the control room. The plant-specific result for power rerate remain wellbelow established regulatory limits. The doses resulting from the accidents analyzed are compared with the applicable dose limits in the following tables.

TABLE 1: LOCA Radiological Consequences

LOCATION	UFSAR DOSE (rem) @ 3458 MWt	DOSE (rem) * @ 3527 MWt	LIMIT
Exclusion Area:			
Whole Body Dose	0.67	0.68	25
Thyroid Dose	0.15	0.15	300
Low Population Zone:			
Whole Body Dose	1.7	1.7	25
Thyroid Dose	0.04	0.04	300
Main Control Room:			
Whole Body Dose	4.6	4.7	5
Thyroid Dose	14.0	14.3	30
Beta	7.6	7.8	30

* Represents 102% of uprated power limit. The 2% conservatism allows for possible instrument error.

TABLE 2: FHA Radiological Consequences

Exclusion Area:			
Whole Body Dose	0.7	0.7	6
Thyroid Dose	0.95	0.98	75
Low Population Zone:			
Whole Body Dose	0.099	0.102	6
Thyroid Dose	0.13	0.135	75

TABLE 3: CRDA Radiological Consequences

Exclusion Area:			
Whole Body Dose	0.04	0.042	6
Thyroid Dose	0.32	0.3	75
Low Population Zone:			
Whole Body	0.014	0.0148	6
Thyroid Dose	0.62	0.63	75

Based on the above information, the analyzed consequences of postulated accidents remain within staff acceptance criteria and are therefore acceptable.

Based on the staff's review of the major assumptions and methodology used in the licensee's reconstituted dose calculations and a review of the staff's original safety evaluation (Reference 19), the staff finds that the off-site radiological consequences and control room operator doses for operation at a rerated power level of 3458 MWt still remain below 10 CFR Part 100 dose reference values and GDC 19 dose limit and therefore, are acceptable.

3.7.3 Anticipated Transients Without Scram (ATWS)

General Electric has performed generic bounding ATWS analyses. The LGS parameter changes for power uprate are within the generic criteria except that the ATWS high pressure setpoint increase is 40 psi rather 20 psi in order to maintain the same relationship between the ATWS high pressure setpoint and the SRV opening setpoints. The previous analysis indicates that this difference would have a minor effect on the analysis results. The only significant change is a slightly higher (on the order of 10 psi) peak vessel pressure.

For additional assurance, LGS specific ATWS analysis for a 5-percent power rerate was performed by the licensee. The events analyzed were:

1. MSIV Closure
2. Pressure Regulator Failure - Open

3. Loss of Feedwater
4. Inadvertent Opening of Relief Valve

The LGS specific analysis performed by the licensee also concluded that the ATWS acceptance criteria for fuel, RPV and containment integrity will be met for a 5-percent power rerate. The staff finds this acceptable.

3.7.4 Station Blackout

The licensee stated in its power uprate submittal that operating the plant at the uprated power level would slightly affect its response and coping capabilities for a station blackout (SBO) because the operating temperature of the reactor coolant system, the decay heat, and the main steam safety relief-valve setpoints would all increase. The licensee analyzed the impact of these increases on the condensate water requirement and the temperature heat-up in the areas which contain equipment necessary to mitigate the SBO event and concluded that no changes to the required coping period or to the systems and equipment used to respond to an SBO event are required. The licensee also determined that emergency diesel generator and Class 1E battery capacities following a loss of power were sufficient to maintain safe shutdown for plant operations at the proposed uprated power level.

Based on its review, the staff finds that the effect on the plant's coping capabilities for an SBO event of plant operation at the proposed uprated power level will be insignificant and that no changes are needed to the required coping time and to systems and equipment used to respond to an SBO event.

3.8 Additional Aspects of Power Uprate

3.8.1 High Energy Line Breaks

To operate the plant at an uprated level, the licensee will need to slightly increase the RPV dome operating pressure to supply more steam to the turbine. The slight increase in the operating pressure and temperature resulting from the plant operations at the proposed uprated power level will cause a small increase in the mass and energy release rates following a high energy line break (HELB) outside the primary containment. This results in a small increase in the subcompartment pressure and temperature profiles. The licensee performed evaluations for the HELB in the piping systems (main steam, feedwater, high pressure coolant injection, reactor core isolation cooling, reactor water cleanup) and concluded that the existing HELB temperature and pressure analyses envelop those resulting from the proposed uprated power operation and that there is no change in postulated break locations due to plant operations at the proposed uprated power level.

The licensee evaluated the existing pipe whip restraints, jet impingement shields, and their supporting structures for the effects of pipe whip and jet impingement from the postulated HELBs and found that in most cases these structures are acceptable for the safe shutdown conditions at uprated power. For those cases where unacceptable analytical results were originally encountered, the licensee performed reanalysis and concluded that the existing structures and structural components are adequate to sustain loads resulting from a HELB at uprated power conditions.

Based on its review of the licensee's information, the staff agrees that the licensee's analysis for high-energy line breaks indicates an acceptably small increase in the compartment temperature and pressure, and that structural restraints used to limit the effects of pipe whip and jet impingement are acceptable for the uprated conditions.

3.8.2 Equipment Qualifications

The licensee evaluated safety-related electrical equipment to assure qualification for the normal and accident conditions expected in the areas where the equipment is located.

The licensee evaluated the effects of the uprated power conditions on equipment qualification and determined that the dynamic loads used in equipment design are bounding for the power uprate. The staff agrees with the licensee's assessment that the power uprate conditions will not adversely affect the safety-related mechanical and electrical equipment for the following reasons:

1. The uprate will not change the seismic loads.

2. Jet impingement will increase only 4 percent and will become negligible when combined with the governing seismic loads.

3. The original LOCA dynamic loads and SRV discharge hydrodynamic loads will be bounding for the power uprate conditions.

4. The uprated conditions will not result in new pipe break locations.

The licensee reevaluated equipment qualification and determined that some equipment located inside and outside the containment would be affected by higher accident temperature and radiation levels resulting from plant operations at the proposed uprated power level. The licensee committed to resolve the qualification of this equipment by refining radiation calculations for the specific location, by reducing qualified life, by replacing specific equipment or by using available EQ test data prior to power uprate implementation (Reference 2).

Based on its review, the staff finds the licensee's commitment and approach to resolve the qualification of safety-related electrical equipment for plant operations at the proposed uprated power level acceptable.

3.8.3 Startup Testing

The licensee committed to a startup testing program as described in Reference 3. The startup test program includes system testing of such process control systems as the feedwater flow and main steam pressure control systems. The licensee will collect steady-state operational data during various portions of the power ascension to the higher licensed power level so that predicted equipment performance characteristics can be verified. The licensee will do the startup testing program in accordance with its procedures. In Reference 10, the licensee committed to include acceptance testing of RCIC and HPCI in the startup test program. The staff finds the licensee's approach in conformance with the test guidelines of Reference 3 and, therefore, acceptable.

3.9 Evaluation of Effect on Responses to Generic Communications

In Reference 5, GE submitted an assessment of the effect of power uprate on licensee responses to generic NRC and industry communications. GE reviewed both NRC and industry communications to determine whether parameter changes associated with power uprate could affect previously made licensee commitments or earlier responses. A large number of documents were reviewed (more than 3000 items); GE noted that only a small number of these would be affected by power uprate. The list of affected topics was then divided into those that could be bounded generically by GE, and those that would require plant-specific reevaluation. The NRC staff audited the GE assessment in December 1991 and approved the assessment in Reference 20. In addition to assessing those items requiring a plant-specific reevaluation, the licensee is also reviewing the potential effects of uprate on internal commitments. The licensee committed to resolve any changes to commitments before beginning uprated operations. The staff may audit these activities after plant startup following the implementation of power uprate modifications. The staff finds this approach acceptable.

4 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

5 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an Environmental Assessment and Finding of No Significant Impact has been prepared and published in the <u>Federal Register</u> on February 13, 1995, (60 FR 8255). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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