Docket File

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

October 18, 1994

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

SUBJECT: REVISED MAXIMUM AUTHORIZED THERMAL POWER LIMIT, PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2 (TAC NO. M86826)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 198 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. This amendment consists of changes to the Facility Operating License and Technical Specifications in response to your application dated June 23, 1993, as supplemented by letters dated April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28, (two letters), September 16, September 30, and October 14, 1994.

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 amendment is effective as of its date of issuance. You are requested to inform the staff when you have implemented the provisions of the amendment. In your application, you proposed that the power uprate amendment apply to both Unit 2 and Unit 3, however, you noted that power uprate would not be implemented on Unit 3 until the fall of 1995. In order to preclude confusion between the effective date for the Unit 3 power uprate amendment and any subsequent amendment requests that might affect the same TS pages, the staff will issue the power uprate amendment for Unit 3 just prior to refueling outage 3R010.

CP-1

DF01

G. Hunger, Jr.

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

Sincerely,

/S/

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-277

Enclosures:
1. Amendment No. 198 to DPR-44
2. Safety Evaluation

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NAME	LCunningham	MZobler	JStolz	SVarga	AThadani	RZimmerman RPZ	FMiraglia	WRussell	
DATE	09/27/94	10/13/94	10/14/94	10/14/94	10/15/94	10/17/94	10/18/94	10/18/94	

#6 10/17

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

Sincerely,

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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WASHINGTON, D.C. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. DPR-44

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated June 23, 1993, as supplemented by letters dated April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28, (two letters) September 16, September 30, and October 14, 1994, 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 or 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. DPR-44 paragraph 2.C.(1) is hereby amended to read as follows:

   (1) Maximum Power Level

   PECo is authorized to operate the Peach Bottom Atomic Power Station, Unit 2 at steady state reactor core power levels not to exceed 3458 megawatts thermal.

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. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 198 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of its date of issuance and is to be implemented prior to startup in Cycle 11 currently scheduled for October 28, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Page 4 of License*
2. Changes to the Technical
   Specifications

Date of Issuance: October 18, 1994

*Page 4 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 198

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Facility Operating License (FOL), the Appendix A Technical Specifications, and the Appendix B Environmental Technical Specifications, with the enclosed pages. The revised areas are indicated by marginal lines.

	Remove	Insert
FOL	4	4
Appendix A	2	2
	6	6
	9	9
	11	11
	16	16
	17	17
	18	18
	24	24
	29	29
	30	30
	37	37
	39	39
	40	40
	49	49
	50	50
	73	73
	74	74
	117	117
	129	129
	130	130
	137	137
	140a	140a
	140c	140c
	157	157
	164d	164d
	189	189
	193	193
	195	195
Appendix B	2	2
	5	5

and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

PECO is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not to exceed 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 198 are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

(3) The licensees may perform modifications to the Low Pressure Coolant Injection System as described in the licensees' application for license amendment dated July 9, 1975. The licensees shall not operate the facility prior to receipt of the Commission's authorization.

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Peach Bottom Atomic Power Station, Units 2 and 3, Physical Security Plan," with revisions submitted through December 16, 1987; "Peach Bottom Atomic Power Station, Units 2 and 3 Plant Security Personnel Training and Qualification Plan," with revisions submitted through July 9, 1986; and "Peach Bottom Atomic Power Station, Units 2 and 3 Safeguards Contingency Plan," with revisions submitted through March 10, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

1.0 DEFINITIONS (Cont'd)

Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Fraction of Limiting Power Density (FLPD) - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

Gaseous Radwaste Treatment System - Any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

High (power) Trip Set Point (HPTS) - The high power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable above 85% reactor thermal power.

Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212 F.

Hot Standby Condition - Hot Standby Condition means operation with coolant temperature greater than 212 F, system pressure less than 1085 psig, and the mode switch in the Startup/Hot Standby position. The main steam isolation valves may be opened to provide steam to the reactor feed pumps.

Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

1.0 DEFINITIONS (Cont'd)

Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

Purge - Purging - Purge or Purging is 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 Power - Rated power refers to operation at a reactor power of 3458 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

Refuel Mode - With the mode switch in the refuel position, the reactor is shutdown and interlocks are established so that only one control rod may be withdrawn.

*Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling

* See the term "Refuel" under the Definition of "Surveillance Frequency" for specific time limits on surveillances with a frequency that includes the term "Refueling Outage."

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation, or 1.08 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66W + 66\% - 0.66\ \Delta W$
(Clamp @ 120%)

where:

S = Setting in percent of rated thermal power (3458 MWt)

W = Loop recirculating flow rate in percent of design.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

B. Core Thermal Power Limit
(Reactor Pressure ≤ 800 psia)

When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

B. APRM Rod Block Trip Setting

$S_{RB} \leq (0.66\ W + 54\% - 0.66\ \Delta W)$
(Clamp @ 108%)

where:

S_{RB} = Rod block setting in percent of rated thermal power (3458 MWt)

W = Loop recirculation flow rate in percent of design.

ΔW = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single loop operation, the reduction in trip setting ($-0.66\ \Delta W$) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM Rod block setpoint and recirculation drive flow or by adjusting the APRM Rod block trip setting. $\Delta W = 0$ for two loop operation.

The APRM rod block trip setting shall not exceed 108% of rated thermal power.

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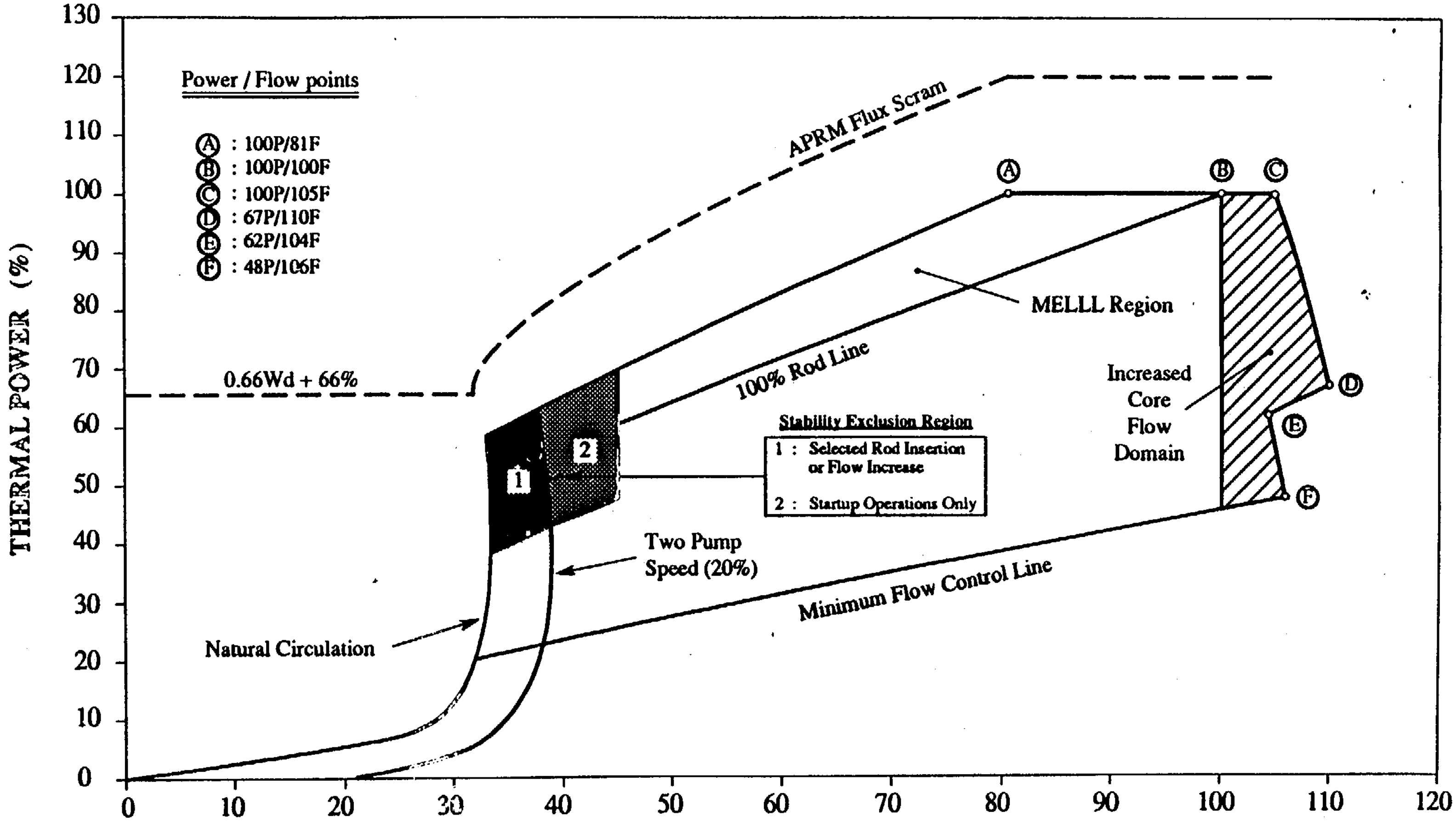
Figure 1.1-1 APRM Flow Biased Scram Relationship to Normal Operating Conditions



Notes : P = Uprated Thermal Power
(100% uprated Power = 3458 MWt)
F = Mlb/hr Core Flow
(100% Core Flow = 102.5 Mlb/hr)

2.1 BASES: FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to or above the thermal power condition required by Regulatory Guide 1.49. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3458 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded. (See Reference 4).

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. Conservatism incorporated into the transient analysis is documented in References 2 and 3.

2.1 BASES (Cont'd)

For analyses of the thermal consequences of the transients, a MCPR equal to or greater than the operating limit MCPR given in Specification 3.5.K is conservatively assumed to exist prior to initiation of the limiting transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady state operation without forced recirculation will not be permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculating pumps.

In summary:

i. The abnormal operational transients were analyzed at or above the maximum power level required by Regulatory Guide 1.49 to determine operating limit MCPR's.

ii. The licensed maximum power level is 3458 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram

The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (3458 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.1 <u>BASES</u>: (Cont'd)

L. <u>References</u>

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO 10802, February 1973.

2. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

3. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

4. "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3," NEDC-32183P, May 1993.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to limits on reactor coolant system pressure.

Objectives:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

1. The reactor vessel dome pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objectives:

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification:

1. The limiting safety system settings shall be as specified below:

Protective Action/Limiting Safety System Setting

A. Scram on Reactor Vessel high pressure

$\leq$1085 psig

B. Relief valve settings

1135 psig (±11 psi)
(4 valves)
1145 psig (±11 psi)
(4 valves)
1155 psig (±12 psi)
(3 valves)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	C. Safety valve settings 1260 psig ± 13 psi (2 valves)
2. The reactor vessel dome pressure shall not exceed 75 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.	2. The shutdown cooling isolation valves shall be closed whenever the reactor vessel dome pressure is >75 psig.

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes In which Function Must Be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		x	x	x	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		x	x	x	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	x	x	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		x	x	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.66W+66%-0.66ΔW) (Clamp @ 120%) (12) (13)			x	6 Instrument Channels	A or B
6	2	APRM Inoperative	(13)	x	x	x	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	x	x		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1085 psig	x(9)	x	x	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	x(8)	x(8)	x	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	x	x	x	4 Instrument Channels	A

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable sensor channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the start up position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% rated.

2. Permissible to bypass, in refuel and shutdown positions of the reactor mode switch.

3. Deleted.

4. Bypassed when turbine first stage pressure is less than that which is equivalent to 30% of rated thermal power.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212 degrees F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge instrument volume high level

8. Not required to be operable when primary containment integrity is not required.

9. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

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NOTES FOR TABLE 3.1.1 (Cont'd)

10. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

11. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 14 LPRM inputs of the normal complement.

12. W = Loop Recirculation flow in percent of design.

    Delta W = The difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

    Trip level setting is in percent of rated power (3458 MWt).

13. See Section 2.1.A.1.

3.1 BASES (Cont'd)

the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph 7.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (reference paragraph 7.5.7 FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRM's and APRM's provide adequate coverage in the start-up and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for Startup and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1.1 operable in the Refuel mode assures that shifting to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. The main condenser low vacuum trip is bypassed except in the run position of the mode switch.

Turbine stop valve closure occurs at 10% of valve closure. When turbine first stage pressure is below that which corresponds to 30% of rated thermal power, the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

3.1 BASES (Cont'd.)

Turbine control valves fast closure initiates a scram based on pressure switches sensing Electro-Hydraulic Control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal EHC oil pressure of 1600 psig gauge that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure. This scram signal is also bypassed when the turbine first stage pressure indicates that reactor power is less than 30% of rated.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	(0.66W+54%-0.66ΔW) (Clamp at 108% max)	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	≤12%	6 Inst. Channels	(10)
4	APRM Downscale	≥2.5 indicated on scale	6 Inst. Channels	(10)
1 (7)(11)	Rod Block Monitor (Power Biased)	(RTP ≥85%), S_{RB} ≤HTSP (65% ≤RTP <85%), S_{RB} ≤ITSP (30% ≤RTP <65%), S_{RB} ≤LTSP	2 Inst. Channels	(1)
1 (7)(11)	Rod Block Monitor Downscale	≥DTSP	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤108 indicated on scale	8 Inst. Channels	(10)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	≤10^5 counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Level	≤25 gallons	1 Inst. Channel	(9)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W = Loop Recirculation flow in percent of design.

   Trip level setting is in percent of rated power (3458 MWt).

   ΔW is the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow. ΔW = 0 for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is ≥ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is ≤ 30%.

8. This function is bypassed when the mode switch is placed in Run.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd.)	4.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd.)
3. The Standby Liquid Control System conditions must satisfy the following equation: $\left(\frac{C}{13\%\ \mathrm{wt.}}\right)\left(\frac{Q}{86\ \mathrm{gpm}}\right)\left(\frac{E}{19.8\%\ \mathrm{atom}}\right)\geq 1$ where, C = Sodium Pentaborate Solution Concentration (% weight)	
Q = Pump Flow Rate (gpm) against a system head of 1255 psig.	3. Pump Flow Rate: At least once per month each pump loop shall be functionally tested by pumping boron solution to the test tank. At least once per quarter check and record pump flow rate against a system head of 1255 psig.
E = Boron-10 Enrichment (% atom Boron-10)	4. Enrichment: Following each addition of boron to the solution tank, calculate enrichment within 8 hours. Verify results by analysis within 30 days.
	5. Solution Volume: At least once per day check and record.

LIMITING CONDITIONS FOR OPERATION

3.5.C HPCI Subsystem (cont'd.)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.C HPCI Subsystem (cont'd.)

Item	Frequency
(b) Pump Operability	Once/month
(c) Motor Operated Valve Operability	Once/month
(d) Flow Rate at approximately 1030 psig Reactor Steam Pressure	Once/3 months
(e) Flow Rate at 150 psig Reactor Steam Pressure	Once/operating cycle

The HPCI pump shall deliver at least 5000 gpm for a system head corresponding to a reactor pressure of approximately 1030 to 150 psig.

2. DELETED

LIMITING CONDITIONS FOR OPERATION

3.5.D <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor steam pressure is greater than 105 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 below.

2. From and after the date that the RCIC Subsystem is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCI Subsystem is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 105 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.D <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

1. RCIC Subsystem testing shall be performed as follows:

Item	Frequency
(a) Simulated Automatic Actuation Test*	Once/Operating Cycle
(b) Pump Operability	Once/Month
(c) Motor Operated Valve Operability	Once/Month
(d) Flow Rate at approximately 1030 psig Reactor Steam Pressure**	Once/3 Months
(e) Flow Rate at approximately 150 psig Reactor Steam Pressure**	Once/Operating Cycle
(f) Verify automatic transfer from CST to suppression pool on low CST water level	Once/Operating*** Cycle

2. DELETED

* Shall include automatic restart on low water level signal.

** The RCIC pump shall deliver at least 600 gpm for a system head corresponding to a reactor pressure of approximately 1030 to 150 psig.

*** Effective at 1st refueling outage after Cycle 7 reload.

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3.5 BASES (cont'd.)

C. HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the HPCI System (Section 6.0).

The HPCIS is provided to assure that the rleactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1150 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in the FSAR, Appendix G, shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC serves as an alternate to the HPCI only for decay heat removal when feed water is lost. Considering the HPCI and the ADS plus RCIC as redundant paths, reference (1) methods would give an estimated allowable repair time of 15 days based on the one month testing frequency. However, a maximum allowable repair time of 7 days is selected for conservatism.

3.5 BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at $\geq$25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). See Reference 12. The transients evaluated are as described in References 7 and 10.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate including MELLL operation with implementation of ARTS restrictions (Ref. 11).

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. Letter, C. O. Thomas (NRC) to J. F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), 'The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident'," June 1, 1984.

5. DELETED.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analyses," NEDC-32163P, January, 1993.

9. DELETED.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

11. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.

12. "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3," NEDC-32183P, May 1993.

3.6.D & 4.6.D <u>BASES</u>

<u>Safety and Relief Valves</u>

The safety/relief and safety valves are required to be operable above the pressure (122 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for each unit at the Peach Bottom APS has been sized to meet two design bases. First, the total capacity of the safety/relief and the safety valves has been established to meet the overpressure protection criteria of the ASME code. Second, the distribution of this required capacity between safety/relief valves and safety valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which show compliance with the ASME code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report presented in Appendix K of the FSAR.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 2 with a total capacity of 75.30% of rated-steam flow. The analysis of the worst overpressure transient demonstrates margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total pressure relief system capacity of 75.30% has been divided into 62.21% safety/relief (11 valves) and 13.09% safety (2 valves). The analysis of the plant isolation transient shows that the 11 safety/relief valves limit pressure at the safety valves below the setting of the safety valves. Therefore, the safety valves will not open.

Experience in safety/relief and safety valve operation shows that a testing of 50 per cent of the valves per cycle is adequate to detect failure or deteriorations. The safety/relief and safety valves are benchtested every second

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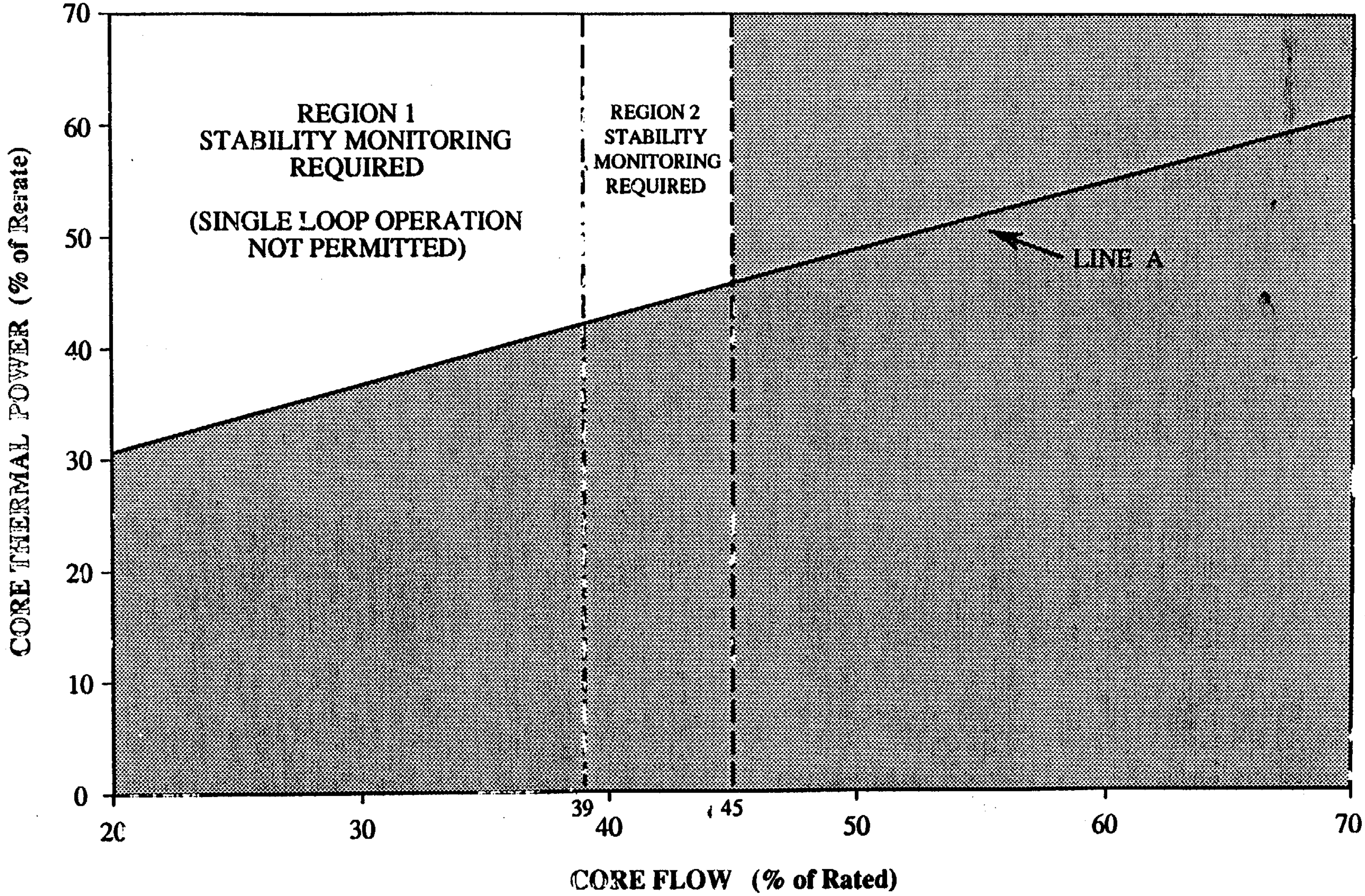
Fig 3.6.5
Thermal Power and Core Flow Limits
of Specifications 3.6.F.3, 3.6.F.4, and 3.6.F.5



3.7.A/4.7.A BASES

Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1038 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed 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 the specification, containment pressure during the design basis accident is approximately 49.1 psig which is below the maximum of 62 psig. Maximum water volume of 127,300 ft^3 results in a downcomer submergence of 4.4 feet and the minimum volume of 122,900 ft^3 results in a submergence approximately 0.4 feet less.

3.7.A & 4.7.A BASES (Cont'd)

The design basis loss-of-coolant accident was evaluated in the SER at the primary containment maximum allowable accident leak rate of 0.5%/day at 56 psig, a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in TID-14844. The SER shows that the maximum two hour dose is about 1.0 REM whole body and 14 REM thyroid at 4500 meters from the stack. The resultant doses in the SER that would occur for the duration of the accident at the low population zone distance of 7300 meters are about 2.5 REM total whole body and 105 REM total thyroid. As a result of uprating the power to 3,458 MWt, the corresponding doses calculated in UFSAR Subsection 14.9 are more conservative since they are based on a containment leak rate of 0.635% per day and larger dispersion (X/Q) values. These UFSAR analyses result in two hour doses at the Exclusion Area Boundary of about 1.0 REM whole body and 15 REM thyroid. The UFSAR analyses also result in doses at the low population zone distance (7300 meters) for the duration of the accident of about 3.9 REM whole body and 239 REM thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design bases loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Drywell Interior

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During the first year of operation the normal inerting nitrogen makeup system will be available for this purpose. After that time the specifically designed CAD system will serve as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 2000 gallons of liquid N_2 in the storage tank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no

3.7.A & 4.7.A BASES (Cont'd)

Due to the nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions, repressurization of the containment will reach 30 psig. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

APPENDIX B

TO

FACILITY OPERATING LICENSE DPR-44 AND FACILITY OPERATING LICENSE DPR-56

ENVIRONMENTAL

TECHNICAL SPECIFICATIONS AND BASES

FOR

THE FULL POWER FULL TERM

OPERATION OF

PEACH BOTTOM ATOMIC POWER STATION

UNIT 2

MAY 31, 1989

YORK COUNTY, PENNSYLVANIA
PHILADELPHIA ELECTRIC COMPANY
DOCKET NO. 50-277

l. Protection Limit - A numerical limit on a plant effluent or operating parameter which, when not exceeded, should not result in an unacceptable environmental impact.

m. Rated Thermal Power - Rated thermal power refers to operation at a reactor power of 3458 MWt.

n. Report Level - The numerical level of an environmental parameter below which the environmental impact is considered reasonable on the basis of available information.

o. Special Study Program - An environmental study program designed to evaluate the impact of plant operation on an environmental parameter.

p. Total Residual Chlorine - The sum of the free chlorine and the combined chlorine.

1.2 ABBREVIATIONS

a. AEC - Atomic Energy Commission

b. BWR - Boiling Water Reactor

c. 10CFR20 - Code of Federal Regulations;
Title 10 - Atomic Energy
Part 20 - Standard for Protection Against Radiation

d. 10CFR50 - Code of Federal Regulations;
Title 10 - Atomic Energy
Part 50 - Licensing of Production and Utilization Facilities

e. FSAR - Final Safety Analysis Report

f. NEPA - National Environmental Policy Act

g. MPC - Maximum Permissible Concentration

h. MSL - Mean Sea Level

i. PBAPS - Peach Bottom Atomic Power Station Units No. 2 and 3

j. POR - Plant Operations Review

k. O&SR - Operation and Safety Review

l. PMF - Probable Maximum Flood

m. PSAR - Preliminary Safety Analysis Report

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 198 TO FACILITY OPERATING

LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

DOCKET NO. 50-277

1 INTRODUCTION

In a letter of June 23, 1993 (Reference 1), as supplemented in letters of April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28 (two letters), September 16, September 30 and October 14, 1994, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Operating License for Peach Bottom Atomic Power Station, Units 2 and 3, and for Appendices A (Technical Specifications [TS]) and B ("Environmental Technical Specifications") to the Operating License. The licensee submitted NEDC-32183P, "Power Rerate Safety Analysis Report For Peach Bottom 2 & 3," Class III, May 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 boiling water reactor (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 April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28 (two letters), September 16, September 30, and October 14, 1994, the licensee submitted clarifying information that did not change the initial proposed no significant hazards determination, which was noticed in the *Federal Register* on August 29, 1994 (59 FR 44432).

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 LTR 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, NRC approved the program proposed in the LTR, provided individual power uprate amendment requests meet certain requirements in the document (Reference 4).

The generic BWR power uprate program was created to give a consistent means for individual licensees to recover additional generating capacity beyond their 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 was generally 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*. Since the design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS), 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 5.8-percent increase in rated steam flow (an increase in vessel steam flow from 13.37 to 14.14 Mlb/h). The planned approach to achieving the higher power level is to (1) increase the core thermal power to increase steam flow, (2) increase the feedwater system flow by a corresponding amount, (3) increase reactor pressure to ensure adequate turbine control margin, (4) not increase the current maximum core flow, and (5) operate 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 30 psi to ensure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

3 EVALUATION

The staff reviewed the request for a Peach Bottom power uprate amendment using applicable rules, regulatory guides, sections of the Standard Review Plan, and NRC staff positions. The staff also evaluated the Peach Bottom 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 impact 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. The map includes the increased core flow (ICF) domain and an uprated Maximum Extended Load Line Limit Analysis (MELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the previously analyzed core flow range when rescaled so that uprated power is equal to 100-percent rated.

3.1.3 Stability

The staff 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 staff reviewed recommendations from GE Service Information Letter SIL-380, Revision 1, NRC Bulletin 88-07, Supplement 1 (Reference 6), and current BWR Owners Group (BWROG) efforts including Interim Corrective Actions (ICAs) recommended by GE and the BWROG. In Reference 7, the licensee clarified the meaning of *ICA* and stated that the potential for BWR core thermal-hydraulic instability is documented in GE SIL-380, Revision 1. GE SIL-380, Revision 1, recommendations have been included in the operating procedures for PBAPS, Units 2 and 3. GE and the BWROG also developed ICAs to further address core stability concerns. In Reference 6, the NRC staff endorsed these ICAs, which have been implemented at PBAPS, Units 2 and 3. The ICAs include *operating exclusion regions* on the PBAPS power/flow map. Inadvertent entry into these regions requires immediate action to exit the region.

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, PBAPS Units 2 and 3 will maintain 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 PBAPS Units 2 and 3.

The licensee will continue following the restrictions recommended by NRC in Reference 6 and NRC Bulletin 88-07 (Reference 8) for uprated operation. The licensee will continue resolving these issues as directed by the joint effort of the BWROG and NRC. Based on the above discussion, the staff concluded that the licensee's actions with regard to thermal hydraulic stability are acceptable.

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 due to uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion is slowed down due to the increased pressure. As the scram continues, 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 continue to monitor scram performance by following various surveillance requirements as required in the plant TS to ensure that the original licensing basis for the scram system is preserved.

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. Using plant CRD pump and system data from the CRD system process instruments, the licensee calculated that with normal maintenance the CRD system will function adequately to insert and withdraw rods at uprate pressure levels. CRD positioning is classified as nonsafety-related. If worst case losses are used (i.e., the drive water filter is clogged and pump is degraded), operating normal CRDs requiring 250 psid may require adjusting the drive water pressure control valve to increase drive pressure. This would temporarily reduce the cooling water flow rate to the CRDs. Adjusting the drive water pressure control valve would be an operational consideration, however, it is not a safety concern.

Based on the above, the staff concludes that the CRD system will continue to perform all its safety-related functions at uprated power with ICF, 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 PBAPS 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 give nuclear system pressure relief to prevent overpressurization 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 accomodate the increased uprate 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. Uprated conditions will increase the peak reactor pressure vessel (RPV) bottom pressure to 1307 psig, but the analyzed pressure will remain below the 1375-psig ASME limit. Therefore, there is no decrease in the margin of safety.

3.2.3 Reactor Vessel and Internals

The licensee evaluated the reactor internals and reactor vessel components for the power uprate conditions in accordance with the requirements of the ASME *Boiler and Pressure Vessel Code*, Section III, Division 1, 1965 Edition with Addenda through Winter 1965 (Reference 9) to ensure compliance with the PBAPS original Code of Record. The design basis load combinations include reactor internal pressure difference (RIPD), seismic, and fuel lift loads as defined by the PBAPS Updated Final Safety Analysis Report (UFSAR). The licensee did the analyses for normal, upset, and faulted conditions. The licensee summarized the maximum stresses at the critical locations for the shroud support legs, steam dryer, and core plate in Table 1 of a letter of July 7, 1994 (Reference 10). The licensee performed fatigue evaluations according to Paragraph NB-3200 of the 1974 Edition of the ASME Section III Code including Summer 1976 Addenda. The limiting fatigue usage factor calculated for the uprated power level was 0.997 and was located at the feedwater nozzle as shown in Table 3-4 of Reference 2, using the design basis cycles of transients defined by the UFSAR. No new assumptions were used in the analysis for the power uprate condition. The maximum stresses and fatigue usage factor provided by the licensee are within the Code-allowable limits and are therefore acceptable.

However, the licensee performed a fatigue reanalysis as documented in Reference 12, based on the actual number of cycles and transients at PBAPS to date and upon the anticipated fatigue usage based on trends in past operating data. The licensee took this action to address an unresolved item discussed in NRC Inspection Report 50-277/90-14 and 50-278/90-14 (Reference 11). The staff evaluated the licensee's fatigue reanalysis as discussed in Section 3.2.11.

3.2.4 Control Rod Drive Mechanism

The licensee evaluated the adequacy of the PBAPS control rod drive mechanism (CRDM) in accordance with the ASME *Boiler and Pressure Vessel Code*, Section III, Division 1, 1968 Edition, with Winter 1970 Addenda (Reference 13). The licensee found the limiting component of the CRDM to be the indicator tube, which has a calculated stress of 20,790 psi (allowable stress: 26,060 psi). The maximum stress was calculated based on a maximum CRD internal water pressure of 1750 psig, which is not affected by the power uprate. The licensee calculated the maximum fatigue usage factor to be 0.15 for the CRD main flange for 40 years of plant operation. The licensee stated that the CRDM has been tested at simulated reactor pressure up to 1250 psig, which bounds the high pressure scram setpoint of 1101 psig for the power uprate.

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. The core reload analyses are performed with the most conservative allowable core flow. A review of the reactor recirculation system (RRS) thermal-hydraulic performance at the uprated power condition shows that the core flow can be maintained at 105 percent.

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 by 30 psi to operate at the uprated power will increase the RRS pump suction pressure by 30 psi and the RRS pump discharge pressure by 32 psi. 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 3 °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 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. In response to a staff question, the licensee stated in Reference 7 that it did a detailed vibration analysis for the RRS piping for uprate conditions and found the uprate resulted in a negligible effect. By letter dated October 14, 1994, the licensee committed to perform recirculation pump vibration monitoring and a reactor building walkdown to detect pump induced structural vibration while the recirculation pump was operating at uprated speeds during power ascension testing. The licensee committed to monitoring for recirculation pump induced structural vibration at the end of the fuel cycle when the recirculation pump is operated at uprated speeds.

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 single feedwater pump is tripped 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 and thus reducing the power level. The RRS pump speed runback setpoint is 45 percent of rated pump speed, which corresponds to at least 54 percent of rated core flow.

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 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 did this evaluation to ensure compliance with requirements of the code of record as specified in Appendix A to the PBAPS UFSAR. For example, USAS B31.1.0, "Power Piping," 1967 Edition (Reference 14) is the code of record for all piping and pipe supports with the exception of the Recirculation System for which the code of record is the ASME *Boiler and Pressure Vessel Code* Section III, 1980 Edition including Winter 1981 Addenda, and portions of torus attached piping which were designed to the ASME Section III, 1977 Edition through Summer 1977 Addenda. 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. The licensee evaluated the maximum stress levels for BOP piping based on the bounding percentage increases in Table 3-5 of Reference 2 and concluded that most BOP systems were originally designed to maximum temperatures and pressures that bounded the increased operating temperature and pressure for the power uprate, and are, therefore, acceptable. 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 main steam isolation valves (MSIVs) and found them consistent with the bases and conclusions of the generic evaluation.

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 reactor core isolation cooling (RCIC) system supplies core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low-pressure core cooling system. The licensee evaluated the RCIC system and found it consistent with the generic evaluation. The licensee stated that the PBAPS RCIC turbines have not experienced the overspeed trips described in SIL 377. PBAPS may not require the SIL 377 modifications based on evaluation of RCIC startup transient plant data. Testing during the first cycle of operation at uprate conditions will verify this evaluation. The modifications will not be needed if testing confirms that the startup transients will not cause excessive peak transient speeds at the increased (uprate) reactor pressure. The staff asked the licensee why it did not use the guidance of SIL 377 for the RCIC system and received the following response:

> The PBAPS RCIC turbine is a Terry turbine, GS-1 model. This model has less inlet nozzles than later RCIC turbine models (GS-2). As a result, the effects of start-ups are less severe. Review of actual PBAPS RCIC start-up transient data shows that there is sufficient margin between the initial speed spike and the overspeed trip setpoint. Following implementation of power uprate at PBAPS, Units 2 & 3, a test will be performed during the start-up of the unit to confirm this margin (Reference 7).

By letter dated October 14, 1994, the licensee committed to perform testing to asssure RCIC injection capability at uprated power as part of its power uprate testing program. The licensee stated that RCIC system reliability is not expected to be impacted by operation at uprated power conditions. Based on the review of the licensee's information and commitments, the staff finds the RCIC system acceptable for operation at uprate power conditions.

3.2.9 Residual Heat Removal System

The residual heat removal (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.

(a) Shutdown Cooling Mode

The licensee evaluated shutdown cooling for when either one or two RHR loops are available. The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. The two-loop analysis was performed to determine the time to reach 125 °F, while the one-loop analysis was to determine the time to reach cold shutdown (<212 °F). The decay heat increases proportionally at the uprated power level, thus increasing the time required to reach the shutdown temperature. The analyses are based on 110 percent of original thermal power and use a conservative combination of the May-Witt and American Nuclear Society (ANS) standard decay heat curves. The two-loop analysis indicates that the cooling time to achieve the 125 °F reactor vessel temperature increases from 20 to 27 hours. The 20-hour shutdown cooling time criterion was established for outage scheduling, and therefore, the additional cooldown time does not affect plant safety. Based on the one-loop analysis at uprated conditions, the RHR system can achieve cold shutdown in less than 24 hours with one RHR loop available. The shutdown cooling analyses use heat transfer coefficients (k-values) based on revised data from the manufacturer of the RHR heat exchangers. The revised data yield lower (more conservative) k-values. The two-loop analysis assumes that the RHR heat exchanger flows are 100-percent rated on both the shell and tube sides; however, the one-loop analysis assumes 80-percent rated shell side flow and 100-percent rated tube side flow.

(b) Suppression Pool Cooling Mode

The functional design basis as stated in the UFSAR for the suppression pool cooling mode (SPCM) 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 Reference 2 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.

3.2.10 Reactor Water Cleanup System

The operating termperature and pressure of the reactor water cleanup (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.

3.2.11 Evaluation of Reactor Vessel Fatigue Reanalysis

In Reference 12, the licensee reevaluated the fatigue cumulative usage factor for limiting reactor vessel components such as the feedwater nozzle, vessel support skirt, vessel closure studs, refueling containment skirt, and recirculation inlet nozzle. The licensee used the actual number of cycles and transients that the plant experienced rather than the assumptions from the original fatigue analyses. The licensee also considered cyclic transients such as excessive heatups, feedwater temperature reduction, HPCI/RCIC injection, excessive cooldown events, safety/relief valve blowdown, and the sudden start of a recirculation pump in a cold recirculation loop. These event transients were not considered in the previous design basis fatigue analyses. The fatigue reanalysis did not include the recirculation outlet nozzle, which was considered the limiting fatigue location for the power uprate in Reference 2, because it is not significantly affected by the modified cycles, as stated by the licensee.

The original fatigue usage was modified for the actual number of cycles using the ratio of actual cycles to the number of cycles assumed in the original analysis. GE calculated the fatigue usage for the uprated power in Appendix C to Reference 12, based on the actual number of cycles and the revised allowable number of cycles corresponding to the revised peak stress for the power uprate. The revised peak stresses were calculated by scaling up the original stresses with the maximum scale factor to account for the increase of pressure, temperature, and flow rate for the uprated conditions.

The licensee reevaluated fatigue in accordance with Paragraph NB-3200 of the 1974 Edition of the ASME Section III Code including the Summer 1976 Addenda. Table 1-1 of Reference 12 summarizes the fatigue usage factors of the limiting components for the original, modified event cycles and power uprate conditions. The table shows that all fatigue usage values are within the Code allowable value of 1.0 except for the vessel closure studs. Using the actual number of cycles, the fatigue usage for the closure studs through 40-years is calculated to be 1.09 (>1.0) for both the rated and uprated power conditions. The licensee's analysis provided a number of options for managing the fatigue usage factor for the closure studs. The appropriate option will be determined in the future depending on future operating history. Using the actual number of cycles and transients, the fatigue usage factor for the vessel support skirt was calculated to be 0.998 for 40 years of operation, consisting of 21 years of uprated power and 19 years of rated power. A detailed finite element model was used for the stress and fatigue analyses of the vessel support skirt, as documented in Appendix A to Reference 12. The licensee committed to perform future reevaluations using actual plant experience and refined calculational methods as necessary.

The staff reviewed and accepts the methodology and results of the licensee's fatigue reanalyses. Based on the licensee's exiting fatigue reanalysis and the licensee's commitment to manage fatigue issues with appropriate reanalyses, the staff concludes that operation of the plant at uprated power levels is acceptable with respect to fatigue.

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

Section 5.10.2 of Reference 3 requires the power uprate applicant to show the uprated power level is acceptable 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 applicant's approach for doing 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 (Reference 15)
net positive suction head (NPSH),
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 16). 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.

3.3.1.2 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 102 percent of 110 percent of original rated power using the SHEX code and ANS 5.1 decay heat assumptions prescribed by Reference 3. The licensee increased the initial drywell temperature in the uprate analysis from 135 °F to 145 °F, and the initial suppression pool temperature was increased from 90 °F to 95 °F extra temperature margin. All other key input parameters for power uprang 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 45.4 psig for the limiting DBA LOCA with break flow from a more detailed RPV model in accordance with 10 CFR Part 50, Appendix K, and 100-percent core flow with a 55 °F feedwater temperature reduction. The containment is designed for a pressure of 56 psig. Therefore, the maximum pressure of 45.4 psig at uprated power remains below the containment design pressure.

Technical specification definitions, limiting conditions for operation, surveillance requirements, and bases relating to the current 49.1 psig value of P_a will not be revised as it remains higher than the maximum containment pressure of 45.4 psig calculated for the power uprate.

Based on the above review, the staff concludes that the Peach Bottom containment pressure response following a postulated LOCA will remain acceptable after power uprate.

3.3.1.3 Containment Dynamic Loads

(1) LOCA Containment Dynamic Loads

Reference 3 requires that the power uprate applicant 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 licensee performed the short-term analyses articulated in Reference 3 and concluded 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 reanalyzed the containment dynamic loads to reflect increased SRV opening setpoints (2.7-percent or 30 psi increase) and changes in SRV discharge line water level at the time of subsequent SRV actuations. The licensee compared the increased SRV loads with the plant-unique design limits in the Mark I Containment LTP and found sufficient conservatism in the original containment dynamic loads definition to accommodate the increased SRV loads. The limiting SRV originally had about 11-percent margin to the load definition before power uprate and about 8 percent after power uprate. The results of the reanalysis indicate that the loads remain below their design values, and are therefore 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 peak pressure in the annulus increases only slightly due to a small increase in the blowdown flow. The mass-energy release rates are not significantly affected by power uprate. 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.

3.3.1.4 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.

3.3.1.5 Primary Containment Atmosphere Control and Dilution System (PostLOCA Combustible Gas Control)

The containment atmospheric dilution system (CADS) maintains an inert mixture of gases in the containment atmosphere after a DBA LOCA. The combustibility of the post-LOCA containment atmosphere is controlled by the concentration of oxygen. The post-LOCA production of oxygen by radiolysis will increase proportionally with the power level and will also increase slightly because of a higher peak temperature in the containment. The licensee stated that the CAD system has sufficient capacity to accommodate the increased oxygen production and that the initiation of CAD is controlled procedurally based on gas concentration in the containment. Based on the above discussion, the staff concludes that the post-LOCA combustible gas control will remain acceptable after uprated 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 uprate increases the calculated peak suppression pool temperature, which could decrease the NPSH available to the ECCS pumps. However, as suppression pool temperature increases, so does the containment pressure, which increases the NPSH available to the ECCS pumps. The NPSH requirements of the ECCS pumps are evaluated at a conservatively high suppression pool temperature and a conservatively low containment pressure. At design conditions, sufficient margin to the required NPSH exists with the RHR and CS systems at rated loop flows. Assuming a LOCA occurs during operation at the uprated power, the calculated suppression pool temperature will remain below the value used in the NPSH analysis. Therefore, power uprate will not affect compliance with NPSH requirements for the ECCS pumps.

3.3.2.1 High-Pressure Coolant Injection System

The licensee evaluated the high-pressure coolant injection (HPCI) system and hardware for power uprate conditions and found the HPCI consistent with the basis and conclusions of the generic evaluation. In response to a staff question, the licensee stated that SIL 480 has been implemented at PBAPS, Units 2 and 3, for the HPCI system (Reference 7).

By letter dated October 14, 1994, the licensee committed to perform testing to asssure HPCI injection capability at uprated power as part of its power uprate testing program. The licensee stated that HPCI system reliability is not expected to be impacted by operation at uprated power conditions. Based on the review of the licensee's information and commitments, the staff finds the HPCI system acceptable for operation at uprate power conditions.

3.3.2.2 RHR System (Low-Pressure Coolant Injection)

Section 3.3.3 addresses the adequacy of the low-pressure coolant injection (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 methodolgies described in Section 3.3.3. The ability to provide 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 licensee used the staff-approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 and Appendix K criteria (Reference 19). The PBAPS 2/3 S/G - LOCA results demonstrate that a sufficient number of plant-specific peak cladding temperature (PCT) points have been evaluated to establish the shape of both the nominal and Appendix K

PCT versus break size curves. The analyses demonstrate that the limiting licensing basis PCT occurs for the recirculation suction line DBA.

The licensee also evaluated ECCS performance for PBAPS 2 and 3 under Single Loop Operation (SLO) using S/G - LOCA calculations for the DBA. With a MAPLHGR multiplier of 0.90, the SLO DBA Appendix K PCT is 1641 °F for BP/P8x8R fuel, which is less than the two-loop DBA Appendix K PCT result. The licensee concluded that the actual PCT for SLO will always be lower than for two-loop operation.

An analysis for the 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). 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. The BP/P8x8R fuel type was chosen as the worst case since its LOCA results are the most sensitive to potential EBT because of the high initial fuel stored energy. The results showed that EBT does not occur at 75-percent initial core flow for the high power node. Next, SAFER calculations were performed at 75-percent initial core flow with BP/P8x8R and GE 11 fuel. For GE 11 fuel, EBT of high power node is conservatively assumed to occur. The results show that the 75-percent flow PCT is 35 °F higher for BP/P8x8R fuel (less for GE 11) compared with the rated flow PCT for Appendix K assumptions (1717 °F compared to 1682 °F). 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 not large 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 consideration.

3.3.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to limit the ground level release from the reactor building, and to release primary and secondary containment air at an elevated release point via the stack. The SGTS is common to both Units 2 and 3 and is located in a shielded room in the radwaste building between the reactor buildings. The SGTS consists of two parallel filter trains connected to three full-capacity exhaust fans. Each filter train can serve either unit during drywell purge at the rate of 8,500 cfm, not to exceed 10,500 cfm while maintaining a negative 1/4-inch water gauge pressure in the reactor building. Each fan is capable of exhausting the rated flow through either filter train and up through the stack. Upon a reactor building isolation signal, the reactor building ventilation isolation valves isolate the reactor building atmosphere in 3 to 10 seconds. At the same time, the SGTS is automatically started to maintain a negative pressure in the reactor building. Potentially contaminated air from the reactor then passes through the SGTS for filtration prior to elevated release from the stack.

In its power uprate submittal, the licensee noted that the design of the charcoal filter system and therefore its capability to meet its design objectives will not be changed by the power uprate. The staff recognizes that iodine loading in the filters will increase marginally (5 percent) due to the proposed power uprate. The increase in dose rates from the worst case accident analyzed by the licensee (3.1 rem from a main steam line break) will also increase by nominally the same margin. However, this worst case dose is still far below 10 CFR Part 100 limits. Additionally, the same percent increase in flow through the SGTS expected from the worst case accident will still be well below the design maximum capability of the system.

Based on the above findings, the staff concludes that the uprated power level operation will have an insignificant effect on the capability 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 is one of the control room habitability systems. The system consists of ventilation air supply fans, emergency air supply fans, air conditioning supply and return fans, filters, heating coils, refrigerant water chillers, chilled water pumps, filters, dampers, duct work, instrumentation, and controls. The emergency makeup air system filter train consists of a pre-filter, electric heaters, and a redundant filtration system consisting of a charcoal adsorber and HEPA filters, one upstream and another downstream of the adsorber. The emergency makeup air filter train filters the radioiodine and radioactive material in particulate form present in the makeup air intake during an emergency situation such as a design basis accident (DBA). The emergency recirculation train consists of a mixture of the control room recirculated air and filtered outside makeup air. The filters are designed in accordance with Regulatory Guide (RG) 1.52 Reference (20) guidelines. The system accomplishes its design objective by bringing in controlled and filtered outside air and mixing it with the recirculated air to keep the control room operator doses within the Genral Design Criteria (GDC) 19 limits during an accident. The staff concludes that the proposed increase in power (5.0 percent) by itself will not cause any increase in unfiltered inleakage of contaminated outside air into the control room during an accident since it does not change the ventilation design aspect of the control room emergency filtration system.

The staff recognizes that iodine loading in the makeup air filters will increase marginally (5.0 percent) due to the proposed power uprate. In Reference 21, the licensee stated that it evaluated the iodine loading on the control room filter for accident releases for the uprated plant. The filter loading based on 102 percent of uprated power was calculated to be 2.46E2 milligrams of iodine per gram of carbon, which is well below the limit of the RG 1.52 acceptance criterion (no more than 2.5 milligrams of iodine (radioactive and stable) per gram of activated carbon). The staff concludes that its earlier conclusion regarding the filters meeting the guidelines of RG 1.52, continues to be valid for the proposed uprated power situation.

In the UFSAR, Section 12.3.4.1, the licensee stated that the design basis accidents defining the protection required for the main control room are the refueling accident and the LOCA. In its power uprate submittal, the licensee made a comparison of the calculated dose resulting from the DBAs and has shown that the increase in exposure is minimal and well below the limits in GDC 19. The licensee used plant-specific radiological analyses based on Atomic Energy Commission methodology which included the use of TID-14844 source terms to perform the analyses at uprated conditions for selected postulated accidents. While a direct comparison between the original and uprated values in the tables provided in the licensee's submittal was not meaningful because the original analyses could not be exactly reconstituted, as further discussed in Section 3.7.2, the analyses were performed in a conservative manner by using the more conservative dose (chosen from the UFSAR dose and the dose from the reconstituted analysis) to adjust for values at the uprated power level.

The staff concludes that the uprated power level will not have any effect on the Control Room Atmosphere Control System meeting its design objectives.

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.

This section provides the basis for acceptance of setpoint changes for several instruments at PBAPS. The conservative design calculations for the initial licensing of PBAPS resulted in setpoints which provided excess reactor coolant flow capacity and corresponding margins in the power conversion system. For PBAPS, 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 following setpoint changes have been proposed by the licensee:

1. APRM Flow Biased Simulated Thermal Power

   a. Flow Biased
      Change trip from 0.66W + 71% - 0.66dW to 0.66W + 66% - 0.66dW.
      Change Analytical Limit from 0.66W + 71% to 0.66W + 66%.

   b. Flow Clamped
      No change in trip setpoint.
      No change in Analytical Limit.

2. Reactor Vessel Steam Dome Pressure High
   Change trip from 1055 psig to 1085 psig.
   Change Analytical Limit from 1071 psig to 1101 psig.

3. Main Steam High Flow
   The instrumentation will be recalibrated for the higher steam flow condition. The Analytical Limit remains at 140% of the uprated steam flow condition.

4. APRM Rod Block - Flow Biased Neutron Flux Upscale
   Change trip from 0.66W + 59% - 0.66dW to 0.66W + 54.0% - 0.66dW.
   Change Allowable Value from 0.66W + 59% to 0.58W + 54.0%.

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.

The licensee's application (Reference 1) did not describe the methodology used for instrument setpoint calculations. Therefore, in a letter of March 29, 1994, (Reference 22), the staff requested additional information regarding instrument setpoint methodology. The licensee, in a letter of May 2, 1994, (Reference 23) confirmed that GE Licensing Topical Report NEDC-31336P (Reference 24) 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 PBAPS changes.

For the turbine valves, the licensee used a PECO-specific instrument setpoint methodology which is consistent with GE setpoint methodology with some minor exceptions. The staff reviewed these exceptions for this application and finds them acceptable. The setpoint calculation for the turbine valves is based on 30% power, considering the uprated power level. This approach maintains the original safety basis for these setpoints. The staff finds this approach acceptable.

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. 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 PBAPS licensing basis and are, therefore, acceptable.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling

The spent fuel pool cooling system is designed to remove the decay heat generated by the stored spent fuel assemblies. Each spent fuel pool cooling system consists of three fuel pool cooling pumps, three heat exchangers, a filter-demineralizer, two skimmer surge tanks, and associated piping, valves, and instrumentation. The three fuel pool pumps are connected in parallel, as are the three heat exchangers. The heat exchangers in the RHR System can be used in conjunction with the fuel pool cooling and cleanup system to supplement pool cooling.

The analyses of the fuel pool cooling system in the UFSAR were performed for normal offload of 1/3 core every 18 months, and a full core offload just before refueling assuming all storage cells are filled, neglecting any use of the RHR heat exchangers. The guidance in the Standard Review Plan (Reference 25) (SRP 9.1.3) for the spent fuel pool states that the temperature of the pool should be kept at or below 140 °F for the maximum normal heat load with normal cooling systems in operation and assuming a single active failure. For the abnormal maximum heat load (full core unload) the temperature of the pool water should be kept below boiling and the liquid level maintained with normal systems in operation (a single active failure need not be considered for the abnormal case). On June 6, 1994, the licensee submitted calculations demonstrating that the guidance of SRP 9.1.3 was met for the maximum normal heat load with a calculated pool outlet temperature of 137.8 °F (Reference 26). The licensee made its determination of the fuel pool cooling adequacy for uprate by assuming a 24-month fuel cycle. The licensee also determined that the fuel pool temperature will remain below the design temperature of 150 °F, for a full core offload just before normal refueling, with all remaining storage spaces filled with used fuel off-loaded at regular intervals. Therefore, the pool temperature will be well below the SRP guidance for the abnormal maximum heat load.

The staff finds that using a 24-month fuel cycle in the analysis to be more conservative as this will result in a larger heat load on the fuel pool cooling system. The results of the licensee's evaluation are below the values of the SRP for the maximum normal heat loads, and well below the limits for the abnormal maximum heat load. With the use of the High Pressure Service

Water System via the RHR heat exchangers as backup, the spent fuel pool cooling system is sufficient for maintaining the spent fuel pool temperature within the guidance in the Standard Review Plan for all refueling scenarios.

The staff concludes that the spent fuel pool cooling system will be acceptable for operation at the uprated power level.

3.5.2 Water Systems

The licensee evaluated the effect of power uprate on the various plant water systems including the safety-related and non-safety-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 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 (ESW) system and the high pressure service water system (HPSW). All heat removed from these systems is rejected to the ultimate heat sink (UHS) except when the pump structure is isolated from Conowingo Pond or when local flooding occurs. Under these circumstances, heat from these systems is rejected to the emergency cooling tower (ECT). The staff's evaluation of the effects of uprated power level operation on each of these systems is provided below.

The ESW system was evaluated for its ability to provide cooling to emergency diesel generators and the emergency cooling equipment and space coolers during a loss of off-site power. The ESW system heat loads include the heat rejected by the residual heat removal (RHR) pump seal water coolers and the room unit coolers for such systems as RHR, HPCI, RCIC, and CS. A change in the heat load from the diesel generator coolers is not anticipated since no new or significantly increased electrical loads are imposed on the emergency diesel generators. The staff recognizes that there will be a slight increase in the heat loads of the room unit coolers as a result of the small increase in the torus temperature due the power uprate. In its power uprate submittal, the licensee stated that this increase in the torus temperature will result in an expected increase in the ESW system return temperature of less than 1 °F. The staff considers this to be an insignificant change in the heat loads and agrees with the licensee that the power uprate does not affect the heat removal capability of the ESW system.

The staff concludes that the uprated power level will not have any effect on the ESW system meeting its design objectives.

The HPSW system provides cooling water for the RHR system during normal reactor shutdown, post-accident shutdown, hot standby, refueling, and normal plant operation. The safety objective of the HPSW system is to provide a reliable supply of cooling water for RHR under post-accident conditions. There will be no significant increase in the HPSW heat load during normal plant operation, hot standby, or refueling, since the operating parameters for the RHR system have not changed during these operating modes. Likewise, there is also no increase in the HPSW heat load when the RHR system is operating in the shutdown cooling mode during normal reactor shutdown, since the RHR shutdown cooling mode initiating pressure and temperature are not changed by uprate. The following functions of the HPSW system are affected to a small degree by uprate mainly due to higher decay heat from the fuel:

a. Increased heat load from the RHR system when operating in the torus cooling mode following a postulated LOCA (HPSW suction from and discharge to Conowingo Pond).

b. Increased heat load from the RHR system when operating in the torus cooling mode following a postulated scram due to a loss of offsite power without a LOCA (HPSW suction from and discharge to the emergency cooling tower (ECT)).

c. Increased heat load from the RHR system when operating in the fuel pool cooling mode (backup system).

In its submittal, the licensee stated that the increased heat loads after uprate result in a maximum HPSW system return temperature increase of approximately 5 °F, and for torus cooling when HPSW is aligned with supply from and return to the ECT, the increase in the HPSW system heat load will result in a system temperature rise of less than 2 °F. The slightly increased heat loads that give rise to these temperature increases remain within the heat removal capacity of the system. The staff agrees with the licensee that the design flow rates and heat removal capacities are acceptable for the proposed power uprate.

Based on the information discussed above, the staff concludes that the uprated power level will not have a significant effect on the HPSW system meeting its design objectives, and is therefore, acceptable.

3.5.2.2 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 is designed to provide screened and chlorinated cooling water to the plant during normal plant operation and shutdown periods. The system is also able to provide a supply of water to the reactor building cooling water heat exchangers in the event of a loss of off-site power through system interconnections. Additionally, the service water system supplies cooling water to the core standby cooling equipment and space coolers during normal plant operation and shutdown periods. The system accomplishes its functions while inhibiting the release of radioactive material into the river.

In its power uprate submittal, the licensee stated that the increase in heat loads due to the power uprate will result in a temperature increase of 1 °F in the bulk outlet temperature which returns to the discharge pond, and further that this is insignificant to the design of the system.

Since the service water system does not perform any safety function, the staff has not reviewed the effect of the uprated power level operation to the service water system design and performance.

3.5.2.3 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 duty of the actual yearly range of circulating water inlet temperatures, and confirms that the condenser and circulating water system are adequate for uprated conditions. The net result of the power uprate is that the difference between the operating pressure and the required minimum condenser vacuum is reduced slightly.

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.

3.5.2.4 Reactor Building Closed Loop Cooling Water System (RBCCW)

The reactor building cooling water system is designed to cool auxiliary plant equipment over the full range of reactor power operation, and to inhibit the release of radioactive material to the environment. The licensee stated in its power uprate submittal that an increase in temperature rise of approximately 2 °F in the bulk RBCCW temperature returning to the RBCCW heat exchangers is expected. The RBCCW heat exchangers were conservatively designed with heat loads which bound those anticipated for operation at the uprated power level. Therefore, there is no effect to the system design.

The staff concluded that the effect of uprated power operation on the RBCCW system is negligible and that there is sufficient operating margin for this system to perform adequately at uprated conditions.

3.5.2.5 Turbine Building Closed Loop Cooling Water (TBCCW) System

The TBCCW system is designed to cool non-nuclear auxiliary plant equipment over the full range of plant operation.

In its submittal, the licensee stated that the heat loads felt by the isolated-phase bus coolers, condensate [pump] thrust bearing oil coolers, and the motor bearing coolers will increase in proportion to the increase in plant electrical power output. However, the flows to these coolers are small enough so that the increased heat load is insignificant. The remaining heat loads are not power dependent and will not be affected by the power uprate. The bulk turbine building closed cooling water system temperature returning to the TBCCW system heat exchangers will increase by approximately 1 °F as a result of the power uprate. The TBCCW system heat exchangers were conservatively designed with heat loads which bound those anticipated for operation at the uprated power level. Therefore the effect of the power uprate to the TBCCW system will be insignificant.

Since the TBCCW system does not perform any safety function, the staff has not reviewed the effect of the uprated power level to the TBCCW system design and performance.

3.5.2.5 Ultimate Heat Sink

The ultimate heat sink (UHS) for PBAPS 2 and 3 is the Conowingo Pond on a once-through cooling basis. While the power uprate will not affect the temperature of water drawn from the UHS, discharges to the UHS will increase due to the 5-percent increase in reactor decay heat. The licensee stated in its power uprate submittal that the increase in discharge temperature due to uprate is small and will have an insignificant effect on the UHS. Additionally the licensee determined that the existing UHS system will continue to provide a sufficient quantity of water following a LOCA. Because of the insignificant impact on UHS temperatures and continued assurance of adequate UHS inventory, the staff agrees with the licensee conclusion that the UHS design is acceptable for the uprated power level operation.

Based on the above discussion, the staff concludes that uprated power operation will have little or no effect on the existing UHS in performing its design objectives and, therefore, is acceptable.

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 30 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 heating, ventilation and air-conditioning (HVAC) system is designed to control the plant air temperatures and the flow of airborne radioactive contaminants to ensure the operability of plant equipment and the accessibility and habitability of plant buildings.

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 2 °F, with the exception of the non-regenerative heat exchanger area which will experience an increase in area temperature of approximately 7 °F.

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, although the peak temperatures increased, 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 were originally designed to use 105 percent of the rated power available from the nuclear steam supply system. The licensee stated in its power uprate proposal that operating the plant at the new power rating will have minimal effect on the balance-of-plant system instruments and control devices. Each of the process control valves and instruments (except for a few nonsafety-related devices) has sufficient range and adjustment capability for use at the expected uprated conditions.

The objective of the pressure control system gives a fast and stable response to pressure and steam flow distrubances 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.

Although the licensee will not need to modfy the turbine control valves or the turbine bypass valves for them to operate at the uprated throttle pressure conditions, operation under these conditions could result in third harmonic steam line resonances. The licensee committed in its power uprate submittal to add an additional harmonic notch filter to each turbine pressure control unit.

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 licensee stated in its power uprate submittal that it will need to collect only slightly more liquid radwaste. The largest contributor to the liquid waste is the backwash of the condensate demineralizers. The power uprate will increase the flow rate through the condensate demineralizers and thus reduce the average time between backwashes. The licensee stated that this reduction does not affect plant safety. Neither the floor drain collector subsystem nor the waste collector subsystem will need to process significantly larger amounts of liquid waste when the plant operates in the uprated condition.

The licensee stated that while the activated corrosion products in liquid wastes will increase proportionally to the power uprate, the total volume of processed waste will not will not increase appreciably since the only significant increase in processed waste will be from the more frequent backwashes of condensate and RWCU demineralizers. However, the licensee analyzed the liquid radwaste system and concluded the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will be met.

The staff agrees that the power uprate will not have a significant effect on the liquid radwaste system which, therefore, remains acceptable for the uprated power level.

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. 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 contains 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 are 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.

The staff agrees with the licensee that the effects of the power uprate on the gaseous waste management system are not significant and the system remains acceptable for the power uprate.

3.6.3 Radiation Sources in the Core and Coolant

Radioactive materials in the reactor core are produced in direct proportion to the fission rate. Thus, the expected increase in the levels of radioactive materials (for both fission and neutron activation products) produced will increase by a maximum of 5 percent. The licensee noted that experience with operation of PBABS indicates that concentrations of fission and activation products in the reactor coolant will not increase significantly. Current experience with operation of PBAPS indicates that the unit operates well below the 0.1 Curie/sec design basis and that current offsite radiological release rates are well below the original design basis. The staff reviewed available plant data and experience with previous power uprates and concludes that the power increase will not significantly affect radiation sources in either the core or reactor coolant.

3.6.4 Radiation Levels

The licensee evaluated the affects of the power uprate on in-plant radiation levels in the Peach Bottom 2 and 3 facility during normal conditions. The radiation levels during periods of normal operation and post-operation are expected to increase by no more than the percentage increase in power level. However, because many areas of the plant were designed for higher than expected radiation sources, the small increase in radiation levels expected due to power uprate will not affect radiation zoning or shielding in the plant.

During periods of normal and post-operation conditions, individual worker exposures will be maintained within acceptable limits by the existing ALARA program, which controls access to radiation areas. The ALARA program at Peach Bottom has been instrumental in the lowering of annual collective doses at the plant over the past several years. Since 1985, the 3-year average dose at Peach Bottom 2 and 3 has decreased by approximately 70 percent.

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 list the limiting event(s) to be considered in each category of events, the analytical methods, the operating conditions to be assumed, and the criteria to be applied (Reference 3).

The following sections address each event, summarize the resulting transient safety evaluations for a representative core (based on PBAPS Unit 2 Cycle 10), and show the overall capability of the design to meet all transient safety criteria for uprated operation. Reference 3 lists the specific events to be analyzed for power uprate, the power level to be assumed, and the computer models to be used. The licensee used the GEMINI transient analysis methods listed in Reference 3.

Table 9-1 of Reference 2 summarizes the reactor operating conditions that apply most directly to the transient analysis and compares them to the conditions used for the UFSAR and the most recent reload fuel cycle (Unit 2 Cycle 10) analyses. The licensee used the Cycle 10 core as the representative fuel cycle for power uprate and analyzed most of the transient events at the full uprated power and maximum allowed core flow operating point on the power/flow map. The licensee included direct or statistical allowance for 2-percent power uncertainty in the analysis. The Safety Limit MCPR (SLMCPR) was used to calculate the MCPR operating limits for the analyzed events. The licensee assumed no SRV will be out of service for each pertinent event. GE generically evaluated the effect of power uprate on the SLMCPR as documented in Reference 5.

GE analyzed the limiting events for each limiting transient category to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. GE used the results from these analyses in establishing the new licensing basis for transient analyses at uprated power. The power uprate will not change the basic characteristics of any of the limiting events.

The licensee analyzed applicable events and concluded that turbine/generator trip and feedwater controller failure are the limiting events that would cause the largest change in CPR and the MCPR operating limits. If an additional single failure such as a loss of RCIC or HPCI occurred during a loss of feedwater flow (LOFW) transient, the RCIC or the HPCI system would automatically maintain the reactor water level above the top of the active fuel (TAF) without any operator action. If both of these high-pressure systems failed, ADS would automatically initiate on low water level, and the low-pressure ECCS would automatically maintain water without any operator action. The operator would need to act (to control level, reduce pressure, and begin RHR shutdown cooling) only for long-term plant shutdown once water level is restored. The added heat from the power uprate would slightly increase the time required for the automatic systems to restore water level, and thus, the operator would have more time to plan and take manual actions. The sequences of events would not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient would not significantly change for power uprate.

3.7.2 Design Basis Accident

The staff reviewed (1) Reference 1, (2) Reference 2, and (3) the licensee's response of July 28, 1994, to a staff request for additional information (RAI) (Reference 27). The GE report described re-analyzed radiological consequences of DBAs resulting from the power uprate, and the licensee's response to the RAI described major parameters and assumptions of GE's radiological consequence analyses.

The licensee stated that it did the reconstituted analyses using a methodology described in the UFSAR with the original licensing basis assumptions at 3528 MWt (102 percent of the uprated power level) because the analyses could not be exactly reconstituted. The licensee's reconstituted analyses indicate the calculated offsite radiological consequence doses are within the dose reference values in 10 CFR Part 100 and meet the control room operator dose limit in GDC 19.

In August 1972, the staff did independent radiological consequence analyses of the plant at 3440 MWt (105 percent of current power level) (Reference 28). The staff expects offsite and control room operator doses to increase proportionally to the increase in power level. Therefore, the staff did not recalculate the offsite and control room operator doses resulting from a postulated design basis loss of coolant accident (controlling DBA). Instead, the staff proportionally increased the doses based on power levels using the licensing basis assumptions from the 1972 analyses and compared the results with the licensee's reconstituted calculation (See Table 1). The original licensing basis assumptions did not include (1) leakage through the main steamline isolation valve and (2) SGTS fission-product bypass during the reactor building pressure drawdown time after a DBA. Therefore, the staff and the licensee did not include in their analyses the radiation doses from either item.

TABLE 1

		EAB (rem)		LPZ (rem)		
		Thyroid	Whole Body	Thyroid	Whole Body	
SER	3440 MWt	14	1	105	3	(note 1)
	3528 MWt	14	1	108	3	(note 2)
UFSAR	3440 MWt	12.5	0.4	201	1.3	
	3528 MWt	14.8	0.6	239	3.9	
Part 100 Limits		300	25	300	25	

Note 1 Safety Evaluation for Peach Bottom Atomic Power Station Units 2 and 3 (August 1972)

Note 2 Uprated based on power ratio

The staff reviewed the major assumptions and methodology from the licensee's reconstituted dose calculations and the staff's original safety evaluation. The staff finds the offsite radiological consequences and control room operator doses at uprated 3528 MWt acceptable because they will remain below 10 CFR Part 100 dose reference values and GDC 19 dose limit.

3.7.3 Anticipated Transients Without Scram (ATWS)

General Electric has performed generic bounding ATWS analyses. The PBAPS parameter changes for power uprate are within the generic criteria.

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. However, no changes would be needed to the systems and equipment used to respond to an SBO and the required coping time would not change.

The power uprate will not affect the temperature response in the control room, cable spreading room, battery rooms, emergency switchgear room, HPSW/ESW pump room, and invertor areas. The temperature responses of the RCIC and HPCI equipment rooms are bounded because of conservatism in the existing calculation. Conservative assumptions in the existing containment analysis for SBO are bounding for uprate conditions. The systems which are used to respond after power is restored are designed for the uprated torus peak temperature. The licensee also determined that the evaluation of emergency diesel generator and Class 1E battery capacities following loss of power will be sufficient to maintain safe shutdown for uprated conditions.

The staff finds that operating the plant at uprated power will not significantly affect its response during an SBO event 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 vessel pressure and temperature would result in a small increase in the mass and energy release rates following high-energy line breaks (HELB). A break in a high-energy line outside the primary containment would cause the subcompartment pressure and temperature to increase only slightly, while causing a negligible change in the relative humidity. The licensee reviewed the HELB for the subject piping systems (main steam, feedwater, high-pressure coolant injection, reactor core isolation cooling, reactor water cleanup, and high-energy sampling and instrument sensing lines) and concluded that the resulting increases in the peak compartment pressure and temperature would be small and insignificant.

The licensee stated in its power uprate submittal that the existing pipe whip restraints, jet impingement shields, and their supporting structures are sufficient to minimize the effects of pipe whip and jet impingement from the postulated HELBs and will therefore be acceptable for the safe shutdown conditions at the uprated power.

The staff agrees that the analysis for high-energy line breaks submitted by the licensee indicates an acceptably small increase in the compartment temperature and pressure, and that existing structures 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 re-evaluated the equipment qualifications for both electrical and mechanical equipment and found that certain electrical equipment both inside and outside containment will be affected by the higher accident temperature and radiation levels resulting from the power uprate. The licensee committed to resolve the qualification of this equipment by refining radiation calculations for the specific location or by replacing specific equipment before making the uprate (Reference 2).

In analyzing the design qualification of mechanical components, the licensee recognized equipment or components in certain BOP systems that would be affected by the slight increases in temperature, pressure, and in some cases, flow resulting from operation at the uprated power level. In Reference 21, the licensee stated that it reviewed all equipment in the BOP systems affected by the power uprate to determine if they would operate acceptably at power uprate conditions. Systems primarily affected were the steam cycle systems such as main steam, extraction steam, feedwater, and condensate. In all cases, the as-designed and equipment capability bounds the marginal increases in system pressure, temperature, and flow, and all loads associated with the uprate.

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. LOCA dynamic loads and jet impingement will increase only 3 percent and will become negligible when combined with the governing seismic loads.

3. The original 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 staff accepts the licensee's evaluation of equipment qualification for the uprated power levels.

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 conduct the startup testing program in accordance with its procedures. By letter dated October 14, 1994, 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 guidleins of 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 potentially 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 potentially 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 29.

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 amendments. 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 have been prepared and published in the Federal Register on October 17, 1994, (59 FR 52317). 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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Date: October 18, 1994

7 REFERENCES

(1) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Operating License Change Request 93-12-0," June 23, 1993.

(2) NEDC-32183P, GE Nuclear Energy, "Power Rerate Safety Analysis Report For Peach Bottom 2 & 3," Class III, May 1993.

(3) GE Licensing Topical Report NEDC-31897P-1, "Generic Guidelines For General Electric Boiling Water Reactor (BWR) Power Uprate," June 1991. (Proprietary information. Not publicly available.) and GE Nuclear Energy, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

(4) Letter from W. T. Russell, NRC, to P. W. Marriotte, GE, "Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program (TAC NO. M79384)," September 30, 1991.

(5) GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements 1 & 2.

(6) NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)", December 30, 1988.

(7) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Response to Request for Additional Information Regarding Power Uprate Program (RAI-5)", July 6, 1994.

(8) NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)", June 15, 1988.

(9) ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1965 Edition with Addenda through Winter 1965.

(10) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information Regarding Power Uprate Program (RAI-4)," July 7, 1994.

(11) Letter from A. R. Blough, NRC to D. M. Smith, PECo, "Combined Inspection 50-277/90-14; 50-278/90-14," September 14, 1990.

(12) Letter from G. A. Hunger, Jr., PECo, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Response to Request for Additional Information Regarding Power Rerate Program (RAI-8), September 16, 1994.

(13) *ASME Boiler and Pressure Vessel Code*, Section III, Division 1, 1968 Edition with Winter 1970 Addenda.

(14) *United States of America Standard* (USAS) B31.1.0, "Power Piping," 1967 Edition.

(15) NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," September 1991.

(16) Letter from A. Thadani, NRC, to G. L. Sozzi, General Electric, "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993.

(17) NEDO-30832, "Elimination of Limits on Local Suppression Pool Temperature for SRV Discharge with Quenchers," General Electric Company, December 1984.

(18) NEDE-20566-P-A, "General Electric Model for LOCA Analysis In Accordance With 10 CFR 50 Appendix K," General Electric Company, September 1986.

(19) NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," General Electric Company, Revision 1, October 1984

(20) Regulatory Guide 1.52, "Design, Maintenance, Testing Criteria for Atmospheric Cleanup Air Filtration and Adsorption Unit of Light-Water-Cooled Nuclear Power Plants", Revision 2, March 1978.

(21) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information Regarding Power Rerate Request Dated March 9, 1994," April 5, 1994.

(22) Letter from S. Dembek, NRC, to G. A. Hunger, Jr., PECO, "Second Request For Additional Information Regarding Power Rerate Request, Peach Bottom Atomic Power Station, Units 2 and 3 (TAC NOS. M86826 and M86827)," March 29, 1994.

(23) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information Regarding Power Rerate Request Dated March 29, 1994 (RAI-2), May 2, 1994.

(24) GE Topical Report NEDC-31336P, "General Electric Instrument Setpoint Methodology," October 1986. (Proprietary information. Not publicly available.)

(25) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis

Reports for Nuclear Power Plants," July, 1981.

(26) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information Regarding Power Rerate Program (RAI-3)," June 6, 1994.

(27) Letter from G. A. Hunger, Jr., PECO, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information Regarding Power Rerate Program (RAI-7)," July 28, 1994.

(28) "Safety Evaluation of the Peach Bottom Atomic Power Station Units 2 & 3, Docket Nos: 50-277, 50-278" issued by the Atomic Energy Commission, August 11, 1972.

(29) Letter from W. T. Russell, NRC, to P. W. Marriott, General Electric Company, "Staff Safety Evaluation of General Electric Boiling Water Reactor Power Uprate Generic Analyses (TAC No. M81253)," July 31, 1992.