April 10, 2003

Mr. John L. Skolds, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - CORRECTION TO SAFETY EVALUATION FOR AMENDMENT NOS. 247 AND 250 (TAC NOS. MB5192 AND MB5193)

Dear Mr. Skolds:

On November 22, 2002, the Nuclear Regulatory Commission (NRC) issued Amendment Nos. 247 and 250 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consisted of changes to the Facility Operating License and Technical Specifications in response to your application dated May 24, 2002, as supplemented by letters dated June 27, September 11, September 24, and October 16, 2002. These amendments authorized an increase in the licensed power level from 3458 megawatts thermal (MWt) to 3514 MWt.

Your staff noted minor errors in the associated safety evaluation (SE). The corrections have been made in the enclosed SE (Enclosure 1)(pages 15, 20, 21, 30, and 31), and are denoted by vertical lines in the margin. These changes do not affect the conclusion of the SE. Also, we are requesting that you return the incorrect version of the SE. A stamped and addressed envelope is also enclosed (Enclosure 2) for your use. We apologize for any inconvenience this may have caused you.

Sincerely,

/RA/

John P. Boska, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures: 1. SE
2. Stamped and addressed envelope

cc w/encls: See next page

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OFFICE	PM/PD1-2	LA/PD1-2	OGC	SC/PD1-2
NAME	JBoska	MO'Brien	RWeisman	JClifford
DATE	4-2-03	4/8/03	3 April 2003	4/9/03

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 TO FACILITY OPERATING LICENSE NO. DPR-44 AND

AMENDMENT NO. 250 TO FACILITY OPERATING LICENSE NO. DPR-56

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By application dated May 24, 2002 (Reference 1), as supplemented by letters dated June 27, September 11, September 24, and October 16, 2002 (References 2 through 5, respectively), Exelon Generation Company, LLC (EGC or the licensee), submitted a request for changes to the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, Facility Operating License (FOL) and Technical Specifications (TSs). These amendments increase the licensed power level by approximately 1.62% from 3458 megawatts thermal (MWt) to 3514 MWt. The request is based on the installation of the Caldon LEFM✓™ ultrasonic flow measurement system with its ability to achieve increased accuracy in measuring reactor feedwater flow.

The supplemental letters dated June 27, September 11, September 24, and October 16, 2002, provided clarifying information that did not change the scope of the original *Federal Register* notice (67 FR 45568, published July 9, 2002) or the original no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

On June 1, 2000, a revision to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, “ECCS [Emergency Core Cooling System] Evaluation Models,” was issued, to be effective on July 31, 2000. The stated objective of this rulemaking was to reduce an unnecessarily burdensome regulatory requirement. Appendix K was originally issued to ensure an adequate performance margin of the ECCS in the event of the occurrence of a design basis loss-of-coolant accident (LOCA). The margin is provided by conservative features and requirements of the evaluation models and by the ECCS performance criteria. The original regulation did not require the power measurement uncertainty to be demonstrated, but rather mandated a 2% margin. The new rule allows licensees to justify a smaller margin for power measurement uncertainty. Because there will continue to be substantial conservatism in other Appendix K requirements, a sufficient margin to ECCS performance in the event of a LOCA will be preserved.

However, the final rule, by itself, did not allow increases in licensed power levels. Because the licensed power level for a plant is a licensed limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. Reference 1 includes a justification of the reduced power measurement uncertainty and the basis for the modified ECCS analysis.

PBAPS was originally licensed at 3293 MWt and was uprated by 5% to the current licensed thermal power (CLTP) level of 3458 MWt (References 6 and 6a). A 2% margin was added to the CLTP level of 3458 MWt in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR Part 50, Appendix K. Appendix K has since been revised to permit licensees to use an assumed power level for the analyses of less than 1.02 times the licensed power level, provided the assumed power level is demonstrated to account for uncertainties due to power level instrument error. This reduction in power measurement uncertainty does not constitute a significant change to the ECCS evaluation model as defined in 10 CFR 50.46(a)(3)(i). The analyses performed at 102% of the CLTP remain applicable at the proposed higher rated thermal power (RTP), because the new power level, combined with the new power measurement uncertainty, does not exceed 102% of the CLTP.

Attachment 2 (NEDC-33064P, Revision 0) of Reference 1 contains the plant-specific evaluation, written by General Electric Company (GE), for the proposed 1.62% power uprate. Attachment 1 (NEDC-33064P, Revision 1) of Reference 4 supersedes NEDC-33064P, Revision 0; but is substantially the same document except with regard to the treatment of proprietary information. By letter dated October 7, 2002, GE withdrew NEDC-33064P, Revision 0.

The licensee indicated that the GE evaluation follows the scope and content of GE licensing topical report NEDC-32938P (Proprietary), Thermal Power Optimization (TPO) licensing topical report (TLTR) (Reference 7), for up to 1.5% power uprate. Since Reference 7 is based on the generic guidelines and evaluations in the GE licensing topical reports ELTR1 and ELTR2 (References 8 and 9, respectively), which were reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) staff for extended power uprates in GE boiling water reactors (BWRs) of up to 120% of the original licensed thermal power, it can be reasonably referenced for the proposed 1.62% power uprate at PBAPS.

PBAPS has installed the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ (✓+™) System for feedwater (FW) flow measurement on Unit 2, and has scheduled installation for Unit 3 during the following refueling outage in September 2003. Use of the LEFM✓+™ System will reduce the calorimetric core thermal power measurement uncertainty to < ± 0.38%. Based on this, EGC is proposing to reduce the power measurement uncertainty required by 10 CFR Part 50, Appendix K, to permit an increase of 1.62% in the licensed power level. The reduction in power measurement uncertainty does not constitute a significant change to the ECCS evaluation model as defined in 10 CFR 50.46(a)(3)(i).

Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy While Increasing Power Level Using the LEFM System," (Reference 10) and its supplement, Engineering Report ER-157P, "Supplement to Caldon Topical ER-80P: Basis for a Power Uprate With the LEFM or LEFM CheckPlus™ System," Revision 5 (Reference 11), were approved by the NRC staff in March 1999 (Reference 10a) and December 2001 (Reference 11a), respectively. The plant-specific basis for the proposed uprate is provided in the applicable sections of the GE

Nuclear Energy topical report included in Reference 4 and in EGC's responses to NRC staff requests for additional information (RAIs) (References 3 and 5).

2.1 Applicable Regulatory Requirements/Criteria

The staff finds that the licensee, in Sections 3.0, 4.0, and 5.0 of Attachment 1 of Reference 1, identified the applicable regulatory requirements. The review and the basis for staff acceptance included the requirements of PBAPS Updated Final Safety Analysis Report (UFSAR), Section 1.5, "Principal Design Criteria;" proposed General Design Criteria (GDC) dated July 11, 1967 (refer to UFSAR Appendix H); 10 CFR Part 50, Appendix A, GDC-as applicable; 10 CFR Part 50, Appendix G; 10 CFR Part 50, Appendix H; 10 CFR Part 50, Appendix K; 10 CFR Part 50, Appendix R; 10 CFR 50.46; 10 CFR 50.48; 10 CFR 50.49; 10 CFR 50.55a; 10 CFR 50.62; 10 CFR 50.63; 10 CFR Part 100; 10 CFR 50.90; and 10 CFR 50.92 for no significant hazards consideration determinations and TSs. Additional guidance for the NRC staff's review is provided in Regulatory Guide (RG) 1.190 (Reference 12); RG 1.99, Revision 2 (Reference 13); the Standard Review Plan (SRP) (Reference 15); and Branch Technical Position MTEB 5-2, Revision 1 (Reference 15).

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in Sections 3.0, 4.0, and 5.0 of Attachment 1 to Reference 1. The detailed evaluation below will support the conclusions 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.

The NRC staff's review of the licensee's application is organized as follows:

3.1 Reactor - Core and Fuel Performance

The licensee submitted safety analysis report (SAR) NEDC-33064P (Attachment 1 to Reference 4) to support the proposed power uprate. The report evaluated the impact of the increased operating power on the facility's safety analyses and on the capabilities and performance of the nuclear steam supply system (NSSS) and its components. The power-dependent safety analyses, which are based on 102% of the current reactor thermal power, will remain applicable and bounding at the uprated condition; however, analyses and equipment or system qualifications performed at nominal power have to be reevaluated. The power uprate will be achieved by increasing the FW flow to produce higher steam flow from the reactor vessel and by adjusting the turbine control valve position to reduce the main steam (MS) line flow resistance.

NEDC-33064P follows the generic format and content of NEDC-32938P (Reference 7). NEDC-32938P is under staff review and is intended to be used for reference in future plant-specific TPO requests. Reference is made to the TLTR (NEDC-32938P) in several sections of the PBAPS plant-specific TPO report (NEDC-33064P), even though the TLTR report covers power uprates of only up to 1.5%. In response to an NRC staff question regarding the applicability of the TLTR to the 1.62% power uprate, EGC stated that every reference made to the TLTR in the PBAPS TPO SAR is valid. The methodology for the analysis of the PBAPS TPO addresses the following three approaches: (a) the existing analysis conducted at 102% or greater of current licensed thermal power (CLTP) is bounding for the TPO power uprate; (b) new plant-specific analysis was conducted; or (c) the generic analysis presented in the TLTR is applicable. A confirmation was made that the generic analysis at the 1.5% uprate was valid for PBAPS's 1.62% uprate.

The NRC staff finds it acceptable for EGC to refer to Reference 7 and believes that this justification is acceptable for the PBAPS TPO as discussed below.

PBAPS Unit 2 is currently operating in Cycle 15 and PBAPS Unit 3 is operating in Cycle 14 now. The TPO will be implemented during Cycle 15 for each unit and both units will utilize GE13 and GE14 fuel bundles. The PBAPS reload analysis is based on the NRC-approved GE methodology described in NEDE-24011-P-A-14 (GESTAR II) (Reference 16). The NRC-approved codes and methodologies used for the licensing safety analyses are also referred to in Section 5 of the PBAPS TSs. The limiting anticipated operational occurrence (AOO) and accident analyses are reanalyzed for every reload and the safety analyses are documented in Chapter 14 of the PBAPS UFSAR. Limiting AOOs and accidents are events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle in accordance with the NRC-approved GE design criteria, analytical models, and methods described in GESTAR II.

The following sections address the effect of the power uprate on fuel design performance, thermal limits, the power/flow map, and reactor stability.

3.1.1 Fuel Design and Operation

Fuel assemblies are designed to ensure that (1) they are not damaged during normal steady state operation and AOOs, (2) any damage would not be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during accidents is not underestimated, and (4) the coolability of the core is always maintained. For each fuel vendor, the NRC-approved fuel design acceptance criteria and analysis methodology assure that the fuel bundles comply with the objectives of Sections 4.2 and 4.3 of the SRP, and the applicable General Design Criteria (GDC) of 10 CFR Part 50, Appendix A. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

Since the uprated core for PBAPS Unit 2 will consist of 764 GE fuel bundles, the fuel design criteria are based on the NRC-approved methodology described in GESTAR II. A new mechanical fuel design is not needed to achieve the 1.62% power uprate, even though new fuel designs may be used in the future to obtain additional operating flexibility or to maintain the fuel cycle length. The current GE13 and GE14 fuel meets the NRC-approved acceptance criteria, and any new fuel designs that do not comply with the NRC-approved fuel design criteria given in GESTAR II will require NRC review and approval.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. Moreover, the licensee may change the power distribution in the reload design to achieve more operating flexibility or to maintain the fuel cycle length. This would also affect the core and fuel performance. However, the steady-state and transient design linear heat generation rate limits for each fuel bundle ensure that the fuel plastic strain design limit and the fuel centerline melt limit will not be exceeded. The thermal-hydraulic design and the operating limits will also ensure that the probability of boiling-transition fuel failures will not increase at the uprated conditions.

Upon introduction of any new fuel type, numerous evaluations are performed as part of the reload process. These evaluations not only confirm that the approved burnup limits are not exceeded, but also address all other impacts that this new fuel type may have on operation at the TPO power level, including impacts on stability, thermal-hydraulic compatibility, radiological analyses, and hydrogen generation. The licensee will follow the methods and processes described in the NRC-approved fuel vendor topical reports to perform these analyses and evaluations.

3.1.2 Thermal Limits Assessment

GDC-10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with an appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory limits and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The safety limit minimum critical power ratio (SLMCPR) protects 99.9% of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as a result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e., the 1% fuel plastic strain design limit or the no-fuel-centerline-melt criterion) will not be exceeded as a result of an AOO.

The SLMCPR is calculated for every reload at the RTP using NRC-approved methodologies. By letter dated June 10, 2002, Exelon submitted an application to amend the PBAPS Unit 2 TS Section 2.1.1.2 pertaining to the SLMCPR. The NRC staff has concluded that EGC's justification for analyzing and determining the SLMCPR value of 1.07 for two recirculation-loop operation and 1.09 for single recirculation-loop operation is acceptable for PBAPS Unit 2 Cycle 15 since approved methodologies are used (Reference 17).

The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis and this approach will not change. AOOs are analyzed at various points in the allowable operating domain, depending on the type of transient. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR, which ensures that 99% of the rods will not reach boiling transition in the event of an anticipated transient. The licensee will calculate the OLMCPR at the uprated condition for PBAPS.

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated cycle in the reload analysis for future cycles, and these limits will be maintained during operation.

The maximum average planar LHGR (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload the licensee confirms that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

Thus, the licensee will calculate the OLMCPR, the SLMCPR, the LHGR, and the MAPLGHR for the uprated conditions as part of the reload analysis using NRC-approved methodologies. It is expected that the licensee will make appropriate changes to the core operating limits report when implementing the power uprate.

3.1.3 Reactivity Characteristics

The reload core analysis will ensure that the minimum shutdown margin requirements will be met for each core design.

3.1.4 Stability

PBAPS is currently operating under the requirements of reactor stability interim corrective actions (ICAs). An evaluation was performed to determine the effect of TPO on the core stability ICAs per the guidelines of the TLTR. The instability exclusion region boundaries are unchanged with respect to absolute power level.

In June 2001, GE reported in a 10 CFR Part 21 report that licensees that implemented stability detect and suppress trip systems at their plants may be making nonconservative errors in their licensing calculations for reloads, resulting in inadequate MCPR safety limit protection. The Boiling Water Reactor Owners Group (BWROG) is in the process of resolving this generic issue. Since the ICA operation is conservative, PBAPS operation under ICAs is acceptable until the generic issue is resolved.

3.1.5 Reactivity Control

3.1.5.1 Control Rod Drives and Control Rod Drive Hydraulic System

The generic discussions in TLTR (Reference 7), Section 5.6.3, and Appendix J of Section 2.3.3 of the same report apply to PBAPS. The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram and rod insertion/withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulics and the reactor vessel bottom head pressure. EGC determined that following the power uprate, the CRD system is capable of performing its design functions of rapid rod insertion (scram) and rod positioning (insertion/withdrawal).

The NRC staff finds that the proposed power uprate will not have a significant impact on the operation of the CRD system for the following reasons:

(1) The operating dome pressure will not change, and the scram timing at steady-state power conditions will not be affected.

(2) There must be a minimum pressure differential of 250 psid between the hydraulic control unit and the vessel bottom head for normal CRD insertions and withdrawals. Since the operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the NRC staff finds that the CRD system will continue to perform all its safety-related functions at the proposed uprated conditions.

3.2 Reactor Coolant System and Connected Systems

3.2.1 Nuclear System Pressure Relief/American Society of Mechanical Engineers Boiler and Pressure Vessel Code Overpressure Protection

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The steam flow associated with the 1.62% power uprate can be regulated adequately by adjusting the turbine control valve position; therefore, the operating dome pressure will not increase, and the SRV setpoints and the number of valve actuation groups will not be changed.

Table 1-2 of NEDC-33064P provides the thermal-hydraulic parameters for the rated and the proposed uprated conditions. The table shows that, for a core flow of 112.75%, the steam flow rate increases by 1.62% for the uprated conditions. Since the SRVs will actuate at the current setpoints and the current American Society of Mechanical Engineers (ASME) overpressure protection analysis is based on operation at 102% power, the NRC staff accepts the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.2.2 Reactor Pressure Vessel and Internals

The licensee evaluated effects of the 1.62% PBAPS power uprate on the reactor vessel and internal components in accordance with its current design basis.

The effect of the TPO uprate was evaluated to ensure that the reactor pressure vessel (RPV) components comply with the existing structural requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. For the TPO uprate, the RPV design requirements are bounded by the previous PBAPS 5% power uprate. The current basis for the PBAPS 5% power uprate also bounds the normal and upset transient conditions for TPO operation. The component stress reports and design specification were reviewed and the licensee concluded that the current analysis is bounding in the saturated portions of the vessel. The TPO uprate does not change the emergency and faulted conditions for PBAPS. Therefore, the current assessment of the "original" Certified Stress Report applies to PBAPS for the TPO uprate.

The maximum flow rate is not changed for the TPO uprate; therefore, the reactor internal pressure differences (RIPDs) remain bounded by the current analyses for normal/upset and emergency/faulted evaluations that assumed an initial power level of 104.7% and 106.8% of the CLTP, correspondingly. In addition, since the faulted evaluations of the fuel bundle lift margins are bounded by the CLTP analyses, the fuel bundle lift margins are not calculated. The PBAPS licensing basis does not require the combination of hydraulic lift forces and seismic loads.

The reactor internal components were evaluated for structural integrity due to load changes associated with the TPO uprate. The TPO loads were either bounded by the design basis values or the changes were insignificant. Therefore, the reactor internal components remain qualified for the TPO uprate. No additional evaluation is necessary for the steam separator and dryer since the generic evaluation is applicable.

An RPV internal vibration evaluation was performed to determine the effects of flow-induced vibration on the reactor internals at TPO RTP and 110% rated core flow. The vibration levels for the TPO uprate conditions were estimated from vibration data recorded during startup testing of the NRC-designated prototype plant and during other tests. The calculations indicate that vibrations of all safety-related reactor internal components are within the GE acceptance criteria and, therefore, remain within acceptable limits. The safety-related MS and FW piping have increased flow velocities of 1.8% resulting from the TPO uprate. The increased vibration levels are approximately proportional to the square of the flow velocities and in proportion to the increase in fluid density.

Based on its review of the licensee's evaluation, the NRC staff finds that the reactor vessel, internals, and support structure remain in compliance with the existing structural requirements of the ASME B&PV Code at the TPO uprated conditions.

3.2.3 Reactor Vessel Fracture Toughness

In Section 3.2.1 of NEDC-33064, Revision 1, the licensee made two specific conclusions regarding the RPV integrity evaluations required by Appendix G to 10 CFR Part 50. The first conclusion, regarding the issue of RPV beltline material upper shelf energy, stated:

> "The upper shelf energy (USE) is bounded by the BWR Owners Group (BWROG) equivalent margins analysis, thereby demonstrating compliance with 10 CFR 50, Appendix G."

In this context, the analysis that the licensee refers to is that which was submitted by the BWROG in topical report NEDO-32205, Revision 1 (Reference 18). The NRC staff has previously reviewed NEDO-32205, Revision 1 and has approved of BWR licensees utilizing it as the basis for demonstrating that BWR RPV beltline materials maintain, per paragraph IV.A.1.a. of Appendix G to 10 CFR Part 50, "margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code." Based on PBAPS Units 2 and 3 RPV beltline material copper content and end-of-license neutron fluence values provided in Tables 3-1a and 3-2a of NEDC-33064P, Revision 1, the NRC staff was able to independently determine whether the PBAPS Units 2 and 3 RPV beltline materials continued to be bounded by the analyses in NEDO-32205, Revision 1. The NRC staff concluded that the PBAPS Units 2 and 3 RPV beltline materials do continue to be bounded by the NEDO-32205, Revision 1, report. Therefore, the NRC staff agrees with the licensee's conclusion that, with regard to USE issues, the PBAPS Units 2 and 3 beltline materials continue to meet the requirements specified in Appendix G to 10 CFR Part 50.

The second licensee conclusion, regarding the existing pressure-temperature (P-T) limit curves for PBAPS Units 2 and 3, in Section 3.2.1 of NEDC-33064P, Revision 1, stated:

> "The current Technical Specification pressure-temperature (P-T) curves for Unit 2 are non-beltline limited and remain non-beltline limited with TPO conditions up to 32 EFPY [effective full power years]. The current Technical Specification P-T curves for Unit 3 are non-beltline limited. Considering TPO conditions, the Unit 3 P-T curves become beltline limited at 22 EFPY and will require modification for operation beyond 22 EFPY."

Based on this information, the NRC staff informed the licensee that revised TS P-T limit curves for PBAPS Unit 3, which reflected their new limitation to 22 EFPY, had to be submitted as part of the licensee's power uprate submittal. The licensee submitted the revised PBAPS Unit 3 P-T limit curves in Reference 3 and indicated that the pressure vessel fluences were recalculated using the staff-approved method described in TLTR (Reference 7).

The NRC staff has confirmed that the fluence values cited for the PBAPS Units 2 and 3 RPVs in NEDC-33064P, Revision 1, and in Attachment 5 of Reference 3 (where the 1/4T fluence at 22 EFPY for the PBAPS Unit 3 limiting material, intermediate shell plate C2773-2, is given) are acceptable. Based on these fluence values, the staff agrees with the licensee's conclusion that the existing PBAPS Unit 2 P-T limit curves remain bounding through 32 EFPY of operation. The NRC staff also agrees with the licensee's conclusion that the PBAPS Unit 3 P-T limit curves in TS figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 should be limited in applicability to no more than 22 EFPY of operation based on an adjusted reference temperature of 79 °F for the

PBAPS Unit 3 RPV limiting material. Therefore, the NRC staff approves the revised PBAPS Unit 3 TS figures submitted in Attachment 2 of Reference 3, which reflect this limitation to no more than 22 EFPY of operation.

Based on the continued acceptability of the PBAPS Unit 2 P-T limit curves and the licensee's proposed revision of the PBAPS Unit 3 P-T limit curves, the staff concludes that the requirements specified in Section IV.A.2. of Appendix G to 10 CFR Part 50 have been met for the PBAPS Units 2 and 3 RPVs.

In Section 3.2.1 of NEDC-33064, Revision 1, the licensee also made a conclusion regarding the PBAPS Units 2 and 3 RPV material surveillance programs required by Appendix H to 10 CFR Part 50. The licensee stated:

> "The reactor vessel material surveillance program consists of three capsules for each unit. One capsule containing Charpy specimens was removed from the PBAPS Unit 2 vessel after 7.53 EFPY of operation; it was tested, reconstituted, and placed back into the vessel during the 2R08 Refueling Outage. One capsule containing Charpy specimens was removed from the PBAPS Unit 3 vessel after 7.57 EFPY of operation; it was tested, reconstituted, and placed back into the vessel during the 3R08 Refueling Outage. The remaining two capsules in each unit have been in their respective vessels since plant startup. PBAPS Units 2 and 3 are part of the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) and will comply with the withdrawal schedule specified for representative or surrogate surveillance capsules that now represent each unit. Therefore, the 10 CFR 50, Appendix H surveillance capsule schedule for the ISP will govern. Implementation of TPO has no effect on the BWRVIP withdrawal schedule."

The NRC staff disagrees with the licensee's statement that PBAPS Units 2 and 3 are currently part of the BWRVIP ISP. A license amendment request from EGC requesting that the BWRVIP ISP be approved for PBAPS Units 2 and 3 is, however, pending. Based on the fluence information provided, the NRC staff does not expect that any immediate modification to the BWRVIP ISP surveillance capsule withdrawal schedule will be required. If any modification to the BWRVIP ISP surveillance capsule withdrawal schedule is required in the longer term as a result of the recalculated PBAPS Units 2 and 3 fluences, such a change will be submitted by the BWRVIP as a modification to the ISP program for NRC staff approval. Therefore, the licensee's assessment of the compliance of PBAPS Units 2 and 3 with the requirements of Appendix H to 10 CFR Part 50 based on utilization of the ISP, as documented in Section 3.2.1 of NEDC-33064, Revision 1, is acceptable.

3.2.4 Reactor Coolant Piping and Components

3.2.4.1 Reactor Coolant Pressure Boundary Piping

The reactor coolant pressure boundary (RCPB) piping is composed of portions of various systems such as Recirculation, MS, FW, Residual Heat Removal (RHR), Low Pressure Coolant Injection (LPCI), High Pressure Coolant Injection (HPCI), Core Spray (CS), Reactor Water Cleanup (RWCU) and Standby Liquid Control (SLCS). The licensee summarized its evaluation of RCPB piping inside the containment in a table in Section 3.5.1 of NEDC-33064P. The licensee stated that the current licensing basis envelops the TPO uprate conditions with no nominal vessel dome pressure increase for the RCPB portion of all piping except for portions of the FW lines, MS lines and the MS attached piping. The FW, MS, and MS attached piping systems inside the containment were evaluated for compliance with ASME Code, Section III, and American National Standards Institute (ANSI) B31.1 stress criteria. The current licensing basis for these piping systems (inside containment), analyzing for pressure, temperature, and flow, envelops the TPO uprate conditions. Therefore, the NRC staff finds acceptable the licensee's conclusion that all safety aspects of these piping systems (inside containment) are within current licensing basis evaluations.

3.2.4.2 Balance-of-Plant Piping

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design basis conditions with those for the proposed power uprate. The BOP piping systems that are affected were determined from the uprated reactor and BOP heat balances. These systems include all piping, pipe supports and anchorages throughout PBAPS, except for piping evaluated above as RCPB piping.

The Design Basis Accident (DBA) LOCA dynamic loads were originally defined based on analyses at 102% CLTP. For the TPO conditions, the DBA LOCA containment response loads applied to piping systems do not change. In addition, there are no changes in the SRV hydrodynamic loads because the SRV opening setpoints do not increase for the TPO uprate. The TPO uprate may result in a slight reduction in the time between SRV actuations affecting the SRV discharge line water level at the subsequent actuations. However, the TPO uprate has no effect on the maximum reflood height; therefore, the original subsequent actuation SRV load definition is still bounding. The NRC staff concurs that the dynamic loads applicable to BOP piping do not change.

There is no change to pipe stresses due to the TPO uprate since design parameters such as pressure, dead weight, seismic, hydrodynamic load, piping temperature, fluid transient loads, operating temperatures, pipe break loads, and postulated pipe break locations parameters are not changed. Support loads are not affected by the TPO uprate since there are no changes in piping reaction loads, room/ambient temperatures, pipe break loads and postulated pipe break locations.

Small bore piping and supports are not affected by the TPO uprate because they have been conservatively analyzed and the analysis encompasses the TPO condition, and loads/ movements of the large bore piping to which they are attached are not changed. The carbon steel piping can be affected by Flow Accelerated Corrosion (FAC). The changes in fluid velocity, temperature, and moisture content can impact FAC. Operation at the TPO uprate results in some changes to parameters affecting FAC in those systems associated with the turbine cycle. PBAPS has an established program, using the CHECKWORKS™ software, for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. The program is in compliance with NRC Generic Letter (GL) 89-08 (Reference 19). Continued monitoring of the systems provides confidence in the integrity of high energy piping systems susceptible to FAC.

Based on the above discussion, the staff concurs with the licensee's conclusion that the dynamic LOCA loads would not change the calculated pipe stresses, pipe supports and small bore piping/supports at the proposed 1.62% power uprate levels.

3.2.4.3 Safety-Related Valves, Motor-Operated Valves, and Air-Operated Valves

The licensee evaluated the containment system performance for SRV actuations. Since the maximum operating reactor dome pressure remains unchanged for the TPO uprate, the licensee concluded that the SRV setpoints and analytical limits are not affected by the proposed TPO uprate. There are no changes in the SRV hydrodynamic loads. The TPO uprate may result in a slight reduction in the time interval between SRV actuations affecting the SRV discharge line water level at the subsequent actuations. However, the TPO uprate has no effect on the maximum reflood height; therefore, the original subsequent actuation SRV load definition is still bounding and the SRV loads for the SRV discharge line piping will remain unchanged. The staff agrees with the licensee's conclusion that the SRVs and the SRV discharge piping will continue to maintain their structural integrity and provide sufficient over-pressure protection to accommodate the proposed TPO uprate.

The licensee reviewed the motor-operated valve (MOV) requirements in the PBAPS UFSAR against the functional requirements of GL 89-10 (Reference 20). No changes are identified as a result of operating at TPO uprate. The evaluation considered the effect of bounding system pressure and flow rate increases on the capability of the MOVs to perform their safety function except on four occasions. The licensee reviewed the safety-related MOVs in these exceptions and found that no safety-related MOVs are affected in these piping runs. Therefore, the TPO uprate has no effect on the MOV program at PBAPS.

The licensee reviewed its response to GL 96-06 (Reference 21) for the TPO uprate and concluded that the PBAPS response to GL 96-06 remains valid under the TPO uprate conditions. The licensee also reviewed its commitments relating to GL 95-07 (Reference 22), and no changes were identified as a result of operating at TPO uprate conditions. The valves remain capable of performing their design basis function.

Based on its review of the licensee's programs in response to GL 89-10 and GL 95-07, the staff finds the licensee's evaluation of the effect of the proposed power uprate on the capability of safety-related SRVs, MOVs, and air-operated valves (AOVs) at PBAPS to be acceptable. The licensee also concluded that the proposed power uprate has no impact on its evaluation in response to GL 96-06 in regards to potential over-pressurization of isolated piping segments for PBAPS. Based on its review of the information submitted by the licensee describing the scope, extent, and results of the evaluation, the NRC staff concurs with the licensee's evaluation and conclusion.

3.2.4.4 Flow-Accelerated Corrosion in Piping

The MS and associated piping systems and FW system piping are made of carbon steel, which can be affected by FAC. Changes in fluid velocity, temperature, and moisture content can impact FAC. The licensee has established a program for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. This continuing inspection program allows the licensee to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. The licensee stated that this program provides assurance that the

TPO uprate will not have any adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to erosion/corrosion.

The licensee performed a comparison of the changes in wear rates from the current operating conditions to the new operating conditions. The model used for this study was the CHECWORKS FAC model. The modeled lines evaluated were the Condensate, FW, FW Heater Drains, Moisture Separator Drains, FW Minimum Flow Recirculation and Main Steam Bypass Valve Discharge. The licensee's analysis showed the increases in wear rates were less than 10%. These increases correspond to less than 0.001 inches/year.

The NRC staff finds the licensee evaluation and reasoning to be acceptable and, therefore, concludes that the proposed 1.62% TPO uprate for PBAPS Units 2 and 3 will not have a significant impact on FAC.

3.2.5 Reactor Recirculation System

The reactor recirculation system evaluation described in TLTR (Reference 7) Section 5.6.2 of applies to PBAPS.

The power uprate will be accomplished by operating along extensions of the rod and core flow lines on the power/flow map. PBAPS is currently licensed to operate at up to a maximum core flow of 110% of the rated flow or 112.75 Mlb/hr. The power uprate does not require an increase in the maximum allowable core flow. Therefore, the reactor recirculation flow will be maintained within the flow limits of the existing power/flow map, with 100% power corresponding to the power uprate level. The cycle-specific reload analysis will consider the full range of the power and flow operating region.

The NRC staff concludes that the changes associated with the 1.62% power uprate will have an insignificant impact on the function of the recirculation system.

3.2.6 Main Steam Isolation Valves and Main Steamline Flow Restrictors

The generic discussions in TLTR (Reference 7), Appendix J apply to PBAPS. The MS isolation valves (MSIVs) are part of the RCPB and must be able to close within specific limits at all design and operating conditions upon receipt of a closure signal. The licensee states that the requirements for the MSIVs remain unchanged for the 1.62% power uprate and that all safety and operational aspects of the MSIVs are within previous evaluations. Regarding the main steamline flow restrictors, the licensee states that the requirements remain unchanged for the power uprate because no change in steam break flow occurs (since the operating pressure is unchanged), and that all safety and operational aspects of the flow restrictors are within previous evaluations.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant operations at the 1.62% power uprate condition will have an insignificant impact on the ability of the MSIVs and main steamline flow restrictors to meet their design objectives.

3.2.7 Reactor Core Isolation Cooling System

The generic discussion provided in TLTR (Reference 7) Section 5.6.7 of the reactor core isolation cooling (RCIC) system is applicable to PBAPS.

The RCIC system provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures from 150 psig to the maximum pressure corresponding to the lowest opening setpoint for the SRVs. In particular, the loss-of-feedwater (LOFW) flow transient assumes that the RCIC system will maintain sufficient water level inside the core shroud to ensure that the top of the active fuel will be covered throughout the event. The transient analysis also assumes that the low-setpoint SRVs would remove the stored and decay heat since MSIV closure on low water level isolates the reactor from the main condenser. The transient is a power-dependent transient and is more severe at a higher initial power since there is more stored energy and decay heat to be dissipated and the water level drops faster.

The LOFW analysis described in NEDC-31984P (Reference 23) is applicable to PBAPS. Since the proposed 1.62% power uprate does not increase the steady-state operating pressure or the SRV actuation setpoints, the staff accepts that the RCIC performance would not be affected.

3.2.8 Residual Heat Removal System

The generic discussion provided in TLTR (Reference 7) Section 5.6.4, and Appendix J, Section 2.3.13, of the same report, is applicable to PBAPS.

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel, and to provide primary-system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the shutdown cooling mode, the suppression pool cooling mode, and the containment spray cooling mode.

The slightly higher decay heat has negligible effect on the operation of the RHR system in the shutdown cooling mode.

3.2.9 Reactor Water Cleanup System

The generic discussions in TLTR (Reference 7), Section 5.6.6, and Appendix J of Section 2.3.4 of the same report apply to PBAPS. The primary parameters that affect the RWCU system are power transients, RWCU operating temperature and pressure, recirculation flow temperature, and system impurities such as fission and corrosion products. Power transients are the primary challenge to the RWCU system and are independent of the power uprate. The licensee stated that the power uprate conditions will not significantly affect the remaining parameters.

On the basis of the information the licensee provided, the NRC staff concludes that the proposed power uprate is acceptable with respect to the RWCU system because it will not significantly affect the water chemistry performance of the reactor and, therefore, will not significantly affect the performance requirements of the RWCU system.

3.3 Engineered Safety Features

3.3.1 Containment System Performance

The containment system is designed to prevent the release of fission products to the environment in excess of that specified in 10 CFR Part 100, in the event of a design-basis accident. Reference 1 states that the previous containment evaluations are bounding for the 1.62% power uprate because they were performed at greater than or equal to 102% of the current licensed thermal power. Although the nominal operating conditions increase slightly because of the power uprate, the required initial conditions for containment analysis inputs remain the same.

The licensee's review included the short-term pressure and temperature response of the containment, the long-term temperature response of the suppression pool, the containment dynamic loads, and containment isolation.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the containment system performance will not be affected by the 1.62% power uprate.

3.3.2 Emergency Core Cooling Systems

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary-system piping. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA analysis, and the licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for PBAPS includes the high-pressure coolant injection (HPCI) system, the low-pressure coolant injection (LPCI) mode of the RHR system, the low-pressure core spray (LPCS) system, and the automatic depressurization system (ADS).

3.3.2.1 High-Pressure Coolant Injection System

The HPCI system (with other ECCS systems as backups) is designed to maintain reactor water inventory during small- and intermediate-break LOCAs, isolation transients, and LOFW events. The HPCI system is designed to pump water into the reactor vessel over a wide range of reactor operating pressures. The HPCI system also serves as a backup to the RCIC system. The system is designed to operate from battery power backed up by normal offsite auxiliary power or by an emergency diesel generator.

The HPCI system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients, the RCIC maintains water level above the top of the active fuel (TAF). For the MSIV closure, the SRVs open and close as required to control pressure and the HPCI system eventually restores water level.

The licensee evaluated the capability of the HPCI system during operation at the TPO power level to provide core cooling to the reactor to prevent excessive fuel peak cladding temperature (PCT) following small- and intermediate-break LOCAs and keep the core covered up to the TAF in isolation transients and LOFW events. The HPCI evaluation is applicable to and consistent with the evaluation in TLTR (Reference 7) Section 5.6.7. The maximum reactor pressure at which the HPCI system must be capable of injecting into the vessel for the RCIC backup function was selected based on the upper analytical values for the second lowest group of SRVs. The TPO does not decrease the net positive suction head (NPSH) available for the HPCI pump or increase the required NPSH.

The licensee evaluated the capability of the HPCI system to perform as designed and analyzed its performance at the TPO conditions. The licensee determined that the HPCI system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCAs and isolation transients. Since the licensee's ECCS-LOCA analysis is based on the current HPCI capability (as discussed in Section 3.3.3 below) and demonstrates that the system provides adequate core cooling, the NRC staff finds the analysis acceptable.

3.3.2.2 Core Spray System

The core spray system evaluation is applicable to and consistent with the evaluation in TLTR (Reference 7) Section 5.6.10.

The core spray (CS) system initiates automatically in the event of a LOCA. In conjunction with other ECCS systems, the CS system provides adequate core cooling for all LOCA events. The system also provides spray cooling for long-term core cooling after a LOCA. The licensee explained that the existing CS system hardware has the capability to perform its design function at the TPO conditions and that the generic evaluation in Section 5.6.10 of the TLTR is applicable to PBAPS. Since ECCS-LOCA analysis demonstrates that the system provides adequate core cooling, the NRC staff concludes that the CS is acceptable for TPO operation.

3.3.2.3 Low-Pressure Coolant Injection System

The low-pressure coolant injection (LPCI) system evaluation is applicable to and consistent with the evaluation in TLTR (Reference 7) Section 5.6.10.

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. In conjunction with other ECCS systems, the LPCI mode is used to provide adequate core cooling for all LOCA events. The licensee stated that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the TPO conditions.

Since the licensee's ECCS-LOCA analysis for TPO operation is based on the current LPCI capability (as discussed in Section 3.3.3 below) and demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable.

3.3.2.4 Automatic Depressurization System

The ADS uses the safety/relief valves (SRVs) to reduce reactor pressure after a small-break LOCA with HPCI failure, allowing LPCI and CS to provide cooling flow to the vessel. The plant design requires SRVs to have a minimum flow capacity. After a delay, the ADS actuates either on low water level plus high drywell pressure or on low water level alone. The licensee stated that the ADS's ability to perform these functions is not affected by the power uprate, and the NRC staff agrees. Since the small-break LOCA analyses assume that the ADS actuates at a bounding vessel pressure and power, the staff accepts the licensee's assessment that the current power uprate does not affect the capability of the ADS to perform its function.

The LOCA analyses of record demonstrate that the HPCI system, the LPCI mode of RHR, the CS system, and the ADS have the capabilities to provide core cooling during a LOCA. These capabilities do not change for operation at the uprated conditions. Therefore, the ECCS will continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated conditions.

Since the LOCA analysis is based on an NRC-approved methodology and codes and the assumed power is bounding, the NRC staff accepts the licensee's assessment that the ECCS will perform as designed and analyzed at the uprated conditions.

3.3.3 Emergency Core Cooling System Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The licensee stated that the ECCS performance satisfies these requirements under all LOCA conditions. The GE fuel was analyzed with GE's NRC-approved SAFER/GESTR model (References 24 and 25). These analyses were performed at 102% of current licensed power level and remain applicable at the proposed higher power level. The analyses yielded PCTs less than 1645 °F for GE13 fuel and less than 1450 °F for GE14 fuel, peak metal-water reactions less than 1%, and core-wide metal-water reactions less than 0.1% for both fuel types. These results comply with the 10 CFR 50.46 requirements of PCT of less than 2200 °F, cladding oxidation less than 17%, and core wide metal water reaction less than 1%. The NRC staff accepts EGC's ECCS performance evaluation because the analytical models and codes are based on the NRC-approved methodology described in GESTAR II and the ECCS-LOCA analyses are based on bounding power and flow conditions.

3.3.4 Main Control Room Atmospheric Control System

The main control room atmospheric control system minimizes unfiltered in-leakage following a design-basis accident. Habitability (including control room operator doses) following a postulated accident from the 1.62% power uprate condition is unchanged because the main control room atmospheric control system had previously been evaluated for accident conditions from 102% of the CLTP. This evaluation is bounding for the proposed 1.62% power uprate.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the licensee's existing analysis for the main control room atmospheric control system is bounding for the 1.62% power uprate.

3.3.5 Standby Gas Treatment System

The standby gas treatment system (SGTS) minimizes the offsite and control room doses during venting and purging of the containment atmosphere under abnormal conditions. The current capacity of this system was selected to maintain the secondary containment at a slightly negative pressure under such conditions. The charcoal beds in this system can accommodate DBA conditions at 102% of the current licensed thermal power.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the licensee's existing analysis for the SGTS remains valid for the 1.62% power uprate.

3.3.6 Post Loss-of-Coolant Accident Combustible Gas Control System

Hydrogen recombiners are used following a LOCA to maintain containment atmosphere hydrogen levels below combustible levels. The metal available for reaction is unchanged by the 1.62% power uprate, and the hydrogen production assumed due to radiolytic decomposition is unchanged because the system was previously evaluated for accident conditions from 102% of the CLTP.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the licensee's existing analysis bounds the 1.62% power uprate, and the impact on the hydrogen recombiners is negligible.

3.4 Instrumentation and Controls

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system. This calculation is called a "heat balance." The accuracy of this calculation depends primarily upon the accuracy of FW flow, FW enthalpy, and MS enthalpy measurements. Thus, an accurate measurement of FW flow and temperature will result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring the FW flow rate typically consists of a venturi, an orifice plate, or a flow nozzle to generate a differential pressure proportional to the FW velocity in the pipe. Typically, the FW temperature is measured using resistance temperature detectors mounted in the pipe. The major disadvantage of the venturi flow meter is the effect of venturi fouling upon flow meter instrument accuracy. Fouling causes a venturi flow meter to indicate higher differential pressures for equivalent flow velocities, which results in an output signal representing a higher than actual flow rate. Since FW flow rate is directly proportional to calorimetric power, this error in FW flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher than actual core power. This causes the licensee to generate proportionately less electrical power when the plant is operated at its indicated thermal power rating.

The use of the transit time methodology with ultrasonic pulse transmission in multiple acoustic paths across pipe cross sections, as utilized by the Caldon LEFM✓+™ System technology, improves the accuracy of the measurement of FW flow and reduces the uncertainty of the flow measurement.

EGC stated that the Caldon LEFM✓+™ System is designed and manufactured in accordance with the Caldon 10 CFR Part 50, Appendix B, quality assurance program, and the system software and laboratory calibration tests are required to meet 10 CFR Part 50, Appendix B, requirements. The system software was developed under the Caldon Verification and Validation (V&V) program, which meets the criteria of ANSI/Institute of Electrical and Electronic Engineers (IEEE) Standard 7-4.3.2, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Generating Stations," and ASME Standard NQA-2A-1990, "Quality Assurance Requirement for Nuclear Facility Applications." The V&V program is consistent with the guidance of Electric Power Research Institute's Topical Report TR-103291S, "Handbook for Verification and Validation of Digital Systems," and includes requirements for user notification of important deficiencies. All conditions adverse to quality are handled in accordance with the Exelon corrective action program. The licensee also stated that the Caldon LEFM✓+™ System software will be controlled under the PBAPS software quality assurance program, which provides for appropriate vendor notification and error reporting.

In approving Caldon Topical Report ER-80P, the NRC staff included four additional requirements to be addressed by a licensee requesting a power uprate. The licensee's submittal addressed each of the four requirements as follows:

1. The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM✓+™. These procedures should include processes and contingencies for an inoperable LEFM✓+™ and the effect on thermal power measurement and plant operation.

   The licensee stated that implementation of the power uprate amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the power uprate level with the new LEFM✓+™ system. Plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM✓+™ system operable and raising power above 3458 MWt.

   The LEFM✓+™ system features automatic self-checking and a continuously operating on-line test verifies that the digital circuits are operating correctly and within the specified accuracy envelope. If the LEFM✓+™ system becomes inoperable, the control room operators are promptly alerted by control room indications.

   The licensee has developed contingency plans for operation of the plant with an LEFM✓+™ out-of-service, and requested an allowable outage time (AOT) of 72 hours to enact LEFM✓+™ system repairs. During the 72-hour AOT, the licensee proposed that the thermal power will be maintained at 3514 MWt, provided steady state conditions persist during this period. If the LEFM✓+™ is not restored in 72 hours, reactor power will be reduced to the CLTP of 3458 MWt.

The feedwater venturis are calibrated during normal operations using the flow measurement values provided by the LEFM✓+™. If the LEFM✓+™ system becomes inoperable, the feedwater venturi measurements will be used in the plant heat balance calculations.

The licensee calculated the time-dependent effect of feedwater venturi measurement uncertainty on the plant heat balance calculation. In this calculation, the licensee assumed the LEFM✓+™ system was inoperable for 1 year, and calculated the change in feedwater venturi measurement accuracy for that period. The licensee then used the corresponding 1-year value to determine the change in feedwater venturi measurement accuracy for a 72-hour period. The licensee determined that the decrease in feedwater venturi measurement accuracy would result in a maximum flow error of 0.0107%, resulting in a maximum error in core power measurement of less than 0.4 MW during the 72-hour AOT. This error is within the margin of the core power measurement uncertainty calculations for PBAPS. The licensee concluded that there is no overall plant risk impact of continued operation at 3514 MWt as a result of the change in feedwater venturi measurement uncertainty during the 72-hour AOT. The NRC staff finds that the licensee's justification for the 72-hour AOT is consistent with NRC Regulatory Issue Summary (RIS) 2002-03 and NRC regulations and, therefore, is acceptable.

2. For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM✓+™ system and bounds the analysis and assumptions set forth in topical report ER-80P.

   The licensee stated that this criterion is not applicable because at the time of their application, there were no LEFMs installed at PBAPS.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM✓+™ in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

   The licensee stated that the combination of errors within instrument loops is accomplished in accordance with NRC-approved GE Setpoint Methodology as described in NEDC-31336, Class III (Reference 26). The NRC staff reviewed the uncertainty analysis presented in NEDC-33064P, Revision 1, and found the calculation methodology and results to be in accordance with approved setpoint uncertainty guidelines.

4. Licensees for plant installations where the ultrasonic meter (including the LEFM✓+™) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant- specific installation), should provide additional justification for use. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and

the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM✓+™ installation and calibration assumptions.

The licensee stated that this criterion is not applicable to PBAPS. The calibration factor for the PBAPS spool pieces will be established by tests at Alden Research Laboratory. These tests will include tests of a full-scale model of the PBAPS hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon Engineering report evaluating the test data will be maintained by the licensee. The calibration factor used for the LEFM ✓+™ at PBAPS will be based on these two reports.

The licensee stated that its final acceptance of the site-specific uncertainty analyses will occur after the commissioning process. The bounding calibration test data obtained from the Alden tests, and confirmation that the actual performance in the field meets the uncertainty bounds established for the instrumentation (see licensee's response to criterion 3 above) will be verified during the commissioning process. Final commissioning was planned for October 2002 for Unit 2 and October 2003 for Unit 3. The staff finds the licensee's commitment for verifying the calibration test data and confirming the uncertainty bounds to be an appropriate commitment for ensuring the power uprate margin will be maintained consistent with the margin addressed in this safety evaluation.

The NRC staff finds that the licensee's responses sufficiently resolve the plant-specific concerns regarding maintenance and calibration of the LEFM✓+™ system and other instrumentation affecting heat balance, hydraulic configuration of the installed LEFM✓+™, processes and contingencies for an inoperable LEFM✓+™, and methodology for calculating the LEFM✓+™ and plant core power measurement uncertainties.

3.5 Electrical Systems

As described in Reference 1, the main generator for each PBAPS unit is rated at 1280 megavolt amperes (MVA) at a 0.9 power factor. The generator output is generated at 22 kV, is fed through an isolated phase bus to the primary windings of the 1206 MVA main power transformer where it is stepped up to 500 kV voltage. The unit auxiliary transformer (UAT) supplies power to balance of plant (BOP) systems under normal operating conditions. Upon a trip of the main generator, the station auxiliaries are automatically transferred to the preferred offsite power source through the startup and emergency transformers to assure continued power to equipment when the main generator is off-line. The station distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions. The electrical distribution system has been previously evaluated to conform to GDC-17. Also, the plant has been previously evaluated for environmental qualification (EQ) of electrical equipment, 10 CFR 50.49, and station blackout (SBO), 10 CFR 50.63.

The following is the NRC staff's power uprate evaluation of grid stability, main generator, transformers, emergency diesel generators, SBO, and EQ.

3.5.1 Grid Stability

The current generator for each unit is rated at 1159 megawatts electric (MWe) at 0.906 power factor. The uprated analysis evaluated the generator capability at 1240 MVA at 0.95 power factor (1178 MWe). The licensee completed the feasibility study for grid stability on January 31, 2002. The results of the study showed that there is no impact on the grid stability due to this power uprate.

The NRC staff reviewed the licensee's submittal and concluded that there is no impact of the power uprate on the grid stability. Therefore, the plant continues to meet GDC-17 for grid stability with this power uprate.

3.5.2 Main Generator

The current generator for each unit is rated at 1159 MWe at 0.906 power factor. An assessment of the capability of the main generator to support an uprate of 1.62% was performed by GE. Reference 4 concluded that the main generator can achieve the power uprate of 1178 MWe without hardware modifications. The main generator and associated cooling equipment are designed to accept the maximum generator output at the uprated condition. There is no impact of the power uprate on the protective relay settings of the main generator.

The NRC staff reviewed the licensee's submittal and concluded that the main generator can achieve the power uprate without hardware modifications and, therefore, operating the main generator at the uprated power condition is acceptable.

3.5.3 Main Transformer

The main transformer for each unit is rated at 1206 MVA. The licensee performed the review of the main power transformer and associated switchyard components (rated for maximum transformer output). The loading on the main transformer is 1178.9 MVA (main generator output of 1178 MWe minus the 58 MWe house load fed through the unit auxiliary transformers at 0.95 power factor) which is below the main transformer rating of 1206 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the main power transformer and the associated switchyard components (rated for maximum transformer output) are adequate for the uprated generator output and, therefore, operating the main transformer at the uprated power condition is acceptable.

3.5.4 Isophase Bus

The isophase bus duct for each unit connects the main generator to the primary windings of the main transformer and the unit auxiliary transformer and is rated at 35,300 amperes.

The NRC staff reviewed the licensee's submittal and concluded that the power uprate load of 1178 MWe is below the design rating of the isophase bus and, therefore, operating the isophase bus at the uprated power condition is acceptable.

3.5.5 Unit Auxiliary Transformers

The unit auxiliary transformer (UAT) is rated at 45.4 MVA. It typically operates with approximately a 30% margin with the unit at full power. Since there is no significant additional loading on the plant auxiliary systems, the UAT is not impacted by the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the increase in house loads resulting from the power uprate is below the maximum UAT design rating and; therefore, operating the UAT at the uprated power condition is acceptable.

3.5.6 Start-up Transformers

The start-up transformers are rated at 50 MVA each. Since there is no significant additional loading on the plant auxiliary systems, they are not impacted by the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the start-up transformers are not impacted by the power uprate; and, therefore, operating the start-up transformers at the power uprate condition is acceptable.

3.5.7 Emergency Diesel Generators

The emergency diesel generators (EDGs) supply the source of power following a loss of offsite power or degraded voltage conditions. The EDGs automatically supply ac power to the safety related (Class 1E) buses in order to provide motive and control power to equipment required for safe shutdown of the plant and mitigation and control of accidents. There are no modifications associated with the power uprate that would increase the electrical loads associated with the engineered safeguard and selected non-safeguard systems or alter the diesel generator subsystems. Therefore, the performance of the EDG and the 4 kV emergency system is not affected by the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the power uprate does not affect the loading on the EDG; and, therefore, the licensee will continue to meet the GDC-17 requirements with the power uprate.

3.5.8 Environmental Qualification of Electrical Equipment

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive the environment at its specific location during normal and accident operating conditions. Environmental qualification for safety-related electrical equipment located inside the containment is based on MS line break (MSLB) and/or LOCA conditions and their resultant temperature, pressure, humidity and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature and pressure are based on the analyzed conditions initiated from $\geq$ 102% of the CLTP. Normal temperatures may increase slightly near the FW and reactor recirculation lines and will be evaluated through the EQ temperature monitoring program, which tracks such information for equipment aging considerations. The current radiation levels under normal plant conditions also increase slightly. The current plant environmental envelope for radiation is not exceeded by the changes resulting from the power uprate.

Accident temperature, pressure, and humidity environments used for the qualification of equipment outside containment result from an MSLB in the pipe tunnel, or other high energy line breaks (HELB). Some of the HELB pressure and temperature profiles increase by a small amount due to the uprate conditions. The NRC staff requested that the licensee describe the effect of the decrease in margin on leak detection trip avoidance, and the effect of this decreased margin on associated analytical limits (ALs). In Reference 3, the licensee showed that there is sufficient margin between the 192 °F temperature setpoint for leak detection and the normal MS tunnel temperature (140 to 150 °F), such that an increase in the MS tunnel temperature caused by the expected increase in FW temperature would not affect the current setpoint margin. The AL is not affected by the expected increase in MS tunnel temperature caused by an increase in the FW temperature. The staff finds that there is adequate margin in the qualification envelopes to accommodate the small changes.

The NRC staff reviewed the licensee's submittal and concluded that the EQ program bounds the 1.62% power uprate and the plant would meet the requirements of 10 CFR 50.49. Based on these considerations, the program is acceptable.

3.6 Auxiliary Systems

3.6.1 Fuel Pool - Cooling and Design

The fuel pool cooling and cleanup system (FPCCS) removes heat from the spent fuel assemblies stored in the spent fuel pool (SFP) in order to maintain the pool temperature at or below its design temperature during normal plant operations. In addition, the FPCCS reduces activity, maintains water clarity, and maintains the cooling function during and after a seismic event.

The fuel pool heat load increases slightly as a result of the power uprate. However, the new heat load is within the design basis heat load for the FPCCS, and it will not result in a delay in removing the RHR system from service (i.e., the duration of supplemental cooling will not be increased).

Regarding other fuel pool design considerations, the crud activity and corrosion products in the SFP can increase slightly; however, this increase is insignificant and the water quality will be maintained by the FPCCS. In addition, the normal radiation levels around the SFP may increase slightly; however, the increase will not significantly increase the operational doses to personnel or equipment. Also, there is no effect on the design of the spent fuel racks because the original SFP design temperature is not exceeded.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the FPCCS, in combination with the RHR system, can maintain the SFP at or below design limits for all core offload conditions at the proposed 1.62% power uprate level.

3.6.2 Water Systems

The safety-related standby service water (SSW) systems provide cooling to the RHR heat exchangers, diesel generators, and ECCS equipment during and following a design-basis accident. The heat loads generated by the diesel generators and the ECCS equipment are not affected by the power uprate. The 1.62% power uprate will increase the heat loads on the RHR heat exchangers and room coolers due to the increase in suppression pool temperature; however, the increased heat loads are acceptable since the containment response analysis was based on a core power level of ≥ 102% of the CLTP.

Regarding nonsafety-related heat loads, the plant service water (SW) systems are designed to cool plant auxiliary equipment during normal operating and normal shutdown conditions. The 1.62% power uprate will result in slight increases in heat load from the turbine building closed cooling water system and the reactor building closed cooling water (CCW) system; however, the increase in system demand and heat loads from these sources is within the design of the plant SW systems.

The main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and maintain a low condenser pressure. The 1.62% power uprate increases the heat rejected to the condenser and may reduce the difference between the operating condenser vacuum and the required minimum condenser vacuum; however, the licensee's evaluation confirms that the condenser, circulating water system, and heat sink are adequate for the power uprate.

The heat loads on the CCW system do not increase significantly due to the 1.62% power uprate because they depend on either reactor vessel water temperature or flow rates in the systems cooled by CCW. The change in reactor vessel water temperature is minimal and there is no change in nominal reactor operating pressure. Regarding the systems cooled by CCW, the CCW system will experience a slight heat load increase; however, the CCW system has adequate design margin to remove the additional heat load.

The power-dependent heat loads on the turbine building closed cooling water system, which increase due to the 1.62% power uprate, are the coolers for the isophase bus, turbine, and generator. The remaining heat loads are not strongly dependent on reactor power and do not increase significantly. The licensee has determined that the turbine building closed cooling water system has sufficient capacity to remove the additional heat load.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant operations at the proposed 1.62% power uprate level do not change the design aspects and operations of the water systems. Therefore, the NRC staff finds that the impact of plant operations at the proposed power uprate level on these systems is acceptable.

3.6.2.1 Discharge Limits

Operational monitoring of the effluents is required by the Commonwealth of Pennsylvania. The licensee estimated an increase in the temperature of water being discharged to the environment (specifically to the Susquehanna River) to be ~1 °F . The effect of this increase in discharge temperature is considered small when reviewed against the state thermal discharge limit. The licensee indicated that PBAPS will monitor and report any adverse conditions on the discharge water temperature.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant operations at the proposed 1.62%

power uprate level is within the current environmental permit. Therefore, the NRC staff finds that the impact of plant operations at the proposed power uprate level on these systems is acceptable.

3.6.3 Standby Liquid Control System

The standby liquid control (SLC) system evaluation is applicable to and consistent with the evaluation in TLTR (Reference 7) Section 5.6.5.

The SLC system provides an alternate means of adding negative reactivity, besides the control rods, as required by 10 CFR 50.62.

The shutdown capability of the SLC system and the boron solution necessary are evaluated each reload cycle. Since the SRV setpoints are not changed for the proposed power uprate, the uprate will have no effect on the rated injection flow. The licensee determined that the capability of the SLC system to provide its backup shutdown function is unchanged and it will continue to meet the requirements of 10 CFR 50.62. Because the proposed power uprate will not change the operating parameters of the SLC system, the NRC staff concludes that the SLC system will perform acceptably during TPO operation.

3.6.4 Heating, Ventilation, and Air Conditioning Systems

The function of the heating, ventilation, and air conditioning (HVAC) systems is to prevent extreme thermal environmental conditions from impacting personnel and equipment by ensuring that design temperatures are not exceeded. HVAC systems that could potentially be affected by the requested power uprate include heating, cooling, exhaust, and recirculation units in the turbine building, drywell, and the reactor building.

The 1.62% power uprate results in a minor increase in heat load caused by the slightly higher FW process temperature ($< 1\ ^\circ$F). The increased heat load is within the margin originally defined based on HVAC analyses at 102% CLTP. Other areas are unaffected by the power uprate because the process temperatures and electrical heat loads remain constant.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that plant operation at the proposed power uprate level will have an insignificant or no impact on the HVAC systems for the above-cited areas.

3.6.5 Fire Protection and 10 CFR Part 50, Appendix R

Fire detection and suppression is not expected to be impacted by plant operations at the proposed 1.62% power uprate level since there are no physical plant configuration changes or combustible load changes resulting from the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and the operator actions required to mitigate the consequences of a fire are not affected by the uprated conditions.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48 and 10 CFR Part 50, Appendix R, and will not be affected by plant operations at the proposed 1.62% power uprate.

3.7 Power Conversion Systems

The PBAPS power conversion systems and their support systems (including the turbine generator, condenser and steam jet air ejectors, turbine steam bypass, and the FW and condensate systems) were originally designed to utilize the energy available from the NSSS and to accept the system and equipment flows resulting from continuous operation at 105% of the CLTP. Therefore, these systems will not be affected by the proposed 1.62% power uprate.

The NRC staff did not review the impact of plant operations at the power uprate level on the design and performance of these systems, because they perform no safety-related function. Additionally, the failure of these systems will not affect the performance of any safety-related system or component.

3.8 Radwaste and Radiation Sources

3.8.1 Activated Corrosion and Fission Products

The reactor coolant contains activated corrosion products from metallic materials entering the water and being activated in the reactor region. Under the TPO uprate conditions, the FW flow increases with power, increasing the activation rate in the region and decreasing the filter efficiency of the condensate demineralizers. The net result may be an increase in the activated corrosion product production.

The licensee states that although the activated corrosion product and fission product activities are expected to increase approximately proportionally to the TPO power increase, the sum of the total activated corrosion product activity and the total fission product activity due to the TPO uprate is expected to remain a fraction of the original design-basis activity in the reactor water.

The design basis for radioactivity in the reactor water is dose equivalent iodine (DEI). The TS limit for DEI is 0.2 µCi/gm. The actual value for DEI is typically in the 10^{-5} to 10^{-4} µCi/gm range. This is three or four orders of magnitude less than the limit. The licensee evaluation indicated that if a 1.62% increase in activity is expected, there would be virtually no impact due to this increase. The DEI limit would remain 0.2 µCi/gm. Therefore, the activated corrosion product and fission product activities design bases are virtually unchanged for the TPO uprate. The NRC staff reviewed the licensee’s analysis and concurs with its conclusion.

3.9 Reactor Safety Performance Evaluation

3.9.1 Anticipated Operational Occurrences (AOOs) - Reactor Transients

AOOs are abnormal transients, which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC-10, -15, and -20. GDC-10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC-15 requires that sufficient margin be included to ensure that the design conditions of the RCPB are not exceeded during normal operating conditions and AOOs. GDC-20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

The Standard Review Plan (SRP) (Reference 14) provides further guidelines: (1) pressure in the reactor coolant and MS system should be maintained below 110% of the design values according to ASME Code, Section III, Article NB-7000, “Overpressure Protection;” (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active-component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

Chapter 14 of the PBAPS UFSAR contains the design-basis accidents (DBAs) that evaluate the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature, (2) a increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The facility’s responses to the most limiting transients are analyzed each reload cycle and corresponding changes in the MCPR are added to the SLMCPR to establish the OLMCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The licensee will perform the reload analysis at the uprated conditions using an NRC-approved methodology. Since the licensee determined that the thermal limits to ensure the fuel cladding integrity will be maintained for operation at the uprated conditions during AOOs and accidents, applicable acceptance criteria are met.

3.9.2 Radiological Analysis of Design-Basis Accidents (DBAs)

The NRC staff reviewed the impact of the proposed TS changes on DBA radiological analyses, as documented in Chapters 5 and 14 of the PBAPS UFSAR. The staff used the information currently contained in the PBAPS UFSAR, in addition to that submitted by the licensee in References 1 through 5. In Reference 3, the licensee identified the existing DBA radiological analyses of record that bound the conditions expected at the proposed power uprate level, 3514 MWt. The licensee stated that the design basis radiological accidents considered in the PBAPS UFSAR and in this power uprate evaluation are:

- loss-of-coolant accident
- main steam line break accident
- control rod drop accident
- fuel handling accident
- instrument line break accident

The licensee further stated that these five DBAs were the same five radiological accidents that were re-analyzed in support of the previous PBAPS power uprate (5%) in 1996, which used a power level basis of 3528 MWt (References 6 and 6a). The staff verified that the radiological analyses in the PBAPS UFSAR Chapters 5 and 14 are based on a reactor power level of 3528 MWt and, therefore, the staff finds that the current DBA radiological analyses bound the proposed 1.62% power uprate (3514 MWt).

3.9.3 Special Events

3.9.3.1 Anticipated Transient Without Scram

Anticipated Transient Without Scram (ATWS) is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62. The regulation requires BWR facilities to have the following mitigating features for an ATWS event:

- an SLC system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel

- an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device

- equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

PBAPS meets the ATWS mitigation requirements defined in 10 CFR 50.62 using the following:

- an SLC capable of boron injection equivalent to 86 gpm

- an ARI system

- an automatic recirculation pump trip (RPT) logic

BWR facilities are also analyzed against certain ATWS acceptance criteria to demonstrate the ability to withstand an ATWS event. These criteria include maintaining fuel integrity (the core and fuel must maintain a coolable geometry), primary system integrity (the peak reactor vessel pressure remains below the ASME Code, Section III, Service Level C limit of 1500 psig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

TLTR (Reference 7) Section 5.3.5 and Appendix L of the same report present a generic evaluation of an ATWS event after a TPO uprate. This evaluation is applicable to PBAPS for suppression pool temperature. In addition, as discussed in NEDC-33064P (Reference 4, Attachment 1), an evaluation performed by GE demonstrated that the calculated peak vessel bottom pressure meets the ATWS acceptance criteria of less than 1500 psig.

Based on the criteria and justification provided in the TLTR and the analyses performed by GE, the RAI responses, and the available margin for peak ATWS parameters, the NRC staff accepts the licensee evaluation and agrees that PBAPS meets the ATWS rule requirements specified in 10 CFR 50.62.

3.9.3.2 Station Blackout

The methodology and the assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the power uprate. The condensate inventory is not credited as a make-up water source under the current SBO analysis. However, condensate storage tank water inventory is available for SBO coping. The licensee performed an analysis for the SBO event. The existing SBO calculations were reviewed for changes due to the increase in the reactor thermal power. The licensee reviewed the following areas for the effect of power uprate:

- control room, cable spreading room
- RCIC room
- HPCI room
- containment
- condensate water requirement
- suppression pool temperature

The analysis concluded that equipment operability during the SBO event and event mitigation capability under the 1.62% power uprate condition is not compromised.

Based on its review, the NRC staff concludes that the uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event and recovery. Under the uprate conditions, the plant continues to meet the requirements of 10 CFR 50.63 and the design is, therefore, acceptable.

3.10 Other Evaluations

3.10.1 High-Energy Line Break (HELB) Analyses

The licensee stated that since the 1.62% power uprate system operating temperatures and pressures change only slightly there is no significant change in HELB mass and energy release. Also, the existing HELB analyses were performed assuming 102% of the current licensed power level, which bounds the proposed 1.62% power uprate condition. Therefore, the licensee concluded that the existing HELB analysis, break locations, pipe whip, and jet impingement analyses remain unchanged. The existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the proposed 1.62% power uprate condition.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the consequences of any postulated HELB would not change significantly and will be acceptable for plant operations at the power uprate level.

3.10.2 Moderate-Energy Line Crack

Analysis for protection against moderate-energy line crack (MELC) is not required for PBAPS since the operating license was issued prior to July 1, 1975. However, the TPO uprate does not change the process conditions for the moderate energy lines. Therefore, the plant internal flooding protection and safe shutdown consideration under MELC are not affected at TPO uprate.

3.11 Human Factors

3.11.1 Emergency and Abnormal Operating Procedures

The licensee stated that its change control process requires the identification and update of the affected operating procedures associated with a modification. The procedures that impact plant operation have been identified and will be revised prior to operation above the current licensed thermal power level.

The NRC staff finds the licensee's response acceptable because the licensee has identified the plant procedures that will be affected by the 1.62% power uprate and indicated that the procedures will be appropriately revised.

3.11.2 Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated that for the power uprate conditions, operator responses to transient, accident, and special events are not affected. Operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the power uprate.

The NRC staff finds the licensee's response acceptable because the licensee has adequately addressed the question of operator actions sensitive to the power uprate by describing the lack of effect on operator performance and operator response.

3.11.3 Operator Training Program and the Control Room Simulator

Regarding the operator training program, the licensee stated that no additional training (apart from normal training) is required to operate the plant at the uprated conditions. Minor changes to the power/flow map, flow-referenced setpoint, and changes to the TSs will be communicated through routine operator training prior to operation at the power uprate level.

Regarding the control room simulator, the licensee stated that no physical changes are required to the simulator to reflect the power uprate conditions. Simulator software changes and validation are controlled in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination."

The NRC staff finds the licensee's response acceptable because the licensee has adequately addressed the changes to the operator training program and how the simulator will accommodate the changes.

3.11.4 Summary - Human Performance

Based on the evaluation in Sections 3.11.1 through 3.11.3 of this safety evaluation, the NRC staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The NRC staff further concludes that the power uprate should not adversely affect simulation facility fidelity or operator performance.

3.12 Facility Operating License (FOL) and Technical Specification Changes

The licensee proposed to revise the FOL and TSs as follows to reflect the increase in licensed power level from 3458 MWt to 3514 MWt:

- Paragraph 2.C.1, of FOL DPR-44 & DPR-56, is revised to authorize operation at a steady state reactor core power level not in excess of 3514 megawatts.

- The definition of RATED THERMAL POWER in TS Section 1.1 is revised to reflect the increase from 3458 MWt to 3514 MWt.

- The allowable value of Function 2.b, Average Power Range Monitors (APRM) Simulated Thermal Power - High, in TS Table 3.3.1.1-1, is revised to:

    - "≤ 0.65 W + 63.7%" from "≤ 0.66 W + 64.9%" for two loop operation and
    - "≤ 0.65 W + 63.7% -0.65 (delta W)" from "≤ 0.66 W + 64.9% -0.66 (delta W)" for single loop operation

- The percentage RTP is revised to "29.5" from "30" in TS Table 3.3.1.1-1, Applicable Modes or Other Specified Conditions of Functions 8 and 9.

- The percentage RTP is revised to "29.5" from "30" in TS Section 3.3.1.1, Required Action E.1.

- The percentage RTP is revised to "29.5" from "30" in TS Surveillance Requirement (SR) 3.3.1.1.13.

- The percentage RTP is revised to "29.5" from "30" in TS 3.3.4.2 Applicability, Required Action C.2, and SR 3.3.4.2.4.

- Figure 3.4.1-1, "Thermal Power Versus Core Flow Stability Regions," is replaced.

The FOL and TS changes reflect the proposed increase in licensed power level based on installation of the Caldon LEFM✓+™ System for FW flow and temperature measurements. Based on the evaluations discussed in Sections 3.1 through 3.11 of this safety evaluation, the NRC staff concludes that the above-described changes to the FOL and TSs are acceptable.

4.0 REGULATORY COMMITMENTS

The licensee included regulatory commitments in its application and its responses to the NRC staff RAIs. The commitments relevant to the NRC staff evaluations are listed in the following table.

Regulatory Commitments

COMMITMENT	TYPE (Check One) ONE-TIME ACTION	TYPE (Check One) CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If required)
The administrative controls will be added to the PBAPS Technical Requirements Manual for LEFM✓+™ inoperability.		X	upon implementation
Pressure control system (PCS) tests will be performed during the power ascension phase (Section 10.4). (NEDC-33064P Section 5.2.1)	X		upon implementation
Per the guidelines of Appendix L of the TLTR, the performance of the FW level control systems will be recorded at 95% and 100% of the CLTP and confirmed at the TPO RTP during power ascension. These checks will demonstrate acceptable operational capability. (NEDC-33064P Section 5.2.2)	X		upon implementation
In preparation for operation at TPO uprated conditions, routine measurements of reactor and system pressures and flows and vibration measurements on selective rotating equipment will be taken near 95% and 100% of the CLTP, and retaken at 100% of TPO RTP. (NEDC-33064P Section 10.4)	X		upon implementation

COMMITMENT	TYPE (Check One) ONE-TIME ACTION	TYPE (Check One) CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If required)
Demonstration of acceptable fuel thermal margin will be performed prior to power ascension to the TPO RTP at the 100% CLTP steady-state heat balance point. Fuel thermal margin will be calculated for the TPO RTP point after the measurements taken at 95% and 100% of the CLTP to project the estimated margin. (NEDC-33064P Section 10.4)	X		upon implementation
The response of the pressure and FW level control systems will be recorded at each steady-state point defined above to demonstrate acceptable operational capability. (NEDC-33064P Section 10.4)	X		upon implementation
A cycle-specific reload analysis will be performed prior to implementation of power uprate. This analysis will be submitted to the NRC for review, prior to operation at the uprated power level, if deemed necessary by criteria of 10 CFR 50.59.		X	Prior to use of the amendment.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee’s administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67 FR 45568). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 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.

7.0 REFERENCES

1. Michael P. Gallagher, EGC, letter to NRC, "License Amendment Request 01-01190, Power Uprate Request for Appendix K Measurement Uncertainty Recapture,” dated May 24, 2002.

2. Michael P. Gallagher, EGC, letter to NRC, "Transmittal of Non-Proprietary General Electric Topical Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3, NEDO-33064,” dated June 27, 2002.

3. Michael P. Gallagher, EGC, letter to NRC, "Response to Request for Additional Information Regarding License Amendment Request 01-01190, Power Uprate Request for Appendix K Measurement Uncertainty Recapture,” dated September 11, 2002.

4. Michael P. Gallagher, EGC, letter to NRC, "Submittal of Peach Bottom Atomic Power Station, Units 2 & 3, Safety Analysis Report, Revision 1, Appendix K Measurement Uncertainty Recovery Power Uprate (License Amendment Request 01-01190),” dated September 24, 2002.

5. Michael P. Gallagher, EGC, letter to NRC, "Request for Additional Information, Appendix K Measurement Uncertainty Recovery Power Uprate (License Amendment Request 01-01190),” dated October 16, 2002.

6. Joseph W. Shea, NRC, letter to George A. Hunger, PECO Energy Company, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 2,” dated October 18, 1994.

6a. Joseph W. Shea, NRC, letter to George A. Hunger, PECO Energy Company, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 3," dated July 18, 1995.

7. General Electric, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TLTR)," Licensing Topical Report NEDC-32938P, dated July 2000.

8. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1), Licensing Topical Report NEDC-32424P-A, Class III (Proprietary), February 1999; and NEDC-32424, Class I (Nonproprietary), dated April 1995.

9. GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2), Licensing Topical Report NEDC-32523P-A, Class III (Proprietary), February 2000; NEDC-32523P-A, Supplement 1, Volume 1, February 1999, and Supplement 1, Volume II (Proprietary), dated April 1999.

10. Caldon, Inc., Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using The LEFM✓+™ System," dated March 1997.

10a. John N. Hannon, NRC, letter to C. L. Terry, TU Electric, "Staff Acceptance of Caldon Topical Report ER-80P: Improving Thermal Power Accuracy While Increasing Power Level Using The LEFM System," dated March 8, 1999.

11. Caldon, Inc., Engineering Report ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus. System," dated October 2001.

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