Attachment 12

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

Risk Assessment

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Section 1
INTRODUCTION

The Peach Bottom Atomic Power Station (PBAPS) is currently pursuing an increase in reactor power from the current licensed thermal power of 3514 MWt to 3951 MWt, an Extended Power Uprate (EPU) of 120% of the original licensed thermal power (OLTP). The EPU is a constant pressure power uprate (CPPU).

The purpose of this report is to:

(1) Identify any significant change in risk associated with the Extended Power Uprate (EPU) as measured by the PBAPS Probabilistic Risk Assessment (PRA) models.

(2) Provide the basis for the impacts on the risk model associated with EPU.

1.1 BACKGROUND

The 2009A update to the Peach Bottom (PB) PRA model is the most recent evaluation of the risk profile at PB for internal event challenges. The PB PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the PB PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon Generation Company (Exelon) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Peach Bottom PRA.

<u>PRA Maintenance and Update</u>

The Exelon risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk

Management program, which consists of a governing procedure and subordinate implementation procedures. The PRA model update procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites. The overall Exelon Risk Management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, Exelon risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance addresses:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- The process for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- The use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

The PBAPS PRA is derived based on realistic assessments of system capability over the 24 hour mission time of the PRA analysis. Therefore, PRA success criteria may be different than the design basis assumptions used for licensing PBAPS. This analysis uses the PRA to provide insights about how plant risk from postulated accidents, including severe accidents, is impacted by EPU implementation.

1.2 PRA QUALITY

Several assessments of technical capability have been made, and continue to be made, for the PBAPS, Units 2 and 3 PRA models. These assessments are as follows and further discussed in Appendix A of this document.

- An independent PRA peer review was conducted under the auspices of the BWR Owners Group in 1998, following the Industry PRA Peer Review process [1]. This peer review included an assessment of the PRA model maintenance and update process.
- In 2004, a gap analysis was performed against the available version of the ASME PRA Standard [2] and the draft version of Regulatory Guide 1.200, DG-1122 [3]. In 2006, an assessment of the extent to which the previously defined gaps had been addressed was performed in conjunction with a PRA model update.
- During 2005 and 2006 the PBAPS, Units 2 and 3 PRA model results were evaluated in the BWR Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process [4].
- In November of 2010, a BWROG peer review was conducted on the PB209A Unit 2 and PB309A Unit 3 PRA models (that is, the 2009A PRA models used as the basis for the EPU risk assessment). This review was performed using ASME/ANS RA-Sa-2009 [5] and RG 1.200, Rev. 2 [6].

In summary, there are a few identified issues that remain open from the peer review. These deviations do not significantly impact the base PRA model or its ability to support the full range of PRA applications. Appendix A provides more details of this evaluation including an assessment of the peer review findings on the EPU risk assessment. Additionally, sensitivity studies were performed, where warranted, as described in Appendix A and Section 5.7.1 of this report.

<u>Scope and Level of Detail</u>

The PBAPS 2009A PRA model is of sufficient quality and scope to measure the potential changes in plant risk related to EPU implementation. The PBAPS PRA modeling is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, extensive level of detail, operator actions, and common cause events.

External hazards were evaluated in the PBAPS Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) [7]. The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the PBAPS IPEEE study are documented in the PBAPS IPEEE Main Report [8] and related correspondence. The primary areas of external event evaluation at PBAPS were internal fire and seismic. The internal fire events were addressed by using a modified version of the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology [9] and the seismic evaluations were performed in accordance with the EPRI Seismic Margins Analysis (SMA) methodology [10]. As such, there are no comprehensive CDF and LERF values available from the IPEEE to support the EPU risk assessment.

Since the performance of the IPEEE, a Fire PRA was performed. The EPRI FIVE Methodology [9] and Fire PRA Implementation Guide (FPRAIG) [11] screening approaches, EPRI Fire Events Database [12] and plant specific data were used in this 2002 study, to develop the PBAPS Fire PRA. An update to that Fire PRA model was performed in 2007 that included explicit analysis of the main control room (MCR) and cable spreading room (CSR) that had previously not been included. The ignition frequencies for the MCR and CSR were developed using the guidance in NUREG/CR-6850 [13]. The Fire PRA model was also integrated with the PB205C and PB305C internal events models as part of the 2007 update.

In addition to internal fires and seismic events, the PBAPS IPEEE analysis of high winds, floods, and other (HFO) external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Since both PBAPS units were designed (with construction started) prior to the issuance of the 1975 Standard Review Plan (SRP) criteria, PECO [now Exelon] performed a plant hazard and design information review for conformance with the SRP criteria. For seismic and fire events that were not screened out, additional analyses were performed to determine whether or not the hazard frequency was acceptably low. HFO events were screened out by compliance with the 1975 SRP criteria [14]. As such, these hazards were determined in the PBAPS IPEEE to be negligible contributors to overall plant risk.

Although a quantifiable Fire PRA model exists for PBAPS, this model has not been approved for general use in quantified risk applications because there are several areas of conservatism in the current treatment that result in skewing the total reported CDF towards the upper bound.

PBAPS does not maintain a shutdown PRA model. However, insights from other available industry studies were utilized to allow for quantitative comparisons of the likelihood of boiling and fuel damage scenarios based on equipment availability, reliability, and decay heat levels. The magnitude of the changes to shutdown risk resulting from EPU was estimated by examining how the corresponding increased heat load and equipment changes would impact the risk profile at PBAPS. Therefore, the impact on shutdown risk based on EPU conditions is based on more generic shutdown insights and assumptions obtained from a review of other industry BWR shutdown PRA results.

Summary

In summary, it is found that the PBAPS integrated Level 1 and Level 2 PRA model provides the necessary scope and level of detail to allow the calculation of CDF and radioactive release frequency changes due to the EPU. The External Events models will allow for a review of the largest contributors to External Events risk and how they might be impacted by EPU. The information from generic shutdown PRA results will provide the capability to determine the magnitude of the changes to plant shutdown risk that would occur based on EPU implementation.

1.3 PRA DEFINITIONS AND ACRONYMS

Definitions

The following PRA terms are used in this study:

CDF – Core Damage Frequency (CDF) is a risk measure for calculating the frequency of a severe core damage event at a nuclear facility. CDF is calculated in units of events per reactor year. Core damage is the end state of the Level 1 Probabilistic Risk Assessment (PRA). A core damage event is defined in the PBAPS PRA by the following:

> The onset of core damage is defined as the time at which more than two-thirds of the active fuel becomes uncovered, without sufficient injection available to recover the core quickly, i.e., water level below one-third core height and falling plus calculated peak core temperatures from MAAP greater than 1800^{O}F for more than 10 minutes.

LERF – Large Early Release Frequency (LERF) is a risk measure for calculating the frequency of an offsite radionuclide release that is "large" in fission product magnitude and "early" in release timing. LERF is calculated in units of events per reactor year. LERF is one of the end states of the Level 2 PRA. A large (high) and early release is defined in the PBAPS PRA by the following:

> A "large" (high) magnitude release is defined as a radionuclide release of sufficient magnitude to have the potential to cause early fatalities (e.g., greater than 10% of the core inventory of Cesium Iodide in the release). An "early" timing release is defined as the timing in which minimal offsite protective measures can be implemented (e.g., less than 8.25 hours from declaration of general emergency based on PBAPS evacuation studies).

Initiating Event – Any event that causes a scram (e.g., Loss of Feedwater, MSIV Closure) and requires the initiation of mitigation systems to reach a safe and stable state. An initiating event is modeled in the PRA to represent the primary transient event that can lead to a core damage event given failure of adequate mitigation systems (i.e., adequate with respect to the transient in question).

Internal Events – Those initiating events caused by failures internal to the system boundaries. Examples include MSIV Closure, Loss of an AC Bus, Loss of Offsite Power, and internal floods.

External Events – Those initiating events caused by failures external to the system boundaries. Examples include fires, seismic events, and tornadoes.

HEP – Human Error Probability (HEP) is the probabilistic estimate that the operating crew fails to perform a specific action (either properly or within the necessary time frame) to support accident mitigation. The HEP is calculated using industry methodologies and considers a number of performance shaping factors such as:

- training of the operating crew,
- availability of adequate procedures,
- man-machine interface issues,
- time required to perform action,
- time available to perform action.

HRA – Human Reliability Analysis (HRA) is the systematic process used to evaluate operator actions and quantify human error probabilities.

MAAP – The Modular Accident Analysis Program (MAAP) is an industry recognized thermal hydraulic code used to evaluate design basis and beyond design basis accidents. MAAP can be used to evaluate thermal hydraulic profiles within the primary system (e.g., RPV pressure, boildown timing) prior to core damage. MAAP also can be used to evaluate post core damage phenomena such as RPV breach, containment mitigation, and offsite radionuclide release magnitude and timing.

Level 1 PRA – The Level 1 PRA is the evaluation of accident scenarios that begin with an initiating event and progress to core damage. Core damage is the end state for the Level 1 PRA. The Level 1 PRA focuses on the capability of plant systems to mitigate a core damage event.

Level 2 PRA – The Level 2 PRA is a continuation of the Level 1 PRA evaluation. The Level 2 PRA begins with the accident scenarios that have progressed to core damage and evaluates the potential for offsite radionuclide releases. Offsite radionuclide release is the end state for the Level 2 PRA. The Level 2 PRA focuses on the capability of plant systems (including containment structures) to prevent a core damage event to result in an offsite release.

RAW – The Risk Achievement Worth (RAW) is the calculated increase in a risk measure (e.g., CDF or LERF) given that a specific system, component, operator action, etc. is assumed to fail (i.e., failure probability of 1.0). RAW is presented as a ratio of the risk measure given the component is failed divided by the risk measure given the component is assigned its base failure probability.

FV – The Fussell-Vesely (FV) importance is a measure of the contribution of a specific system, component, operator action, etc. to the overall risk. F-V is presented as the percentage of the overall risk to which the component failure contributes. In other words, the F-V importance represents the overall decrease in risk if the component is guaranteed to successfully operate as designed (i.e., failure probability of 0.0).

Acronyms

The following acronyms are used in this study:

AC	Alternating Current
ANS	American Nuclear Society
ARI	Alternate Rod Insertion
ASEP	Accident Sequence Evaluation Program
ASME	American Society Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BEID	Basic Event Identification
BOC	Break Outside Containment
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAD	Containment Atmosphere Dilution
CBDT	Cause-Based Decision Tree
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CLERP	Conditional Large Early Release Probability

CLTP	Current Licensed Thermal Power
CLTR	Constant Pressure Power Uprate LTR
CPPU	Constant Pressure Power Uprate
CRD	Control Rod Drive
CS	Core Spray
CSR	Cable Spreading Room
CST	Condensate Storage Tank
CWG	Conowingo (SBO Line)
DBA	Design Basis Accident
DC	Direct Current
DHR	Decay Heat Removal
DW	Drywell
DWS	Drywell Spray
ECCS	Emergency Core Cooling System
EF	Error Factor
EOC	End of Cycle
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
ESW	Emergency Service Water
FIVE	Fire Induced Vulnerability Evaluation
F&O	Facts and Observations
FPRAIG	Fire PRA Implementation Guide
F-V	Fussell-Vesely (risk importance measure)
FW	Feedwater
GE	General Electric
HCTL	Heat Capacity Temperature Limit
HEP	Human Error Probability
HFE	Human Failure Event
HFO	High Winds, Floods, and Other (External Hazards)
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPSW	High Pressure Service Water
HRA	Human Reliability Analysis
HX	Heat Exchanger
ILRT	Integrated Leak Rate Test
INS	Instrument Nitrogen System
IORV	Inadvertent Open Relief Valve
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Evaluation of External Events

ISLOCA	Interfacing Systems LOCA
LAR	License Amendment Request
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LTR	Licensing Topical Report
MCR	Main Control Room
MAAP	Modular Accident Analysis Program
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MG	Motor Generator
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSPI	Mitigating Systems Performance Indicator
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLTP	Original Licensed Thermal Power
OOS	Out of Service
PB	Peach Bottom
PBAPS	Peach Bottom Atomic Power Station
PIMS	Plant Information Monitoring System
PRA	Probabilistic Risk Assessment
PSL	Pressure Suppression Limit
PSSA	Probabilistic Shutdown Safety Assessment
PUSAR	Power Uprate Safety Analysis Report
R&R	Risk and Reliability
RAW	Risk Achievement Worth (risk importance measure)
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RM	Risk Management
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RTP	Reactor Thermal Power

RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth (risk importance measure)
RWCU	Reactor Water Clean-Up
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SDC	Shutdown Cooling
SLC	Standby Liquid Control
SMA	Seismic Margins Analysis
SORV	Stuck Open Relief Valve
SPC	Suppression Pool Cooling
SR	Supporting Requirement
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSC	Systems, Structures, and Components
SV	Safety Valve
SW	Service Water
TAF	Top of Active Fuel
TBCCW	Turbine Building Closed Cooling Water
TDT	Torus Dewatering Tank
TF	Transient – Loss of Feedwater
THERP	Technique for Human Error Rate Prediction
TT	Transient – Turbine Trip
URE	Updating Requirements Evaluation

1.4 GENERAL ASSUMPTIONS

The extended power uprate (EPU) risk evaluation includes a limited number of general assumptions as follows:

- This analysis is based on all the inputs provided by Exelon [15] in support of this assessment. For systems where no hardware or procedural changes have been identified, the risk evaluation is performed assuming no impact as a result of the EPU.

- Replacement of components with enhanced like components does not result in any supportable significant increase in the long-term failure probability for the components. Equipment reliability can be postulated theoretically to behave as a "bathtub" curve (i.e., the beginning and end of life phases being associated with higher failure rates than the steady-state period); however, no significant impact on the long term average of component reliability is supportable at this time and no modifications to the PRA are suggested for these types of changes.

- The PRA success criteria are different than the success criteria used for design basis accident evaluations. The PRA success criteria assume that systems that can realistically perform a mitigation function (e.g., main condenser or containment venting for decay heat removal) are credited in the PRA model. In addition, the PRA success criteria are based on the availability of a discrete number of systems or trains (e.g., number of pumps for RPV makeup).

Section 2
SCOPE

The scope of this risk assessment for the Extended Power Uprate at PBAPS addresses the following plant risk contributors:

- Level 1 Internal Events At-Power (CDF)
- Level 2 Internal Events At-Power (LERF)
- External Events At-Power
  - Seismic Events
  - Internal Fires
  - Other External Events
- Shutdown Assessment

Risk impacts due to internal events are assessed using the PBAPS Level 1 and Level 2 2009A1 PRA Models for Pre-EPU and EPU conditions. External events are evaluated using the insights and results from the PBAPS Individual Plant Examination for External Events (IPEEE) Submittal [8] and more recent fire PRA investigations. The impacts on shutdown risk contributions are evaluated on both qualitative and a quantitative bases.

The PBAPS IPE [17] and IPEEE [8] submittals were reviewed for identification of vulnerabilities, outliers, anomalies or weaknesses that would impact the PBAPS EPU risk assessment. The IPE submittal noted that no plant vulnerabilities leading to core damage or a large release were uncovered in the IPE process and the results of the IPE were comparable to the NRC sponsored NUREG/CR-4550 study [28]. Additionally, the IPEEE did not identify any vulnerabilities associated with seismic, fire or other external events. However, a number of areas for improvement were identified with respect to seismic and fire risk. Actions to address these and their closure are documented in the Exelon PIMS Action Request System [29]. Based on this review, there are no vulnerabilities, outliers, anomalies or weaknesses that would impact the results and conclusions of the PBAPS EPU risk assessment. In summary, all of the commitments resulting from the PBAPS IPE and IPEEE Programs have been adequately resolved.

As is discussed in Section 3, all the PRA elements are reviewed to ensure that identified EPU plant, procedural, or training changes that could affect the risk profile are addressed. The information input to this process is based on the PBAPS EPU modification list developed by Exelon [15].

Section 3
METHODOLOGY

This section of the report addresses the following:

- Analysis approach used in this risk assessment (Section 3.1)
- Identification of principal elements of the risk assessment that may be affected by the Extended Power Uprate and associated plant changes (Section 3.2)
- Plant changes used as input to the risk evaluation process (Section 3.3)
- PRA Scoping assessment (Section 3.4)

3.1 ANALYSIS APPROACH

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate", Class III, July 2003 [18], (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of Constant Pressure Power Uprates. Section 10.5 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Individual Plant Evaluation.

The approach used to examine the risk profile changes and confirm the conclusions from the CLTR for PBAPS is described in the following subsections.

3.1.1 Identify PRA Elements

This task is to identify the key PRA elements to be assessed as part of this analysis for potential impacts associated with plant changes. The identification of the PRA elements stems from the ASME/ANS PRA Standard review elements [5]. Section 3.2 summarizes the PRA elements assessed for the PBAPS EPU.

3.1.2 Gather Input

The input required for this assessment includes the identification of all plant hardware modifications, operational changes, and procedure updates that are implemented as part of the extended power uprate. This includes changes such as instrument setpoint changes, added equipment, and procedural modifications.

3.1.3 Scoping Evaluation

This task is to perform a scoping evaluation by reviewing the plant input against the key PRA elements. The purpose is to identify those items that require further quantitative analysis and to screen out those items that are estimated to have negligible or no impact on plant risk as modeled by the PBAPS PRA model.

3.1.4 Qualitative Results

The result of this task is a summary which dispositions all the risk assessment elements regarding the effects of the extended power uprate. The disposition consists of three Qualitative Disposition Categories:

Category A: Potential PRA change due to power uprate. PRA modification desirable or necessary

Category B: Minor perturbation, negligible impact on PRA, no PRA changes required

Category C: No change

A short explanation providing the basis for the disposition is provided in Section 4.

3.1.5 Implement and Quantify Required PRA Changes

This task is to identify the specific PRA model changes required to address the EPU, implement them, and quantify the models. Section 4.1 summarizes the review of PRA analysis impacts associated with the increased power level. These effects and other effects related to plant or procedural changes are identified and documented in Section 4.

3.2 PRA ELEMENTS ASSESSED

The PRA elements to be evaluated and assessed can be derived from a number of sources. The ASME/ANS PRA Standard [5] provides a convenient division for the internal events "elements" to be examined.

Each of the major risk assessment elements is examined in this evaluation. Most of the risk assessment elements are anticipated to be unaffected by the Extended Power Uprate. The risk assessment elements addressed in this evaluation for impact due to the EPU (refer to Section 4 for impact evaluation) include the following:

- Initiating Events
- Accident Sequence Modeling
- Systemic/Functional Success Criteria, e.g.:
  - Time to Boil-off
  - RPV Inventory Makeup
  - Heat Load to the Suppression Pool
  - Blowdown Loads
  - RPV Overpressure Margin
  - SRV Actuations
  - SRV Capacity for ATWS
- System Modeling
- Component Reliability / Failure Data
- Human Reliability Analysis
- Internal Flooding
- LERF Analysis
- Quantification

3.3 INPUTS (PLANT CHANGES)

This section summarizes the inputs to the risk evaluation, which include hardware modifications, setpoint changes, procedural and operational changes associated with the extended power uprate.

3.3.1 Hardware Modifications

The hardware modifications associated with the extended power uprate have been identified by Exelon as input to this assessment [15]. These hardware modifications were reviewed to determine their potential impact on the PRA model. This assessment is based on review of the plant hardware modifications and engineering judgment based on knowledge of the PRA models. The majority of the changes are characterized by either:

- Replacement of components with enhanced like components, or
- Upgrade of existing components

For many of the identified changes, there is either no direct PRA impact or the impact is encompassed within sensitivity cases that increase various initiator frequencies. First, the transient initiator frequencies are increased to conservatively bound the potential impact from various changes to the BOP side of the plant. Additionally, potential operational issues were taken into account in a sensitivity case for the loss of feedwater transient initiating event frequency. Finally, this analysis doubles the LOCA initiating event frequency in a quantitative sensitivity case, which is assumed to address any potential changes in the LOCA frequency related to the EPU changes. Refer to Section 5.7.1 for results of the sensitivity cases.

Given this, however, the review did identify that the following set of changes were determined to have an impact on the PRA model.

- Install a Third Spring Safety Valve on Main Steam Line
- Install Residual Heat Removal (RHR) Heat Exchanger Cross-Tie Modifications
- High Pressure Service Water (HPSW) Cross-connect
- CST Standpipe and Swapover Point
- SLC Boron Enrichment

A scoping evaluation for the changes identified above is summarized in Section 3.4-1.

3.3.2 Procedural Changes

In order to ensure the plant is operated safely, adjustments to the PBAPS Emergency Operating Procedures/Severe Accident Management Guidelines (EOPs/SAMGs) will be made consistent with EPU operating conditions. The full set of anticipated changes is documented in the Human Factors Evaluation discussed in Section 2.11 of the PUSAR. In almost all respects, the EOPs/SAMGs are expected to remain unchanged because they are symptom-based; however, certain parameter thresholds and curves are dependent upon power and decay heat levels and will require procedure modification.

Based on generic EPU evaluations by the nuclear steam and supply system (NSSS) vendor, General Electric [16], EOP variables that play a role in the PRA and which may require adjustment for the EPU include:

- Heat Capacity Temperature Limit (HCTL)
- Pressure Suppression Limit (PSL)

These variables may require adjustment to reflect the change in power level, but will not be adjusted in a manner that involves a change in accident mitigation philosophy. The HCTL and PSL relate to long-term scenarios and any perturbations in the scenario timings associated with EPU changes to these curves will be minor. However, because the human reliability analysis (HRA) methodology can be sensitive to these types of changes, the timing data have been explicitly addressed in the event tree evaluations and the HRA for EPU conditions in the PB209A1 PRA model.

Additional required PRA model changes have also been identified to account for changes to implement the RHR cross-tie modification when needed and to throttle the flow to meet the associated revised NPSH curves.

3.3.3 Setpoint Changes

The RPV operating pressure and the operating temperature are not being changed as part of the EPU. In addition, changes to the following setpoints which could have an impact on the PRA model are not anticipated for the EPU:

- Recirculation Pump Trip/Anticipated Transient without Scram (RPT/ATWS) high dome pressure
- Safety Valve/Safety Relief Valve (SV/SRV) setpoints

Setpoint changes for the EPU that have been identified include:

- Main Steam hi-flow instrument setpoint changes
- Cross Around Relief setpoint changes (steam piping for HP and LP turbines)
- Power Range Neutron Monitoring setpoint changes

Other minor setpoint changes may be made to various systems for operational margin purposes. Such minor setpoint changes have no direct quantifiable impact on the plant risk.

3.3.4 Plant Operating Conditions

The key plant operational modifications to be made in support of the EPU are:

- Increase in the current licensed thermal power from 3514 to 3951 MWt (general change, not identified in the Modification List)
- Corresponding increase in the FW/Condensate flow and steam flow rates (general change, not identified in the Modification List)
- Following EPU (prior to MELLLA+), the acceptable region of operating flows at 100% Reactor Thermal Power (RTP) will be narrowed.

RPV pressure will remain unchanged for the EPU, and the maximum core flow will also remain unchanged.

3.4 PRA SCOPING EVALUATION

The scoping evaluation examines the hardware, procedural, setpoint, and operating condition changes to assess whether there are PRA impacts that need to be considered in addition to the increase in power level. These changes will also be examined in Section 4 relative to the PRA elements that may be affected. The scoping evaluation conclusions reached are discussed in the following subsections.

3.4.1 Hardware Changes

The hardware changes required to support the EPU (see Section 3.3.1) were reviewed and determined not to result in new accident types or a measurable increase in the frequency of challenges to plant response. This assessment is based on review of the plant hardware modifications and engineering judgment based on knowledge of the PRA models. The majority of the changes are characterized by either:

- Replacement of components with enhanced like components, or
- Upgrade of existing components

Extensive changes to plant equipment have been shown by operating experience to result in an increase in system unavailability or failure rate during the initial testing and break-in period. It can be expected that there will be some short-term increase in such events at PBAPS but the frequency and duration of such events cannot be predicted. Nevertheless, it is expected that a steady state condition equivalent to (or potentially better than) current plant performance would result.

Given this, however, the review did identify that the following set of changes were determined to have an impact on the PRA model. Reference to the PRA change discussion is noted in parenthesis.

- Install a Third Spring Safety Valve on Main Steam Line (refer to Section 4.1.2.5)
- Install Residual Heat Removal (RHR) Heat Exchanger Cross-Tie Modifications (refer to Section 4.1.2.3)
- High Pressure Service Water (HPSW) Cross-connect (refer to Section 4.1.2.3)
- CST Standpipe and Swapover Point (refer to Section 4.1.4)
- SLC Boron Enrichment (refer to Section 4.2 under the ATWS heading)

3.4.2 Procedure Changes

Final changes to the EOPs/SAMGs as a result of the EPU were not available prior to completion of the PRA evaluation. However, the list of anticipated changes documented in the Human Factors Evaluation discussed in Section 2.11.1.1 of the PUSAR was reviewed for applicability to the PRA model. Based on this review, for the most part it is assumed that the procedural changes (e.g., modification to HCTL curve) have a minor impact on the PRA results, and are encompassed within the timing changes associated with EPU conditions that have been directly factored into the risk assessment (refer to Section 4.1.6 of this report). However, specific representation for implementation of the RHR cross-tie when needed and throttling the RHR flow has been specifically included in the PRA model for EPU conditions. Refer to Section 4.1.2.3 of this report for more details on the PRA modeling of the RHR cross-tie.

3.4.3 Setpoint Changes

Most of the planned setpoint changes will not result in any quantifiable impact to the PRA. Key setpoints that play a role in the PRA are planned to remain unchanged, such as:

- Main Steam SRV opening and closing setpoints
- RPV pressure setpoint (e.g., ATWS RPT high pressure setpoint)

The analyses discussed in Sections 2.8.4.2 and 2.8.5.7 of the PUSAR show that the above existing current license thermal power (CLTP) setpoints remain adequate for EPU conditions, which results in no required changes to the PRA model.

3.4.4 Normal Plant Operational Changes

The Feedwater/Condensate flow rates will be increased to support the EPU. Despite the increase in flow, there is no indication modeling-wise that this operational change will significantly impact component failure rates or initiating event frequencies in the long term.

Section 4
PRA CHANGES RELATED TO EPU CHANGES

Section 3 has examined the plant changes (hardware, procedural, setpoint, and operational) that are part of the extended power uprate (EPU). Section 4 examines these changes to identify PRA modeling changes necessary to quantify the risk impact of the EPU. This section discusses the following:

- Individual PRA elements potentially affected by EPU (4.1)
- Level 1 PRA (4.2)
- Internal Fires Induced Risk (4.3)
- Seismic Risk (4.4)
- Other External Hazards Risk (4.5)
- Shutdown Risk (4.6)
- Radionuclide Release (Level 2 PRA) (4.7)

4.1 PRA ELEMENTS POTENTIALLY AFFECTED BY POWER UPRATE

A review of the PRA elements has been performed to identify potential effects associated with the extended power uprate. The result of this task is a summary which dispositions all the PRA elements regarding the effects of the extended power uprate. The disposition consists of three qualitative disposition categories.

Category A: Potential PRA change due to power uprate. PRA modification desirable or necessary

Category B: Minor perturbation, negligible impact on PRA, no PRA changes required

Category C: No change

Table 4.1-1 summarizes the results from this review. Based on Table 4.1-1, only a small number of the PRA elements are found to be potentially influenced by the power uprate.

Table 4.1-1

REVIEW OF PRA ELEMENTS FOR POTENTIAL RISK MODEL EFFECTS

PRA Elements	Disposition Category	Basis
Initiating Events	B	No new initiators or increased frequencies of existing initiators are anticipated to result from the PBAPS EPU. However, quantitative sensitivity cases that increase the transient and LOCA frequencies are performed as part of this analysis.
Success Criteria	B	There are a number of potential effects that could alter success criteria. These are discussed in the text. They include the following: • Timing • RPV Inventory Makeup • Heat Load to the Suppression Pool • Blowdown Loads • RPV Overpressure Margin (number of SRVs/SVs required) • SRV Actuations post-trip • RPV Depressurization (number of SRVs required) • Structural Evaluations

Accident Sequences (Structure, Progression)	C	For the most part, the EPU does not change the plant configuration or operation in a manner such that new accident sequences or changes to existing accident scenario progressions result. The one exception is the incorporation of the requirement to align the new RHR cross-tie valve under certain conditions to avoid the need for crediting containment accident pressure. Additionally, the accident progression is slightly modified in timing. The majority of these changes are incorporated in the Human Reliability Analysis (HRA). One additional aspect is the impact on long term LOOP recovery probabilities due to timing differences to reach the containment vent pressure to prevent overpressure failure. See Section 4.1.3.
System Modeling	B	For the most part, no new system failure modes or significant changes in system failure probabilities due to the EPU. The exceptions for PBAPS are the incorporation of the new RHR cross-tie valve and the addition of a CST stand-pipe.
Data Analysis	C	No change to component failure probabilities.
Human Reliability Analysis	A	The change in initial power level in turn results in decreases in the time available for operator actions. See discussion of operator actions in Section 4.1.6.
Internal Flooding	C	No changes in the internal flooding modeling are anticipated based on EPU. The initiating event frequencies and impact vectors (i.e., the affected equipment from the flood event) from the flooding analysis are unchanged from EPU. Any changes in the overall contribution from flooding would be related to other modeling changes (e.g., HEP changes). However, quantitative sensitivity cases that increase the internal flood initiating event frequencies are performed as part of this analysis.

Quantification	C	No changes in PRA quantification process (e.g., truncation limit, flag settings, etc.) due to EPU.
Level 2	B	Slight changes in accident progression timing result from the increased decay heat. This resulted in slightly different release category magnitude and timing results. The release magnitude and timing category assignments were unchanged, however, since the PRA release categories are defined based on the percentage of CsI released to the environment.

The PRA elements from Table 4.1-1 are discussed to summarize whether they may be affected by the extended power uprate and the associated changes.

4.1.1 Initiating Events

The CLTR states that the increase in power level results in the plant operating closer to limits which can potentially increase event frequency and affect CDF and LERF results. However, although experience indicates that major changes to equipment can increase equipment unavailability in the short-term due to break-in ("bathtub curve"), this impact cannot be easily quantified and steady state conditions are expected to be equivalent or better than current plant performance. Therefore, the evaluation of the plant and procedural changes indicates no new initiators or increased frequencies of existing initiators are anticipated to result from the PBAPS EPU.

The PBAPS PRA initiating events can be categorized into the following:

- Transients
- Loss of Offsite Power (LOOP)
- Loss of Coolant Accidents (LOCAs)
- Support System Failures
- Internal Floods

Additionally, external event initiators are also discussed for completeness.

Transients

The evaluation of the EPU plant and procedural changes do not result in any new transient initiators, nor is there anticipated any direct impact on transient initiator frequencies due to the EPU (i.e., no changes are being made for the EPU to the number of normally operating pumps and equipment in BOP systems). The Peach Bottom transient initiating event frequencies are calculated by performing a Bayesian update of generic industry frequencies obtained from NUREG/CR-6928 supplemented with information from NUREG/CR-5750 with Peach Bottom specific experience over the dates January 1, 2003 to December 31, 2008. This method establishes an accepted basis for the applicability of the transient initiating event frequencies utilized in the Peach Bottom PRA model.

However, sensitivity quantifications were performed that increase the turbine trip initiator frequency and loss of condenser vacuum initiating event frequency to bound the various changes to the BOP side of the plant (e.g., main turbine modifications). Additionally, potential operational issues were taken into account in the sensitivity case for the loss of feedwater transient scenario (refer to Table 5.7-1).

Loss of Offsite Power (LOOP)

No change in the Loss of Offsite Power initiating event frequency is expected. Analysis described in Sections 2.3.2 and 2.3.3 of the PUSAR indicated that the existing off-site power and on-site power systems were determined to be adequate for operation with the EPU related electrical output. The isolated phase bus duct was modified to accommodate the additional power output. Based on this analysis, there is no significant impact on grid stability due to the PBAPS EPU.

LOCAs

No changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of the EPU; as such, no impact on LOCA frequencies due to the EPU can be postulated. However, acknowledging that increased flow rates of the EPU can result in increased piping erosion/corrosion rates, a risk sensitivity case quantification is performed that increases the LOCA initiating event frequencies including main steam and feedwater line breaks (refer to Table 5.7-1).

Support System Initiators

No significant changes to support systems (e.g., AC, DC, Service Water, etc.) are planned in support of the EPU; as such, no impact on support system initiating event frequencies due to the EPU can be postulated.

Internal Flood Initiators

Since the methodology used in calculating the initiating event frequency for internal flooding is based on the length of piping found within a system and the fact that the geometry and most of the flow rates associated with the major flooding sources are not changing, the internal flooding initiator frequencies remained the same. The addition of the RHR cross-tie piping was examined for potential impact and was determined to be a very negligible contributor to the internal flood frequency in those areas. However, since the higher flow rates associated with EPU could have an impact on some of the internal flooding initiating event frequencies (e.g., steam and feedwater flow rates), a separate sensitivity evaluation was explored which conservatively increased all of the internal flood frequencies. Refer to Section 5.7.1 for results of the sensitivity cases.

External Event Initiators

The frequency of external event initiators (e.g., fires, seismic events, extreme winds) is not linked to reactor power or operation; as such, no impact on external event initiator frequencies due to the EPU can be postulated.

The CLTR states that the increase in power level could have an impact on the PBAPS PRA external events, which could impact the CDF and LERF results. However, since the frequency of external events is not affected by EPU, the potential impacts on their mitigation (fire, seismic, and other external events) are discussed in more detail in Sections 4.3, 4.4, and 4.5, respectively.

Internal Events Summary

No planned operational modifications as part of the PBAPS EPU include operating equipment beyond design ratings. However, sensitivity cases that increase transient initiating event frequencies are quantified in this EPU risk analysis to bound the various changes to the BOP side of the plant and potential operational issues (refer to Section 5.7.2).

In summary, it is anticipated that the long-term initiating event frequency is unchanged and no change is being made to the PRA initiating events in the base case analysis as a result of EPU. This is consistent with CLTR conclusions on this issue:

> *"Based on PRA experience for uprated BWRs, EPU is not expected to have a major effect on the initiating event frequencies, as long as equipment operating limits, conditions, and/or ratings are not exceeded."*

4.1.2 Success Criteria

The success criteria for the 2009A pre EPU PRA are derived based on realistic evaluations of system capability over the 24 hour mission time of the PRA analysis. These success criteria therefore may be different than the design basis assumptions used for licensing PBAPS. PRA Analyses are required to consider all proceduralized plant capabilities not limited to those credited as part of plant's design basis to obtain an accurate evaluation of risk. For example, CRD flow for injection to the RPV is credited after initial injection from HPCI or RCIC to avoid core damage in the PRA model, but this is not credited in any design basis analysis. This analysis uses the PRA to provide insights about how plant risk from postulated accidents, including severe accidents, is impacted by EPU implementation. The following subsections discuss different aspects of the success criteria as used in the PRA. Both the PBAPS EPU task reports performed by General Electric and MAAP 4.0.6 runs [19] performed for the PBAPS EPU risk assessment were used to assess impacts on success criteria.

4.1.2.1 Timing

Shorter times to boil-off are likely on an absolute basis due to the increased power levels. The reduction in timings can impact the human error probability calculations, especially for short-term operator actions. This has been directly factored into revised HEP values for EPU conditions (See HRA discussion in Section 4.1.6).

4.1.2.2 RPV Inventory Makeup Requirements

The PRA success criteria for RPV makeup remains the same for the post-uprate configuration. Both high pressure (e.g., FW, HPCI, and RCIC) and low pressure (e.g., LPCI, CS, and condensate) injection systems have more than adequate flow margin for the post-uprate configuration. This includes the EPU reduction in the maximum RHR flow rate to 10,600 GPM. RPV injection systems that were considered marginal in the pre-uprate configuration (e.g., CRD) as an independent RPV makeup source during the initial stages of an accident are still deemed marginal and are not adequate in the post-uprate configuration. However, following initial operation of another injection system, CRD remains a viable RPV makeup source at high and low pressures in the post-EPU configuration (i.e., late injection source) for certain accidents. All success criteria have been verified with MAAP4.0.6 runs for both pre-EPU and EPU conditions.

4.1.2.3 Heat Load to the Pool

Energy to be absorbed by the pool during an isolation event or RPV depressurization increases for the EPU case relative to the original license basis power level. For non-ATWS scenarios, the RHR heat exchangers, the main condenser, and the containment vent all have capacities that exceed the increase in heat load due to extended power uprating. The heat removal capability margins are sufficiently large such that the changes in power level associated with EPU do not affect the success criteria for these systems. By design, the main condenser and RHR SPC systems are sufficient for containment heat removal for the EPU condition [Refer to Section 2.6.5 of the PUSAR]. With respect to containment venting, MAAP run PB0010 shows that the emergency containment vent is clearly sufficient for the EPU conditions. Note that run PB0010 assumes loss of all injection at the time of the vent for the purposes of evaluating other accident issues, but the vent is successful in controlling containment pressure.

One change to the RHR system has been implemented regarding eliminating the need to credit containment accident pressure for design basis LOCA calculations. That is, a split flow alignment of the heat exchangers is employed in response to LOCA conditions. This has been factored into the risk assessment in the following fashion:

a) The drag valves or orifices between the RHR pump and the RHR heat exchanger are replaced with MOVs with divisional power dependencies.
b) A cross-tie MOV between the A and C RHR pumps (and the B and D RHR pumps) is included to allow for split flow from one RHR pump to discharge to both heat exchangers in the RHR loop.
c) A human error probability (HEP) has been developed to represent the human failure rate associated with aligning suppression pool cooling in a timely fashion given the conditions exist that require the cross-tie to be implemented for success of systems taking suction from the suppression pool. The initial HEP value has been derived at 6.0E-2 for implementation in the PRA model. A longer term action is also included to reflect the need to align the RHR cross-tie and throttle the flow to maintain NPSH. The HEP value associated with this action is much lower since it only includes the remaining execution steps (i.e., the cognitive contribution to the initial HEP evaluation dominates the failure probability)
d) Logic has been added to the model to include the requirement for success of the cross-tie with flow through both RHR heat exchangers in a loop for the scenarios of interest (i.e., large break LOCA initiator with coincident containment isolation failure).
e) The success criteria for other scenarios (i.e., non DBA type LOCA scenarios) remain the same in the model.

For the HPSW cross-connect, representation of the cross-connect valve was already included in the PRA model for beyond design basis events. Use of the HPSW cross-connect will come into play for those scenarios where flow from the opposite HPSW loop is required to meet the RHR heat exchanger service water flow requirements (including the cases when flow through two heat exchangers is now required).

Additionally, changes for EPU will be made to install a manual power supply transfer switch for the HPSW cross-connect valve to be powered from an alternate power supply and replace existing MOV actuators with a larger size. Credit for this manual transfer switch was conservatively not included in the risk evaluation. Otherwise, these modifications are considered to be upgrades and enhancements to existing components, which are expected to have a positive risk impact.

4.1.2.4 Blowdown Loads

Dynamic loads would increase slightly because of the increased stored thermal energy. This change would not quantitatively influence the PRA results. Analyses for LOCA under EPU conditions indicate that dynamic loads on containment remain acceptable for the EPU case [Refer to Section 2.6 of the PUSAR].

4.1.2.5 RPV Overpressure Margin

The RPV dome operating pressure will not be increased as a result of the power uprate. However, the RPV pressure following a failure to scram is expected to increase slightly. For transient scenarios, Section 2.8.4.2 of the PUSAR indicates that there is sufficient overpressure protection for transient response especially since an additional SV is being added for ATWS considerations as part of the EPU modifications. Since the dominant failure mechanism will remain as common cause failure of the SRVs (as data for group-sizes larger than eight is typically not available), there would be no change to the common cause failure contribution and any increase in the independent failure contributions to risk (not modeled) would be extremely negligible.

For ATWS scenarios, Section 2.8.5.7 of the PUSAR indicates that with the incorporation of an additional safety valve and with changes to the RPT system that allow for quicker trip of the recirculation pump trips, there is actually more margin to the ASME Service Level C peak RPV pressure criterion. As such, there is no change warranted to the overpressure success criteria for ATWS scenarios.

The 2009A pre EPU PBAPS PRA does not require any SRVs for initial RPV overpressure control for LOCA initiators. This success criterion also remains unchanged for the EPU.

As such, no model changes to the PBAPS PRA regarding this function are required for this EPU risk assessment.

4.1.2.6 SRV Actuations

The SRV setpoints have not been changed as a result of the PBAPS EPU. Given the power increase of the EPU, one may postulate that the probability of a stuck open relief valve given a transient initiator would increase due to an increase in the number of SRV cycles.

The 2009A PRA base stuck open relief valve probability may be modified using different approaches to consider the effect of a postulated increase in valve cycles. The following three approaches are considered:

1. The upper bound approach would be to increase the stuck open relief valve probability by a factor equal to the increase in reactor power (i.e., a factor of 1.125 in the case of the PBAPS EPU). This approach assumes that the stuck open relief valve probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power increase.

2. A less conservative approach to the upper bound approach would be to assume that the stuck open relief valve probability is linearly related to the number of SRV cycles, but the number of cycles is not necessarily directly related to the reactor power increase. In this case the postulated increase in SRV cycles due to the EPU would be determined by thermal hydraulic calculations (e.g., MAAP runs).

3. The lower bound approach would be to assume that the stuck open relief valve probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate. In this approach the base stuck open relief valve probability could be assumed to be insignificantly changed by a postulated increase in the number of SRV cycles.

Approach #1 is used to modify the PRA stuck open relief valve probability. The SORV probability basic events in the PBAPS PRA are increased 12.5% for the EPU base case risk evaluation:

BE ID	Description	Pre-EPU Probability	EPU Probability
APHSRVTMDX I2	SRVS FAIL TO RECLOSE / MSIV CLOSURE EVENTS	1.90E-3	2.14E-3
APHSRVTTDXI 2	SRVS FAIL TO RECLOSE / TT OR TF EVENTS	2.40E-4	2.70E-4

Note that for ATWS scenarios, even though an extra safety valve is available, the fail to reclose probability of the SRVs is based on all 11 SRVs being challenged. This would not change for EPU such that a PRA model change is not warranted. Additionally, it is noted that the stuck open relief valve (SORV) probability for ATWS scenarios is not a risk significant contributor to the PRA model results. As such, any postulated change to the SORV probability for ATWS scenarios due to EPU would result in a negligible change to the CDF and LERF risk metrics.

4.1.2.7 RPV Emergency Depressurization

The current 2009A PRA requires two SRVs for RPV emergency depressurization. MAAP cases performed in support of this EPU risk assessment show that this success criterion remains unchanged by the EPU. Therefore, the PRA success criterion of 2 SRVs is maintained in this analysis. Note however, that there are some timing differences related to maintaining this requirement, and these have been factored into the human error probabilities for emergency depressurization as described in Section 4.1.6 below.

4.1.2.8 Structural Evaluations

This assessment did not identify issues associated with postulated impacts from the EPU on the PRA modeling of structural (e.g., piping, vessel, containment) capacities. This is consistent with CLTR conclusions on this issue [16]:

> *"The RPV is analyzed for power uprate conditions. Transients, accident conditions, increased fluence, and past operating history are considered to recertify the vessel. Plant specific analyses at power uprate conditions demonstrates that containment integrity will be maintained."*
>
> *"... no significant effect on LOCA probability. Increase in flow rates is addressed by compliance with Generic Letter 89-08, Erosion/Corrosion in Piping..."*

4.1.2.9 Success Criteria Summary

The PRA success criteria are affected by the increased boil off rate, the increased heat load to the suppression pool, and the increase in containment pressure and temperatures.

MAAP runs demonstrate the significant margins associated with the installed systems. However, MAAP runs did indicate the impact of EPU on timing for achieving success. The impact of these timing changes is then reflected in the human error probabilities developed for the PB209A1 EPU PRA model. The impact of these changes on the human reliability analysis is described in more detail in Section 4.1.6 below.

Besides the change to the RHR heat exchanger alignment under certain situations, the changes to the SORV probabilities, the changes for the CST standpipe, and the timing issues described above, no other changes in the modeled success criteria have been identified for the Level 1 or Level 2 PRA.

This assessment is consistent with CLTR conclusions on this issue:

> *"Based on PRAs done for other uprated plants, EPU is not expected to have a major impact on the PRA success criteria."*

The changes described above and in the operator response section below are directly factored into the risk assessment and the changes to CDF and LERF are reported.

4.1.3 Accident Sequence Modeling

For the most part, the EPU does not change the plant configuration or operation in a manner such that new accident sequences or changes to existing accident scenario progressions result.

This assessment for PBAPS is consistent with CLTR conclusions on this issue [16]:

> *"The basic BWR configuration, operation and response is unchanged by power uprate. Generic analyses have shown that the same transients are limiting. ... Plant-specific analyses demonstrate that the accident progression is basically unchanged by the uprate."*

One exception is the reduction in available accident progression timing for some scenarios and the associated impact on operator action HEPs (this aspect is addressed in the Human Reliability Analysis section). The other exception for PBAPS is the need to align the cross-tie valve for the

RHR system to eliminate the need for crediting containment accident pressure under certain conditions as described in Section 4.1.2.3 above.

Another aspect of the accident sequence modeling to consider is the impact on LOOP recovery times. Note that the short term LOOP response is driven by things such as battery capacity that are not being affected by the EPU. However, the longer term LOOP response is sensitive to EPU since it is partially based on the time to reach containment venting conditions which is a direct function of the decay heat level. There are two longer term time frames utilized in the PRA model (10 hours for LOCA cases and 20 hours for non-LOCA cases). These broad categories are not based on any one MAAP run, but a series of potentially representative runs. MAAP runs [19] for EPU conditions indicate that about 10-15% timing change can be anticipated depending on various aspects of the accident sequence progression. To account for this, the LOOP failure to recover probability basic events in the PBAPS PRA are adjusted to account for 15% less time available for the EPU base case risk evaluation. Note that the reported times in the basic event descriptions are not changed for this initial EPU assessment, but the values are changed to reflect the reductions in times that would be available.

BE ID	Description	Pre-EPU Probability	EPU Probability
NOOSP2010-GRID	FAILURE TO RECOVER OSP AT 20 HRS / NO RECOVERY AT 10 HRS - GRID RELATED LOOP	0.189	0.205
NOOSP2010-PLANT	FAILURE TO RECOVER OSP AT 20 HRS / NO RECOVERY AT 10 HRS - PLANT RELATED LOOP	0.203	0.214
NOOSP2010-SWYD	FAILURE TO RECOVER OSP AT 20 HRS / NO RECOVERY AT 10 HRS - SWITCHYARD RELATED LOOP	0.219	0.232
NOOSP2010-WTHR	FAILURE TO RECOVER OSP AT 20 HRS / NO RECOVERY AT 10 HRS - WEATHER RELATED LOOP	0.596	0.612
NOOSP105-GRID	FAILURE TO RECOVER OSP IN 10 HRS / NO RECOVERY IN 5 HRS - GRID RELATED LOOP	0.275	0.388
NOOSP105-PLANT	FAILURE TO RECOVER OSP IN 10 HRS / NO RECOVERY IN 5 HRS – PLANT RELATED LOOP	0.264	0.373

NOOSP105-SWYD	FAILURE TO RECOVER OSP IN 10HRS / NO RECOVERY IN 5 HRS - SWITCHYARD RELATED LOOP	0.287	0.398
NOOSP105-WTHR	FAILURE TO RECOVER OSP IN 10HRS / NO RECOVERY IN 5 HRS - WEATHER RELATED LOOP	0.656	0.731

Similarly, the AC power non-recovery probabilities utilized in the Level 2 analysis are also adjusted to account for less time available to recover off-site power to prevent vessel failure. The adjusted conditional Level 2 model non-recovery probabilities are shown below.

BE ID	Description	Pre-EPU Probability	EPU Probability
2RX-IBE-OPFPLANT	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT HIGH PRESSURE) FOR CLASS IBE – PLANT RELATED LOOP	0.224	0.305
2RX-IBE-OPF-SWYD	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT HIGH PRESSURE) FOR CLASS IBE – SWITCHYARD RELATED LOOP	0.263	0.350
2RX-IBE-OPF-GRID	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT HIGH PRESSURE) FOR CLASS IBE – GRID RELATED LOOP	0.314	0.410
2RX-IBE-OPF-WTHR	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT HIGH PRESSURE) FOR CLASS IBE – WEATHER RELATED LOOP	0.630	0.694
2RX-IBE-OPSPLANT	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT LOW PRESSURE) FOR CLASS IBE – PLANT RELATED LOOP	0.170	0.224
2RX-IBE-OPS-SWYD	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT LOW PRESSURE) FOR CLASS IBE – SWITCHYARD RELATED LOOP	0.205	0.263
2RX-IBE-OPS-GRID	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT LOW PRESSURE) FOR CLASS IBE – GRID RELATED LOOP	0.246	0.314

2RX-IBE-OPS-WTHR	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE (RPV AT LOW PRESSURE) FOR CLASS IBE – WEATHER RELATED LOOP	0.579	0.630
2RX-IBL-OSPPLANT	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE FOR CLASS IBL – PLANT RELATED LOOP	0.329	0.418
2RX-IBL-OSP-SWYD	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE FOR CLASS IBL–SWITCHYARD RELATED LOOP	0.354	0.442
2RX-IBL-OSP-GRID	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE FOR CLASS IBL – GRID RELATED LOOP	0.342	0.434
2RX-IBL-OSP-WTHR	OFFSITE POWER NOT RECOVERED IN TIME TO PREVENT VESSEL FAILURE FOR CLASS IBL – WEATHER RELATED LOOP	0.703	0.759

4.1.4 System Modeling

For the most part, the PBAPS plant changes associated with the EPU do not result in the need to change any system modeling in support of this risk assessment. One exception is the addition of the cross-tie valve for the RHR system to eliminate the need for crediting containment accident pressure under certain conditions. The impact on the PRA modeling from this modification was described above in Section 4.1.2.3.

Another system modification that required changes to the PRA system modeling was the addition of the CST standpipe for the hotwell reject/makeup line nozzle. This impacted the system modeling as follows:

a) First, it eliminates the potential scenario that inadvertently drains the CST to the hotwell. This is implemented in the EPU PRA model by eliminating the logic gates that included this diversion path for water in the CST.

b) Second, however, since this line also provides the suction source for CRD, the available CST inventory to support CRD injection is reduced. Since the standpipe elevation would require that CST makeup be provided from the RWST or TDT via use of the refueling water transfer pumps (i.e. gravity drain from those tanks would not be viable to support extended CRD injection to the RPV), then credit for gravity drain was removed from the model.

4.1.5 Data Analysis (Component Reliability)

The CLTR states that the minimum acceptable required system or component capability may increase as a result of the increased power level, which may affect the system or component reliability and CDF and LERF results.

However, EPU will not significantly impact the reliability of equipment. The majority of the hardware changes in support of the EPU may be characterized as either:

- Replacement of components with enhanced like components
- Upgrade of existing components

Although equipment reliability as reflected in failure rates can be theoretically postulated to behave as a "bathtub" curve (i.e., the beginning and end of life phases being associated with higher failure rates than the steady-state period), no significant effect on the long-term average of initiating event frequencies, or equipment reliability during the 24 hr. PRA mission time due to the replacement/modification of plant components is anticipated, nor is such a quantification supportable at this time. No planned operational modifications as part of the PBAPS EPU include operating equipment beyond design ratings. Therefore, no significant effect on the long-term average failure rates (initiating events and equipment reliability) due to replacement/modification of components is anticipated. If any degradation were to occur as a result of EPU implementation, existing plant monitoring programs would address any such issues. This assessment is consistent with CLTR conclusions on this issue [18]:

> *"...CPPU is not expected to have a major effect on component or system reliability, as long as equipment operating limits, conditions, and/or ratings are not exceeded."*

Additionally, it is noted that minor variations in system or component design response times that may be postulated or planned due to the EPU would not impact the PRA risk profile. A review of the PBAPS EPU System Task Reports that affect systems modeled in the PRA was performed. These task reports identify the EPU effects on the subject system. There are no significant changes to system and component response times due to the EPU for any of these systems, and thus, there is no impact on the PRA risk profile or EPU risk assessment.

4.1.6 Human Reliability Analysis (Operator Response)

The CLTR states that the increase in power level results in changes to event dynamics.

The PBAPS risk profile, like other plants, is dependent on the operating crew actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance shaping factor that is principally influenced by the power uprate is the time available within which to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions.

MAAP calculations for the PBAPS EPU configuration were performed to determine how the operator action timelines were impacted. All the post-initiator human error probabilities (HEPs) in the model were then re-calculated using the same human reliability analysis (HRA) methods used in the PBAPS HRA document. Refer to Table 4.1-2 for a summary of the changes in operator action timings and associated HEPs due to the EPU. Table 4.1-3 includes the corresponding changes to the human reliability dependency analysis. Application specific model documentation, PB-ASM-001 [19] provides detailed documentation of the impact of EPU on the HRA. The methodology employed for the derivation of the post-initiator HEPs in the Peach Bottom HRA is described below.

One additional action was added to the model related to aligning an RHR cross-tie to two RHR heat exchangers supported by two HPSW pumps to allow operation of the pumps with suction from the suppression pool without crediting containment accident pressure. Specific control room indications that will be made to support the use of the RHR cross-tie and HPSW cross-connect modifications are subsumed in the human reliability assessment for those actions.

Other than that, no significant changes are to be made to the Control Room for the EPU that would impact the existing actions included in the PBAPS PRA human reliability analysis. Potential changes to be made to the Control Room displays for the EPU are re-scaling certain indicators/recorders and/or replacement of certain indicators with digital units. None of these Control Room display changes will have a measurable impact on the human reliability analysis for the PBAPS PRA. However, the changes that were identified to the HEP values as identified in Tables 4.1-2 and 4.1-3 are factored directly into the risk assessment and the changes to CDF and LERF are reported.

Post-Initiator HRA Methodology

The human error probability values for the new Human Failure Events (HFEs) for aligning the RHR cross-tie and all other risk significant post-initiator actions (where significant HFEs are defined here as having a risk achievement worth (RAW) greater than or equal to 2.0 or a risk reduction worth (RRW) greater than or equal to 1.005) are derived from a combination of analytical methods. The EPRI Cause Based Method [25] and the ASEP HRA time reliability correlation [26] procedure have been chosen as the bases for determining the non-response probabilities (P_C) for Peach Bottom analysis. The execution error (P_E) is derived using the NUREG/CR-1278 [27] HRA procedure called Technique for Human Error Rate Prediction (THERP). The final HEP value utilized in the PRA model includes both components of the post-initiator HEPs ($P_C + P_E$).

The cause-based approach involves the identification of situation-specific error factors. The approach is one of decomposition, consisting of identifying potential failure mechanisms and, for each mechanism, identifying specific causes of human error, evaluating the impact of certain performance shaping factors on a human action specific basis, and also allowing for potential recovery mechanisms. This is essentially an analytical approach, as opposed to the empirical approach represented by the use of human reliability curves. The actions for which the cause-based approach is used exclusively are generally not time limited. Available time is considered primarily in the application of the recovery factors, whose impact is considered to be time dependent. The cause-based evaluation is performed for each Peach Bottom post-initiator human action identified for evaluation.

The ASEP HRA Time Reliability Correlation Procedure is a shortened version of the procedure, models, and data for HRA that are presented in the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (NUREG/CR- 1278). This procedure was developed to enable system-knowledgeable personnel to perform an effective analysis. The time dependent non-response probabilities (Pc) from the methodology are applied according to its basic principles for short-term actions (time available for diagnosis, T_d <1 hour) in order to compensate for possible non-conservative estimates produced by the cause-based method. The non-response probability for short-term action is taken to be the sum of the cause-based and ASEP results; longer term actions (time available for diagnosis, T_d >=1 hour) do not include the ASEP component.

4.1.7 Internal Flooding

No changes in the internal flooding modeling were incorporated based on EPU. The initiating event frequencies and impact vectors (i.e., the affected equipment from the flood event) from the flooding analysis are unchanged from EPU. Any changes in the overall contribution from flooding would be related to other modeling changes (e.g., HEP changes).

4.1.8 Quantification

No changes in the PBAPS PRA quantification process (e.g., truncation limit, etc.) due to the EPU have been identified (nor were any anticipated). Changes in the quantification results (accident sequence frequencies) were realized as a result of the minor modeling changes described above.

4.1.9 Level 2 PRA Analysis

The Level 2 PRA framework, functional fault trees, and Level 2 basic event failure probabilities remain unchanged in the transition from pre-EPU to EPU.

Fission product inventory in the reactor core is higher as a result of the increase in power due to the EPU. The increase in fission product inventory results in an increase in the total radioactivity available for release given a severe accident. The total activity available for release is approximately 12.5% higher. However, this does not impact the definition or quantification of the LERF risk measure used in Regulatory Guide 1.174, and as the basis for this risk assessment. The PBAPS PRA release categories are defined based on the percentage (as a function of EOC inventories) of CsI released to the environment, this is consistent with most industry PRAs.

Given the minor change in Level 1 results, minor changes in the Level 2 release frequencies can be anticipated. Such changes are directly attributable to the changes described previously and the minor changes in short term accident sequence timing and the impact on HEPs. The structure of the accident sequence modeling in the Level 2 PRA is not impacted by the EPU. MAAP4.0.6 calculations for pre-EPU and EPU conditions showed that although variations in the absolute magnitude of the releases may occur and reductions in the calculated times between the declaration of a General Emergency and the time of first fission product release to the environment may occur, neither of the differences would be sufficient to alter the assigned release categories in the Level 2 containment event trees.

Although radiological source terms might be higher from EPU power levels, the definition of LERF in the PBAPS PRA is based on fractional releases which do not change. The PBAPS PRA does not include a Level 3 model and this is not explicitly required to be evaluated for EPU.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A1	AHU--650DXI2	OPERATOR FAILS TO DEPRESSURIZE TO 650 PSIG FOR CONDENSATE INJECTION	3.8E-03	**2.2E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 23 to 19 minutes).
A2	AHUALTDPDXI2	FAILURE TO OPEN NON-ADS SRVS OR TBP VALVES	2.5E-02	**3.5E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 12.4 to 10.9 minutes).
A3	AHU-ATWSDXI2	FAILURE TO EMERG DEPRESSURIZE AFTER HPI FAILS IN ATWS	1.0E-02	**1.3E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 12.4 to 10.9 minutes).
A50	AHU--BCIDXI2	OPERATOR FAILS TO BYPASS CONTAINMENT ISOLATION	3.5E-03	**2.3E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 23 to 19 minutes).
A4	AHUBTL-RDXI2	OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR)	1.3E-03* (1.9E-03)	**1.4E-03**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes). Also, MAAP basis changed from PB0005 to PB0007 as more representative.

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[1] Pre-EPU HEPs marked with an "*" symbol have been updated to be consistent with the EPU HRA and to maintain an appropriate basis for comparison. Prior values are shown in parenthesis.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A4	AHUBTL-RDXD2	OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) - LATE, CONDITIONAL.	1.0* (0.68)	**9.3E-01**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes). Also, MAAP basis changed from PB0005 to PB0007 as more representative.
A5	AHU--CADDXI2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	1.1E-02* (3.8E-02)	**2.3E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes). Also, removed over conservatism of doubling the manipulation time estimate.
A6	AHU--FINDXI2	OPERATOR FAILS TO INHIBIT ADS	2.1E-03* (2.4E-03)	**2.9E-03**	EPU conditions reduce system window (i.e., the end of system window reduced from 21.5 to 17.6 minutes). Also, removed over conservatism of delay time to recognize cue.
A7	AHU--INFDXI2	FAILURE TO INHIBIT ADS IN ATWS WITH FEEDWATER AVAILABLE	1.5E-02* (2.1E-03)	**1.6E-02**	On re-examination of SP conditions, it was determined that level reduction to -172" would be required on high SP temp. Action to inhibit ADS now required by 12.5 minutes for pre EPU and 12.2 minutes for EPU.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A8	AHU--INXDXI2	FAILURE TO INHIBIT ADS IN ATWS W/O FEEDWATER AVAILABLE	2.1E-02* (1.3E-02)	**2.2E-02**	EPU conditions reduced the time available for response, but the basis was also changed from high DWP/low level to low level with high DWP bypass (10.5 minutes) as it is more limiting.
A10	AHU--SS1DXI2	OPERATOR FAILS TO INITIATE EMERGENCY DEPRESSURIZE (STEAM MEDIUM LOCA)	3.7E-04	**4.4E-04**	EPU conditions reduce system window (i.e., the end of system window reduced from 37 to 32 minutes).
A10	AHU--WS1DXI2	OPERATOR FAILS TO INITIATE EMERGENCY DEPRESSURIZE (WATER MEDIUM LOCA)	1.8E-02	No change	PBAPS MAAP calculations bound the relevant LGS calculation that is used as the system window basis. EPU conditions would impact the scenario slightly, but the break size is considered to dominate the results and the PBAPS bounding cases imply the LGS calc is still valid, so the LGS calc has been retained as the basis (10 minutes to CD).
A11	AHU--XTEDXI2	FAILURE TO INITIATE MANUAL DEPRESSURIZATION (LOOP CASES)	3.2E-04	**3.3E-04**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes).

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A11	AHU--XTRDXI2	FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES)	3.2E-04	**3.3E-04**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes).
A9	BHU--MAXDXI2	FAILURE TO MAXIMIZE CRD FLOW PER T-246	2.1E-02	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A12	BHU-PUMPDXI2	FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP	6.9E-03	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A13	DHU--DWSDXI2	OPERATORS FAIL TO UTILIZE DWS FOR DHR	7.6E-04	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A45	DHU-LEAKDXI2	OPERATOR FAILS TO STOP LEAK FLOOD BY TRIPPING PUMP	9.4E-04	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A45	DHU-RUPTDXI2	OPERATOR FAILS TO STOP RUPTURE FLOOD BY TRIPPING PUMP	5.7E-02	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A14	DHU--SDCDXI2	FAILURE OF OPERATOR TO INITIATE RHR/SDC	2.9E-03	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A15	DHU--SPADXI2	FAILURE OF OPERATOR TO INITIATE RHR/SPC(ATWS)	2.0E-03	**2.3E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 30 to 25 minutes).
A16	DHU--SPCDXD2	CONDITIONAL FAILURE OF OPERATOR TO INITIATE SPC/SDC - LATE	2.5E-02	**5.0E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 22.9 to 18.8 hours.) This changes the long term HEP that is the basis for this conditional HEP.
A17	DHU--SPCDXI2	FAILURE OF OPERATOR TO INITIATE RHR/SPC	2.2E-04	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A56	DHU--SPXDXD2	OPERATORS FAIL TO ALIGN RHR PUMP DISCHARGE X-TIE FOR SPC	NA	**5.5E-5**	New action to allow operation of pumps taking suction from the SP without crediting containment accident pressure for NPSH.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A57	DHU--SPXDXI2	OPERATORS FAIL TO INITIATE ONE TRAIN OF RHR IN SUPPRESSION POOL COOLING MODE BEFORE CROSS-TIE	NA	**6.0E-2**	New action to allow operation of pumps taking suction from the SP without crediting containment accident pressure for NPSH.
A54	EHUCHGERDXI0	OPERATOR FAILS TO PERFORM FAST TRANSFER OF BATTERY CHARGERS	2.0E-03	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A18	EHUCWGCNDXI0	CWG OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	4.4E-02	No change	The timing basis governed by the AC power recovery interval and the HEP was not impacted.
A19	EHUCWGPBDXI0	PB OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	2.7E-02	No change	The timing basis governed by the AC power recovery interval and the HEP was not impacted.
A20	EHU-LOCADXI0	FAILURE TO START DIESEL AFTER NO AUTO-INIT (LOCA)	1.2E-01	**2.0E-01**	EPU conditions reduce system window (i.e., the end of system window reduced from 7.1 to 6.4 minutes).
A20	EHU-LOOPDXI0	FAILURE TO START DIESEL AFTER NO AUTO-INIT (NO LOCA)	4.5E-04	**4.7E-04**	EPU conditions reduce system window (i.e., the end of system window reduced from 47.9 to 40.6 minutes).
A21	EHU-SE11DXI0	FAILURE TO X-TIE EMERGENCY AC POWER PER SE-11	1.9E-02	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A22	FHUBLMSVDXI2	FAIL TO BYPASS THE MSIV RPV LOW LEVEL INTERLOCK (LEVEL 1)	1.0E+00	No change	No credit taken for the action.
A55	FHULEVELDXI2	OPERATORS FAIL TO TAKE MANUAL CONTROL OF FW	3.2E-02	No change	For overfill prevention, higher power would increase the time available for action. In this case, the assumed 15 minute system window was not increased and the HEP was not impacted.
A23	HHUCSTSPDXI2	FAILURE OF OPERATOR TO MANUALLY TRANSFER WATER SOURCES	2.3E-02	No change	Assumed HPCI makeup rate was not changed for EPU conditions and HEP was not impacted.
A24	HHU--HLTDXI2	OPERATOR FAILS TO TRIP HPCI ON HIGH LEVEL	8.4E-02	**2.1E-02**	EPU conditions increase the system window (i.e., the end of system window increased from 13.7 to 20.1 minutes). This system window is increased because the higher power results in an increased boil off rate, which competes with the flow into the RPV.
A47	JHU-2344DXI2	OPERATOR FAILS TO CORRECTLY ALIGN CROSS CONNECT	2.6E-03	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A25	JHU--ECTDXI2	OPERATOR FAILS TO CORRECTLY ALIGN HPSW FOR COOLING TOWER FLOW	1.7E-03	3.4E-3	EPU conditions reduce system window (i.e., the end of system window reduced from 10.3 to 7.9 minutes). In addition, it was determined that the pre-EPU time to HCTL was 9.4 hours instead of 10.3 hours. Update of the pre-EPU timing does not impact the HEP.
A26	JHUHWINJDXD 2	OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE)	4.8E-02	**2.0E-02**	EPU conditions result in earlier containment failure (41.1 hours instead of 51.2 hours). As this is the cue for the action, the system window is significantly expanded. CRD remains operable to the time of core damage, which is reduced from 59.6 to 53.5 hours. The net impact of the changes is an increased diagnosis time and lower HEP.
A26	JHUHWINJDXI2	OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY)	4.4E-02	**5.6E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 37 to 32 minutes).
A27	MHUNOLPWDXI 2	TSC FAILS TO GUIDE OPS TO SE-11, ATTACHMENT W WHEN NO LOOP	1.0E+00	No change	No credit taken for the action.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A28	MHUSE11WDXI2	OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W	5.8E-02	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A29	OHU-ECCIDXI2	OPERATOR FAILS TO LINE UP INJECTION BEFORE VENTING	1.0E+00	No change	No credit taken for the action.
A23	RHUCSTSPDXI2	HUMAN ERROR FAILURE TO TRANSFER IN TIME CST/SP	2.3E-02	No change	Assumed RCIC makeup rate was not changed for EPU conditions and HEP was not impacted.
A30	RHU--HLTDXI2	OPERATOR FAILS TO TRIP RCIC ON HIGH LEVEL	9.9E-04	No change	EPU conditions altered both the time of the cue (1.5 hr to 1.8 hr) to and the end of the system window (2 hr to 2.3 hr), but the diagnosis time remained constant at 0.5 hours.
A31	SHU--SLCDXD2	FAILURE TO INITIATE SLC / ISOLATE RWCU (LATER)	1.0E-01	No change	This is a short term action and EPU conditions did reduce the diagnosis time, but this action represents only the execution portion of the action. The CBDT timeframe, which governs execution recovery, was not changed, so the HEP was not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A31	SHU--SLCDXI2	FAILURE TO INITIATE SLC / ISOLATE RWCU (EARLY)	4.3E-03	No change	EPU conditions did reduce the diagnosis time, but this action represents only the execution portion of the action. The CBDT timeframe, which governs execution recovery, was not changed, so the HEP was not impacted.
A49	UHU-2803LPD2	OPERATORS FAIL TO LOCALLY OPEN MOV-2803 TO ALIGN HPSW TO ECT COOLING	2.7E-03	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A32	UHU--ECTDXI0	OPERATOR FAILS TO PROPERLY ALIGN FOR ECT OPERATION	2.3E-03	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A48	UHUSLUCELND0	OPERATORS FAIL TO LOCALLY CLOSE SLUICE GATES 23427A(B) OR 33427A(B)	2.0E-02	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A33	VHU--LCEDXI2	FAILURE TO CONTROL RPV LEVEL WITH LP ECCS W/ HPCI FAILED	1.1E-01	**1.6E-01**	EPU conditions reduce system window (i.e., the end of system window reduced from 5.8 to 5.0 minutes).
A33	VHU--LCLDXI2	FAILURE TO CONTROL RPV LEVEL WITH LP ECCS AFTER HPCI OK	9.8E-03	**1.2E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 14.9 to 13.3 minutes).

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A34	VHU-VENTDXI2	OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	1.3E-02	**1.4E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 22.9 to 18.9 minutes).
A32	WHU-2209DXI0	OPERATOR ERROR ESW PUMP BAY CROSSTIE FTO	2.6E-04	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A35	WHU--ESWDXD0	FAILURE TO START ESW PUMP LATER	9.5E-02	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A35	WHU--ESWDXI0	FAILURE TO START ESW PUMP EARLY	7.3E-02	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A36	WHU--NSWDXD2	OPERATORS FAIL TO START STANDBY SW PUMP (LATER)	4.0E-03	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A36	WHU--NSWDXI2	OPERATORS FAIL TO START STANDBY SW PUMP (EARLY)	1.0E+00	No change	The timing basis is unrelated to reactor power and the HEP was not impacted.
A37	YHU--CSTDXI2	OPERATORS FAIL TO REFILL CST FROM RWST (RW PUMPS)	9.1E-03	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A38	YHU--GRFDXI2	OPERATORS FAIL TO REFILL CST FROM RWST (GRAVITY FEED)	1.4E-02	No change; however set to 1.0 based on incorporation of CST standpipe that would make gravity feed unfeasible	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A52	YHUGRTDTDXI 2	OPERATOR FAILS TO REFILL UNIT 2 CST VIA TDT GRAVITY FEED	2.1E-02	No change; however set to 1.0 based on incorporation of CST standpipe that would make gravity feed unfeasible	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A53	YHUGRTDTDXI 3	OPERATOR FAILS TO REFILL UNIT 3 CST VIA TDT GRAVITY FEED	5.4E-03	No change; however set to 1.0 based on incorporation of CST standpipe that would make gravity feed unfeasible	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.
A51	YHU--TDTDXI2	OPERATORS FAIL TO REFILL CST FROM TDT (RW PUMPS)	3.3E-02	No change	EPU conditions reduce system window; however, the diagnosis time remains in the same CBDT time frame and the HEP is not impacted.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A39	ZHUFWHPLDXD2	EXECUTION ERROR FOR LEVEL / POWER LATER IN AN ATWS	1.0E-01	No change	This is a short term action and EPU conditions did reduce the diagnosis time, but this action represents only the execution portion of the action. The CBDT timeframe, which governs execution recovery, was not changed, so the HEP was not impacted.
A39	ZHUFWHPLDXI2	EXECUTION ERROR FOR LEVEL / POWER EARLY IN AN ATWS	2.3E-02	No change	This is a short term action and EPU conditions did reduce the diagnosis time, but this action represents only the execution portion of the action. The CBDT timeframe, which governs execution recovery, was not changed, so the HEP was not impacted.
A40	ZHU-HIGHDXI2	FAILURE TO MANUALLY INITIATE HPCI/RCIC INJECTION	2.2E-03	**3.2E-03**	EPU conditions reduce system window (i.e., the end of system window reduced from 21.5 to 17.6 minutes).
A41	ZHU--HRLDXI2	OPERATOR FAILS TO TAKE MANUAL CONTROL OF HPCI/RCIC -EARLY	4.6E-02	**4.0E-02**	EPU conditions altered both the time of the cue (3 to 1.6 min) to and the end of the system window (9.8 to 9 min), but the diagnosis time increase, so the HEP was slightly reduced.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A42	ZHU--LPADXI2	FAILURE TO MANUALLY INITIATE LOW PRESS ECCS (ATWS OR LOCA)	4.6E-02	**6.0E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 7.1 to 6.4 minutes).
A42	ZHU--LPIDXI2	FAILURE TO MANUALLY INITIATE LOW PRESS ECCS (TRANSIENT)	8.3E-04	**1.2E-03**	EPU conditions reduce system window (i.e., the end of system window reduced from 49.6 to 41.3 minutes).
A46	ZHULVCLCDXI2	OPERATOR FAILS TO CONTROL RPV LEVEL ADEQUATELY WITH LPI (NOT TOO LOW, LLOCA)	1.5E-03	**1.7E-03**	EPU conditions reduce system window (i.e., the end of system window reduced from 22.1 to 21.4 minutes).
A43	ZHULVCTRDXI2	OP FAILS TO CONTROL LEVEL IN A TRANS W/ ECCS INJECTION	2.6E-04	No change	EPU conditions reduce system window (i.e., the end of system window reduced from 51.8 to 42.1 minutes); however, the contribution from ASEP is negligible and the change does not impact the total HEP.
A44	ZHUPWLVLDXD2	COGNITIVE ERROR FOR LEVEL / POWER LATER IN AN ATWS	6.7E-02	**5.8E-02**	There was no change to the system window for this action, but because it is developed as a conditional probability and because the early time frame HEP changed, this HEP also changed.

TABLE 4.1-2
PEACH BOTTOM INDEPENDENT POST-INITIATOR HEP RESULTS SUMMARY FOR PRE-EPU AND EPU CONDITIONS

CALC NUMBER	PBAPS BE ID	PBAPS ACTION DESCRIPTION	PRE-EPU HUMAN ERROR PROBABILITY[1]	EPU HUMAN ERROR PROBABILITY	EPU BASIS
A44	ZHUPWLVLDXI2	COGNITIVE ERROR FOR LEVEL / POWER EARLY IN AN ATWS	4.3E-02	**5.0E-02**	EPU conditions reduce system window (i.e., the end of system window reduced from 6.0 to 5.6 minutes).

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• AHU--XTEDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (LOOP CASES), • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC	• 3.2E-04 • (zero) 2.2E-04	ZHU--ADLDXI	1.0E-6	No Change
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES), • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC	• 3.2E-04 • (zero) 2.2E-04	ZHU--ADTDXI	1.0E-6	No Change
• AHU--XTEDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (LOOP CASES), • ZHU-HIGHDXI: FAILURE TO MANUALLY INITIATE HPCI/RCIC INJECTION	• 3.2E-04 • (medium) 2.2E-03	ZHU--AHLDXI	4.8E-5	No Change
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES), • ZHU-HIGHDXI: FAILURE TO MANUALLY INITIATE HPCI/RCIC INJECTION	• 3.2E-04 • (medium) 2.2E-03	ZHU--AHTDXI	4.8E-5	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• YHU--CSTDXI: OPERATORS FAIL TO REFILL CST FROM RWST (RW PUMPS) • YHU--GRFDXI: OPERATORS FAIL TO REFILL CST FROM RWST (GRAVITY FEED) • YHUGRTDTDXI: OPERATOR FAILS TO REFILL UNIT 2 CST VIA TDT GRAVITY FEED • YHU--TDTDXI: OPERATORS FAIL TO REFILL CST FROM TDT (RW PUMPS)	• 9.1E-03 • (high) 1.4E-02 • (high) 5.4E-03 • (complete) 3.3E-02	ZHU--CSTDXI	2.4E-3	No Change
• AHU--CADDXI: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B', • AHUBTL-RDXI: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) • AHUBTL-RDXD: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) - LATE, CONDITIONAL • IHURESETDXI: OPERATORS FAIL TO RESET COMPRESSOR AFTER TRIP ON LOOP • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC	• 1.1E-02 • (high) 1.3E-03 • (conditional) 1.0 • (medium) 1.0E-01 • (zero) 2.2E-04	ZHU-AADIDXI	1.0E-6	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• AHU--XTEDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (LOOP CASES) • BHU-PUMPDXI: FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC	• 3.2E-04 • (zero) 6.9E-03 • (zero) 2.2E-04	ZHU--ABDDXI	1.0E-6	No Change
• AHU--XTEDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (LOOP CASES) • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC, • MHUSE11WDXI: OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W	• 3.2E-04 • (zero) 2.2E-04 • (zero) 5.8E-02	ZHU--ADMDXI	1.0E-6	No Change
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU--XTEDXI) • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC, • ZHU--CSTDXI: OPERATORS FAIL TO REFILL CST FROM RWST (ANY MEANS)	• 3.2E-04 (or 3.2E-04) • (zero) 2.2E-04 • (zero) 3.3E-03	ZHU--ADYDXI	1.0E-6	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU--XTEDXI) • JHU--ECTDXI: OPERATOR FAILS TO CORRECTLY ALIGN HPSW FOR COOLING TOWER FLOW	• 3.2E-04 (or 3.2E-04) • (zero) 1.7E-03	ZHU---AJDXI	1.0E-6	No Change
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU--XTEDXI) • DHU-LEAKDXI: OPERATOR FAILS TO STOP LEAK FLOOD BY TRIPPING PUMP	• 3.2E-04 (or 3.2E-04) • (zero) 9.4E-04	ZHU-ADLKDXI	1.0E-6	No Change
• ZHU--HRLDXI: OPERATOR FAILS TO TAKE MANUAL CONTROL OF HPCI/RCIC -EARLY • HHU--HLTDXI: OPERATOR FAILS TO TRIP HPCI ON HIGH LEVEL (or RHU--HLTDXI)	• 4.6E-02 • (high) 8.4E-02	ZHU--HRZDXI	2.5E-2	2.0E-2

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • BHU-PUMPDXI: FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP • JHUHWINJDXI: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY) • JHUHWINJDXD: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE)	• 2.2E-04 • (zero) 6.9E-03 • (medium) 4.4E-02 • (conditional) 4.8E-02	ZHU--BDJDXI	5.0E-7	No Change
• BHU-PUMPDXI: FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	• 6.9E-03 • (zero) 2.2E-04 • (medium) 1.3E-02	ZHU--BDVDXI	5.0E-7	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • JHUHWINJDXI: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY) • JHUHWINJDXD: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE) • MHUSE11WDXI: OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W	• 2.2E-04 • (zero) 4.4E-02 • (conditional) 4.8E-02 • (zero) 5.8E-02	ZHU--DJMDXI	5.0E-7	No Change
• DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • JHUHWINJDXI: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY) • JHUHWINJDXD: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE) • ZHU--CSTDXI: OPERATORS FAIL TO REFILL CST FROM RWST (ANY MEANS)	• 2.2E-04 • (zero) 4.4E-02 • (conditional)4.8E-02 • (zero) 3.3E-03	ZHU--DJYDXI	5.0E-7	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• MHUSE11WDXI: OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	• 5.8E-02 • (zero) 2.2E-04 • (medium) 1.3E-02	ZHU--DMVDXI	5.0E-7	No Change
• THU--THXDXI: OPERATOR FAILS TO ALIGN STANDBY TBCCW HX • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	• 1.0E-02 • (zero) 2.2E-04 • (medium) 1.3E-02	ZHU--DTVDXI	5.0E-7	No Change
• DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • DHU--SPCDXD: CONDITIONAL FAILURE OF OPERATOR TO INITIATE SPC/SDC - LATE • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	• 2.2E-04 • (conditional) 2.5E-02 • (medium) 1.3E-02	ZHU---DVDXI	8.3E-7	1.8E-6

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE • ZHU--CSTDXI: OPERATORS FAIL TO REFILL CST FROM RWST (ANY MEANS)	• 2.2E-04 • (medium) 1.3E-02 • (zero) 3.3E-03	ZHU--DVYDXI	5.0E-7	No Change
• JHUHWINJDXI: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY) • JHUHWINJDXD: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE) • VHU-VENTDXI: OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	• 4.4E-02 • (conditional) 4.8E-02 • (low) 1.3E-02	ZHU---JVDXI	1.3E-4	6.6E-5
• EHU-SE11DXI0: FAILURE TO X-TIE EMERGENCY AC POWER PER SE-11 • EHUCWGPBDXI0: PB OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	• 1.9E-02 • (medium) 2.7E-02	ZHUALTACDXI0	3.2E-3	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU--XTEDXI) • BHU--MAXDXI: FAILURE TO MAXIMIZE CRD FLOW PER T-246 (or BHU-PUMPDXI)	• 3.2E-04 (or 3.2E-04) • (medium) 2.1E-02	ZHU---ABDXI	5.1E-5	5.3E-5
• AHUBTL-RDXI: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) • AHU--BCIDXI: OPERATOR FAILS TO BYPASS CONTAINMENT ISOLATION • AHU--CADDXI: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	• 1.3E-03* • (medium) 3.5E-03 • (medium) 1.1E-02*	ZHU--AAADXI	9.9E-5* (1.5E-4)	1.1E-4
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU--XTEDXI) • THU--THXDXI: OPERATOR FAILS TO ALIGN STANDBY TBCCW HX	• 3.2E-04 (or 3.2E-04) • (low) 1.3E-02	ZHU---ATDXI	1.9E-5	2.0E-5
• AHU--BCIDXI: OPERATOR FAILS TO BYPASS CONTAINMENT ISOLATION • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC	• 3.5E-03 • (zero) 2.2E-04	ZHU---ADDXI	1.0E-6	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• BHU--MAXDXI: FAILURE TO MAXIMIZE CRD FLOW PER T-246 (or BHU-PUMPDXI) • FHULEVELDXI: OPERATORS FAIL TO TAKE MANUAL CONTROL OF FW	• 2.1E-02 • (low) 3.2E-02	ZHU---BFDXI	1.7E-3	No Change
• FHULEVELDXI: OPERATORS FAIL TO TAKE MANUAL CONTROL OF FW • AHU--650DXI: OPERATOR FAILS TO DEPRESSURIZE TO 650 PSIG FOR CONDENSATE INJECTION • JHUHWINJDXI: OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY)	• 3.2E-02 • (zero) 3.8E-03 • (medium) 4.4E-02	ZHUA650JDXI	6.8E-4	4.2E-3
• AHUBTL-RDXI: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) • AHU--CADDXI: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' • DHU--SPCDXI: FAILURE OF OPERATOR TO INITIATE RHR/SPC • WHU—NSWDXD: OPERATORS FAIL TO START STANDBY SW PUMP (LATER)	• 1.3E-03* • (medium) 1.1E-02 • (zero) 2.2E-04 • (zero) 4.0E-03	ZHU--ADWDXI	1.0E-6	No Change

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• FHULEVELDXI: OPERATORS FAIL TO TAKE MANUAL CONTROL OF FW • ZHU--HRLDXI: OPERATOR FAILS TO TAKE MANUAL CONTROL OF HPCI/RCIC -EARLY	• 3.2E-02 • (high) 4.6E-02	ZHU--FHRDXI	1.7E-2	No Change
• AHU--XTRDXI: FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES) (or AHU—XTEDXI) • AHU--BCIDXI: OPERATOR FAILS TO BYPASS CONTAINMENT ISOLATION	• 3.2E-04 or (3.2E-04) • (medium) 3.5E-03	ZHU--AAXDXI	4.8E-5	5.3E-5

TABLE 4.1-3
SUMMARY OF CHANGES IN POST-INITIATOR DEPENDENT HEPS DUE TO EPU

INDEPENDENT HFE BEID AND DESCRIPTION	PRE-EPU INDEPENDENT HEPS AND ASSUMED DEPENDENCE LEVEL	JOINT HEP BEID	PRE-EPU JOINT HEP FAILURE PROBABILITY(1)	EPU JOINT HEP FAILURE PROBABILITY
• AHUBTL-RDXI: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) • AHUBTL-RDXD: OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR) - LATE, CONDITIONAL • AHU--CADDXI: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	• 1.3E-03* • (conditional) 1.0* • (high) 1.1E-02*	ZHU---AADXI	6.6E-4* (6.8E-4)	6.6E-4

[1] Pre-EPU HEPs marked with an "*" symbol have been updated to be consistent with the EPU HRA and to maintain an appropriate basis for comparison. Prior values for the dependent HEPs are shown in parenthesis.

4.2 LEVEL 1 PRA

Section 4.1 summarized possible effects of the EPU by examining each of the PRA elements. This section examines possible EPU effects from the perspective of accident sequence progression. The dominant accident scenario types (classes) that can lead to core damage are examined with respect to the changes in the individual PRA elements discussed in Section 4.1.

Loss of Inventory Makeup Transients

The loss of inventory accidents (non-LOCA) are determined by the number of systems, their success criteria, and operator actions for responding to their demands. The following bullets summarize key issues:

- FW, HPCI, RCIC, and Low Pressure Makeup System[(1)] flow rates - all of these systems have substantial margin in their success criteria relative to the EPU power increase to match the coolant makeup flow required for postulated accidents.
- CRD - CRD is not initially an adequate makeup source to the RPV at the current PBAPS power rating for events initiated from full power. CRD is considered successful in the PBAPS PRA for late RPV injection given initial RPV injection from another source. MAAP cases PB044a and PB044b indicate that the timing requirements for initiating CRD are reduced for EPU conditions compared to pre-EPU conditions. However, the diagnosis time remains in the same Cause-Based Decision Tree (CBDT) time frame and the HEP is not impacted.
- HPSW Injection to the RPV - this system also has substantial margin in its success criteria relative to the EPU power increase to match the coolant makeup flow required for postulated accidents.
- The success criterion used in the 2009A PRA for the number of SRVs required to open to assure RPV emergency depressurization is two (2). Based on the MAAP evaluations (e.g., MAAP case PB005a), the 2 SRVs success criterion remains adequate for the EPU condition. However, timing differences associated with this requirement have been factored into the HEP analysis for operator actions to depressurize (refer to Table 4.1-2 above).
- The SRV setpoints are not changed for the PBAPS EPU. Given the power increase of the EPU, one may postulate that the probability of a stuck-open relief valve given a transient initiator would increase due to an increase in the

(1) Core Spray, LPCI, and Condensate.

- number of SRV cycles. This change has been incorporated into the PBAPS 2009A1 EPU model.

Operator actions include emergency depressurization and system control and initiation. The injection initiation/recovery and emergency depressurization timings are slightly impacted by the EPU. As such, changes to the existing risk profile associated with loss of inventory makeup accidents result.

ATWS

Following a failure to scram coupled with additional failures, a higher power level and increase in suppression pool temperature would result for the EPU configuration compared with the current PBAPS configuration (assuming similar failures).

The number of SRVs that must fail to open during an isolation ATWS in order to overpressurize the RPV is two (2) for the EPU case. This is consistent with the 2009A pre-EPU model (given the installation of an additional SV that is planned as part of the EPU hardware modifications) such that no change to the common cause failure contribution for this event is required for use in the 2009A1 EPU model.

The increased power level reduces the time available to perform operator actions. Given the shorter time frames associated with ATWS scenarios, this time reduction has an impact on ATWS scenarios. Refer to Table 4.1-2 for changes in ATWS related HEPs. Given these ATWS HEP changes, changes to the existing risk profile associated with ATWS accidents result.

Note that for EPU conditions, the use of enriched boron is anticipated to reduce the time required to shutdown the reactor, and increase the time available to the operators to initiate SLC to prevent containment overpressurization and/or core damage. However, for the 2009A1 EPU model, it was still assumed that the HEP values associated with SLC initiation should be based on shorter available times. This may provide a slight conservative bias to the calculated delta risk for the EPU assessment results. That is, the EPU PRA model assumes that a pro-rated shorter time is available to initiate SLC pumps compared to the pre-EPU available times. This assumption combined with the future use of enriched boron after EPU implementation ensures that the PRA results are not overly optimistic, and will show the maximum net increase from EPU.

LOCAs

The blowdown loads may be slightly higher because of the higher initial power. The GE task analyses confirm that the blowdown loads and SSCs remain acceptable after EPU. This includes the assessment that containment accident pressure is no longer required to ensure NPSH is satisfied for the pumps taking suction from the torus. However, this is contingent upon implementation of the RHR cross-tie and associated HEP to perform the alignment within one hour of a large break LOCA initiator coincident with a containment isolation failure as described in Section 4.1.2.3 above. The net result is actually a risk reduction in this very low likelihood scenario because the strategy now exists to avoid the need for overpressure credit that didn't exist before (i.e. prior to implementation of the EPU modifications).

Other than the RHR cross-tie issue described above, the success criteria for the systems to respond to a LOCA are delineated by system trains. Sufficient margin is available in these success criteria to allow adequate core cooling for EPU. MAAP 4.0.6 cases were used to verify that the success criteria did not change. However, since some timing values are impacted, slight changes to the existing risk profile associated with LOCA accidents result.

SBO

Station Blackout represents a unique subset of the loss of inventory accidents identified above. The station blackout scenario response is almost totally dominated by AC and DC power issues. In all other respects, SBO sequences are like the transients discussed above. Extended power uprate will not increase the loads on diesel-generators or batteries. As discussed earlier, the success criteria for mitigating systems is largely unchanged for the EPU. However, the LOOP recovery times are adjusted in the EPU analysis as discussed in Section 4.1.3 to account for shorter available recovery times that would be present for EPU conditions given an SBO occurs.

Additionally, a few operator actions are impacted by the reduced available timings of the EPU, and are propagated through the SBO accident sequences.

As such, minor changes to the existing risk profile associated with SBO accidents result.

Loss of Containment Heat Removal

Sequences that involve the loss of containment heat removal are affected slightly in terms of the time to reach the containment venting pressure or ultimate pressure. The impact on long-term LOOP non-recovery probabilities has been factored into the assessment by assuming that 15% less time is available in these scenarios as described in Section 4.1.3. However, the success criteria for the key systems (RHR, main condenser, and torus hard-piped vent) in the loss of containment heat removal accident sequences are not affected. Other systems (e.g., DW coolers) are considered marginal or inadequate for containment heat removal for the current PBAPS power level. Such systems would remain inadequate for the EPU.

The time available to initiate containment heat removal is over 15 hours in the PRA. The reduction in this very long time frame due to the EPU has no quantifiable impact on the HEPs for containment heat removal initiation.

In summary, only minor changes to the risk profile associated with loss of decay heat removal accidents result.

ISLOCA / BOC

Similar to the LOCA analysis, the success criteria for the systems to respond to an ISLOCA or BOC are delineated by system trains. Sufficient margin is available in these success criteria to allow adequate core cooling for EPU. Since the risk from these events is dominated by failure of early isolation or failure of injection within 1-2 hours from an external source, there is little or no change to the existing risk profile associated with ISLOCA and BOC accidents.

4.3 INTERNAL FIRES INDUCED RISK

The frequency of fires is not dependent on reactor power or operation. Thus, no impact on fire initiating event frequency is postulated.

Since the performance of the IPEEE, a Fire PRA was performed. The EPRI FIVE Methodology [9] and Fire PRA Implementation Guide (FPRAIG) [11] screening approaches, EPRI Fire Events Database [12] and plant specific data were used in this 2002 study, to develop the PBAPS Fire

PRA. An update to that Fire PRA model was performed in 2007 that included explicit analysis of the main control room (MCR) and cable spreading room (CSR) that had previously not been included. The ignition frequencies for the MCR and CSR were developed using the guidance in NUREG/CR-6850 [13]. The Fire PRA model was also integrated with the PB205C and PB305C internal events models as part of the 2007 update.

While the fire analysis did yield a CDF, the intent of the analysis was to identify the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios. The screening attributes of the fire PRA are summarized below.

4.3.1 Attributes of Fire PRA

Fire PRAs are useful tools to identify design or procedural items that could be clear areas of focus for improving the safety of the plant. Fire PRAs use a structure and quantification technique similar to that used in the internal events PRA.

Historically, since less attention has been paid to fire PRAs, conservative modeling is common in a number of areas of the fire analysis to provide a "bounding" methodology for fires. This concept is contrary to the base internal events PRA which has had more analytical development and is closer to a realistic assessment (i.e., not conservative) of the plant.

There are a number of fire PRA topics involving technical inputs, data, and modeling that prevent the effective comparison of the calculated core damage frequency figure of merit between the internal events PRA and the fire PRA. These areas are identified as follows:

Initiating Events: The frequency of fires and their severity are generally conservatively overestimated. A revised NRC fire events database indicates the trend toward both lower frequency and less severe fires. This trend reflects the improved housekeeping, reduction in transient fire hazards, and other improved fire protection steps at nuclear utilities. The database used in the PBAPS fire assessment used significantly older data that is conservative compared to more current data.

System Response: Fire protection measures such as sprinklers, CO_2, and fire brigades may be given minimal (conservative) credit in their ability to limit the spread of a fire. Therefore, the severity of the fire and its impact on requirements is exacerbated.

In addition, cable routings are typically characterized conservatively because of the lack of data regarding the routing of cables or the lack of the analytic modeling to represent the different routings. This leads to limited credit for balance of plant systems that are extremely important in CDF mitigation.

Sequences: Sequences may subsume a number of fire scenarios to reduce the analytic burden. The subsuming of initiators and sequences is done to envelope those sequences included. This causes additional conservatism.

Fire Modeling:	Fire damage and fire propagation are conservatively characterized. Fire modeling presents bounding approaches regarding the fire immediate effects (e.g., all cables in a tray are always failed for a cable tray fire) and fire propagation.

The fire PRA is subject to more modeling uncertainty than the internal events PRA evaluations. While the fire PRA is generally self-consistent within its calculational framework, the fire PRA calculated quantitative risk metric does not compare well with internal events PRAs because of the number of conservatisms that have been included in the fire PRA process. Therefore, the use of the fire PRA figure of merit as a reflection of CDF may be inappropriate. Any use of fire PRA results and insights should properly reflect consideration of the fact that the "state of the technology" in fire PRAs is less evolved than the internal events PRA.

Relative modeling uncertainty is expected to narrow substantially in the future as more experience is gained in the development and implementation of methods and techniques for modeling fire accident progression and the underlying data.

4.3.2 EPU Impact on Fire Risk

A qualitative impact on the PBAPS fire risk profile due to the EPU is estimated here based on review of the PBAPS fire PRA results. This estimate is performed as follows:

- As the dominant change in the internal events model is related to the change in operator error terms, examine the fire PRA model results and make similar changes to those defined in Tables 4.1-2 and 4.1-3 in the fire PRA model.
- The set of applicable fire PRA model human error probability changes are shown in Table 4.3-1. This includes an evaluation first to update the HEP values to be consistent with the 2009A model values and then an evaluation to provide the updated values for EPU conditions as well.
- Based on making the changes to the applicable HEP values for both pre-EPU and EPU conditions, the fire CDF increase and dominant scenarios are listed in Table 4.3-2.

The fire impact calculation estimate is summarized in Table 4.3-2. As can be seen from Table 4.3-2, it is estimated here that the PBAPS fire PRA CDF would increase by approximately 2.5E-07 due to the EPU. This represents less than 1% of the calculated fire CDF which on a percentage basis is much less than that calculated for the internal events CDF. Given that the success criteria did not change in going from pre-EPU to EPU conditions, then it is reasonable to assume that the timing differences associated with EPU conditions would have a small impact on the risk from fire events. The small increase in CDF makes sense since the dominant fire scenarios are more related to the experienced equipment failures due to the fire initiating event rather than being related to the operator actions required to respond to the fire events. This is evident in Table 4.3-2 which shows that the majority of the dominant fire scenarios were not impacted by the changes to the HEP values for EPU conditions. Qualitatively, then, regardless of the actual total CDF that is calculated, it is concluded that the risk increase due to EPU on fire risk is negligible.

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	**Proba-bility**	**Fussell-Vesely**	**Description**	**Equivalent 2009A HEP**	**Description**	**Pre-EPU**	**EPU**	**Comment**	**Fire Pre-EPU**	**Fire EPU**
AHU--600DXI2	1.90E-03	7.49E-06	OPERATOR FAILS TO DEPRESSURIZE TO 600 PSIG FOR CONDENSATE INJECTION	AHU--650DXI2	OPERATOR FAILS TO DEPRESSURIZE TO 650 PSIG FOR CONDENSATE INJECTION	3.80E-03	**2.20E-02**	Make pre-EPU and EPU Change	3.80E-03	**2.20E-02**
AHUALTDPDXI2	2.30E-02	1.53E-03	FAILURE TO OPEN ALTERNATE DEPRESSURIZATION	AHUALTDPDXI2	FAILURE TO OPEN NON-ADS SRVS OR TBP VALVES	2.50E-02	**3.50E-02**	Make pre-EPU and EPU Change	2.50E-02	**3.50E-02**
AHUBTL-RDXI2	1.50E-03	2.82E-02	OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR)	AHUBTL-RDXI2	OPERATOR FAILS TO VALVE IN N2 BOTTLES (FROM MCR)	1.3E-03*	**1.40E-03**	Make pre-EPU and EPU Change	1.30E-03	**1.40E-03**
AHU--CADDXI2	1.00E+00	2.11E-01	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	AHU--CADDXI2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	1.1E-02*	**2.30E-02**	1.0 in Fire PRA - Leave as is	No Change	No Change
AHU--FINDXI2	2.50E-03	6.60E-03	OPERATOR FAILS TO INHIBIT ADS	AHU--FINDXI2	OPERATOR FAILS TO INHIBIT ADS	2.1E-03*	**2.90E-03**	Make pre-EPU and EPU Change	2.10E-03	**2.90E-03**

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
AHU--XTRDXI2	2.70E-04	1.52E-02	FAILURE TO INITIATE MANUAL DEPRESS. (NON-LOOP CASES)	AHU--XTRDXI2	FAILURE TO INITIATE MANUAL DEPRESSURIZATION (NON-LOOP CASES)	3.20E-04	**3.30E-04**	Make pre-EPU and EPU Change	3.20E-04	**3.30E-04**
BHU--MAXDXI2	1.60E-02	7.55E-03	FAILURE TO MAXIMIZE CRD FLOW PER T-246	BHU--MAXDXI2	FAILURE TO MAXIMIZE CRD FLOW PER T-246	2.10E-02	No change	Make pre-EPU Change	2.10E-02	No change
BHU-PUMPDXI2	5.80E-03	3.36E-03	FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP	BHU-PUMPDXI2	FAILURE TO START STANDBY CRD PUMP OR RESTART RUNNING PMP	6.90E-03	No change	Make pre-EPU Change	6.90E-03	No change
DHU-OVERDXI2	1.00E-01	4.91E-04	OPERATOR FAILS TO OVERRIDE SHROUD LOW LEVEL PERMISSIVE					0.1 in Fire PRA - Leave as is	No Change	No Change
DHU-RUPTDXI2	1.00E+00	4.97E-06	OPERATOR FAILS TO STOP RUPTURE FLOOD BY TRIPPING PUMP	DHU-RUPTDXI2	OPERATOR FAILS TO STOP RUPTURE FLOOD BY TRIPPING PUMP	5.70E-02	No change	1.0 in Fire PRA - Leave as is	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
DHU--SDCDXI2	2.50E-03	6.31E-05	FAILURE OF OPERATOR TO INITIATE RHR/SDC	DHU--SDCDXI2	FAILURE OF OPERATOR TO INITIATE RHR/SDC	2.90E-03	No change	Make pre-EPU Change	2.90E-03	No change
DHU--SPCDXD2	4.10E-01	3.40E-04	COND. FAILURE OF OPERATOR TO INIT. SPC/SDC - LATE	DHU--SPCDXD2	CONDITIONAL FAILURE OF OPERATOR TO INITIATE SPC/SDC - LATE	2.50E-02	**5.00E-02**	Make pre-EPU and EPU Change	2.50E-02	**5.00E-02**
DHU--SPCDXI2	2.10E-05	3.47E-04	FAILURE OF OPERATOR TO INITIATE RHR/SPC	DHU--SPCDXI2	FAILURE OF OPERATOR TO INITIATE RHR/SPC	2.20E-04	No change	Make pre-EPU Change	2.20E-04	No change
DHU-TRANPX2	1.00E+00	2.46E-03	RHR PUMP SYSTEM REPAIRS NOT COMPLTD(TRANSIENT)					1.0 in Fire PRA - Leave as is	No Change	No Change
EHUCWGCNDXI0	1.60E-02	1.43E-03	CWG OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	EHUCWGCNDXI0	CWG OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	4.40E-02	No change	Make pre-EPU Change	4.40E-02	No change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
EHUCWGPBDXI0	1.00E+00	1.00E-01	PB OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	EHUCWGPBDXI0	PB OPERATOR FAILS TO ESTABLISH CONOWINGO LINE	2.70E-02	No change	1.0 in Fire PRA - Leave as is	No Change	No Change
EHU-LOOPDXI0	4.00E-04	7.02E-04	FAILURE TO START DIESEL AFTER NO AUTO-INIT (NO LOCA)	EHU-LOOPDXI0	FAILURE TO START DIESEL AFTER NO AUTO-INIT (NO LOCA)	4.50E-04	**4.70E-04**	Make pre-EPU and EPU Change	4.50E-04	**4.70E-04**
EHU-SE11DXI0	1.00E+00	7.94E-02	FAILURE TO X-TIE EMERGENCY AC POWER PER SE-11	EHU-SE11DXI0	FAILURE TO X-TIE EMERGENCY AC POWER PER SE-11	1.90E-02	No change	1.0 in Fire PRA - Leave as is	No Change	No Change
EHU-SE11DXI0-0.15	1.50E-01	5.81E-02	EHU-SE11DXI0 modified to 0.15					10x used in Fire PRA - Do same for pre-EPU	1.90E-01	No change
HHU--HLTDXI2	1.30E-02	9.72E-07	OPERATOR FAILS TO TRIP HPCI ON HIGH LEVEL	HHU--HLTDXI2	OPERATOR FAILS TO TRIP HPCI ON HIGH LEVEL	8.40E-02	**2.10E-02**	Make pre-EPU and EPU Change	8.40E-02	**2.10E-02**
HHU-HPCIDMI2	1.20E-04	6.90E-05	MAINTENANCE ERROR DISABLES HPCI					Pre-initiator - Leave as is	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
IHUTRAINDXI2	1.00E+00	7.06E-05	OPERATOR FAILS TO CROSSTIE U2 INSTRUM AIR TRAINS					1.0 in Fire PRA - Leave as is	No Change	No Change
JHU-2344DXI2	1.00E-01	7.96E-04	OPERATOR FAILS TO CORR. ALIGN CROSS CONNECT	JHU-2344DXI2	OPERATOR FAILS TO CORRECTLY ALIGN CROSS CONNECT	2.60E-03	No change	Make pre-EPU Change	2.60E-03	No change
JHU--ECTDXI2	1.50E-03	3.09E-05	OPERATOR FAILS TO CORR ALIGN HPSW FOR COOLING TOWER FLOW	JHU--ECTDXI2	OPERATOR FAILS TO CORRECTLY ALIGN HPSW FOR COOLING TOWER FLOW	1.70E-03	**3.40E-03**	Make pre-EPU and EPU Change	1.70E-03	**3.40E-03**
JHUHWINJDXD2	3.40E-02	2.18E-03	OPERATOR FAILS TO INJECT WITH HPSW THRU RHR (LATE)	JHUHWINJDXD2	OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (LATE)	4.80E-02	**2.00E-02**	Make pre-EPU and EPU Change	4.80E-02	**2.00E-02**
JHUHWINJDXI2	5.30E-02	2.27E-03	OPERATOR FAILS TO INJECT WITH HPSW THRU RHR (EARLY)	JHUHWINJDXI2	OPERATOR FAILS TO INJECT WITH HPSW THROUGH RHR (EARLY)	4.40E-02	**5.60E-02**	Make pre-EPU and EPU Change	4.40E-02	**5.60E-02**

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
KHUDGFANDXI0	1.00E+00	2.16E-02	OPERATOR FAILS TO MANUALLY INITIATE SUPPLEMENTAL FAN					1.0 in Fire PRA - Leave as is	No Change	No Change
MHUSE11WDXI2	1.00E+00	4.55E-02	OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W	MHUSE11WDXI2	OPERATORS FAIL TO IMPLEMENT SE-11 ATTACHMENT W	5.80E-02	No change	1.0 in Fire PRA - Leave as is	No Change	No Change
MHUSE11WDXI2-0.043	4.00E-02	1.15E-05	MHUSE11WDXI2 modified to 0.043					1x used in Fire PRA - Do same for pre-EPU	5.80E-02	No change
RHU--LCLDXI2	1.00E+00	4.81E-05	FAILURE TO LOCALLY RESET TURBINE AFTER OVERSPEED TRIP					1.0 in Fire PRA - Leave as is	No Change	No Change
RHU-RCICDMI2	1.20E-04	2.25E-06	MAINTENANCE ERROR DISABLES RCIC					Pre-initiator - Leave as is	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
THU--THXDXI2	1.00E+00	1.86E-05	OPERATOR FAILS TO ALIGN STANDBY TBCCW HX					1.0 in Fire PRA - Leave as is	No Change	No Change
VHU-VENTDXI2	1.80E-03	2.37E-03	OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILUR	VHU-VENTDXI2	OPERATOR FAILS TO INITIATE VENT GIVEN RHR HARDWARE FAILURE	1.30E-02	**1.40E-02**	Make pre-EPU and EPU Change	1.30E-02	**1.40E-02**
WHU--ESWDXD0	1.00E+00	2.64E-02	FAILURE TO START ESW PUMP LATER	WHU--ESWDXD0	FAILURE TO START ESW PUMP LATER	9.50E-02	No change	Make pre-EPU Change	9.50E-02	No change
WHU--ESWDXI0	1.30E-02	2.66E-02	FAILURE TO START ESW PUMP EARLY	WHU--ESWDXI0	FAILURE TO START ESW PUMP EARLY	7.30E-02	No change	Make pre-EPU Change	7.30E-02	No change
WHU--NSWDXD2	1.00E+00	3.64E-02	OPERATORS FAIL TO START STANDBY NSW PUMP (LATER)	WHU--NSWDXD2	OPERATORS FAIL TO START STANDBY SW PUMP (LATER)	4.00E-03	No change	1.0 in Fire PRA - Leave as is	No Change	No Change
WHU--NSWDXD2-0.015	1.00E-02	2.27E-04	WHU--NSWDXD2 modified to 0.015					5x used in Fire PRA - Do same for pre-EPU	2.00E-02	No change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
WHU--NSWDXI2	1.00E+00	3.91E-02	OPERATORS FAIL TO START STANDBY NSW PUMP (EARLY)	WHU--NSWDXI2	OPERATORS FAIL TO START STANDBY SW PUMP (EARLY)	1.00E+00	No change	1.0 in Fire PRA - Leave as is	No Change	No Change
YHU--CSTDXI2	6.60E-03	2.15E-04	OPERATORS FAIL TO REFILL CST FROM RWST (RW PUMPS)	YHU--CSTDXI2	OPERATORS FAIL TO REFILL CST FROM RWST (RW PUMPS)	9.10E-03	No change	Make pre-EPU Change	9.10E-03	No change
YHU--GRFDXI2	1.10E-02	2.78E-04	OPERATORS FAIL TO REFILL CST FROM RWST (GRAVITY FEED)	YHU--GRFDXI2	OPERATORS FAIL TO REFILL CST FROM RWST (GRAVITY FEED)	1.40E-02	No change	Make pre-EPU Change	1.40E-02	No change
ZHU-AADIDXI2	1.00E-06	1.20E-04	FLOOR HEP FOR AHU--CAD, AHUBTL-R, IHURESET, DHU--SPC	ZHU-AADIDXI	FLOOR HEP FOR AHU--CAD, AHUBTL-R, IHURESET, DHU--SPC	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU--ABDDXI2	1.00E-06	6.25E-05	FLOOR HEP FOR AHU-XTE, BHU-PUMP, AND DHU--SPCDXI	ZHU--ABDDXI	FLOOR HEP FOR AHU-XTE, BHU-PUMP, AND DHU--SPCDXI	1.00E-06	No Change	Same value applies	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
ZHU-ADFVDXI2	5.00E-07	9.58E-04	FLOOR HEP FOR AHUXTR, DHUSPC*, MISC FW, AND VENT					Deleted - Set to zero	0.00	0.00
ZHU-ADJYDXI2	5.00E-07	9.94E-04	FLOOR HEP FOR AHU--XTR, DHU--SPC, JHUHWINJ, ZHU--CST					Deleted - Set to zero	0.00	0.00
ZHU--ADLDXI2	1.00E-06	9.73E-06	FLOOR HEP FOR AHU--XTEDXI AND DHU--SPCDXI	ZHU--ADLDXI	FLOOR HEP FOR AHU--XTEDXI AND DHU--SPCDXI	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU-ADLKDXI2	1.00E-06	2.55E-05	FLOOR HEP FOR AHU--XT*DXI AND DHU-LEAKDXI	ZHU-ADLKDXI	FLOOR HEP FOR AHU--XT*DXI AND DHU-LEAKDXI	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU--ADMDXI2	1.00E-06	9.73E-06	FLOOR HEP FOR AHU--XTE, DHU--SPC, AND MHUSE11WDXI	ZHU--ADMDXI	FLOOR HEP FOR AHU--XTE, DHU--SPC, AND MHUSE11WDXI	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU--ADTDXI2	1.00E-06	9.94E-04	FLOOR HEP FOR AHU--XTRDXI AND DHU--SPCDXI	ZHU--ADTDXI	FLOOR HEP FOR AHU--XTRDXI AND DHU--SPCDXI	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU-ADVYDXI2	5.00E-07	9.94E-04	FLOOR HEP FOR AHU--XTR, DHU--SPC, VHU-VENT, ZHU--CST					Deleted - Set to zero	0.00	0.00

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
ZHU--ADYDXI2	1.00E-06	2.84E-04	FLOOR HEP FOR AHU--XTE, DHU--SPC, VHU-VENT, ZHU--CST	ZHU--ADYDXI	FLOOR HEP FOR AHU--XTE, DHU--SPC, VHU-VENT, ZHU--CST	1.00E-06	No Change	Same value applies	No Change	No Change
ZHU--AHLDXI2	1.50E-03	5.90E-06	JOINT HEP FOR AHU--XTEDXI AND ZHU-HIGHDXI	ZHU--AHLDXI	JOINT HEP FOR AHU--XTEDXI AND ZHU-HIGHDXI	4.80E-05	No Change	Make pre-EPU Change	4.80E-05	No Change
ZHU--AHTDXI2	1.50E-03	9.66E-03	JOINT HEP FOR AHU--XTRDXI AND ZHU-HIGHDXI	ZHU--AHTDXI	JOINT HEP FOR AHU--XTRDXI AND ZHU-HIGHDXI	4.80E-05	No Change	Make pre-EPU Change	4.80E-05	No Change
ZHU---AJDXI2	1.00E-06	2.55E-05	FLOOR HEP FOR AHU--XTRDXI AND JHU--ECTDXI	ZHU---AJDXI	FLOOR HEP FOR AHU--XTRDXI AND JHU--ECTDXI	1.00E-06	No Change	Same value applies	No Change	No Change
ZHUALTACDXI0	4.60E-03	2.00E-03	JOINT HEP FOR EHU-SE11DXI0 AND EHUCWGPBDXI0	ZHUALTACDXI0	JOINT HEP FOR EHU-SE11DXI0 AND EHUCWGPBDXI0	3.20E-03	No Change	Make pre-EPU Change	3.20E-03	No Change
ZHU--BDJDXI2	5.00E-07	4.40E-04	FLOOR HEP FOR BHU-PUMP, DHU--SPC, AND JHUHWINJDXI	ZHU--BDJDXI	FLOOR HEP FOR BHU-PUMP, DHU--SPC, AND JHUHWINJDXI	5.00E-07	No Change	Same value applies	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
ZHU--BDVDXI2	5.00E-07	4.57E-04	FLOOR HEP FOR BHU-PUMP, DHU--SPC, AND VHU-VENTDXI	ZHU--BDVDXI	FLOOR HEP FOR BHU-PUMP, DHU--SPC, AND VHU-VENTDXI	5.00E-07	No Change	Same value applies	No Change	No Change
ZHU--CSTDXI2	3.30E-03	1.15E-02	OPERATORS FAIL TO REFILL CST FROM RWST (ANY MEANS)	ZHU--CSTDXI	OPERATORS FAIL TO REFILL CST FROM RWST (ANY MEANS)	2.40E-03	No Change	Make pre-EPU Change	2.40E-03	No Change
ZHU--DFVDXI2	5.00E-07	6.30E-04	FLOOR HEP FOR SDC/SPC, MISC FW, AND VENT					Deleted - Set to zero	0.00	0.00
ZHU--DJMDXI2	5.00E-07	4.15E-04	FLOOR HEP FOR DHU--SPC, JHUHWINJ, AND MHUSE11WDXI	ZHU--DJMDXI	FLOOR HEP FOR DHU--SPC, JHUHWINJ, AND MHUSE11WDXI	5.00E-07	No Change	Same value applies	No Change	No Change
ZHU--DMVDXI2	5.00E-07	4.32E-04	FLOOR HEP FOR DHU--SPC, MHUSE11W, AND VHU-VENTDXI	ZHU--DMVDXI	FLOOR HEP FOR DHU--SPC, MHUSE11W, AND VHU-VENTDXI	5.00E-07	No Change	Same value applies	No Change	No Change
ZHU--DTVDXI2	5.00E-07	4.32E-04	FLOOR HEP FOR DHU--SPC, THU--THX, AND VHU-VENTDXI	ZHU--DTVDXI	FLOOR HEP FOR DHU--SPC, THU--THX, AND VHU-VENTDXI	5.00E-07	No Change	Same value applies	No Change	No Change

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
ZHU--DVYDXI2	5.00E-07	9.94E-04	FLOOR HEP FOR DHU--SPC, VHU-VENT, AND ZHU--CSTDXI	ZHU--DVYDXI	FLOOR HEP FOR DHU--SPC, VHU-VENT, AND ZHU--CSTDXI	5.00E-07	No Change	Same value applies	No Change	No Change
ZHU-HIGHDXI2	2.10E-03	9.13E-06	FAILURE TO MANUALLY INITIATE HPCI/RCIC INJECTION	ZHU-HIGHDXI2	FAILURE TO MANUALLY INITIATE HPCI/RCIC INJECTION	2.20E-03	**3.20E-03**	Make pre-EPU and EPU Change	2.20E-03	**3.20E-03**
ZHU--HRLDXI2	7.20E-02	5.44E-04	OPERATOR FAILS TO TAKE MANUAL CONTROL OF HPCI/RCIC - EARLY	ZHU--HRLDXI2	OPERATOR FAILS TO TAKE MANUAL CONTROL OF HPCI/RCIC - EARLY	4.60E-02	**4.00E-02**	Make pre-EPU and EPU Change	4.60E-02	**4.00E-02**
ZHU--HRZDXI2	1.90E-01	2.76E-03	JOINT HEP FOR HHU--HLT/ RHU--HLT, AND ZHU--HRLDXI2	ZHU--HRZDXI	JOINT HEP FOR HHU--HLT/ RHU--HLT, AND ZHU--HRLDXI2	2.50E-02	**2.00E-02**	Make pre-EPU and EPU Change	2.50E-02	**2.00E-02**
ZHU---JVDXI2	9.00E-05	1.03E-04	JOINT HEP FOR JHUHWINJDXD AND VHU-VENTDXI	ZHU---JVDXI	JOINT HEP FOR JHUHWINJDXD AND VHU-VENTDXI	1.30E-04	**6.60E-05**	Make pre-EPU and EPU Change	1.30E-04	**6.60E-05**
ZHU--LPIDXI2	6.20E-04	6.40E-07	FAILURE TO MANUALLY INITIATE LOW PRESS ECCS (TRANSIENT)	ZHU--LPIDXI2	FAILURE TO MANUALLY INITIATE LOW PRESS ECCS (TRANSIENT)	8.30E-04	**1.20E-03**	Make pre-EPU and EPU Change	8.30E-04	**1.20E-03**

Table 4.3-1
Estimate of EPU Impact on Fire Human Error Probabilities

Fire PRA HEP	Proba-bility	Fussell-Vesely	Description	Equivalent 2009A HEP	Description	Pre-EPU	EPU	Comment	Fire Pre-EPU	Fire EPU
ZHU--LVCDXI2	1.40E-04	5.36E-04	OP FAILS TO CNTRL LEVEL IN A TRANS W/ ECCS INJ.	ZHULVCTRDXI2	OP FAILS TO CONTROL LEVEL IN A TRANS W/ ECCS INJECTION	2.60E-04	No change	Make pre-EPU Change	2.60E-04	No change

Table 4.3-2
Estimate of Impact on Fire CDF Due to EPU

Scenario Description	Pre EPU CDF	EPU CDF	Delta CDF
Unit 2 Reactor Recirculation Pump MG Set room	4.23E-06	4.23E-06	~0.00
4kV Switchgear Bus 20A17, Breaker 1708	3.91E-06	3.91E-06	~0.00
4kV Switchgear Bus 20A018, Breaker 1801	3.44E-06	3.44E-06	~0.00
Cable Spreading Room Relay Cabinet 20C32	3.43E-06	3.43E-06	~0.00
4kV Switchgear Bus 20A018, Breaker 1808	2.37E-06	2.37E-06	~0.00
2AC043 portion of the Remote Shutdown Panel	2.22E-06	2.23E-06	+1.0E-08
Same as scenario 38-F with Alternate Battery Charger Success	2.05E-06	2.05E-06	~0.00
Main Control Room Abandonment Back Panel Fire	1.82E-06	1.82E-06	~0.00
4kV Switchgear Bus 20A015	1.54E-06	1.58E-06	+4.0E-08
MCR Cabinet Fire - 00C29B	1.52E-06	1.52E-06	~0.00
MCR Cabinet Fire - 00C29C	1.27E-06	1.27E-06	~0.00
MCR Cabinet Fire - 00C29A	1.21E-06	1.21E-06	~0.00
All other Areas	1.45E-05	1.47E-05	+2.0E-07
Total Fire CDF	4.35E-05	4.38E-05	+2.5E-07

4.4 SEISMIC RISK

The frequency of earthquakes is not dependent on reactor power or operation. Thus, no impact on the seismic initiating event frequency is postulated.

The PBAPS seismic risk analysis was performed as part of the Individual Plant Examination for External Events (IPEEE) [8]. PBAPS performed a seismic margins assessment (SMA) following the guidance of EPRI NP-6041 [10]. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequencies were quantified as part of the seismic risk evaluation.

The IPEEE submittal [8] identified several areas for seismic margin improvement (Refer to Table 7.2-1b of the IPEEE submittal). These changes have all been subsequently addressed and in effect will reduce the seismic risk at the site.

Based on the efforts to correct the seismic issues that were identified as part of the IPEEE program and the ongoing process to monitor seismic issues at the plant, no additional measures are considered to be required based on the implementation of EPU. The EPU has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event will not alter the results of the SMA.

The decrease in time available for operator actions, and the associated increases in calculated HEPs, will have a non-significant impact on seismic-induced risk. Industry BWR seismic PRAs have typically shown (e.g., Peach Bottom NUREG/CR-4550 study [20]; and the Limerick Generating Station Severe Accident Risk Assessment [21]) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures.

Based on the above discussion, the increase in the PBAPS seismic risk due to the EPU is much less than that calculated for internal events.

4.5 OTHER EXTERNAL EVENTS RISK

In addition to internal fires and seismic events, the PBAPS IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents

The PBAPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, and nearby facility accidents was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that PBAPS meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards.

Based on the other external events being low risk contributors and the fact that the EPU changes would not significantly change the risk from these types of events, the increase in the PBAPS other external events risk due to the EPU is much less than that calculated for internal events.

4.6 SHUTDOWN RISK

The impact of the Extended Power Uprate (EPU) on shutdown risk is similar to the impact on the at-power Level 1 PRA. Based on the insights of the at-power PRA impact assessment, the areas of review appropriate to shutdown risk are the following:

- Initiating Events
- Success Criteria
- Human Reliability Analysis

The following qualitative discussion applies to the shutdown conditions of Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), and Refueling (Mode 5). The EPU risk impact during the transitional periods such as at-power (Mode 1) to Hot Shutdown and Startup (Mode 2) to at-power is subsumed by the at-power Level 1 PRA. This is consistent with the U.S. PRA industry, and with NRC Regulatory Guide 1.174, which states that not all aspects of risk need to be addressed for every application. While higher conditional risk states may be postulated during these transition periods, the short time frames involved produce an insignificant impact on the long-term annualized plant risk profile.

4.6.1 Shutdown Initiating Events

Shutdown initiating events include the following major categories:

- Loss of RCS Inventory
  - Inadvertent Draindown
  - LOCAs
- Loss of Decay Heat Removal (includes LOOP)

No new initiating events or increased potential for initiating events during shutdown (e.g., loss of DHR train) can be postulated due to the EPU.

4.6.2 Shutdown Success Criteria

The impact of the EPU on the success criteria during shutdown is similar to the Level 1 PRA. The increased power level decreases the time to boildown. However, because the reactor is already shutdown, the boildown times are much longer compared to the at-power PRA. The estimated time to uncover the core with the existing power level (CLTP) is 11.2 hours (10.0 hours for the EPU) at one day into the outage with the RPV level at the flange. The estimated time to uncover the core exceeds 24 hours when the water level is flooded up into the refueling cavity for both pre-EPU and EPU conditions.

The increased decay heat loads associated with the EPU impacts the time when low capacity decay heat removal (DHR) systems can be considered successful alternate DHR systems. The EPU condition delays the time after shutdown when low capacity DHR systems may be used as an alternative to Shutdown Cooling (SDC). However, this reduction in time for alternate decay heat removal system success minimally impacts shutdown risk.

Other success criteria are marginally impacted by the EPU. The EPU has a minor impact on shutdown RPV inventory makeup during loss of decay heat removal scenarios in shutdown because of the low decay heat level compared to at-power heat loads. The heat load to the suppression pool during loss of decay heat removal scenarios in shutdown (i.e., during shutdown phases with the RPV intact) is also lower because of the low decay heat level such that the margins for suppression pool cooling capacity are adequate for the EPU condition.

The EPU impact on the success criteria for blowdown loads, RPV overpressure margin, and SRV actuation is estimated to be negligible because of the low RPV pressure and low decay heat level during shutdown.

4.6.3 Shutdown HRA Impact

The primary impact of the EPU on risk during shutdown operations is the decrease in allowable operator action times in responding to off-normal events. However, as can be seen in Tables B-2 through B-4 of Appendix B, the reduction in times to core damage (i.e., CLTP case compared to EPU case) is on the order of 10%. Such small changes in already lengthy allowable operator response times result in negligible changes (<<1%) in calculated human error probabilities.

The allowable operator action times to respond to loss of heat removal scenarios during shutdown operations are many hours long. Very early in an outage the times are approximately 5-10 hours; later in an outage the times are dozens of hours. A reduction from 7.1 hours to 6.4 hours (refer to "1 Day After Shutdown" case in Table B-2 of Appendix B) in allowable action times would not result in a significant increase in human error probabilities for most operator actions using current human reliability analysis methods. The allowable timing reductions for times later in the outage would result in indiscernible changes in HEPs using current human reliability analysis methods.

4.6.4 Shutdown Risk Summary

Based on a review of the potential impacts on initiating events, success criteria, and HRA, the EPU is assessed to have a non-significant impact (delta CDF of roughly one percent per calculations in Appendix B) on shutdown risk.

This assessment is consistent with CLTR conclusions on this issue [18]:

> *"The shutdown risks for BWR plants are generally low and the impact of CPPU* [constant pressure power uprate] *on the CDF and LERF during shutdown is expected to be negligible."*

PBAPS Outage Risk Management Process

The plant uses a computerized risk monitor (PARAGON) and site-specific management guidelines as tools for controlling outage risk. The impact of the outage activities upon key safety functions is assessed as follows:

- Identify key safety functions affected by the SSC planned for removal from service.
- Consider the degree to which removing the SSC from service will impact the key safety functions.
- Consider degree of redundancy, duration of out-of-service condition, and appropriate compensatory measures, contingencies, or protective actions that could be taken if appropriate for the activity under consideration.

The Key Safety Function Matrices were developed consistent with guidance provided by NUMARC 91-06. The shutdown key safety functions are achieved by using systems or combinations of systems. The scope of the Systems, Structures and Components (SSCs) to be addressed by the assessment for shutdown conditions are those SSCs necessary to support the following shutdown key safety functions (from Section 4 of NUMARC 91-06):

- Decay heat removal capability
- Inventory Control
- Power Availability
- Reactivity control
- Containment (primary/secondary)

Managing the risk involves invoking some or all of the following elements:

- Pre-job briefs of operating and maintenance crews
- System engineering oversight
- Management oversight
- Outage management approval of the proposed activity
- Pre-staged parts and materials
- Walkdown of tagouts and maintenance activity prior to conducting the maintenance
- Mockup training
- Reduce OOS time through overtime or additional shift coverage.
- Contingency plans for returning equipment to service in a timely manner if needed.

- Compensatory measures to minimize initiators and/or mitigate the consequences.
- Reschedule or minimize work on functionally related equipment.
- Proceduralize other success paths of the safety function affected.

4.7 RADIONUCLIDE RELEASE (LEVEL 2 PRA)

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. In the process of modeling severe accidents (i.e., the MAAP code), the complex plant structure has been reduced to a simplified mathematical model that uses basic thermal hydraulic principles and experimentally derived correlations to calculate the radionuclide release timing and magnitude [22]. Changes in plant response due to EPU represent relatively small changes to the overall challenge to containment under severe accident conditions.

Approximately 125 Level 1 and Level 2 MAAP runs were performed in support of the PBAPS EPU risk assessment. The Level 2 MAAP runs were focused on the assessment of any significant changes in release categories. No changes to the PBAPS PRA Level 2 accident progression logic modeling or release magnitude assignment were evaluated to be necessary for EPU.

The following aspects of the Level 2 analysis are briefly discussed:

- Level 1 input
- Accident Progression
- Human Reliability Analysis
- Success Criteria
- Containment Capability
- Radionuclide Release Magnitude and Timing

<u>Level 1 Input</u>

The front-end evaluation (Level 1) involves the assessment of those scenarios that could lead to core damage. The subsequent treatment of mitigating actions and the inter-relationship with the containment after core damage (Level 2) is then treated in the PBAPS Containment Event Trees (CETs).

In the PBAPS Level 1 PRA, accident sequences are postulated that lead to core damage and potentially challenge containment. The PBAPS Level 1 PRA has identified discrete accident sequences that contribute to the core damage frequency and represent the spectrum of possible challenges to containment.

The Level 1 core damage sequences are also propagated through the Level 2 CETs. Therefore, changes to the Level 1 PRA modeling directly impact the Level 2 PRA results. However, the percentage increase in total CDF due to the EPU is not a direct translation to the percentage increase in total LERF. For example, a change to long-term core damage accidents would not impact the LERF results as much as early-term core damage accidents that have a larger potential to result in a Level 2 large and early release sequence.

Therefore, the Level 2 at-power internal events PRA model is also re-quantified as part of this EPU risk assessment.

Accident Progression

The EPU does not change the plant configuration and operation in a manner that produces new accident sequences or changes accident sequence progression phenomenon. This is particularly true in the case of the Level 2 post-core damage accident progression phenomena. The minor changes in decay heat levels have a minor impact on Level 2 PRA safety functions, such as containment isolation, ex-vessel debris coolability and challenges to the ultimate containment strength. No Level 2 safety function success criteria (e.g., gpm of coolant required for in-vessel or ex-vessel debris cooling) would be changed due to the EPU (although the timing requirements may be shifted somewhat).

Regarding energetic phenomena occurring at or near the time of core slump or RPV breach, such accident progression scenarios are appropriately modeled in the PBAPS Level 2 PRA as leading directly to High magnitude releases. This is a reasonable and standard PRA industry approach. This approach would not be changed due to the EPU.

Therefore, no changes are made as part of this assessment to the Level 2 models (either in structure or basic event phenomenon probabilities) with respect to accident progression modeling.

Human Reliability Analysis

Since the PBAPS PRA employs a fully integrated Level 1 transfer to the Level 2 PRA model, changes to HEP values (refer to Section 4.1.6) have a direct effect on both the Level 1 and Level 2 results. In other words, changing HEPs can affect the outcome of core damage, which then provides the input to the sequences responsible for calculating release categories.

Success Criteria

No changes in success criteria have been identified with regard to the Level 2 containment evaluation. The slight changes in accident progression timing and decay heat load has a minor or negligible impact on Level 2 PRA safety functions, such as containment isolation, ex-vessel debris coolability and challenges to the ultimate containment strength. (Refer to Section 4.1.2.8 of this report). Therefore, no changes to Level 2 modeling with respect to success criteria are made as part of this analysis.

Containment Capability

As discussed in Section 4.1.2.8 earlier in this report, no issues have been identified with respect to the EPU that have any impact on the capacity of the PBAPS containment as analyzed in the PRA.

The PBAPS containment capacity with respect to severe accidents is analyzed in the PRA using plant specific structural analyses as well as information from industry studies and experiments. The minor changes to the plant from the EPU have no impact on the definition of these containment loading profiles or the likelihood of containment isolation failure. The slightly higher decay heat levels associated with the EPU will result in a minor reduction in times to reach loading challenges; however, the time frames are long (many hours) and the accident timing reductions of 10-15% due to the EPU will typically have a small impact on the Level 2 results.

Release Magnitude and Timing

The following issues can substantially increase or decrease the ability to retain fission products or mitigate their release:

- Radionuclide removal processes
- Containment failure modes
- Phenomenology
- Accident sequence timings

Each of these issues is considered and analyzed in the PBAPS Level 2 PRA.

The PBAPS Level 2 PRA release categorization scheme uses both release magnitude and timing. Release categories were assigned to the PBAPS 2009A pre EPU PRA based on results of representative MAAP runs for many accident scenarios, and based on judgment and standard industry approaches for selected scenarios.

The PBAPS release magnitude classification is based on the percentage (as a function of the initial EOC inventory in the core) of CsI released to the environment; this approach is consistent with the majority of US BWR PRAs and standard industry techniques. Changes to the release magnitude categories assigned to individual accident sequences in the PBAPS Level 2 PRA are not necessary; this was confirmed by MAAP runs.

Level 2 Impact Summary

Based on the above discussion, the impact of the EPU on the PBAPS Level 2 PRA results, independent of the Level 1 analysis, is estimated to be small. The change in the Level 2 is due primarily to changes in the Level 1 accident sequences propagated through to the Level 2 quantification. That is, an increase in a Level 1 accident sequence gave rise to a proportional increase in the Level 2 result that was associated with that core damage state, i.e., the Level 2 results are coupled to the Level 1 results.

Section 5
CONCLUSIONS

The Extended Power Uprate (EPU) for PBAPS has been reviewed to determine the net impact on the risk profile associated with operation at an increase in power level to 3951 MWt. This examination involved the identification and review of plant and procedural changes, plus changes to the risk spectrum due to changes in the plant response.

The change in plant response, procedures, hardware, and setpoints associated with the increase in power have been investigated using the 2009A pre-EPU and 2009A1 EPU PRA models; a focused analysis of fire risk; the IPEEE study for seismic and other external events; and a qualitative evaluation of shutdown events. This section provides overall conclusions with respect to success criteria, the Level 1 PRA, the Level 2 PRA, internal fires, seismic events, other external events, and shutdown events. The review has indicated that small perturbations on individual inputs could be identified.

This section summarizes the risk impacts of the EPU implementation on the following areas:

- Level 1 Internal Events PRA
- Fire Induced Risk
- Seismic Induced Risk
- Other External Events Risk
- Shutdown Risk
- Level 2 PRA

In addition, the guidelines from the NRC (Regulatory Guide 1.174) are followed to assess the change in risk as characterized by core damage frequency (CDF) and Large Early Release Frequency (LERF)

5.1 LEVEL 1 PRA

Qualitative engineering insights regarding the adequacy of procedures and systems to prevent postulated core damage scenarios are among the principal results of the Level 1 portion of the PRA. These insights deal with the adequacy of, or improvements to, PBAPS procedures or systems (frontline or support) to accomplish their safety mission of preventing core damage. The severe accident scenarios that have been identified in the Level 1 PRA have been reviewed and the relatively small perturbations due to power uprate do not affect the scenario development or the qualitative insights.

The PRA model changes incorporated for the power uprate evaluation are:

- Representation of the RHR cross-tie (as discussed in Section 4.1.2.3)
- Representation of the CST standpipe (as discussed in Section 4.1.4)
- Revised SORV probabilities (as discussed in Section 4.1.2.6)
- Revised HEPs (as presented in Tables 4.1-2 and 4.1-3)
- Revised LOOP recovery probabilities based on reduced time available (as discussed in Section 4.1.3)

Other than the representation of the RHR cross-tie and CST standpipe, no additional modeling structure changes to the PB209A PRA model were necessary to reflect the EPU in the PB209A1 PRA model. Only basic event value changes and LOOP recovery time impacts were incorporated into the PB209A1 PRA model to represent the remaining EPU impacts.

Based on the model impact discussed previously, the EPU is estimated to increase the PBAPS Unit 2 internal events PRA CDF from the base value of 3.60E-06/yr to 3.70E-06/yr, an increase of 1.0E-07/yr (2.8%). The composition and comparative distribution of the EPU results remain basically unchanged with respect to the base PBAPS PRA. Table 5.1-1 shows quantitative CDF comparisons categorized by initiating event and Table 5.1-2 compares various accident sequences. The at-power internal events LERF increased from the base value of 4.58E-07/yr to 4.74E-07/yr, an increase of 1.6E-08/yr (3.5%) for Unit 2. Table 5.1-3 shows quantitative LERF comparisons categorized by initiating event. Since the base case CDF and LERF are slightly lower for Unit 3 and based on a review of the changes in CDF and LERF for the EPU assessment, the EPU is expected to provide very similar impacts for Unit 3. Shutdown risk and external event risk was also evaluated and determined to be impacted to a similar or lesser degree than the internal events risk (refer to Sections 5.2 through 5.5).

Table 5.1-1

Comparison of PBAPS CLTP CDF vs. EPU CDF by Initiator

Initiator Description	CLTP Value (1/yr)	EPU Value (1/yr)	%Increase by Initiator	Relative % of CDF Increase
LOSS OF CONDENSER VACUUM (%TCV)	6.41E-07	6.59E-07	+2.8%	+0.5%
MSIV CLOSURE (%TMSIV)	4.50E-07	4.59E-07	+2.0%	+0.2%
TURBINE TRIP (%TTR)	3.16E-07	3.35E-07	+6.1%	+0.5%
GRID CENTERED LOOP INITIATING EVENT (%LOOP-GRID)	2.80E-07	2.73E-07	-2.6%(1)	-0.2%
LOSS OF U2 SW INITIATING EVENT (%SW2)	2.71E-07	2.73E-07	+0.6%	+0.1%
MEDIUM LOCA (%S1)	2.53E-07	2.68E-07	+6.0%	+0.4%
LOSS OF 4KV AC BUS E12 (%TACBUSE12)	1.95E-07	2.04E-07	+4.5%	+0.2%
LOSS OF FEEDWATER (%TF)	1.82E-07	1.89E-07	+3.6%	+0.2%
SMALL LOCA (%S2)	1.34E-07	1.45E-07	+8.3%	+0.3%
OTHER LOCA CONTRIBUTORS (%A, %VMSL, %VFW)	8.94E-08	8.04E-08	-10.0%(2)	-0.2%
ALL OTHER INITIATORS	7.88E-07	8.15E-07	+3.5%	+0.8%
TOTAL:	**3.60E-06**	**3.70E-06**	**N/A**	**+2.8%**

(1) Note that this reduction is related to the incorporation of the CST standpipe for EPU. This eliminated some LOOP scenarios that resulted in inadvertent draindown of the CST.

(2) Note that this reduction is due to the new success path that now exists with the alignment of the RHR cross-tie given a Large LOCA and containment isolation failure occurs.

Table 5.1-2

Comparison of PBAPS CLTP CDF vs. EPU CDF by Sequence

Sequence Designator	Description	CLTP Value (1/yr)	EPU Value (1/yr)	Relative % of CDF Increase
RCVSEQ-TM-38	MSIV Closure, Sequence 38: Initial failures of HPCI and RCIC with failure to depressurize	4.50E-07	4.57E-07	+0.2%
RCVSEQ-TF-13	Loss of Feedwater, Sequence 13: Loss of pond or SW with failure of long term DHR	2.99E-07	3.00E-07	+0.0%
RCVSEQ-TM-26	MSIV Closure, Sequence 26: CCF or Dependent HEP failures lead to intermediate time frame CD	2.55E-07	2.55E-07	--
RCVSEQ-TT-10	Turbine Trip, Sequence 10: Loss of 4 kV AC bus, failure of HPCI, and failure to depressurize	2.41E-07	2.54E-07	+0.4%
RCVSEQ-TT-38	Turbine Trip, Sequence 38: Initial failures of FW, HPCI, and RCIC, and failure to depressurize	1.82E-07	1.90E-07	+0.2%
RCVSEQ-S1-23	Medium LOCA, Sequence 23: LOCA below TAF, failure of HPCI, and failure to depressurize	1.64E-07	1.68E-07	+0.1%
RCVSEQ-LP2-18	Loss of Offsite Power, Sequence 18: Station blackout with failure to recover offsite power	1.48E-07	1.48E-07	--
RCVSEQ-A-06	Large LOCA, Sequence 6: Large LOCA with failure of all ECCS (including due to loss of CAP)	1.41E-07	1.31E-07	-0.3%
RCVSEQ-TM-15	MSIV Closure, Sequence 15: Loss of long term DHR mostly from dependent HEP failures	1.37E-07	1.46E-07	+0.3%

Table 5.1-2

Comparison of PBAPS CLTP CDF vs. EPU CDF by Sequence

Sequence Designator	Description	CLTP Value (1/yr)	EPU Value (1/yr)	Relative % of CDF Increase
RCVSEQ-TM-13	MSIV Closure, Sequence 13: Loss of injection after venting containment at PCPL	1.14E-07	1.14E-07	--
OTHER SEQUENCES	Miscellaneous	1.48E-06	1.54E-06	+1.7%
	TOTAL:	**3.60E-06**	**3.70E-06**	+2.8%

Table 5.1-3

Comparison of PBAPS CLTP LERF vs. EPU LERF by Initiator

Initiator Description	CLTP Value (1/yr)	EPU Value (1/yr)	%Increase by Initiator	Relative % of LERF Increase
V SEQUENCE THRU LPCI LINES (%VLPCI)	9.21E-08	9.24E-08	+0.4%	+0.1%
TURBINE TRIP (%TTR)	6.96E-08	7.73E-08	+11.0%	+1.7%
LOSS OF CONDENSER VACUUM (%TCV)	5.27E-08	5.50E-08	+4.4%	+0.5%
GRID CENTERED LOOP INITIATING EVENT (%LOOP-GRID)	4.25E-08	4.88E-08	+14.9%	+1.4%
WEATHER CENTERED LOOP INITIATING EVENT (%LOOP-WTHR)	3.70E-08	3.97E-08	+7.2%	+0.6%
MSIV CLOSURE (%TMSIV)	2.83E-08	2.94E-08	+4.0%	+0.2%
SMALL LOCA (%S2)	2.74E-08	2.80E-08	+2.1%	+0.1%

LARGE LOCA (%A)	2.09E-08	6.92E-09	-66.9%[1]	-3.1%
LOSS OF FEEDWATER (%TF)	1.79E-08	1.87E-08	+4.3%	+0.2%
OTHER LOCA CONTRIBUTORS (%S1, %VMSL, %VFW)	1.06E-08	1.19E-08	+12.2%	+0.3%
ALL OTHER INITIATORS	5.91E-08	6.61E-08	+11.7%	+1.5%
TOTAL:	**4.58E-07**	**4.74E-07**	**N/A**	**+3.5%**

(1) Note that this reduction is due to the new success path that now exists with the alignment of the RHR cross-tie given a Large LOCA and containment isolation failure occurs.

5.1.1 Startup Testing CCDPs

An additional assessment was performed to calculate the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) associated with startup tests that will simulate a Turbine Trip and an MSIV Closure event. This information is provided below in Table 5.1-4. It is obtained from the base case EPU analysis by dividing the CDF (or LERF) associated with each of the initiators by the initiating event frequency to obtain the conditional probabilities.

Table 5.1-4
Conditional Probabilities for PBAPS EPU Startup Testing

Initiating Event	Initiating Event Frequency	Initiating Event Core Damage Frequency (CDF)	Conditional Core Damage Probabilities (CCDP)	Initiating Event Large Early Release Frequency (LERF)	Conditional Large Early Release Probabilities (CLERP)
Turbine Trip	0.754/yr	3.4E-7	4.4E-7	7.7E-8	1.0E-7
MSIV Closure	0.075/yr	4.6E-7	6.1E-6	2.9E-8	3.9E-7

The calculated CCDPs are 4.4E-7 and 6.1E-6 for non-isolation (turbine trip) and isolation (MSIV closure), respectively. Also, the calculated CLERPs are 1.0E-7 and 3.9E-7 for the non-isolation (turbine trip) and isolation (MSIV closure), respectively. These CCDPs and CLERPs represent the additional probabilities of core damage and large early release, caused by performing the proposed tests (i.e., the initiating events occur). If both tests are performed, the total additional probabilities would thus be 6.5E-6 (CCDP) and 4.9E-7 (CLERP). Note the analyses do not credit compensatory measures that may reduce the risk of core damage given that extra operators may be staged for the proposed tests.

5.2 FIRE INDUCED RISK

The fire impact calculation estimate is summarized in Section 4.3. It is estimated that the PBAPS fire PRA CDF would increase by approximately 2.5E-07 due to the EPU. This represents less than 1% of the calculated fire CDF which on a percentage basis is much less than that calculated for the internal events CDF. Given that the success criteria did not change in going from pre-EPU to EPU conditions, then it is reasonable to assume that the timing differences associated with EPU conditions would have a small impact on the risk from fire events. The small increase in CDF makes sense since the dominant fire scenarios are more related to the experienced equipment failures due to the fire initiating event rather than being related to the operator actions required to respond to the fire events. This is evident in the results summary table which shows that the majority of the dominant fire scenarios were not impacted by the changes to the HEP values for EPU conditions. Qualitatively, then, regardless of the actual total CDF that is calculated, it is concluded that the risk increase due to EPU on fire risk is negligible.

5.3 SEISMIC RISK

Based on a review of the PBAPS IPEEE, the conclusions of the seismic margins assessment (SMA) will be unaffected by the EPU. The power uprate has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, will not alter the results of the SMA. Refer to Section 4.4 of this report for further discussion.

5.4 OTHER EXTERNAL HAZARDS

Based on review of the PBAPS IPEEE, the power uprate has no significant impact on the plant risk profile associated with tornadoes, external floods, transportation accidents, and other external hazards. Refer to Section 4.5 of this report for further discussion.

5.5 SHUTDOWN RISK

The impact of the Extended Power Uprate (EPU) on shutdown risk is similar to the impact on the at-power Level 1 PRA. Shutdown risk is affected by the increase in decay heat power. However, the lower power operating conditions during shutdown (e.g., lower decay heat level, lower RPV pressure) allow for additional margin for mitigation systems and operator actions. Based on a review of the potential impacts on initiating events, success criteria, and HRA, the EPU implementation will have a minor impact on shutdown risk. Refer to Section 4.6 and Appendix B of this report for further discussion which indicate that the EPU is assessed to have a non-significant impact (delta CDF of roughly one percent).

5.6 LEVEL 2 PRA

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. As described in Section 4.7, the change in the Level 2 is due primarily to changes in the Level 1 accident sequences propagated through to the Level 2 quantification. Therefore, the majority of the impact on LERF from EPU is related to the timing differences associated with EPU compared to CLTP conditions. These timing differences have been factored into the EPU risk assessment for LOOP recovery times as described in Section 4.1.3 and the HEP value changes as described in Section 4.1.6.

The end result for EPU is an estimated increase of the PBAPS at-power internal events LERF (see Table 5.7-1) from the base value of 4.58E-7/yr to 4.74E-7/yr, an increase of 1.6E-8 (3.5%). As such, the EPU change in power represents a relatively small change to the key figure of merit for measuring containment adequacy, LERF.

5.7 QUANTITATIVE BOUNDS ON RISK CHANGE

5.7.1 Sensitivity Studies

As discussed in the previous sections, the best estimate change in the PBAPS risk profile due to the EPU is a 2.8% increase in CDF and a 3.5% increase in LERF. One of the methods to provide valuable input into the decision-making process is to perform sensitivity calculations for situations with different assumed conditions to bound the results.

These sensitivity studies investigated the impact on the at-power internal events CDF and LERF. As the change in CDF and LERF is minor, only conservative sensitivity cases (i.e., those that will tend to increase the calculated risk increases) are analyzed here. Table 5.7-1 displays the calculated results with an explanation and description of each of the sensitivity cases performed. The results of the sensitivity cases indicate that although increases in the calculated risk metrics could occur, they are not significant enough to change the conclusions of the risk assessment.

One additional evaluation was performed to address the potential for increased internal flood initiating event frequencies. This is not included in the set of sensitivity cases provided in Table 5.7-1 since the majority of the internal flood initiators are from systems that are not experiencing an increase in system flow (e.g., fire protection and service water). Therefore, the potential impact from the increased EPU flow rates is better represented and encompassed with the LOCA frequency changes identified in Sensitivity Case #3. In any event, to determine the potential impacts from an increase to the internal flood frequencies, it is noted that the total internal flood contribution to CDF is less than 7% and the total contribution to LERF is less than 2%. Therefore, even if all of the internal flood initiating event frequencies were to double (which is not credible given the flow rates for most of the flooding initiators are not changing), there would not be a significant change to the calculated risk metrics.

Table 5.7-1

Results of PBAPS EPU PRA Sensitivity Cases

Parameter	CLTP	EPU	Case #1a	Case #1b	Case #2	Case #3	Case #4
Post-Initiator HEPs	Base CLTP values	Calculated using EPU Timings	Calculated using EPU Timings	Calculated using EPU Timings	Calculated using EPU Timings	Calculated using EPU Timings	Calculated using EPU Timings
SORV Probabilities	Base CLTP values	Increased 13%(1)	Increased 13%	Increased 13%	Increased 13%	Increased 13%	Increased 13%
Turbine Trip w/Bypass (%TTR, with units of 1/yr)	Base CLTP value (0.754)	Base CLTP value	0.904(2)	Base CLTP value	Base CLTP value	Base CLTP value	0.904
Loss of Feedwater (%TF, with units of 1/yr)	Base CLTP value (0.050)	Base CLTP value	Base CLTP value	0.200(2)	Base CLTP value	Base CLTP value	0.200
Loss of Condenser Vacuum Initiator (%TC) with units of	Base CLTP value (0.139)	Base CLTP value	Base CLTP value	Base CLTP value	0.289(2)	Base CLTP value	0.289
LOCA Initiators (%A, %S1, %S2, with units of 1/yr)	Base CLTP values (5.20E-5	Base CLTP values	Base CLTP values	Base CLTP values	Base CLTP values	Increased 2x(2) (1.04E-4, 3.40E-3	Increased 2x
FW/MSL Initiators (%VFW, %VMSL, with units of 1/yr)	Base CLTP values (1.77E-9, 1.53E-8)	Base CLTP values	Base CLTP values	Base CLTP values	Base CLTP values	Increased 2x(2) (3.54E-9, 3.06E-8)	Increased 2x
Core Damage Frequency (CDF	3.60E-06	3.70E-06 (+1.0E-7	3.76E-06 (+6.0E-8)	4.28E-06 (+5.8E-7)	4.42E-06 (+7.2E-7)	4.20E-06 (+5.0E-7)	5.57E-06 (+1.97E-6
Large Early Release Frequency (LERF, with	4.58E-07	4.74E-07 (+1.6E-8	4.90E-07 (+1.6E-8)	5.32E-07 (+5.8E-8)	5.35E-07 (+6.1E-8)	5.22E-07 (+4.8E-8)	6.58E-07 (+2.00E-7

(1) The CDF and LERF contributions from the SORV probabilities are 2.1E-9 and 9.9E-10, respectively.
(2) Refer to the Notes to Table 5.7-1 which follow.

Table 5.7-2 summarizes the delta risk impact assessment results for the base case and for the sensitivity cases from several key categories. A discussion of each category is then provided.

Table 5.7-2

Summary of PBAPS EPU PRA Delta Risk From the Base Case and Sensitivity Cases

Impact	Δ CDF	Δ LERF	Comment
Operator Reliability	+1.0E-7	+1.6E-8	New Risk (Included in base case assessment)
Turbine Trip Initiator	+6.0E-8	+1.6E-8	Sensitivity (Case #1a)
Loss of Feedwater Initiator	+5.8E-7	+5.8E-8	Sensitivity (Case #1b)
Loss of Condenser Initiator	+7.2E-7	+6.1E-8	Sensitivity (Case #2)
SORV Probability	<1.0E-8	<1.0E-9	New Risk (Included in base case assessment)
LOCA Initiators	+5.0E-7	+4.8E-8	Sensitivity (Case #3)
Total New Risk	**1.0E-7**	**1.6E-8**	**New Risk**
Total with Sensitivity	**2.0E-6**	**2.0E-7**	**New Risk + Sensitivity (Case #4)**

<u>Operator Reliability</u>

The impact of increased decay heat and reduced time available for operator actions was evaluated. The impact on CDF and LERF associated with reduced times for these actions was calculated on a consistent basis as follows:

- The HEPs for EPU were estimated using the same technique as utilized for pre EPU (refer to Tables 4.1-2 and 4.1-3)

The differences between the CDF and LERF values for the pre-EPU and EPU configurations were calculated. The resulting changes in CDF and LERF are summarized in the above table.

Turbine Trip Sensitivity

Because of the various changes to the BOP side of the plant for EPU, the frequency of turbine trip could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of turbine trips based on EPU; however, the potential sensitivity of an increase was evaluated.

The revision to the turbine trip initiating event frequency (%TTR) uses an approach that assumes an additional turbine trip is experienced in the first year following start-up in the EPU condition and an additional 0.5 event in the second year. The change in the long-term average of the turbine trip initiating event frequency is calculated as follows for this sensitivity case:

- Base long-term turbine trip frequency is 0.754/yr
- 10 years is used as the "long-term" data period
- End of 10 years does not reach the end-of-life portion of the bathtub curve
- Assuming 1.5 additional trips in the first and second years as described above, the revised Turbine Trip w/Bypass frequency for this sensitivity case is calculated as:

$$\%TTR = \frac{(10 \times 0.754) + 1.5}{10} = 0.904/yr$$

All other parameters are maintained the same as the EPU base case.

Initiating Event (IE)	Base Case EPU PRA			Sensitivity
	IE Frequency	CDF	LERF	
%TTR – Turbine Trip	0.754/yr	3.35E-7	7.73E-8	If it is assumed that %TTR increases by 20%, the change in CDF would be 6.0E-8, and the change in LERF would be 1.6E-8, which represents only a few percent increase in risk when compared to the total pre EPU CDF and LERF values.

Loss of Feedwater Sensitivity

Because feedwater margins are also affected by EPU, the frequency of a loss of feedwater initiator could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of loss of feedwater based on EPU; however, the potential sensitivity of an increase was evaluated.

A similar assessment was performed assuming that the EPU changes would manifest into an increase to the loss of feedwater initiating event frequency (%TF).

- Base long-term loss of feedwater frequency is 0.050/yr
- 10 years is used as the "long-term" data period
- End of 10 years does not reach the end-of-life portion of the bathtub curve
- Assuming 1.5 additional loss of feedwater events in the first and second years, the revised Loss of Feedwater frequency for this sensitivity case is calculated as:

$$\%TF = \frac{(10 \times 0.050) + 1.5}{10} = 0.200/yr$$

All other parameters are maintained the same as the EPU base case.

Initiating Event (IE)	Base Case EPU PRA			Sensitivity
	IE Frequency	CDF	LERF	
%TF – Loss of Feedwater	0.05/yr	1.89E-7	1.87E-8	If it is assumed that %TF increases by 300% to 0.20/yr, the change in CDF would be 5.8E-7, and the change in LERF would be 5.8E-8. This represents an additional ~16% increase in CDF risk and ~13% increase in LERF risk when compared to the total pre EPU CDF and LERF values.

Loss of Condenser Sensitivity

Because condenser margins are also affected by EPU, the frequency of a loss of condenser initiator could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of loss of condenser based on EPU; however, the potential sensitivity of an increase was evaluated by assuming that the EPU changes would manifest into an increase to the loss of condenser initiating event frequency.

- Base long-term loss of condenser frequency is 0.139/yr
- 10 years is used as the "long-term" data period
- End of 10 years does not reach the end-of-life portion of the bathtub curve
- Assuming 1.5 additional loss of condenser events in the first and second years, the revised loss of condenser initiating event frequency is calculated as:

$$\%TCV = \frac{(10 \times 0.139) + 1.0 + 0.5}{10} = 0.289/yr$$

All other parameters are maintained the same as the EPU base case.

Initiating Event (IE)	Base Case EPU PRA			Sensitivity
	IE Frequency	CDF	LERF	
%TCV – Loss of Condenser	0.139/yr	6.59E-7	5.50E-8	If it is assumed that %TCV increases by 108% to 0.289/yr, the change in CDF would be 7.2E-7, and the change in LERF would be 6.1E-8. This represents an additional ~20% increase in CDF risk and ~13% increase in LERF risk when compared to the total pre EPU CDF and LERF values.

Stuck Open SRV (SORV)

The SRV setpoints will not be changed as a result of the PBAPS EPU. Given the power increase of the EPU, however, one may postulate that the probability of a stuck open relief valve given a transient initiator would increase due to an increase in the number of SRV cycles.

In the base case risk assessment, the PBAPS PRA for EPU conditions was modified by increasing the stuck open relief valve probability by a factor equal to the increase in reactor power (i.e., a factor of 1.125 in the case of EPU). This approach assumes that the stuck open relief valve probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power. The results of this change which is included in the base case EPU assessment are shown below.

Event	Base Case EPU PRA			Sensitivity
	Probability	CDF	LERF	
SORV – Stuck Open SRV (APHSRVTMDXI2, APHSRVTTDXI2)	2.14E-3 2.70E-4	2.1E-9	9:9E-10	That portion of the reported CDF and LERF values due to the SORV probability increases by ~13% is only 2.3E-10 for CDF and 1.1E-10 for LERF. This represents a negligible increase in risk when compared to the total pre EPU CDF and LERF values.

Loss of Coolant Accident (LOCA) Sensitivity

Because of increased flow rates it is assumed that increased reactor energy could result in LOCA frequency increases. The initiating event frequency task for the PRA update will not increase the frequency of LOCAs based on EPU; however, the potential sensitivity of an increase was evaluated.

This sensitivity case conservatively doubles the LOCA initiator frequencies for the small, medium and large LOCA categories. The initiating event frequencies for feedwater high energy line breaks were also doubled due to increased flow in this system as a result of EPU.

Large LOCA:	%A	= 5.20E-5 * 2	= 1.04E-4/yr
Medium LOCA:	%S1	= 1.70E-3 * 2	= 3.40E-3/yr
Small LOCA:	%S2	= 8.49E-3 * 2	= 1.70E-2/yr
FW Line Break:	%VFW	= 1.77E-9 * 2	= 3.54E-9/yr
MS Line Break:	%VMSL	= 1.53E-8 * 2	= 3.06E-8/yr

All other parameters are maintained the same as the EPU base case.

Initiating Event (IE)	Base Case EPU PRA			Sensitivity
	IE Frequency	CDF	LERF	
LOCA – Loss of Coolant	See above	4.93E-7	4.68E-8	If it is assumed that the LOCA frequency increases by100%, the change in CDF would be 5.0E-7, and the change in LERF would be 4.8E-8. This represents an additional ~14% increase in CDF risk and ~10% increase in LERF risk when compared to the total pre EPU CDF and LERF values.

5.7.2 Results Summary

The key result of the PBAPS EPU risk evaluation is the following:

> Minor risk increases were calculated for both CDF and LERF. The risk increase is associated with reduced times available for certain operator actions and AC power recovery, and the assumed increase in the SORV probability.

The best estimate of the risk increase for at-power internal events due to the EPU is a delta CDF of 1.0E-7/yr (an increase of 2.8% over the base CDF of 3.6E-6/yr). The best estimate at-power internal events LERF increase due to the EPU is a delta LERF of 1.6E-8 (an increase of 3.5% over the base LERF of 4.6E-7/yr).

Using the NRC guidelines established in Regulatory Guide 1.174 and the calculated results from the Level 1 and 2 PRA, the best estimate for the PBAPS CDF risk increase due to the EPU (1.0E-7/yr) is in Region III (i.e., "very small" risk changes). The best estimate for the LERF increase (1.6E-8/yr) is also in the lower range of Region III. (See Figures 5.7-1 and 5.7-2.). Additionally, based on the information available for external events impacts, it is estimated that the incorporation of these contributors would not change this conclusion.

The quantitative sensitivity cases performed in this analysis also showed that the delta CDF and the delta LERF remain within or very close to the lower region of Region III except for the combined sensitivity case (#4) which pessimistically increased all of the initiator frequencies at once. Even in that case, the above increase in risk meets the acceptance guidelines described in Regulatory Guide 1.174, which states that an increase in CDF in the range of 1E-6 to 1E-5 will be considered when it can be reasonably shown that the total CDF is less than 1E-4. Similarly, an increase in LERF in the range of 1E-7 to 1E-6 will be considered when it can be reasonably shown that the total LERF is less than 1E-5.

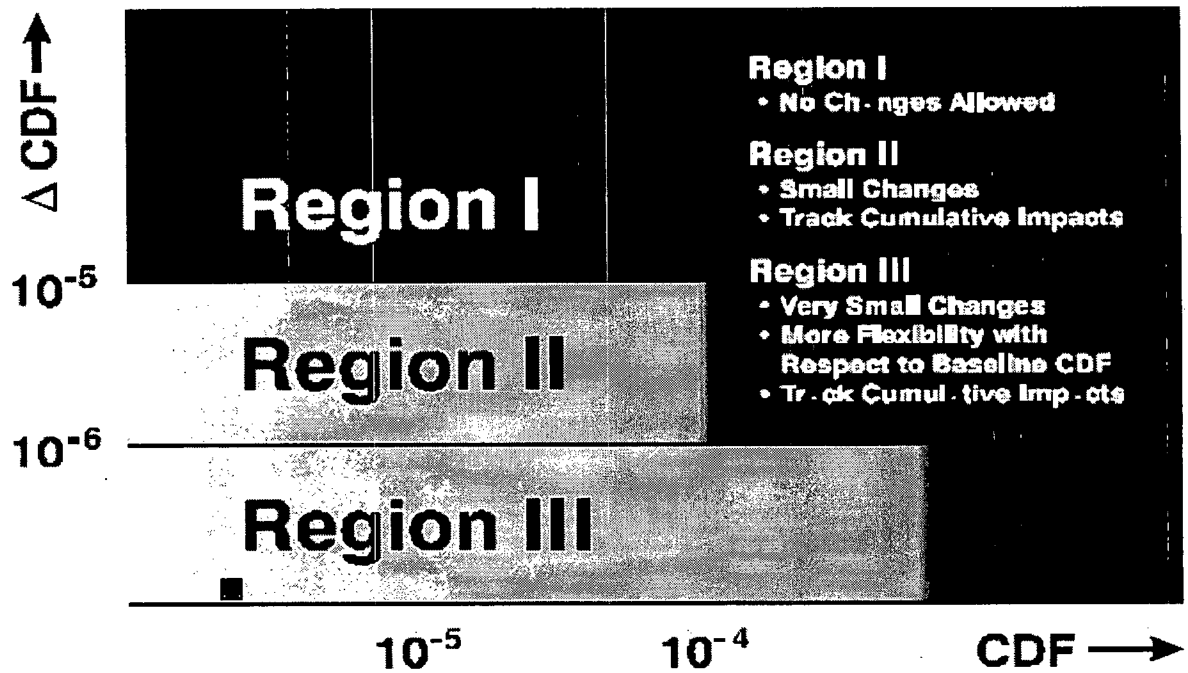

Figure 5.7-1 PBAPS EPU Risk Assessment CDF Result Versus RG 1.174 Acceptance Guidelines* for Core Damage Frequency (CDF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

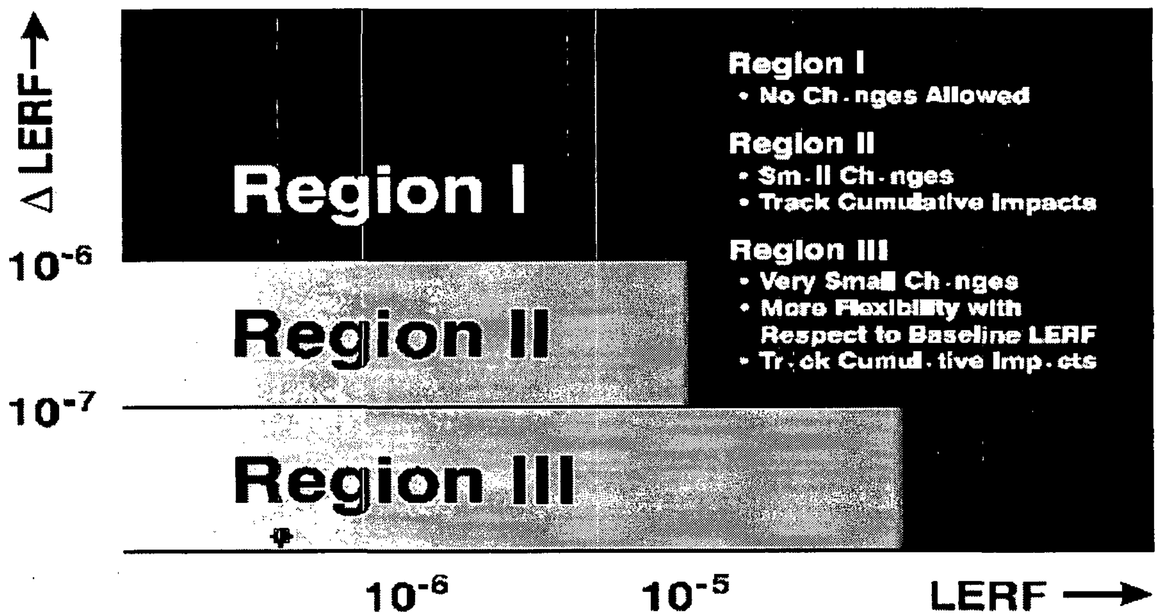

Figure 5.7-2 PBAPS EPU Risk Assessment LERF Result Versus RG 1.174 Acceptance Guidelines* for (LERF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

A bounding assessment is provided to demonstrate that the total CDF is less than 1E-4 and the total LERF is less than 1E-5. As shown in Table 4.3-2, the CDF contribution due to internal fires in the unscreened fire areas is calculated at 4.4E-5/yr for Unit 2 EPU conditions. The fire PRA does not quantify the LERF risk measure, however, review of NUREG-1742 [24], indicates that the fire CDF for BWRs is primarily determined by plant transient type of events such that the LERF distribution from the fire CDF can be assumed to be similar to that from the internal events model. The reported fire PRA CDF value is approximately a factor of 12 higher than the internal events CDF values. The fire CDF values are estimated to be very conservative given the methods employed in developing the fire PRA for Peach Bottom when compared to the best estimate CDF and LERF values obtained from the internal events models. Given this, it is reasonable to assume that the total impact from external events risk is bounded by assuming a factor of 12 on the internal events evaluation. With this conservative assumption, the bounding assessment for total CDF and total LERF is shown in Table 5.7-3.

Table 5.7-3
Bounding Estimate of Total CDF and Total LERF for EPU

Description	EPU CDF	EPU LERF
Internal Events Contribution – Bounding Sensitivity Case	5.6E-06	6.6E-07
External Events Contribution – Bounding factor of 12x internal Events Contribution	<6.7E-05	<8.0E-06
Total:	**<7.3E-05**	**<8.7E-06**

As shown in Table 5.7-3, since the bounding total CDF is less than 1E-4 and the bounding total LERF is less than 1E-5, and the maximum quantified new risk in the pessimistic sensitivity case is less than 1E-5 for CDF and less than 1E-6 for LERF, the quantified impact of EPU is acceptable.

As the combined sensitivity case falls into Region II from RG 1.174 (i.e., for "small" risk changes), it is also worth noting that this pessimistic case is also very close to still meeting the EPRI PSA Applications Guide "Non Risk Significant" criteria for permanent plant changes [23] (i.e., the 55% and 44% increase demonstrated in Case 4 are close to the allowable limits of 53% for CDF and 47% for LERF, respectively, based on a base CDF of 3.6E-6 and a base LERF of 4.6E-7).

Other sensitivity cases presented in Appendix A as part of the identification of potential key sources of model uncertainty also lead to the same conclusions. That is, the results of the model uncertainty sensitivity studies indicated that although some alternative assumptions could challenge the acceptance guidelines for "very small" changes in risk, there is no one issue that would result in exceeding the acceptance guidelines for "small" changes in risk. One notable sensitivity case is that with no credit for the RHR cross-tie to eliminate the need for containment overpressure, the EPU CDF increases to 3.72E-6 (compared to 3.70E-6 for the base case EPU estimate) and the LERF increases to 4.94E-7 (compared to 4.74E-7 for the base case EPU estimate). These increases represent a fairly negligible change compared to the base case best estimate results.

The PBAPS EPU is assessed to result in a small impact on the plant risk profile and thus is acceptable from a risk evaluation perspective.

Section 6

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Appendix A
PRA Technical Adequacy

A.1 Overview

The guidance provided in Regulatory Guide 1.200, Revision 2 [6], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" is used for the EPU risk assessment. The guidance in RG-1.200 indicates that the following steps should be followed when performing PRA assessments:

1. Identify the parts of the PRA used to support the application
   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model
   - A definition of the acceptance criteria used for the application
2. Identify the scope of risk contributors addressed by the PRA model
   - If not full scope (i.e., internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not adequately met, it will not unduly impact the results.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 were incorporated into the main body of this report. The purpose of the remaining portion of this appendix is to provide a PRA model evolution summary and to address the requirements identified in Item 4 above

A.2 PRA Model Evolution and Review Summary

The 2009A versions of the PBAPS PRA models are the most recent evaluations of the Unit 2 and Unit 3 risk profile at PBAPS for internal event challenges. The PBAPS PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the PBAPS PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon Generation Company, LLC (Exelon) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the PBAPS PRA.

<u>PRA Maintenance and Update</u>

The Exelon risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure and subordinate implementation procedures. The PRA model update procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites. The overall Exelon Risk Management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, Exelon risk management procedures provide the guidance for particular risk management maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. The 2009A models were completed in July of 2010.

As indicated previously, RG 1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, relevant peer review findings, and consistency with applicable PRA Standards) will be discussed in turn in this section. An uncertainty analysis including the identification of key assumptions is provided in Section A.3.

A.2.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE- Exelon PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model.

A review of the open UREs indicates that there are no plant changes that have not yet been incorporated into the PRA model that would affect this application. However, it is noted that the proposed changes for EPU have been fully implemented into the risk assessment as described in the main body of this report.

A.2.2 Consistency with Applicable PRA Standards

Several assessments of technical capability have been made for the PBAPS internal events PRA models. These assessments are as follows and further discussed in the paragraphs below.

- An independent PRA peer review was conducted under the auspices of the BWR Owners Group in 1998, following the Industry PRA Peer Review process [1]. This peer review included an assessment of the PRA model maintenance and update process.
- In 2004, a gap analysis was performed to assess gaps between the peer review scope/detail of the Industry PRA Peer Review results relative to the available version of the ASME PRA Standard [2] and the draft version of Regulatory Guide 1.200, DG-1122 [3]. In 2006, an assessment of the extent to which the previously defined gaps had been addressed was performed in conjunction with a PRA model update.
- During 2005 and 2006 the PBAPS, Units 2 and 3, PRA model results were evaluated in the BWR Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process [4].
- After the completion of the most recent PRA update, an industry peer review in accordance with the combined ASME/ANS PRA Standard [5] and Regulatory Guide 1.200, Revision 2 [6] was performed in November 2010. The results of that assessment are used as the basis for the capability assessment provided in Tables A-1 and A-2.

A summary of the disposition of the 1998 Industry PRA Peer Review facts and observations (F&Os) for the PBAPS, Units 2 and 3, PRA models was documented as part of the statement of PRA capability for MSPI in the PBAPS MSPI Basis Document [4]. As noted in that document, there were no significance level A F&Os from the peer review, and all significance level B F&Os were addressed and closed out with the completion of the current PB205C and PB305C models of record. Also noted in that submittal was the fact that, after allowing for plant-specific features, there are no MSPI cross-comparison outliers for PBAPS (refer to the third bulleted item above).

A Gap Analysis for the 2002 PBAPS, Units 2 and 3, PRA models (PB202 and PB302, respectively) was completed in January 2004. This Gap Analysis was performed against PRA Standard RA-S-2002 [2] and associated NRC comments in draft regulatory guide DG-1122 [3], the draft version of Regulatory Guide 1.200 Revision 0. This gap analysis defined a list of 83 supporting requirements from the Standard for which potential gaps to Capability Category II of the Standard were identified. For each such potential gap, a PRA URE was documented for resolution.

A PRA model update was completed in 2006, resulting in the PB205C and PB305C updated models. In updating the PRA, changes were made to the PRA to address most of the identified gaps, as well as to address other open UREs. Following the update, an assessment of the status of the gap analysis relative to the new model and the updated requirements in Addendum A of the ASME PRA Standard concluded that 59 of the gaps were fully resolved (i.e., are no longer gaps), and another seven were partially resolved.

As indicated above, a PRA model update was completed in 2010, resulting in the PB209A and PB309A updated models. This model was subject to a peer review in November 2010 [A-3]. In general, the peer review results supported the high quality of the PRA model as approximately 95% of all the supporting requirements were characterized as meeting Capability Category II or better. Those supporting requirements that were assessed as not meeting Capability Category II are described in Table A-1 with their impact on this application noted.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
2-2	Section 2.4 documented a number of special initiators based on comparisons or review. However, it's not evident that a structured approach has been performed.	IE-A5	Assessed as meeting Capability Category I.	Not significant as the PBAPS PRA model includes a full range of special initiators which are consistent with many BWRs (e.g. loss of SW, loss of IA, loss of RBCCW, loss of TBCCW, loss of individual 4kV ac buses, and loss of individual 125V dc buses). These are sufficient to determine the EPU impacts.
6-2	The Initiating Event NB PB-PRA-001, Rev.2 addresses grouping in Section 2.5 and summarizes the events in Table 2.6-2. Many events have been subsumed into other events as discussed in Section 2.5. The subsuming is based on simple statements rather than a discussion of event progression, success criteria, timing and operator action. Certain items are not even discussed, but summarized in Table 2.6-2. For example, there is no discussion of Turbine Trip without Bypass, or Pressure Regulator Fails Open or Pressure Regulator Fails Closed (except for some foot notes).	IE-B3	Assessed as meeting Capability Category I.	Not significant as the PBAPS PRA model includes a full range of initiating events which are comparable with many BWRs. These are sufficient to determine the EPU impacts.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
2-5	ISLOCA was analyzed and documented in the IE notebook, but it was based on IPE and no particular consideration of protective interlocks, relevant surveillance test, check valve, etc. The newer failure data from NUREG/CR-6928 could be considered. In addition, the RHR shutdown cooling discharge line appears missing in the analysis.	IE-C14	Assessed as not met. The ISLOCA update has not yet been performed. However, the current ISLOCA values are conservative compared to other sites that have utilized the more detailed methodology.	Not significant given that the current approach is reasonably conservative, and ISLOCA scenarios would be very minimally impacted by EPU.
6-5	The SR calls for Peach Bottom's success criteria to be compared with those for other similar plants. There is no evidence such a comparison was performed.	SC-B5	Assessed as not met. Although a formally documented comparison has not been performed, in practice this type of comparison is done as the results of the model are analyzed and reviewed.	Not significant given that the current success criteria have been validated based on plant-specific MAAP runs or other comparable generic sources.
3-1	Alignment pre-initiators are included for some risk significant systems (i.e., HPCI, RCIC, LPCS, and SLC), but these were not included as a result of a review of procedures and practices. Refer to Sections 2.3.3, 4.3, 5.1, and Appendix B of the HRA Notebook (PB-PRA-004).	HR-A1	Assessed as not met.	Not significant given that several pre-initiators are included in the model, and in any event, the pre-initiators would not be impacted by EPU.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
3-3	As described in Sections 2.3.3, 4.3, 5.1,and Appendix B of the HRA Notebook (PB-PRA-004), the process for the identification of misalignment of modeled equipment does not address common misalignment.	HR-A3	Assessed as not met.	Not significant given that several pre-initiators are included in the model, and in any event, the pre-initiators would not be impacted by EPU.
3-4	The process described in the HRA Notebook (PB-PRA-004) does not establish any rules for screening individual activities. Some System Notebooks (PB-PRA-005) (e.g., HPCI, RCIC, LPCS, SLC) include pre-initiators and identify appropriate screening rules in Section 6.1.5 but do not identify activities which might have been screened.	HR-B1	Assessed as not met.	Not significant given that several pre-initiators are included in the model, and in any event, the pre-initiators would not be impacted by EPU.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
5-8	Table 5.1-4 of the HRA Notebook (PB-PRA-004) includes a number of pre-initiators types (e.g., flow, delta-temperature, steam leak) that are not documented in Table 5.1-2 or Appendix B.	HR-D2	Assessed as meeting Capability Category I. Not all significant pre-initiators were evaluated with an individual detailed HEP analysis. Rather, the event was assigned a 'type' based on the transmitter it is associated with, and the types were assigned an HEP value based on the limited set of detailed pre-initiator evaluations that were performed as described in Appendix B of the HRA notebook (PB-PRA-004).	Not significant given that several pre-initiators are included in the model, and in any event, the pre-initiators would not be impacted by EPU.
1-4	No evidence was found for using plant-specific operational records to determine the time that components were configured in their standby status.	DA-C8	Assessed as meeting Capability Category I. The standby status times are estimated based on the anticipated equipment rotation moving forward. This provides an appropriate level of accuracy for the model.	Not significant given that the estimates utilized are sufficient for determining the EPU impacts.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
6-11	The data from the maintenance Rule is used directly without checking to see if it includes only those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded as required by the SR.	DA-C11	Assessed as not met. This level of refinement would have a very small impact on the actual unavailability values utilized in the model.	Not significant given that the unavailability values utilized are sufficient for determining the EPU impacts.
6-9	Inter-system unavailability, (e.g., HPCI/RCIC systems) data was not evaluated and a value of 1.0E-5 was arbitrarily assigned.	DA-C14	Assessed as not met. RHR, HPSW, and CS loop maintenance terms included for intra-system unavailability terms. However, the SR is not met for inter-system unavailability terms as the model includes coincident outage times for a few pertinent combinations (e.g. HPCI/RCIC, RHR Loops), but since no known overlap existed for these combinations, an arbitrarily small value (1.0E-5) was assigned.	Not significant given that the coincident unavailability values utilized are sufficient for determining the EPU impacts.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
4-5	Per PB-PRA-015 R0 "L2 PRA Analysis Notebook", ISLOCA is classified as Class V "Unisolated LOCA outside containment" per Table 4.3-2 of PB-PRA-015 R0 "L2 PRA Analysis Notebook" detailed assessment and frequency analysis of the ISLOCA was not performed, but rather a simplified approach for determining the ISLOCA frequencies as discussed in section 3.3.3 of PB-PRA-001 R2 "Initiating Events Notebook".	LE-D4	Assessed as meeting Capability Category I.	Not significant given that the current approach is reasonably conservative, and ISLOCA scenarios would be very minimally impacted by EPU.
2-14	No documentation is identified for model uncertainty associated with the plant partitioning.	IFPP-B3	Assessed as not met. The sources of model uncertainty and related assumptions are documented based on the guidance provided in EPRI 1016737 (as endorsed in NUREG-1855). This assessment did address the items to consider per the EPRI guidance which did not include any specific items related to the IFPP plant partitioning category. This indicates that there are no sources of model uncertainty for the IFPP category that need to be considered.	None.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
3-19	Plant-specific experience was gathered as shown in Appendix H of the Internal Flood Evaluation Summary Notebook (PB-PRA- 012). However, only generic flood frequencies were used.	IFEV-A6	Assessed as meeting Capability Category I. It is clear that the available pipe failure data is extremely sparse and the associated uncertainties are quite large. There is essentially no PBAPS specific evidence of internal flooding of the size comparable to that used in the EPRI analysis. As such, any Bayesian update of the generic data would not improve this already sparse data set. Specifically, it is further postulated that it is inappropriate to introduce the false rigor of the Bayesian update process given the unknowns introduced by the failure mechanisms, the generic uncertainty distribution size, and the age related effects. In other words, the past PBAPS specific evidence i.e., past 35+ years of operation is not necessarily characteristic of future performance of the piping systems. This exercise of judgment in the use of relevant data is allowed by the Bayesian update process.	None.

TABLE A-1
STATUS OF GAPS TO CAPABILITY CATEGORY II FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
1-7	Note: This SR is modified by the notes in the RG 1.200. Following those notes, this SR can only be judged to be met at CC I. No assessment was done relating to factors such as pipe whip, humidity, condensation, etc., as required by the RG 1.200 notes.	IFSN-A6	Assessed as meeting Capability Category I. This additional level of refinement would have minimal impact on the internal flooding analysis results.	Not significant given that the overall impact would be minimal and therefore would be very minimally impacted by EPU.
6-15	Inter-area propagation has been addressed in the scenario development. However, flow path via drain lines, and areas connected via backflow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays) do not appear to be addressed.	IFSN-A8	Assessed as meeting Capability Category I. This additional level of refinement would have minimal impact on the internal flooding analysis results.	Not significant given that the overall impact would be minimal and therefore would be very minimally impacted by EPU.

A.2.3 Applicability of Peer Review Findings and Observations

The remaining set of findings from the recent 2010 peer review related to the current ANS/ASME PRA Standard for internal events and internal flood associated with supporting requirements that are otherwise met at Capability Category II are described in Table A-2 with their impact on this application noted.

A.2.4 PRA Quality Summary

Based on the above, the PBAPS PRA is of sufficient quality and scope for this application. The modeling is detailed; including a comprehensive set of initiating events (transients, LOCAs, and support system failures) including internal flood, system modeling, human reliability analysis and common cause evaluations.

TABLE A-2
STATUS OF OPEN FINDINGS FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
3-6	No evidence was found that plant testing procedures were used to define pre-initiator activities that would cause system unavailability or plant trips.	HR-C2 QU-D6	Open – However, the failure modes identified in the SR are already included in the generic or plant-specific data utilized for each system, component, and initiating event modeled.	None.
3-13	Random checks in Appendices B and H in the Component Data Notebook (PB-PRA-004, Volume 2) showed that in some cases the CCF applied was not directly applied in the associated file.	DA-D5	Closed – A separate check was performed on all of the CCF values utilized in the model. The few discrepancies were corrected in the models used for this assessment.	None.

TABLE A-2
STATUS OF OPEN FINDINGS FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
3-14	No confirmation that experience from the plant was used to confirm the applicability of the generic CCF alpha factors to the plant specific conditions.	DA-D6	Assessed as meeting Capability Category I. It is agreed that specific documentation related to the applicability of the use of generic alpha factors was not provided. However, a review of the plant-specific failures listed in Table B-3 of the Component Data Notebook (PB-PRA-004, Volume 1) indicates that there is no evidence of significant common cause failure activity at PBAPS that would render the use of the generic alpha factors questionable.	None.
4-4	Section 3.5.5 of PB-PRA-001, Revision 2 estimates value for recovery from 'loss of DC bus' BUT without the detailed analysis.	HR-G1	Open – Detailed analysis not yet performed. A conservative recovery value of 0.5 is applied in model.	Not significant given that the current recovery value utilized of 0.5 is conservative and the loss of DC bus initiators were not significant contributors to the delta risk assessment.

TABLE A-2
STATUS OF OPEN FINDINGS FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
5-9	Of the four systems (i.e., RPS, ARI, RPT and SLC) identified in Table 2.3-2 of the Event Tree Notebook (PB-PRA-002) as needed to support the reactivity control function only SLC has a System Notebook (PB-PRA-005). Except for SLC, modeling for these systems is primarily point estimates in the Data Notebook (PB-PRA- 010). Without more developed system modeling, system interactions may not be evident. (ARI typically uses RPT for an initiation signal and requires DC power for actuation of air pilot valves.) The inclusion of operator actions (e.g., scram the reactor, trip the recirculation pumps, or initiate ARI) may provide more realistic risk. Some BWRs have associated spurious operation of the reactor mode switch with a failure to scram.	SY-A1 SY-A7	Open – Further refinement could be employed for these systems but is not required for every application of the model.	Not significant given that the current treatment is adequate for the determination of the EPU impacts.

TABLE A-2
STATUS OF OPEN FINDINGS FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
6-3	There is adequate documentation to meet the SR. The treatment of four categories of LOOP is an improvement. However, there is room for further improvement: 1. See F&O written in response to IE-B3 to improve documentation. 2. There are a lot of pages written up to calculate the frequency of Large LOCA, but it does not look like the value is used in the PRA. The documentation can be simplified by just referring to the value used and eliminating the text relating to the unused value. 3. The steam LOCA and liquid LOCA seem be getting lumped together. It is not clear if these LOCAs are treated in the same manner (i.e., using the same success criteria). 4. The ISLOCA analysis has not been updated from the IPE days. 5. It might be useful to document why certain events such as the following are excluded from the PRA: Multiple IORV, Multiple SORV, Stuck-open safety valve.	IE-D2	Open – These comments are either documentation issues or reference issues that addressed by other findings.	None.

TABLE A-2
STATUS OF OPEN FINDINGS FROM THE 2010 PEER REVIEW

FINDING NO.	DESCRIPTION OF FINDING	APPLICABLE SRs	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
6-14	Plant walkdown was conducted to identify flood sources. No specific walkdown was conducted to identify the SSCs in the flood areas or the pathways. These were identified through drawings, and verified by mini-walkdowns at the discretion of the PRA analysts. The walkdown documentation is very sketchy. A lot more information needs to be collected during walkdown to help flood scenario development. The location of drains, curbs, doors, sills need to be identified. The paths through stairwells need to be identified. The flood pathways developed in the flood scenarios need to be verified by walkdown.	IFPP-A4 IFSO-A6 IFSN-A17	Open – This finding relates to providing additional detail in the walkdown sheets which would enhance the fidelity of the model documentation. However, it is not expected to change the results of the internal flood analysis.	None.

A.3 Uncertainty Analysis

RG-1.174 [A-1] identifies three high level types of uncertainties — parameter, model, and completeness uncertainty. These are each discussed in the context of the EPU risk assessment in the sections which follow.

A.3.1 Parameter Uncertainty

The cutset results for the different CDF assessments were reviewed to determine if the epistemic correlation could influence the mean value determination. From the review of the cutsets, it was determined that the dominant contributor cutsets do not involve basic events with epistemic correlations (i.e. the probabilities of multiple basic events within the same cutset for the dominant contributors are not determined from a common parameter value). Per Guideline 2b from EPRI 1016737 [A-2], then it is acceptable to use the point estimate directly in the risk assessment.

To verify that the use of the point estimate is acceptable in these four cases, a detailed Monte Carlo calculation using EPRI R&R workstation UNCERT software was performed to compare the mean value determined from the Monte Carlo simulation as compared to the point estimate. The parametric uncertainty assessment directly takes into account the state-of-knowledge correlations since the basic event database for Peach Bottom is fully populated with the appropriate correlations and corresponding uncertainty parameters. The uncertainty in the HEP estimates was characterized via the application of error factors based on the following HEP ranges. These error factor (EF) assignments are consistent with those determined by the EPRI HRA Calculator using the same cognitive and execution quantification methods.

- HEP < 0.001, assigned EF = 10
- HEP between 0.001 and 0.1, assigned EF = 5
- HEP > 0.1, assigned EF = 1

The results of the parametric uncertainty assessments are provided in Table A-3 below. Figures displaying the probability density function for all of the cases appear after the table. Based on the minimal difference in the comparison of the mean value with the point estimate values provided, the use of the CDF point estimate for this assessment is deemed acceptable.

Note that a similar assessment was performed for the LERF figure of merit and the trend was similar. That is, the parametric mean values were very close to the point estimate mean values.

The results of those assessments are also provided in Table A-3 below. Again, figures displaying the probability density function for all of the cases appear after the table. Based on the minimal difference in the comparison of the mean value with the point estimate values provided, the use of the LERF point estimate for this assessment is deemed acceptable.

Table A-3
PARAMETRIC UNCERTAINTY EVALUATIONS AND COMPARISON TO POINT ESTIMATE RESULTS

Result	**CDF**		**LERF**	
	Pre-EPU	**EPU**	**Pre-EPU**	**EPU**
Propagated Mean Values(1)				
CDF(1) or LERF(1)	3.63E-06/yr	3.73E-06/yr	4.60E-07/yr	4.78E-07/yr
Point Estimate Mean Values(2)				
CDF(2) or LERF(2)	3.60E-06/yr	3.70E-06/yr	4.58E-07/yr	4.74E-07/yr

(1) Developed based on the parametric mean value for each case from a Monte Carlo simulation with 25,000 samples.

(2) Developed based on the point estimate value for each case.

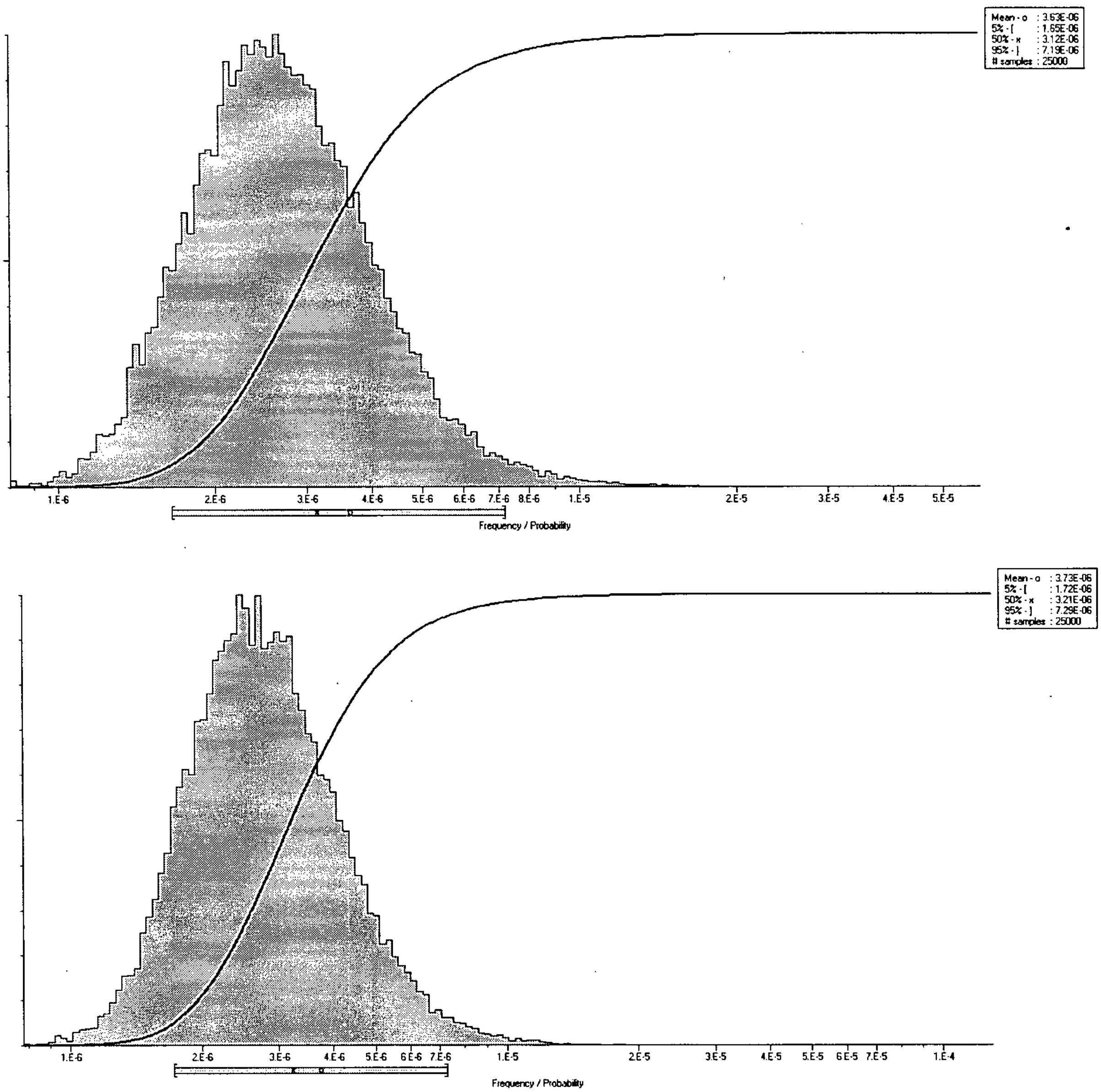
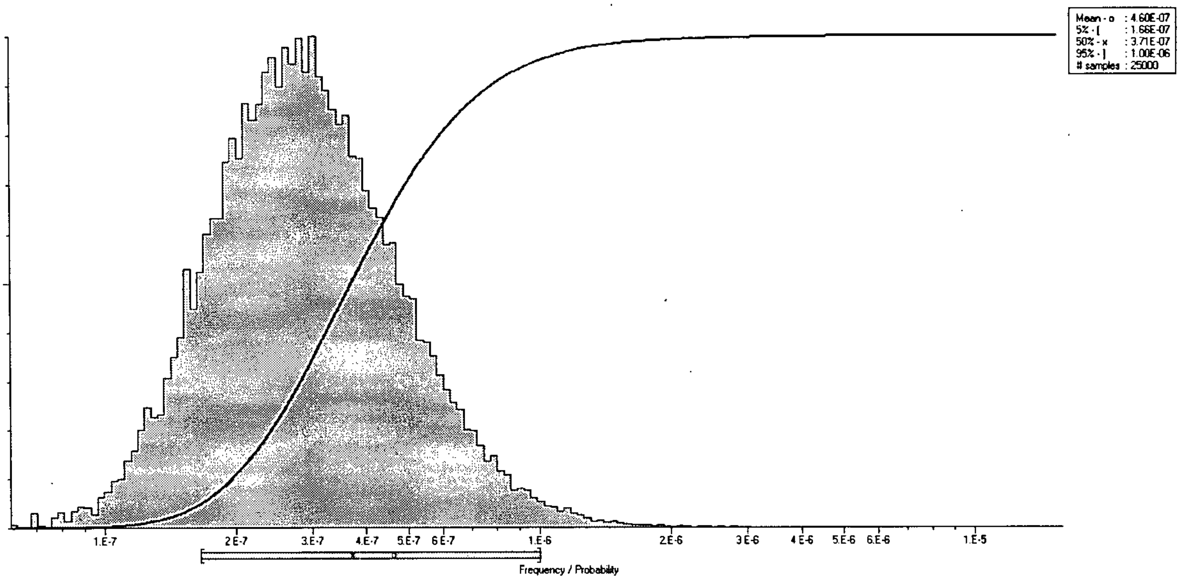


Figure A-1 Pre-EPU and EPU CDF Cases



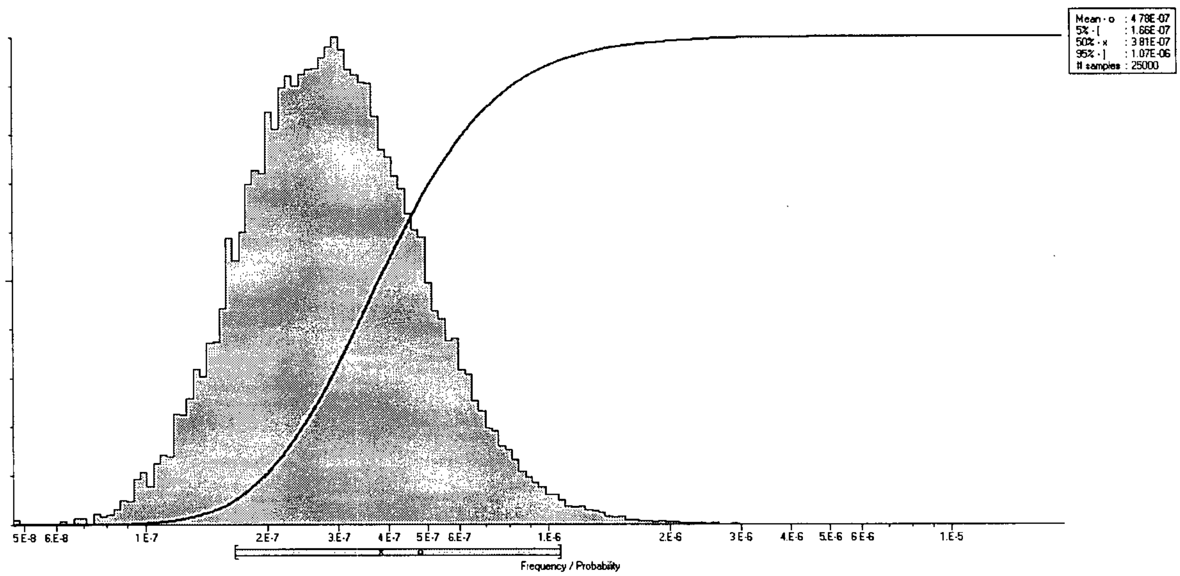


Figure A-2 Pre-EPU and EPU LERF Cases

A.3.2 Model Uncertainty

The assessment of model uncertainty utilizes the guidance provided in EPRI 1016737 [B-2] and in NUREG-1885 [B-3] and considers the following:

1. Characterize the manner in which the PRA model is used in the application
2. Characterize modifications to the PRA model
3. Identify application-specific contributors
4. Assess sources of model uncertainty in the context of important contributors
    a. Also consider other sources of model uncertainty from the base PRA model assessment for the identification of candidate key sources of uncertainty
    b. Screen based on relevance to parts of PRA needed or based on relevance to the results
5. Identify sources of model uncertainty and related assumptions relevant to the application
    a. This involves the formulation of sensitivity studies for those sources of uncertainty that may challenge the acceptance guidelines and an interpretation of the results

A.3.2.1 Characterize the Manner in which the PRA Model is Used in the Application

The manner in which the PRA model is used in this application is fully described in the main body of this report and will not be reproduced here.

A.3.2.2 Characterize Modifications to the PRA Model

There were a few changes made to the model as described in Section 4 of this report. These are summarized below. Additional details are also provided in the application specific model documentation [19].

- The RHR cross-tie mod has been included in the system logic model as described in Section 4.1.2.3.
- The SRV stuck open probabilities have been increased as described in Section 4.1.2.6.
- The LOOP non-recovery probabilities have been adjusted to account for less time available for EPU conditions as described in Section 4.1.3.
- The incorporation of the CST standpipe has been included in the system logic model as described in Section 4.1.4.

- The human reliability analysis has been completely updated based on EPU conditions as described in Section 4.1.6. Table 4.1-2 provides the complete set of independent human error probability values for both pre EPU and EPU conditions. Table 4.1-3 provides the complete set of dependent human error probability values for both pre EPU and EPU conditions.

A.3.2.3 Identify Application-Specific Contributors

Based on the detailed review of the results, the following items are the important contributors to the change compared to the base case results:

- Various independent operator actions – refer to Table 4.1-2
- Various dependent operator actions – refer to Table 4.1-3
- Large LOCA initiating event frequency
- SORV failure probability
- Probability that lack of containment overpressure leads to failure of ECCS from the suppression pool
- Containment isolation failure probability
- Human error probability for implementation of the RHR cross-tie (or effectiveness of RHR cross-tie)

A.3.2.4 Assess Sources of Model Uncertainty in Context of Important Contributors

A review of the identified sources of model uncertainty from the base model assessment as identified by implementing the process outlined in EPRI 1016737 for Peach Bottom was then performed to determine which of those items are potentially applicable for this assessment even though they did not appear as a dominant contributor in the base assessment for the application. Based on this review, some of the items were already identified and many of the items were easily screened, but the following items were added for investigation since they were evaluated to be potentially applicable for this application.

- LOOP frequency and fail to recover probabilities
- The assumption that an RPV Overpressure Protection failure event results in an event equivalent to a Large LOCA
- Common cause failure values

Based on the identified important contributors as summarized in Section A.3.2.3 and the addition of applicable base PRA model sources of uncertainty identified in Section A.3.2.4, the next step is to perform a qualitative assessment or semi-quantitative screening assessment to determine if sources of uncertainty have been utilized in the PRA that affects the important contributors for the application. Since the EPU risk assessment does not readily lend itself to a quantitative screening assessment, a qualitative assessment is then provided for each of the previously identified important contributors or potential sources of uncertainty.

The results of this assessment are shown in Table A-4.

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Various independent and dependent operator actions	Yes	Yes	The credited actions are procedurally directed with the calculated HEP values derived from an accepted methodology that has been peer reviewed to the ASME/ANS PRA standard. Although variations to the HEP values may lead to changes in the risk assessment results, only very bounding assumptions regarding the appropriate HEP values for these individual actions would lead to exceeding the risk metric acceptance guidelines. In any event, the independent and dependent post-initiator HEPs are identified as potential key sources of uncertainty for this application as part of the HEP development as a global source of uncertainty.	Yes – include as part of HEP development as a class

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	**Source of Model Uncertainty for Base Model**	**Application Important Contributor**	**Source of Model Uncertainty Assessment**	**Potential Key Source of Uncertainty**
Large LOCA initiating event frequency	No	Yes	The large LOCA initiating event frequency for PBAPS is based on NRC estimated values. However, the large LOCA frequency is still higher than that reported in even more recent studies (e.g. NUREG-6928). In any event, Section 5.7 includes a separate sensitivity study for the LOCA frequencies, not due to the uncertainty in the frequency, but to account for the potential EPU impacts on piping failure mechanisms.	Yes
SORV failure probability	No	Yes	The SORV failure contribution is just 2.1E-9 for CDF and 9.9E-10 for LERF. The probability increase by ~13%only resulted in a 2.3E-10 increase in CDF and a 1.1E-10 increase in LERF. Therefore, any reasonable variations to this value will not lead to challenging the acceptance guidelines. Therefore, the SORV failure probability is not retained as a potential key source of uncertainty.	No

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Probability that containment isolation failure leading to lack of containment overpressure also leads to failure of ECCS from the suppression pool	Yes	Yes	The base pre EPU model considered a 10% likelihood that given containment overpressure conditions do not exist in certain scenarios, then all ECCS pumps taking suction from the suppression pool would fail. The EPU modifications included the implementation of an RHR cross-tie to eliminate the need for containment overpressure in certain scenarios. An alternative less optimistic assumption for this event (e.g. assuming containment overpressure failure would always lead to loss of ECCS) could actually result in a larger risk reduction than that calculated in the base case for the EPU assessment. Therefore, the probability that lack of containment overpressure leads to failure of ECCS from the suppression pool is identified as a potential key source of uncertainty.	Yes

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	**Source of Model Uncertainty for Base Model**	**Application Important Contributor**	**Source of Model Uncertainty Assessment**	**Potential Key Source of Uncertainty**
Containment isolation failure probability	No	Yes	The dominant contributor to the containment isolation failure probability is based on information from EPRI that has been utilized for various ILRT extension requests. This is conservative for this application and the uncertainty associated for this issue is bounded by the probability that lack of containment overpressure leads to failure of ECCS from the suppression pool described above. Therefore, the containment isolation failure probability is not retained as a potential key source of uncertainty.	No
LOOP frequency and fail to recover probabilities	Yes	No	Uncertainty in the LOOP frequency and recovery probabilities will lead to some change in the calculated deltas since LOOP scenarios comprise a portion of the calculated ΔCDF, but the overall assessment is not limited to only LOOP events. Additionally, the loop initiating event frequency and fail to recover values are fairly well accepted (being based on NUREG-6890). As such the LOOP recovery values are not retained as a potential key source of uncertainty.	No

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Human error probability for implementation of the RHR cross-tie (or effectiveness of RHR cross-tie)	No	Yes	The implementation of the RHR cross-tie does not have a significant impact on the overall CDF and LERF results, but is noted as a potential source of model uncertainty due to its importance in the context of the EPU changes.	Yes
Postulated overpressure failure mode being equivalent to a Large LOCA	Yes	No	The source of model uncertainty from the base model assessment is derived from the assumption that the success criteria for overpressure failures of the RPV are equivalent to the large LOCA success criteria. For this application, however, given that the success criteria are assumed to be the same, the source of model uncertainty is more related to the frequency of overpressure failures (especially with respect to the increased pressures that may arise post-trip from EPU conditions). Therefore, the overpressure failure probability (leading to large LOCAs) is identified as a potential key source of uncertainty.	Yes

Table A-4
Identification of Potential Key Sources Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Common Cause Failure Values	Yes	No	Due to the nature of the EPU evaluation, the change in the risk metrics tended to be dominated by the changes to HEP values and as such CCF values do not play a big role in the risk assessment. Therefore, it is not identified as a potential key source of uncertainty for this application.	No

A.3.2.5 Identify Sources of Model Uncertainty and Related Assumptions Relevant to the Application

Based on the evaluation of important contributors shown in Table A-4, several sensitivity cases were prepared for further exploration. This includes the following cases:

- Human error probability development as a class
- Large LOCA initiating event frequency
- Likelihood that containment isolation failure leading to lack of containment overpressure also leads to failure of all ECCS taking suction from the suppression pool.
- Human error probability for implementation of the RHR cross-tie (or effectiveness of RHR cross-tie)
- RPV overpressure failure probability

The first sensitivity case involves the Human Error Probability (HEP) development as a class. For this sensitivity study, all post-initiator independent and dependent HEP events are set to their 95th percentile values. The results of this sensitivity case are presented in Table A-5.

Table A-5
HEP SENSITIVITY CASE FOR THE EPU RISK ASSESSMENT

CASE	CDF	LERF
Pre-EPU	1.31E-05/yr	1.07E-06/yr
EPU	1.37E-05/yr	1.26E-06/yr
Delta	6.0E-07/yr	1.9E-07/yr
Exceeds Acceptance Guideline for Small Change	No (1.0E-05/yr)	No (1.0E-06/yr)
Exceeds Very Small Acceptance Guideline	No (1.0E-06/yr)	Yes (1.0E-07/yr)

As expected, the results of the sensitivity case show that significant changes to the HEPs have a profound impact on the calculated risk metrics. These results are similar to most BWR PRA uncertainty evaluations when this sensitivity case is performed and is not unexpected. However, even when the extreme assumptions are utilized for the HEP values, only the LERF delta risk results exceed the acceptance guidelines for "very small" changes in risk, and neither the CDF or LERF results exceed the acceptance guidelines for "small" changes in risk.

The second postulated sensitivity case is bounded by the LOCA frequency sensitivity case described in Section 5.7 which includes a variation on all the LOCA frequencies. Therefore, the large LOCA frequency uncertainty is bounded by that case which showed a 16.7% increase in CDF and a 14.0% increase in LERF compared to a 2.8% change in CDF and 3.5% change in LERF for EPU conditions when no changes to the LOCA frequencies are employed. The results of this sensitivity case would not change the conclusions from this analysis.

The third sensitivity case involves the likelihood that containment isolation failure leading to a lack of containment overpressure also leads to failure of all ECCS taking suction from the suppression pool. The base case analysis assumes that this is dictated by a 10% failure likelihood (ZPH-NPSHDXI2 = 0.1). In this sensitivity case, the assumed likelihood value is raised to 100% (i.e., lack of containment overpressure will always fail ECCS, ZPH-NPSHDXI2 =1.0). The results are summarized in Table A-6 which indicates that the base case CDF and LERF would increase but the delta risk metric would actually decrease. This provides an estimate of the maximum benefit of implementing the RHR cross-tie modification that eliminates the need for containment overpressure.

Table A-6
LIKELIHOOD THAT LACK OF CONTAINMENT OVERPRESSURE FAILS ECCS SENSITIVITY CASE FOR THE EPU RISK ASSESSMENT

CASE	CDF	LERF
Pre-EPU	3.88E-06/yr	7.32E-07/yr
EPU	3.71E-06/yr	4.93E-07/yr
Delta	-1.7E-07/yr	-2.4E-07/yr
Exceeds Acceptance Guideline for Small Change	No (1.0E-05/yr)	No (1.0E-06/yr)
Exceeds Very Small Acceptance Guideline	No (1.0E-06/yr)	No (1.0E-07/yr)

The fourth sensitivity case involves taking no credit for implementation of the RHR cross-tie. This is accomplished in the sensitivity case by setting the operator action for implementing the cross-tie to 1.0 (DHU--SPXDXI2). This change leads to a 3.3% increase in CDF and a 7.9% increase in LERF compared to a 2.8% change in CDF and 3.5% change in LERF for EPU conditions when nominal credit is taken for implementation of the RHR cross-tie (i.e., DHU--SPXDXI2=0.06). The results of this sensitivity case would not change the conclusions from this analysis.

The final sensitivity case involves the RPV overpressure failure probabilities for both ATWS and non-ATWS conditions. Based on the discussion in Section 4.1.2.5 no changes to these parameters were warranted (i.e., ARV--COMCPI2 for non-ATWS scenarios and ARV-ATWSCPI2 were both left at their pre EPU values). However, this sensitivity case explored factor of 2 increases to both of these values. Doubling the RPV overpressure failure probabilities resulted in a 5.0% increase in CDF and a 4.6% increase in LERF compared to a 2.8% change in CDF and 3.5% change in LERF for EPU conditions when no changes to the RPV overpressure failure probabilities are employed. The results of this sensitivity case would not change the conclusions from this analysis.

A.3.3 Completeness Uncertainty

The interim Fire PRA model was utilized to obtain quantitative risk metric results as described in Section 4.3. The seismic hazard group was demonstrated to be an insignificant contributor based on qualitative reasoning in Section 4.4. In Section 4.5, the disposition of those hazard groups not included in the PRA for the EPU risk assessment is provided. As discussed there, the majority of those hazard groups were screened based on qualitative considerations from the IPEEE. Section 4.6 presents a consideration of shutdown risk from the EPU changes. Additionally, there are no open items from the recent industry peer review related to model completeness associated with the internal events PRA model.

Therefore, there is no major form of completeness uncertainty that would impact the results of this assessment.

A.4 UNCERTAINTY ANALYSIS CONCLUSIONS

As previously indicated, the uncertainty analysis addresses the three generally accepted forms of uncertainty - parameter, model, and completeness uncertainty. The conclusions from these assessments are as follows.

Parameter Uncertainty

The parameter uncertainty assessment indicated that the use of the point estimate results directly for this assessment is acceptable.

Model Uncertainty

Several sensitivity cases were explored to consider the potential impacts of potential key sources of uncertainty in the risk assessment. The results of the sensitivity studies indicated that although some alternative assumptions could challenge the acceptance guidelines for very small changes in risk, there is no one issue that would result in exceeding the acceptance guidelines for small changes in risk.

Therefore, it is concluded that there are no key sources of model uncertainty that would challenge the characterization of the EPU impacts on the plant as not risk significant.

Completeness Uncertainty

There is no major form of completeness uncertainty that would impact the results of this assessment.

ADDITONAL REFERENCES

[A-1] Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Revision 1, November 2002.

[A-2] *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments*, EPRI Report 1016737, Palo Alto, CA, December 2008.

[A-3] *Peach Bottom Atomic Power Station PRA Peer Review Report*, BWROG Final Report, May 2011.

Appendix B

IMPACT OF EPU ON SHUTDOWN OPERATOR ACTION RESPONSE TIMES

This appendix describes the thermal hydraulic analyses performed to support the assessment that the PBAPS EPU has a non-significant impact on human response times during plant shutdown accident scenarios.

B.1 INTRODUCTION

The risk due to accidents during shutdown is strongly dependent upon the time available from the start of the event to the onset of core damage. As time elapses after shutdown, accidents leading to boiling of coolant within the RPV and consequential inventory losses take more time to evolve. The burden on plant systems decreases as well, introducing the chance of accident mitigation with non-safety, low capacity systems.

The effect of decreasing decay heat on the times to boil and core damage is accounted for in two ways. The first is the calculation of decay heat present at a particular point in the outage. The second takes into consideration the heat capacity of the water and structures in the system available to absorb decay heat before boiling and core damage occur. Both of these aspects are addressed in this appendix to support the assessment of the relationship of decay heat levels and times available in which to perform human actions to prevent core damage during shutdown accident scenarios.

B.2 ASSUMPTIONS

The following assumptions were used in the following calculation of the times to boil off the fuel coolant and reach core damage. These assumptions allow for some simplifications in the calculation, and also allow for an appropriate degree of conservatism in the results.

- The time to boil and time to core damage calculations are appropriate for conditions of RPV vented and maintained at atmospheric pressure.

- The time to core damage is conservatively estimated by calculating the time to reach 2/3 core height, and then extrapolating the time to gap release based on decay heat level ratios by assuming that gap release occurs 0.5 hours after 2/3 core height is reached one day after shutdown. Gap release is the release of fission products in the fuel pin gap, which occurs immediately after failure of the fuel cladding and is the first radiological indication of core damage. This approach is based on calculations performed by Sandia and summarized in SECY-93-190. [B-4]

- There is no heat loss from the system to the surroundings via the water surface or through the vessel walls.

- The calculation of decay heat levels and times to boiling and core damage in this assessment conservatively do not include removal of spent fuel out of the core.

- The decay heat as a function of time after shutdown is derived from a curve fit to the ASB 9-2 Branch Technical Position methodology assuming 100% initial power and 16,000 hours of power operation.

B.3 DECAY HEAT LEVEL CALCULATION

There are several methods available to calculate decay heat as a function of time after shutdown. The NRC has provided an acceptable method of calculating the decay heat rate in Branch Technical Position ASB 9-2 [B-1]. This method uses the following equation:

$$P_s = P_o \left[(1+K)(1/200) \sum_{n=1}^{11} A_n \exp(-a_n t_s) - (1/200) \sum_{n=1}^{11} A_n \exp[-a_n(t_s + t_o)] \right] \qquad \text{(B-1)}$$

Where:	P_s	=	decay heat level (MBtu/hr)
	P_o	=	normal operating power (MBtu/hr)
	t_s	=	time after shutdown (seconds)
	t_o	=	operating history
	K	=	uncertainty factor
		=	0.2 for $t_s < 10^3$, 0.1 for $10^3 < t_s < 10^7$
	A_n, a_n	=	fit coefficients as specified in Reference B-1.

Other less complex formulas have been developed and provide reasonable estimates of decay heat rates. Reference B-2 provides the simplest of these, assuming an infinite power history:

$$P_S(t) = P_O (0.0950)\, t_S^{-0.26} \qquad \text{(B-2)}$$

where $P_s(t)$, P_O and t_S are as defined above. A comparison of Equation B-2 to Equation B-1, assuming 16,000 hours of power operation, shows that Equation B-2 underestimates the decay heat in the first day or two by 10-20%, and it overestimates the decay heat thereafter (by 10-75%). At 70 days after shutdown, the decay heat calculated by Equation B-2 is about 75% higher than that calculated using the ASB 9-2 method [B-1].

Another abbreviated formula is found in Reference B-3. This formula, called the Wigner-Way formula, also includes a factor for the power history:

$$P_S(t) = P_O (0.0622)\, [t_S^{-0.2} - (t_O + t_S)^{-0.2}] \qquad \text{(B-3)}$$

As with Equation B-1, t_O is the operating history in seconds, also assumed to be 16,000 hours for comparison purposes. Equation B-3 shows a better correlation late in the outage, but the first twenty to thirty days after shutdown are under predicted (by 10-20% compared to the ASB 9-2 formula). A separate curve fit to the ASB 9-2 equation can be developed of the form:

$$P_S(t) = P_O (0.02561)\, t_{S(hrs)}^{-0.42371} \qquad \text{(B-4)}$$

where $t_{S(hrs)}$ is the time since shutdown in hours. This simple equation is considered to have an advantage over Equations B-2 and B-3 because it agrees with the ASB 9-2 data to within about 10% over the full time period of interest. Although the agreement is not as accurate as the Wigner-Way formula after about 40 days, the agreement at the critical earlier times is much better. Equation B-4 is often used in industry BWR PSSAs to support boil-off timing calculations.

Using Equation B-4, the decay heat level as a function of time after shutdown is given as:

PBAPS CLTP: $P_S(t) = (3514 \text{ MWt}) (3.4118E6 \text{ Btu/hr} / 1 \text{ MWt}) (0.02561)\, t_{S(hrs)}^{-0.42371}$

$$P_S(t) = (3.07E8)\, t_{S(hrs)}^{-0.42371} \text{ Btu/hr} \qquad \text{(B-5a)}$$

PBAPS EPU: $P_S(t) = (3951 \text{ MWt}) (3.4118E6 \text{ Btu/hr} / 1 \text{ MWt}) (0.02561)\, t_{S(hrs)}^{-0.42371}$

$$P_S(t) = (3.45E8)\, t_{S(hrs)}^{-0.42371} \text{ Btu/hr} \qquad \text{(B-5b)}$$

B.4 RPV HEATUP AND BOILOFF CALCULATIONS

Once the core decay heat rate has been calculated using Equation B-5, the times to fuel coolant boiling and core damage can be calculated using simple heat transfer formulas based on the volume of water available. The principal shutdown states are represented by the following water level configurations:

- normal level
- at the flange level
- reactor cavity flooded

Nominal water volumes and associated heat capacities for use in this calculation are summarized in Table B-1, [B-5].

Time to Boil

The time required for the vessel water to reach the boiling temperature (given loss of coolant decay heat removal) is represented by the following equation:

$$t_b = E_{boil} / P_s(t) \quad \text{hrs.} \tag{B-6}$$

where:

t_b	=	time to boil (hours)
E_{boil}	=	$E_{water} + E_{struct}$
E_{water}	=	energy absorbed by heated water volume to reach saturation (MBtu)
E_{struct}	=	energy absorbed by fuel and clad (MBtu)
$P_s(t)$	=	decay heat level (MBtu/hr),

and

E_{water}	=	$V/v * (h_{Tsat} - h_{Tinit})$
V	=	volume of water that heats up to the saturation temperature (ft^3)
v	=	specific volume of water at T_{init} (assumed constant at 0.0167 ft^3/lb_m over the temperature range of interest)
h_{Tsat}	=	enthalpy of water at T_{sat}, 212°F (Btu/lb_m),
h_{Tinit}	=	enthalpy of water at the initial RPV temperature, T_{init} (Btu/lb_m);

and

E_{struct}	=	$MCp_{struct} (T_{sat} - T_{init})$
MCp_{struct}	=	configuration specific structure heat capacity (Btu/°F - See Table B-1)

Since the specific heat of water is 1.0 Btu/(lb_m°F), the difference in the enthalpies in the E_{water} expression above ($h_{Tsat} - h_{Tinit}$) is equivalent to the temperature difference in the E_{struct} expression ($T_{sat} - T_{init}$). This allows the complete expression for E_{boil} to simplify to:

$$E_{boil} = [(V/v) + MCp_{STRUCT}] * [T_{SAT} - T_{init}] \tag{B-7}$$

Substituting in the appropriate constant values, Equation B-7 can be rewritten as:

$$E_{boil} = C * [212 - T_{init}] \qquad \text{(B-8)}$$

where the constant C is calculated for each of the water volumes and structure capacities given in Table B-1. Thus, with the initial temperature, T_{init} in °F and the decay heat load, $P_s(t)$ in Btu/hr, the time to reach saturation for the different configurations are given by Equations B-9 through B-13.

$$t_{b,\,2/3\ core\ height} = 0.32E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \qquad \text{(B-9)}$$

$$t_{b,TAF} = 0.43E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \qquad \text{(B-10)}$$

$$t_{b,Normal\ Level} = 0.80E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \qquad \text{(B-11)}$$

$$t_{b,Flange\ Level} = 1.11E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \qquad \text{(B-12)}$$

$$t_{b,Cavity\ Flooded} = 7.56E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \qquad \text{(B-13)}$$

where $P_s(t)$ is the decay heat level (refer to Equation B-5) and T_{init} is the initial water temperature (e.g., 140°F early in the outage before cavity flooded and 100°F later in the outage after the cavity flooded).

Table B-1

PBAPS NOMINAL WATER VOLUMES AND HEAT CAPACITIES FOR THE TIME TO BOIL AND TIME TO CORE DAMAGE CALCULATIONS

Water Level	Water Volume (ft^3)	Heat Capacity (Btu/°F) [1]	
		Water	Structure
2/3 Core Height	5272	0.32E6	(2)
Top of Active Fuel	7252	0.43E6	(2)
Normal Level	13,412	0.80E6	(2)
Flange Level	18,617	1.11E6	(2)
Cavity Flooded	126,229	7.56E6	(2)

NOTES:

(1) The term heat capacity is used in Eq. B-8. The water heat capacity is defined as Volume/v (where v is the specific volume of water and is assumed constant at 0.0167 ft3/lbm). The specific heat for water was assumed to be 1.0 Btu/(lb_m°F). Refer to text on preceding pages for further details.

(2) Structural heat capacities are conservatively not credited in this calculation.

Time to Uncover Fuel (Boil Off) and Core Damage

The time to uncover the core due to boil off (due to loss of coolant decay heat removal) is the sum of the time required to bring the full heated water volume to saturation and the time to boil off an equivalent volume of water that lies above the core. This can be represented by an equation similar in format to the time to boil equation (Equation B-6):

$$t_{cu} = E_{total} / P_S(t) \qquad \text{(B-14)}$$

where:

t_{cu} = time to uncover the core (hours)

E_{total} = $E_{boil} + E_{boiloff}$

E_{boil} = energy absorbed to reach saturation as defined for Equation B-6 (MBtu)

$E_{boiloff}$ = energy absorbed by the water that vaporizes during boiloff (MBtu),

and

$E_{boiloff}$ = $V_b / v_{sat} * (h_{fg})$

V_b = equivalent volume of water that must vaporize for the collapsed level to reach TAF (ft^3)

v_{sat} = specific volume of water at saturation (T_{sat} = 212°F), or 0.0167 ft^3/lb_m

h_{fg} = heat of vaporization at 212°F and 14.7 psia, or 970.32 Btu/lb_m.

With constant values again assumed where appropriate, Equations B-15 through B-17 below provide the time to uncover the core for the different shutdown water level configurations:

$$t_{cu,Normal\ Level} = [0.80E6 * (212 - T_{init}) + 3.58E8] / P_S(t) \quad \text{hours} \qquad \text{(B-15)}$$

$$t_{cu,Flange\ Level} = [1.11E6 * (212 - T_{init}) + 6.60E8] / P_S(t) \quad \text{hours} \qquad \text{(B-16)}$$

$$t_{cu,Cavity\ Flooded} = [7.56E6 * (212 - T_{init}) + 6.91E9] / P_S(t) \quad \text{hours} \qquad \text{(B-17)}$$

where $P_s(t)$ is the decay heat level (refer to Equation B-5)

This analysis assumes the initial bulk water temperature is 140F for days 0 through 5; 120F for days 6 through 10; and 100F for days 11 and beyond.

The time to uncover the core with the existing power level (CLTP) is 9.3 hours (8.2 hrs for the EPU case) at one day into the outage if the water level is at the RPV flange elevation at the time of a loss of decay heat removal event. The available time greatly exceeds 24 hours at one day into the outage when the water level is flooded up into the refueling cavity (over 84 hours for the EPU case).

For the impact on shutdown human error probabilities, it is necessary to know the approximate time of core damage so that this time can be used as the maximum allowable time window rather than conservatively estimating the time to reach an uncovered core. For this PBAPS EPU evaluation, the time to core damage is estimated by incorporating the additional time available from boiloff down to 2/3 core height, and then extrapolating the time to gap release by assuming that gap release occurs 0.5 hours after 2/3 core height is reached one day after shutdown [B-4]. The resulting equation for core damage, t_{cd}, is:

$$t_{cd} = t_{cu} + [1.16E8 + 0.5 * P_S(1d)] / P_S(t) \quad \text{hours} \qquad \text{(B-18)}$$

where:

- 1.16E8 represents the amount of decay heat required to boildown from TAF to 2/3 core height (i.e., [(7252-5262)/0.0167] x 970.32).
- $P_S(1d)$ is the decay heat 1 day after shutdown (refer to Eq. B-5)
- $P_S(t)$ is the decay heat as a function of time after shutdown (refer to Eq. B-5)

This equation for estimating the time to core damage during refueling incidents is the approach typically used in U.S. industry BWR PSSAs. This equation was developed in the BWR PSSA industry to reflect BWR fuel heatup timing estimates provided in NSAC-169 and SECY-93-190 [B-4, B-8]. SECY-93-190 reports that fuel heatup calculations performed for Grand Gulf by Sandia show that at 4 days after shutdown approximately 5 hours are available between reaching TAF and before fuel pin gap release occurs; and almost 9 hours is available at 15 days after shutdown.

Given the nature of shutdown risk, the time to core damage due to boil-off is not static but increases with increasing times after shutdown. An equation is used for ease of modeling shutdown incidents. Although one may use MAAP runs to estimate the time to core damage (as is done in the at-power PRA), it is not practical given that numerous different runs would be required for different times after shutdown.

Comparisons of the time to core damage due to boil off (given loss of coolant decay heat removal) for the normal, RPV flange, and cavity flooded water level configurations for the CLTP and the EPU cases are provided in Tables B-2 through B-4. Note that the times to core damage for the flood-up configuration are in the range of multiple days (much longer than the time frames considered in PRAs).

B.5 EPU IMPACT ON SHUTDOWN RISK

Impact Due to Changes in HEPs

The primary impact of the EPU on risk during shutdown operations is the decrease in allowable operator action times in responding to off-normal events.[1] However, as can be seen from Tables B-2 through B-4, the reduction in times to core damage (i.e., CLTP case compared to EPU case) is on the order of 10%. Such small changes in already lengthy allowable operator response times result in negligible changes (<<1%) in calculated human error probabilities.

The allowable operator action times to respond to loss of heat removal scenarios during shutdown operations are many hours long. Very early in an outage the times are approximately 5-6 hours; later in an outage the times are dozens of hours. A reduction from 7.1 hours to 6.4 hours (refer to "1 Day After Shutdown" case in Table B-2) in allowable action times would not result in a significant increase in human error probabilities for most operator actions using current human reliability analysis methods. The allowable timing reductions for times later in the outage would result in indiscernible changes in HEPs using current human reliability analysis methods.

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[1] Another postulated impact is any changes to system success criteria during shutdown operations (specifically with respect to decay heat removal systems) that may result from the EPU. A postulated impact would be that the time into the outage at which backup low capacity heat removal options would be sufficient to prevent coolant boiling would be extended a number of hours. Such a postulated impact is estimated to result in an insignificant change in shutdown risk (e.g., 1% or less change in shutdown CDF).

Impact Due to Changes in Offsite AC Recovery Failure Probabilities

In addition to traditional human error probabilities, offsite AC recovery failure probabilities can be influenced by changes in allowable action times. An approximate calculation is performed here to estimate the impact on shutdown risk due to changes in the offsite AC recovery failure probability. The calculation is described as follows:

- A 30-day refueling outage is assumed and is divided into the following four (4) phases:

    - Day 1 of the outage
    - Day 2 of the outage
    - Days 3-29 of the outage
    - Day 30 of the outage

- These phases are defined to address the higher decay heat in the beginning days of the outage, the "flooded-up" days in the middle of the outage when decay heat issues are not the main risk contributor, and the end of the outage when the coolant level is lowered back down into the vessel.

- The following initial water level configurations are assumed for the phases:

    - Day 1 of the outage (NORMAL RPV LEVEL)
    - Day 2 of the outage (RPV FLANGE LEVEL)
    - Days 3-29 of the outage (FLOODED UP)
    - Day 30 of the outage (NORMAL)

- A review of industry BWR PSSAs (Cooper, Duane Arnold, Dresden, Fermi, LaSalle, Nine Mile Pt. U-2, Quad Cities, and Columbia) was performed to assist in defining the contribution of LOOP/SBO accident scenarios to the CDF of each of the above general phases. Based on the review, the CDF contribution from LOOP/SBO scenarios is significant in the first few days of the outage when the decay heat is higher, it drops significantly in the middle of the outage when decay heat is lower and the cavity is flooded (draindown events dominate these periods), and then it increases at the end of the outage when the coolant level is lowered back down into the vessel.

  Based on the above information, the LOOP/SBO contributions to CDF as a function of outage phase are assumed here as follows:

Outage Phase	LOOP/SBO Contribution to Phase CDF
- Day 1	30%
- Day 2	30%
- Day 3-29	10%
- Day 30	20%

- The review of industry PSSAs also supported the estimation of the contributions to overall shutdown CDF from the different phases of the outage. This review indicates the majority of the outage CDF is comprised of the middle portion of the outage (primarily due to the fact that this is the longest period of the outage, and involves significant equipment outages).

  Based on the above information, the phase CDF contributions to overall outage CDF are assumed here as follows:

Outage Phase	Contribution to Outage CDF
- Day 1	10%
- Day 2	15%
- Day 3-29	70%
- Day 30	5%

- INEEL/EXT-04-02326 is used here to estimate changes in offsite AC recovery failure probabilities due to reductions in allowable timings. [B-6] The INEEL "composite" (i.e., integrated data for plant, switchyard, grid, and weather related LOOP events) LOOP non-recovery curve for LOOP events experienced during shutdown conditions is used. The LOOP non-recovery probabilities are determined for CLTP and EPU core damage times, and then ratioed to model the impact of the EPU on the LOOP recovery failure probabilities.

- The assessment is performed on a normalized CDF basis.

This calculation is summarized In Table B-5. As can be seen from Table B-5, the increase in shutdown CDF due to increases in AC power recovery failure probabilities due to the EPU is estimated at approximately 1%.

Summary

Based on the above discussions and calculations, the qualitative conclusions of this assessment is that the PBAPS EPU has an non-significant impact on shutdown risk (approximately 1% increase in shutdown CDF). This estimate considers the EPU impact on the shutdown post-initiator HEPs, offsite AC recovery failure probabilities, and decay heat removal systems success criteria.

Table B-2

TIME TO CORE DAMAGE DUE TO BOIL OFF
(Initial Water Level: Normal Level)

Days After Shutdown	Time to Core Damage (hrs.)	
	CLTP	EPU
1	7.1	6.4
5 (1)	14.1	12.7
10 (1)	19.5	17.5
15 (1)	23.8	21.3
20 (1)	26.9	24.1
25 (1)	29.5	26.5
30	31.9	28.6

NOTE:

(1) The days marked with the footnote are not directly applicable to the modeled outage schedule for the water level configuration of this table (i.e., during the first couple days of the outage the water level is low, but then for the majority of the outage the water level is at the spent fuel pool level, and then is lowered again at the end of the outage), but are provided to illustrate the increasing trend in time available.

Table B-3

TIME TO CORE DAMAGE DUE TO BOIL OFF
(Initial Water Level: RPV Flange Level)

Days After Shutdown	Time to Core Damage (hrs.)	
	CLTP	EPU
1	11.2	10.0
5 [(1)]	22.2	19.8
10 [(1)]	30.5	27.3
15 [(1)]	37.1	33.2
20 [(1)]	41.9	37.5
25 [(1)]	46	41.2
30	49.7	44.5

<u>NOTE:</u>

(1) The days marked with the footnote are not directly applicable to the modeled outage schedule for the water level configuration of this table (i.e., during the first couple days of the outage the water level is low, but then for the majority of the outage the water level is at the spent fuel pool level, and then is lowered again at the end of the outage), but are provided to illustrate the increasing trend in time available.

Table B-4

TIME TO CORE DAMAGE DUE TO BOIL OFF
(Initial Water Level: Cavity Flooded)

Days After Shutdown	Time to Core Damage (hrs.)	
	CLTP	EPU
1 (1)	95.3	84.8
5	188.5	167.7
10	257.9	229.5
15	312.1	277.8
20	352.6	313.8
25	387.6	344.9
30 (1)	418.7	372.6

NOTE:

(1) The days marked with the footnote are not directly applicable to the modeled outage schedule for the water level configuration of this table (i.e., during the first couple days of the outage the water level is low, but then for the majority of the outage the water level is at the spent fuel pool level, and then is lowered again at the end of the outage), but are provided to illustrate the increasing trend in time available.

Table B-5

ESTIMATED IMPACT ON SHUTDOWN RISK DUE TO OFFSITE AC RECOVERY FAILURE PROBABILITY INCREASES DUE TO EPU

Outage Phase	Initial Water Level	Phase Contribution to Overall SD CDF (CLTP)(1)	LOOP/SBO Contribution to Phase CDF(1)	Time to Core Damage (hrs)		Factor Increase in Offsite AC Recovery Failure Probability(4)	Phase Contribution to Overall S/D CDF (EPU)(5)
				CLTP(2)	EPU(2)		
Day 1	Normal	0.10	0.30	7.1	6.4	1.14	0.104
Day 2	RPV Flange	0.15	0.30	15.0	13.5	1.15	0.157
Days 3-29	Flooded	0.70	0.10	312.1(3)	277.8(3)	negligible	0.700
Day 30	Normal	0.05	0.20	31.9	28.6	1.16	0.050
Normalized CDF (CLTP):		1.00				Normalized CDF (EPU):	1.011

<u>Notes to Table B-5</u>:

(1) Approximated based on review of industry BWR PSSAs (Cooper, Duane Arnold, Dresden, Fermi, LaSalle, Nine Mile Pt. U-2, Quad Cities, and Columbia).

(2) Calculated using Eq. B-18.

(3) The time to core damage for the "Days 3-29" phase of the outage is based on the 15th day.

(4) Based on use of generic offsite AC recovery failure probability information from INEEL/EXT-04-02326 ("composite" shutdown LOOP duration curve). For example, at 6.1 hours the INEEL/EXT-04-02326 'composite" shutdown LOOP AC recovery failure probability is approximately 6.23E-2 and at 6.4 hours it is 5.88E-2 (an increase of 1.14x).

(5) Calculated as:

= [LOOP/SBO fraction of outage phase (impacted)] + [Non-LOOP/SBO fraction of outage phase (not impacted)]

= [3rd Column x 4th Column x 7th Column] + [3rd Column x (1.0 - 4th Column)]

REFERENCES

[B-1] U. S. Nuclear Regulatory Commission, *Residual Decay Heat Energy for Light-Water Reactors for Long-Term Cooling,* Branch Technical Position 9-2.

[B-2] M.M. El-Wakil, Nuclear Heat Transport, International Textbook Company, 1971.

[B-3] K. Way, E. Wigner, *The Rate of Decay of Fission Products*, (Phys. Rev., 73, 1948, pp. 1318-1330)

[B-4] U. S. Nuclear Regulatory Commission, *Regulatory Approach to Shutdown and Low Power Operations*, SECY-93-190, July 12, 1993, Enclosure: Draft Regulatory Analysis in Accordance with 10CFR50.109, dated February 1993.

[B-5] ERIN Engineering and Research, Inc., *PBAPS Probabilistic Shutdown Safety Assessment (PSSA)*, 1999.

[B-6] INEEL/EXT-04-02326, *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003*, Draft, October 2004.

[B-7] Electric Power Research Institute, *Safety Assessment of BWR Risk During Shutdown Operations*, NSAC-175L, Final Report, August 1992.

[B-8] Electric Power Research Institute, *Analysis of BWR Fuel Heatup During a Loss of Coolant While Refueling*, NSAC-169, September 1991.

Attachment 13

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

Flow Induced Vibration

1.0 INTRODUCTION

This Attachment to the submittal provides a detailed discussion of the analyses and testing program undertaken to provide assurance that unacceptable flow induced vibration (FIV) issues are not experienced at Peach Bottom Atomic Power Station (PBAPS) due to extended power uprate (EPU) implementation for affected piping systems.

Increased flow rates and flow velocities during operation at EPU conditions are expected to produce increased FIV levels in some systems. As discussed in Section 3.4.1 of Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," the Main Steam (MS) and Feedwater (FW) system piping vibration levels should be monitored because their system flow rates will be significantly increased [2].

In December 2008, the Boiling Water Reactor Owners' Group (BWROG) issued NEDO-33159, Revision 2, "Extended Power Uprate (EPU) Lessons Learned and Recommendations," based on operating experience (OE) and evaluations from Boiling Water Reactor (BWR) plants that have previously implemented EPUs and from plants currently performing pre-EPU evaluations [1]. NEDO-33159 states:

> "Since the majority of EPU-related component failures involve flow induced vibration, the BWROG EPU Committee held a vibration monitoring and evaluation information exchange meeting of industry experts in June 2004. The committee determined that the current process of monitoring large bore piping systems in accordance with the requirements of American Society of Mechanical Engineers (ASME) Operation and Maintenance (O&M) Part 3 is sufficient to preclude challenges to safe shutdown. Increases in large bore piping vibration levels are a precursor to increased vibration levels in attached small bore piping and components."

Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," was revised in 2007 to Revision 3. In addition to guidance for vibration assessment of reactor internals, this regulatory guide provides helpful information on methods for evaluating the potential adverse effects from pressure fluctuations and vibrations in piping systems for boiling water reactor (BWR) nuclear power plants. However, additional guidance is provided with regard to piping vibration. The guidance is primarily directed to initial start-up of new plants, with general guidance interpreted for use in power uprate power ascension testing. Where applicable, this guidance has been incorporated into the EPU monitoring program for piping vibration at PBAPS.

In addition to MS and FW, the related Extraction Steam (ES), Condensate (CD) and Heater Drain (HD) systems also experience similar flow increases under EPU conditions and are included in the EPU vibration monitoring program. Other systems experience insignificant or no increase in flow and; therefore, are not included in this program. Piping system segments which have been analyzed for flow induced vibrations were selected such that the remaining piping system segments which were not specifically analyzed are bounded by the evaluations.

This document describes the piping vibration monitoring program to be implemented at PBAPS during power ascension to confirm acceptable vibration levels at EPU power. It addresses systems impacted by EPU and identifies locations on those systems where monitoring equipment will be installed. This document also describes the techniques to be used for collecting and storing the vibration data.

2.0 SUSCEPTIBILITY AND MONITORING

The MS and FW piping will experience higher mass flow rates and flow velocities under EPU conditions. When power is increased from current licensed thermal power (CLTP) to EPU conditions, steady state FIV levels are conservatively expected to increase in proportion to the flow velocity squared. Thus, the vibration levels of the MS and FW piping are expected to increase by approximately 35% and 30%, respectively, based on a steam flow velocity increase of up to 16% and a feedwater flow velocity increase of 14%. Other possible sources of increased vibration, such as flow instabilities or acoustic resonance as a result of increased flow velocities, may contribute to EPU vibration levels.

Flow rates in portions of the CD, ES and HD systems increase similarly to MS and FW, and are therefore susceptible to increased vibrations at EPU conditions.

Based on the potential for significantly increased vibrations on the systems identified above, a confirmatory test program will be implemented to monitor piping and attached component vibration levels on the identified systems during initial power ascension to EPU conditions.

Piping in the drywell and inaccessible piping outside containment will be monitored using accelerometers installed at selected locations on the piping and attached components. The accelerometers will be wired to remote data acquisition systems located in the reactor and turbine buildings. Piping outside containment that is included in the monitoring program and is accessible during plant operation will be monitored either remotely or by performing visual observations and by taking vibration measurements using hand-held vibration instruments during power ascension to EPU conditions.

Small bore branch piping is susceptible to the effects of the associated large bore piping FIV. Small bore piping assessments will be performed to identify potentially susceptible configurations, and any modifications required to reduce vibration susceptibility will be made prior to EPU power ascension. The assessments will be conducted in accordance with the guidance and tools provided in EPRI's Fatigue Management Handbook (FMH). Walkdowns of the systems impacted by EPU flow increases will be performed to identify if there are any additional potentially susceptible small bore line configurations. Any necessary small bore line modifications will be made prior to EPU power ascension.

3.0 EPU VIBRATION MONITORING PROGRAM

3.1 Overview

The portions of the MS, FW, CD, HD and ES system included in the EPU vibration monitoring program have been selected based on evaluation of the flow increases resulting from EPU implementation. Analyses using detailed methods have been performed to establish the specific EPU vibration monitoring locations and associated acceptance criteria. The EPU flow increase evaluation and vibration analysis results form the bases for EPU vibration monitoring.

Several MS-associated components will also be monitored. Although PBAPS does not have a history of safety-relief valve maintenance issues due to vibration, selected relief valves will be instrumented with accelerometers, as well as four other power-operated valves. This is in response to industry OE from an earlier EPU project. Valves selected for monitoring will make a representative sample of the effect of EPU flow changes on the vibration levels at the primary valves in the system with symmetry between trains, loops and units considered to remove unnecessary redundancies.

3.2 Vibration Monitoring Location and Acceptance Criteria Development

3.2.1 MS and FW Piping (Drywell and Turbine Building)

Detailed models of the MS and FW piping for both inside containment and outside containment were developed for this evaluation. A flat "1g" response spectrum with increases at potential flow induced vibration frequencies was applied up to 250 Hz in each of the three orthogonal directions for MS and 200 Hz in each of the three orthogonal directions for FW piping. Static loads, such as weight and thermal expansion, are not considered since these loads do not contribute to cyclic vibratory loading of the piping system. Additionally, seismic (inertia and anchor movements) and turbine stop valves loads are not considered, since these loads are transient dynamic loads that do not contribute to the steady-state cyclic vibratory loading of the system.

The results of the piping analysis are provided in terms of the accelerations, displacements, and the stresses at each node. The overall values at each node were obtained by combining the results for all three orthogonal directions using the Square Root of the Sum of Squares (SRSS) method. Adjustment factors were calculated using the maximum endurance stress values and the guidance of 2009 ASME O&M-S/G, Part 3 (OM-3) [3] and the maximum stress values from the piping analysis for each of the maximum alternating stress intensity locations.

Allowable displacement (mils peak-peak, primary) and acceleration (g's-peak, secondary) limits at the selected measurement locations were calculated based on the analysis results and ASME endurance stress limits for steady state vibration per OM-3. The primary acceptance criteria are in terms of displacement, which is proportional to pipe stress. The secondary criteria, when provided, are in terms of acceleration, which are native units to transducers used for monitoring.

The MS displacement and acceleration limits are applicable for vibration frequencies up to 250 Hz, which covers the frequency range in which the most significant structural displacement responses are expected, as well as safety relief valve and safety valve standpipe frequencies. Piping displacements due to excitation frequencies above 50 Hz are typically insignificant relative to the lower frequency displacements, and thus significant piping displacements at safety relief valve standpipe frequencies are not expected.

The vibration monitoring locations were selected based on the vibration response from the modal analysis results, composite vibration displacements and effective vibration accelerations that occurred at these points relative to other locations. The measurement locations were also selected such that the general overall piping vibration responses would be sufficiently reflected and the effects of significant vibration would not be missed. Symmetry between trains, loops and units was considered to remove unnecessary redundancy and to minimize the overall number of analyses performed while selecting a representative number of monitoring locations. The EPU vibration monitoring locations determined for the MS and FW piping inside containment from the vibration analyses are summarized in Tables 3-1 and 3-3 for Units 2 and 3, respectively. Tables 3-2 and 3-4 provide the same information for the MS and FW piping outside containment.

Table 3-1
Drywell EPU Monitoring Locations for MS and FW, PBAPS Unit 2

System	Location[1]	Direction	Allowable Peak-to-Peak Displacement, mils	Description
MS	4	X	48	At support M2191-2-HB3, in the X (north-south) and Z (east-west) directions.
MS	4	Z	32	
MS	10J	X	24	At support M2191-2-HB4 in the X (north-south), Y (vertical) direction and Z (east-west) directions.
MS	10J	Y	16	
MS	10J	Z	22	
MS	80	X	22	At support M2191-2-HA3 in the X (north-south), Y (vertical) and Z (east-west) directions
MS	80	Y	38	
MS	80	Z	24	
MS	15	X	26	At support M2191-2-HA1 in the X (north-south), and Z (east-west) directions.
MS	15	Z	94	
FW	400	X	252	At pipe support 6DDNL-H33, in the X (east-west), Y (vertical), and Z (north-south) directions.
FW	400	Y	278	
FW	400	Z	272	
FW	200	X	143	At pipe support 6DDNL-H42, in the X (east-west), Y (vertical), and Z (north-south) directions.
FW	200	Y	40	
FW	200	Z	90	

Note (1): Since the Unit 2 and 3 piping geometries are similar, only the Unit 3 piping was modeled. Therefore, the Unit 2 locations are identified with the corresponding Unit 3 node numbers.

Table 3-2
Turbine Building EPU Monitoring Locations for MS and FW, PBAPS Unit 2

System	Location[1]	Direction	Allowable Peak-to-Peak Displacement, mils	Description
MS	24	X	500	At support IDB-H10, in the X (north-south) direction, Y (vertical) direction, and Z (east-west) direction
MS	24	Y	130	
MS	24	Z	284	
MS	52	X	190	At support IDB-H33, in the X (north-south) direction, Y (vertical) direction, and Z (east-west) direction
MS	52	Y	234	
MS	52	Z	202	
MS	942	X	284	At the low point drain line branch connection to the turbine lead, in the X (north-south) direction and Z (east-west) direction.
MS	942	Z	500	
MS	78	X	136	At support IDB-H77, in the X (north-south) direction and Y (vertical) direction
MS	78	Y	216	
MS	922	X	126	At the low point drain line branch connection to the turbine lead, in the X (north-south) direction and Z (east-west) direction.
MS	922	Z	500	
FW	59	Y	294	At support 2-6DD-S2, in the Y (vertical) and Z (east-west) directions
FW	59	Z	276	
FW	175	X	306	At support 2-6DD-H71 in the X (north-south) and Y (vertical) directions
FW	175	Y	398	
FW	310	Y	336	At support 2-6DD-H15 in the Y (vertical) and Z (east-west) directions
FW	310	Z	330	

FW	435	X	216	At support 2-18GF-H372, in the X (north-south) and Z (east-west) directions
FW	435	Z	286	

Note 1: Since the Unit 2 and 3 piping geometries are similar, only the Unit 3 piping was modeled. Therefore, the Unit 2 locations are identified with the corresponding Unit 3 node numbers.

Table 3-3

Drywell EPU Monitoring Locations for MS and FW, PBAPS Unit 3

System	Location	Direction	Allowable Peak-to-Peak Displacement, mils	Description
MS	4	X	48	At support M2191-3-HB3, in the X (north-south) and Z (east-west) directions.
MS	4	Z	32	
MS	10J	X	24	At support M2191-3-HB4 in the X (north-south), Y (vertical) direction and Z (east-west) directions.
MS	10J	Y	16	
MS	10J	Z	22	
MS	80	X	22	At support M2191-3-HA3 in the X (north-south), Y (vertical) and Z (east-west) directions.
MS	80	Y	38	
MS	80	Z	24	
MS	15	X	26	At support M2191-3-HA1 in the X (north-south), and Z (east-west) directions
MS	15	Z	94	
FW	400	X	252	At pipe support 3-6DDNL-H33, in the X (east-west), Y (vertical), and Z (north-south) directions
FW	400	Y	278	
FW	400	Z	272	
FW	200	X	143	At pipe support 3-6DDNL-H42, in the X (east-west), Y (vertical), and Z (north-south) directions.
FW	200	Y	40	
FW	200	Z	90	

Table 3-4

Turbine Building EPU Monitoring Locations for MS and FW, PBAPS Unit 3

System	Location	Direction	Allowable Peak-to-Peak Displacement, mils	Description
MS	24	X	500	In the X (north-south) direction, Y (vertical) direction, and Z (east-west) direction, at support 3-IDB-H10
MS	24	Y	130	
MS	24	Z	284	
MS	52	X	190	In the X (north-south) direction, Y (vertical) direction, and Z (east-west) direction, at support 3-IDB-H33
MS	52	Y	234	
MS	52	Z	202	
MS	942	X	284	In the X (north-south) direction and Z (east-west) direction, at the low point drain line branch connection to the turbine lead
MS	942	Z	500	
MS	78	X	136	In the X (north-south) direction and Y (vertical) direction, at support 3-IDB-H77
MS	78	Y	216	
MS	922	X	126	In the X (north-south) direction and Z (east-west) direction, at the low point drain line branch connection to the turbine lead
MS	922	Z	500	
FW	59	Y	294	at support 3-6DD-S2, in the Y (vertical) and Z (east-west) directions
FW	59	Z	276	
FW	175	X	306	at support 3-6DD-H71 in the X (north-south) and Y (vertical) directions
FW	175	Y	398	
FW	310	Y	336	at support 3-6DD-H15 in the

FW	310	Z	330	Y (vertical) and Z (east-west) directions
FW	435	X	216	at support 3-18GF-H372, in the X (north-south) and Z (east-west) directions
FW	435	Z	286	

3.2.2 CD, ES and HD Piping (Turbine Building)

Significant flow increases occur in portions of the condensate, extraction steam and heater drain systems as a result of EPU. Monitoring locations were selected on the basis of percent flow increase due to EPU, projected EPU flow rates, piping configuration and similarity between trains and units.

Condensate:

The condensate system experiences flow increases similar to FW as a result of EPU. Two locations between the 5th stage feedwater heaters and the reactor feedwater pumps were selected for EPU vibration monitoring in each Unit. Those locations experience the highest percent increase (14%) in flow rates under EPU conditions.

Extraction Steam:

The extraction steam system will experience the most significant flow increases in the piping from the high pressure (HP) turbine to the 5th stage feedwater heaters and the piping from the low pressure (LP) turbine to the 3rd stage feedwater heaters. The flow velocity increases to the 5th and 3rd stage heaters are 21% and 33 %, respectively.

The extraction steam lines from the HP turbine to the 5th stage feedwater heaters will be instrumented with accelerometers at two locations in each Unit. The extraction steam lines from the LP turbine to the 3rd stage feedwater heaters will be instrumented with accelerometers at three locations in each Unit.

Heater Drain:

The heater drain system will experience the most significant flow increases (35%) in the normal drain piping between the 4th and 5th stage feedwater heaters. Because, the piping configurations of the three trains are similar, only the drain piping between the 'A' 4th and 5th stage feedwater heaters is selected for monitoring. This piping will be instrumented with accelerometers at three locations in each Unit.

Allowable displacement limits at the selected measurement locations were calculated using the acceptance criteria delineated in ASME O&M-S/G Part 3 [3].

The EPU vibration monitoring locations determined for the condensate, extraction steam and heater drain piping for Units 2 and 3 are summarized in Tables 3-5 and 3-6 respectively.

Table 3-5
Turbine Building EPU Monitoring Locations for CD, ES and HD, PBAPS Unit 2

System	Location[1]	Direction	Allowable Peak-to-Peak Displacement, mils	Description
CD	95	X	500	At support 2-18GFH-238
CD	95	Y	337	
CD	95	Z	500	
CD	930	X	417	At support 2-18GFH-261
CD	930	Y	172	
CD	930	Z	389	
ES	42	X	500	At support 2-16GA-H60
ES	42	Y	500	
ES	42	Z	500	
ES	23D	X	397	At support 2-16GA-H51
ES	23D	Y	329	
ES	23D	Z	500	
ES	174	X	209	At Support 2-16HA-H33
ES	174	Y	305	
ES	74	Y	231	At Support 2-16HA-H27
ES	74	Z	144	
ES	117	X	500	At Support 2-16HA-H37
ES	117	Z	311	
HD	110	X	500	At support 2-17GE-H9
HD	110	Z	500	

HD	140	Y	70	At support 2-17GE-394
HD	40	X	332	At support 2-17GE-H2
HD	40	Z	64	

Note 1: Since the Unit 2 and 3 piping geometries are similar, only the Unit 3 piping was modeled. Therefore, the Unit 2 locations are identified with the corresponding Unit 3 node numbers.

Table 3-6

Turbine Building EPU Monitoring Locations for CD, ES and HD, PBAPS Unit 3

System	Location	Direction	Allowable Peak-to-Peak Displacement, mils	Description
CD	95	X	500	At support 3-18GFH-238
CD	95	Y	337	
CD	95	Z	500	
CD	930	X	413	At support 3-18GFH-260-A
CD	930	Y	252	
CD	930	Z	473	
ES	42	X	500	At support 3-16GA-H60
ES	42	Y	500	
ES	42	Z	500	
ES	23D	X	397	At support 3-16GA-H51
ES	23D	Y	329	
ES	23D	Z	500	
ES	174	X	209	At support 3-16HA-H33
ES	174	Z	305	
ES	74	Y	231	At support 3-16HA-H27
ES	74	Z	144	
ES	117	X	500	At support 3-16HA-H33
ES	117	Z	311	
HD	110	X	500	At support 3-17GF-H9
HD	110	Z	500	

HD	140	Y	70	At support 3-17GF-H394
HD	40	X	332	At support 3-17GF-H2
HD	40	Z	64	

3.2.3 MS Components (Drywell and Steam Tunnel)

PBAPS operating history indicates that excessive component vibrations are not expected at EPU conditions. In order to provide confirmation that component vibrations will be within acceptable limits at EPU conditions, selected components will be instrumented with accelerometers. The selected components include two safety-relief valves (SRV), one spring safety valve (SSV), two main steam isolation valves (MSIV) and one motor operated valve (MOV) each for the reactor core isolation cooling (RCIC) turbine steam supply line and the high pressure coolant injection (HPCI) turbine steam supply line. Both the RCIC and HPCI lines are attached to the MS piping. The EPU component vibration monitoring locations are summarized in Table 3-7. Vibration acceptance criteria for the selected locations will be based on valve seismic qualification reports as well as on the past experience and test data.

Table 3-7
EPU Component Monitoring Locations, PBAPS Units 2 and 3

System	Valve ID	Direction	Allowable RMS Acceleration, g's	Description
MS	RV-70A	X	0.30	Dresser SSV on MSL A
MS		Y	0.40	
MS		Z	0.30	
MS	RV-71E	X	0.15	Target Rock SRV on MSL B
MS		Y	0.35	
MS		Z	0.15	
MS	RV-71K	X	0.15	Target Rock SRV on MSL D
MS		Y	0.35	
MS		Z	0.15	
MS	AO-80D	X	0.15	Inboard MSIV on MSL D
MS		Y	0.10	
MS		Z	0.15	
MS	AO-86D	X	0.15	Outboard MSIV on MSL D
MS		Y	0.10	
MS		Z	0.15	
HPCI	MO-15	X	0.40	Inboard HPCI MOV
HPCI		Y	0.60	
HPCI		Z	0.40	
RCIC	MO-	X	0.40	Inboard RCIC

RCIC	15	Y	0.60	MOV
RCIC		Z	0.40	

3.3 Data Acquisition and Reduction Methodology

The accelerometer data will be collected during EPU power ascension at pre-determined power levels using two PC-based digital data acquisition systems (DAS's). One DAS will be located in the reactor building and another DAS will be located in the turbine building. Each data set will be recorded using a minimum sample rate of 2000 samples per second per channel for a minimum duration of two minutes.

The raw time history data for each power level will be processed for comparison to applicable acceptance criteria. The data processing will include integration, determination of peak, peak-to-peak and root mean square (rms) values, and high and low pass filtering, as applicable for specific monitoring locations and acceptance criteria bases. Additional data processing, such as frequency analysis, will be performed to aid data analysis, as required.

4.0 SUMMARY

A confirmatory test program will be implemented to perform vibration monitoring during power ascension to EPU conditions. Piping and attached components on systems experiencing significant flow increases as a result of EPU will be included in the monitoring program. Piping vibration acceptance criteria will be based on ASME OM-S/G Part 3. Component vibration acceptance criteria will be based on component-specific dynamic characteristics and industry experience. Small bore piping assessments will be performed to identify potentially susceptible configurations, and any modifications required to reduce vibration susceptibility will be made prior to EPU power ascension.

Monitoring of inaccessible piping and components will be accomplished using accelerometers wired to data acquisition systems located in the reactor and turbine buildings. Accessible piping included in the monitoring program will be monitored either remotely or by performing visual observations and by taking vibration measurements using hand-held vibration instruments during power ascension to EPU conditions.

5.0 REFERENCES

1. BWR Owners' Group EPU Committee, “Extended Power Uprate (EPU) Lessons Learned and Recommendations”, NEDO-33159 Revision 2, December 2008, BWR Owners' Group EPU Committee.
2. GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, Class I (Nonproprietary), July 2003.
3. ASME OM-S/G, Standards and Guides for Operation and Maintenance of Nuclear Power Plants, Part 3, 2009 Edition, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems."

Attachment 14

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

EPU Related Changes to TSTF-493 Instrument Setpoints

Purpose

The purpose of this Attachment is to (1) describe adherence to Technical Specification Task Force (TSTF) – 493, Revision 4 for LSSS setpoints changed by the EPU; (2) provide an overview of the safety system setpoint control program pertaining to EPU; and (3) provide the NRC staff with the calculation for a setpoint impacted by EPU and annotated in accordance with the TSTF – 493, Revision 4 in fulfillment of an NRC request for a sample calculation. This request came in a public meeting on December 7, 2011 (see Summary of December 7, 2011 Meeting With Exelon Re: Proposed Amendment Request to Implement Extended Power Uprate (ML120270288))

TSTF – 493 Revision 4 Adherence for EPU-Changed Setpoints

The requirements of 10 CFR 50.36 for safety-related LSSS functions are provided in RIS 2006-17 and further clarified by TSTF – 493, Revision 4. Attachment A to TSTF – 493, Revision 4 identifies the setpoint functions under Option A that are to be annotated with the TSTF – 493 notes to the Technical Specifications or Bases unless an exception is taken in accordance with TSTF – 493, Section 4.0, Technical Analysis. For the EPU License Amendment request, Exelon applies the TSTF – 493 notes and completes the supporting setpoint and uncertainty calculations for setpoints that meet the following criteria:

- They are changed by the EPU;
- They are identified in Attachment A to TSTF – 493,
- They do not fall under one of the exclusions described in TSTF – 493, Revision 4, Section 4.0 (Technical Analysis)

There are three Peach Bottom setpoints that meet these criteria and are annotated in accordance with TSTF-493, Revision 4. They are:

- Average Power Range Monitor Simulated Thermal Power – High
- Main Steam Line Flow – High.
- Main Steam Line Pressure - Low

The Torus High Level Swapover Setpoint is also being changed as part of the EPU LAR (see Enclosure 9e of Attachment 9). However, this setpoint is based on an instrument that derives its input from a float switch with no associated sensor or adjustable device and is therefore excluded from application of the notes.

The TSTF-493 notes use site-specific terminology. The following is the relationship of the site terminology to that of TSTF-493:

Nominal Trip Setpoint (NTSP) is the site term for the TSTF – 493 Limiting Trip Setpoint (LTSP). It is a predetermined limiting value for the trip setpoint so that the trip or actuation will occur before the Analytical Limit is reached, regardless of the process or environmental conditions affecting the instrumentation. This value is determined by calculation. The actual calibrated setpoint may be more conservative than the calculated NTSP obtained from the setpoint calculation.

Actual Trip Setpoint (ATSP) is the site term for the TSTF-493 Nominal Trip Setpoint (NTSP). The ATSP is a predetermined value that is established by providing additional margin above and beyond any considerations accounted for in determining the Peach Bottom Nominal Trip Setpoint (NTSP). When no additional margin is desired, the ATSP is equivalent to the site NTSP. The ATSP is the value to which the field bistable devices are calibrated.

Leave Alone Zone (LAZ) is the site term equivalent to the TSTF-493 As-Left Tolerance (ALT). The LAZ is a range of acceptable values around a nominal value established by adding or subtracting the required accuracy from the nominal value. When an instrument reading (cardinal point of calibration or trip setpoint) is found within this band during Surveillance Testing or calibration checks, no calibration adjustment is required. In special cases, the LAZ can be established as a non-uniform band around the nominal value. In surveillance procedures this can be called the Acceptable Limit.

Average Power Range Monitor Simulated Thermal Power – High
The following notes are added to the channel calibration surveillance in the Surveillance Requirements column of the Average Power Range Monitor Simulated Thermal Power – High setpoint (Technical Specification Table 3.3.1.1-1 Item 2.b)

1. If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

2. The instrument channel setpoint shall be reset to a value that is within the Leave Alone Zone (LAZ) around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found tolerance and LAZ apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP methodologies used to determine the as-found tolerance and the LAZ are specified in the Bases associated with the specified function.

Main Steam Line Flow – High and Main Steam Line Pressure - Low
The Surveillance Requirements Technical Specification Bases for Main Steam Line Flow – High (Technical Specification Table 3.3.6.1-1 Item 1.c) and Main Steam Line Pressure - Low (Technical Specification Table 3.3.6.1-1 Item 1.b) shall be modified with the addition of the following statement to surveillances that include verification of the setpoint:

"There is a plant specific program which verifies that this instrument channel functions as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology."

The markups to the Technical Specifications and Technical Specification Bases related to these setpoints can be found in Attachments 2 and 3 to this EPU LAR, respectively.

Overview

The instrument setpoint methodology currently implemented at PBAPS is based on the GEH Instrument Setpoint Methodology specified in NEDC-31336P-A, General Electric Instrument Setpoint Methodology (Proprietary). This methodology is procedurally-controlled and performed only by qualified personnel.

Setpoint calculations begin by identifying the applicable Safety or Design Limit. The effects of transient overshoot, response times, and any modeling uncertainties are taken into account to obtain the Analytical Limit. Exelon calculates setpoints from the Analytical Limit, establishing margins between the Analytical Limit and the Allowable Value based on performance specifications for instruments being used. Independent instrument uncertainties are quantified, and then combined using the square-root-of-the-sum-of-the-squares method. Other non-device uncertainties are added algebraically.

There is additional margin based on loop drift that is applied between the Allowable Value and the Nominal Trip Setpoint. Additional margin may be assigned between the Nominal Trip Setpoint and the Actual Trip Setpoint that takes into account the instrument As-Found Tolerance (AFT) and Leave Alone Zone, and any unique requirements for that device. If no additional margin is required, then the Actual Trip Setpoint is equal to the Nominal Trip Setpoint. The LAZ or ALT and the AFT are always around the ATSP.

At the start of each calibration, instruments controlled by Technical Specifications are declared inoperable and removed from service. Upon completion, the Operations Shift Supervisor or Manager reviews the results of the surveillance and determines whether the results are acceptable based on Technical Specification operability requirements prior to returning the instrument to service.

During calibration checks, if the as-found setpoint is outside the Leave Alone Zone, the condition is documented for trending purposes and appropriate corrective actions are taken before the instrument is returned to service. Once actions have been taken to correct the condition, the instrument setpoint is reset to as close to the Actual Trip Setpoint value as practicable (i.e. within the Leave Alone Zone) and the instrument is returned to service. For cases in which the as-found setpoint value is within its Leave Alone Zone, the instrument is adjusted if desirable to as close to the Actual Trip Setpoint value as practicable.

At PBAPS, trip setpoints are typically verified via channel calibration procedures.

Calculations

The following calculation, provided as Enclosure 14a to this Attachment, is a sample of the setpoint calculations associated with EPU related Technical Specification setpoint changes.

PE-0251 (Enclosure 14a) – includes the calculations for Allowable Value, Nominal Trip Setpoint, As-Left Tolerance and As-Found Tolerance for the Average Power Range Monitor Simulated Thermal Power – High as a result of EPU related changes.