Attachment 44

Probabilistic Risk Assessment

Executive Summary

The Browns Ferry Nuclear Plant (BFN) is currently pursuing an increase in reactor power from the Current Licensed Thermal Power (CLTP) of 3458 Megawatts Thermal (MWt) to 3952 MWt, an Extended Power Uprate (EPU) of 120% of the original licensed thermal power (OLTP).

The enclosed assessment of the power uprate impacts on risk as characterized as a change in the core damage frequency (CDF) and large early release frequency (LERF), has been performed. The guidelines from the Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.174, are followed to assess these changes and to determine which risk acceptance region the change to EPU is in.

The methodology consists of an examination of the elements of the BFN Probabilistic Risk Assessment (PRA) to assess the impact of the following EPU changes on the PRA elements:

- Hardware changes
- Procedural changes
- Set point changes
- Power level change

These changes are interpreted in terms of their PRA model effects, which can then be used to assess whether there are any resulting risk profile changes. The scope of this report includes the complete risk contribution associated with the EPU at BFN. Risk impacts due to internal events, internal flooding, and internal fires are assessed using the BFN PRA Revision 6 model of record. External events are evaluated using the analyses of the BFN Individual Plant Examination of External Events (IPEEE) Submittal. The impacts on shutdown risk contributions are evaluated on a qualitative basis. All commitments resulting from the BFN IPE and IPEEE programs have been resolved. The results of the PRA evaluation are the following:

- Detailed thermal hydraulic analyses of the plant response using the EPU configuration indicate manageable reductions in the time allowable to perform some operator actions.
- The reduced operator action "allowable" times resulted in minor increases in the assessed Human Error Probabilities (HEPs) in the PRA model.
- Only very small risk increases were identified for the changes associated with the EPU.
- The risk impact due to implementation of the EPU is low and acceptable. The risk impact is in the "small" category (i.e., Region II of Regulatory Guide 1.174 Guidelines) for ΔCDF and for ΔCDF.

The EPU is estimated to increase the Total CDF from 5.91E-5/yr to 6.08E-05/yr for Unit 1 (ΔCDF=1.69E-06), from 5.96E-05 to 6.14E-05 for Unit 2 (ΔCDF=1.74E-06), and from 6.47E-05/yr to 6.64E-05/yr for Unit 3 (ΔCDF=1.67E-06). The EPU is estimated to increase the Total LERF from 7.96E-06/yr to 8.73E-06/yr for Unit 1 (ΔLERF=7.74E-07), from 7.99E-06/yr to 8.65E-06/yr for Unit 2 (ΔLERF=6.63E-07), and from 7.18E-06/yr to 7.72E-06/yr for Unit 3 (ΔLERF=5.45E-07).

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1.0 <u>Introduction</u>

BFN is currently pursuing an increase in reactor power from the current licensed thermal power of 3458 MWt to 3952 MWt, an EPU of 120% of the OLTP. The EPU is a constant pressure power uprate (CPPU) and retains the Maximum Extended Load Limit Line Analyses (MELLLA) power/flow map.

The purpose of this report is to:

1. Identify any significant change in risk associated with EPU as measured by the BFN PRA models.
2. Provide the basis for the model changes associated with EPU.
3. Demonstrate that the risks associated with the proposed EPU are acceptable per the risk acceptability guidelines in RG 1.174 (Reference 1).

1.1 Background

"The method for achieving higher power at GE BWRs is to retain the MELLLA or Maximum Extended Operating Domain (MEOD) power/flow map and to increase core flow (and power) along the existing flow control rod line. The proposed CPPU will not increase reactor operating pressure or the current licensed maximum core flow. CPPU operation will not require an increase in reactor vessel dome pressure because the plant will make modifications to the power generation equipment pressure controls and turbine flow capabilities to control the pressure at the turbine inlet" (Reference 3).

The BFN Model of Record (MOR) Rev. 6 (Reference 4) is the most recent evaluation of the risk profile for internal event challenges, and it is used as the bases for the EPU analyses. The BFN 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 BFN PRA is based on the event tree / fault tree linking methodology, which is a well-known methodology in the industry. It should be noted that TVA has chosen to use the BFN Fire Probabilistic Risk Assessment (FPRA) model (Reference 5) in this Attachment to quantify and assess the impact of EPU on severe fire risk.

The Tennessee Valley Authority (TVA) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating TVA nuclear generation sites. This approach includes both a procedurally controlled 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 BFN PRA.

1.2 PRA Quality

1.2.1 <u>PRA Maintenance and Update</u>

The TVA process for controlling updates to the PRA is documented in TVA procedure NPG-SPP-09.11, "The Probabilistic Risk Assessment Program" (Reference 6) and

NEDP-26, “Probabilistic Risk Assessment” (Reference 7). NPG-SPP-09.11 covers the management of PRA applications, periodic updates and interdepartmental PRA documentation. This procedure provides definitions for PRA model update, PRA model application, and PRA evaluation. This procedure also defines responsibilities of other departments such as operations and system engineering for review of the PRA.

NEDP-26 describes the process used by the PRA staff to perform applications, model PRA model updates and review. An update of the PRA model can either be an upgrade or normal model maintenance. The terms PRA upgrade and maintenance are defined in the procedures using the definitions provided in the ASME/ANS PRA Standard (Reference 8). The procedure requires that updates should be completed at least once every other fuel cycle (for the lead unit at multi-unit sites) or sooner if estimated cumulative impact of plant configuration changes exceeds +/-10% of CDF or LERF. Changes in PRA inputs or discovery of new information is required to be evaluated to determine whether such information warrants a PRA update. In accordance with NEDP-26, items exceeding the above threshold shall be tracked in the Corrective Action Program (CAP). Potential and/or implemented plant configuration changes that do not meet the threshold for an immediate update are required to be tracked.

PRA updates are required to follow the guidelines established by the ASME/ANS PRA Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications for a minimum of a Category II assessment. NRC RG 1.200 Revision 2 (Reference 9) endorses with clarifications and qualifications the ASME/ANS RA-Sa-2009 (Reference 8) standard and is the governing regulatory document for PRA Maintenance and Upgrade. RG 1.200 and the associated ASME/ANS PRA standard define the technical elements required for an acceptable internal events at-power PRA.

NEDP-26 also defines the requirements for PRA documentation supporting the model of record and PRA applications. The model of record is composed of 1) A PRA computer model and supporting documentation, 2) A Modular Accident Analysis Program (MAAP) model and supporting documentation, and 3) other supporting computer evaluation tools. The purpose of the PRA MOR is to provide a prescriptive method for quality, configuration, and documentation control. PRA applications and evaluations are referenced to a MOR and therefore, the pedigree of PRA applications and evaluations is traceable and verifiable.

All current PRA notebook revisions are documented in accordance with the TVA design engineering calculation procedure (NEDP-2). The NEDP-2 (Reference 10) calculation process requires calculations to be prepared, independently checked, and approved.

NEDP-26 also specifies the periodic review of the model. To ensure that the current PRA models remain an accurate reflection of the as-built, as-operated plants, NEDP-26 requires that the following activities are routinely performed:

- Plant-specific design, procedure, and operational changes are required to be reviewed for risk impact. Additional reviews to identify information which could impact the PRA models should be completed, including comparison of the PRA model with the knowledge of industry and plant experiences, information, and data with the purpose of identifying inputs pertinent to the PRA. This PRA information may also include modeling errors discovered during routine use of the PRA or new information that could impact PRA modeling assumptions.

- Various information sources shall be monitored by the Corporate/Site PRA Specialist on an ongoing basis to determine changes or new information that could potentially affect the model, model assumptions, or quantification. Information sources include Operating Experience (OE), Technical Specification (TS) changes, plant modifications, Maintenance Rule changes, engineering calculation revisions, procedure changes, industry studies, NRC information and Problem Evaluation Reports (PERs). NPG-SPP-09.11 “Probabilistic Risk Assessment (PRA) Program” provides the requirements for interdepartmental data collection.

The FPRA model represents the as-built, as-operated and maintained plant as it will be configured at the completion of the transition to EPU. The FPRA model includes credit for the planned implementation of plant modifications. Following installation of modifications and the as-built installation details, additional refinements surrounding the modifications may need to be incorporated into the FPRA model (the FPRA will verify the validity of the reported change-in-risk on as-built conditions after the modification is completed). However, these changes are not expected to be significant. No other significant plant changes are outstanding with respect to their inclusion.

The BFN 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 BFN. This analysis uses the PRA to provide insights about how plant risk from postulated accidents, including severe accidents, is impacted by EPU implementation.

1.2.2 Internal Events PRA

The BFN Units 1, 2 and 3 Internal Events PRA Peer Review was performed in May 2009 (Reference 12) at the TVA offices in Chattanooga, TN, using the process described in NEI 05-04 (Reference 13), the ASME/ANS (American Society of Mechanical Engineers/American Nuclear Society) PRA Standard (ASME/ANS RA-Sa-2009) (Reference 8), and Regulatory Guide 1.200, Revision 2. A separate review was performed for the Internal Flooding portion of the BFN PRA in October 2009 (Reference 14). The Internal Flooding Peer Review also used the NEI 05-04 process, the ASME/ANS PRA Standard, and Regulatory Guide 1.200, Revision 2. A team of independent PRA experts from nuclear utility groups and PRA consulting organizations carried out these Peer Review Certifications. The PRA was not reviewed for Fires, External Flooding, Seismic, High Winds, or other external events in 2009.

The purpose of these reviews was to provide a method for establishing the technical adequacy of the BFN PRA for the spectrum of potential risk-informed plant licensing applications for which the BFN PRA may be used. The 2009 BFN PRA Peer Reviews provided a full-scope review of the Technical Elements of the internal events and internal flooding, at-power PRA.

These intensive peer reviews involved over two person-months of engineering effort by the review team and provided a comprehensive assessment of the strengths and limitations of each element of the PRA model. These Peer Review Certifications of the BFN PRA models performed by the Boiling Water Reactor Owner's Group (BWROG) resulted in a total 125 findings for the three unit model for internal events and internal flooding. All findings from these assessments have been dispositioned. This resulted in a number of enhancements to the BFN PRA model prior to its use to support PRA applications. The certification team determined that with these proposed changes incorporated, the quality of all elements of the BFN PRA model is sufficient to support "risk significant evaluations with deterministic input." As a result of the effort to incorporate the latest industry insights into the BFN PRA model upgrades and certification peer reviews, TVA has concluded that the results of the risk evaluation are technically sound and consistent with the expectations for PRA quality set forth in Regulatory Guide 1.174.

The PRA Update process defined in NEDP-26 ensures that the BFN PRA model adequately reflects the as-built and as-operated plant configurations. The PRA Update process addresses those activities associated with maintaining and upgrading the PRA models and documentation. PRA Updates include a general review of the entire BFN PRA model, incorporation of recent plant data and physical plant changes, conversion to new software versions, implementation of new modeling techniques, and a comprehensive documentation effort. The PRA Update process is applied to the Level 1/2, full power, internal events PRA and FPRA models. However, the process may be applied to other risk related applications. The BFN PRA model updates are scheduled for 48-month intervals; however, additional revisions have been made due to the discovery of new information and plant/procedure changes.

Appendix A of this Attachment presents the resolution of the F&Os from the peer reviews and their impact on EPU application. Some of responses to the F&Os were modified to reflect the NFPA 805 RAI response submittal. No changes have been made to the Internal Events PRA model since the 2009 peer review that constitute an upgrade as defined by the ASME/ANS PRA Standard. (Reference 8).

1.2.3 FPRA Quality

The development of the FPRA is consistent with and satisfies the requirements of ASME/ANS RA-Sa-2009 (Reference 8). In addition, the requirements of RG 1.200 includes Peer Review of the FPRA. In support of this Peer Review, a self-assessment was performed as described in NEI 07-12 (Reference 15). As described in detail in Appendix H, a full peer review of all elements of the FPRA standard and a focused follow-on review were performed (References 16 and 17). The purpose of these reviews was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The final conclusion of the peer reviews was that the BFN FPRA meets Capability Category II following final resolution and closure of all of the Facts and Observations (F&Os). Most of the F&Os from the full-scope peer review were resolved in the follow-on peer review. In addition, a subsequent focused scope peer review was conducted in May 2015. The focused peer review's primary purpose was to formally close all outstanding F&O's with the exception of those F&O's pertaining to future procedure or plant modification development and implementation. The F&Os from the focused scope peer review, some of which remain unresolved, are listed and discussed with respect to the proposed EPU application, in Appendix A, .

The TVA process for controlling updates to the FPRA is documented in TVA procedure NPG-SPP-09.11, “The Probabilistic Risk Assessment Program” and NEDP-26, “Probabilistic Risk Assessment”. NPG-SPP-09.11 covers the management of PRA applications, periodic updates and interdepartmental PRA documentation. A detailed discussion of FPRA quality is presented in Appendix A.

1.2.4 Level of Detail

The BFN PRA MOR Rev. 6 is of sufficient quality and scope to measure the potential changes in plant risk related to EPU implementation. The PRA modeling is highly detailed, including a wide variety of initiating events (e.g., transients, IORV, internal floods, LOCAs inside and outside containment, interfacing system LOCA (ISLOCA), support system failure initiators), modeled systems, operator actions, and common cause events.

External hazards were evaluated in the BFN Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) (Reference 18). 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 IPEEE analysis for the three BFN operating units was done in phases. References 19, 20, 21, and 22 document the IPEEE submittals made to the NRC. Seismic evaluations for the BFN Units were also performed in accordance with the Electric Power Research Institute (EPRI) Seismic Margins Analysis (SMA) methodology (Reference 24). Since the performance of the IPEEE, a FPRA was performed for all three units (see Section 4.4, 5.2 and Appendix D of Reference 26). A seismic PRA has not been completed for BFN. As such, there are no comprehensive CDF and LERF values available from the IPEEE to support the EPU risk assessment.

In addition to internal fires and seismic events, the BFN IPEEE analysis of high winds, floods, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. A screening approach as described in Generic Letter (GL) 88-20 supplement 4 was used for evaluation of high winds, external floods and nearby facility/transportation events. No other external events (volcanic activity, etc.) are applicable to BFN. The screening approach used in analysis of external floods, and nearby facilities/transportation accidents demonstrates that they meet NRC Standard Review Plan (SRP) 1975 (Reference 25) criteria and have adequate defense against these threats. Since Browns Ferry does not meet the SRP 1975 criteria for high winds, a bounding analysis was performed. This analysis showed the contribution to core damage frequency due to high winds to be less than the IPEEE screening criteria of 1E-6 (Reference 19).

BFN 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. 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.

It is important to recognize that during the process of developing the FPRA, the existing internal events PRA model was used as a starting point to construct an application-specific model. The treatment of fire has been integrated into the internal events model and the integrated model is used for the EPU calculation. Detailed information on the construction of the FPRA model can be found in TVA FPRA - Task 7.5 Fire-Induced Risk Model (Reference 28).

1.2.5 Summary

In summary, the BFN integrated Level 1 and Level 2 Internal Events PRA model and FPRA model provide 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 analyses 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 Definitions and Acronyms

1.3.1 Definitions (All taken from Reference 27)

Core Damage

For the BFN Level 1 PRA, a definition of core damage is selected to distinguish between short–term losses of adequate core cooling and core damage as defined in the ASME standard. The objective criterion selected for the BFN PRA is the hottest core node temperature, as reported by parameter TCRHOT in MAAP calculations, exceeding 2500°F for any amount of time.

Core Damage Frequency

Expected number of core damage events per unit of time.

Containment Integrity

Maintaining containment integrity is defined as follows

- Containment is isolated
- The primary containment pressure capacity is temperature dependent. The drywell knuckle 95% confidence failure pressures are used to model the temperature dependent pressure capacity and vary from 144 psig at low drywell temperatures to 125 psig at high drywell temperatures.
- Containment integrity under ATWS conditions is assumed to be limited by the suppression pool temperature and condensation capability, i.e., 260°F bulk fluid temperature.

Large Early Release

This is a radioactive release from the containment which is both large and early. Large is defined as involving the rapid, unscrubbed release of airborne aerosol fission products to the environment. Early is defined as occurring before the effective implementation of the off-site emergency response and protective actions.

An early release is defined as any offsite release occurring before the effective implementation of off-site emergency response and protective actions. This can be generically defined by considering the time required to evacuate the population within a 10 mile Emergency Planning Zone (EPZ) using assumptions described in NUREG-1150 (Reference 26, USNRC 1990, pp. 11-2, 11-5).

Large atmospheric releases occurring within four hours of offsite notification can be considered early for purposes of the LERF evaluation (Reference 46).

Large Early Release Frequency

Expected number of large early releases per unit of time.

Initiating Event

Events that challenge normal plant operation and that require successful mitigation to prevent core damage. Initiating events are grouped according to the mitigation requirements to facilitate the efficient modeling of plant response.

Internal Events

A hazard group that encompasses events that result from or involve mechanical, electrical, structural, or human failures from causes originating within a nuclear power plant that directly or indirectly cause an initiating event and may cause safety system failures or operator errors that may lead to core damage and possibly large early release. By historical convention, Loss of Offsite Power (LOOP), which may result from causes within or outside the plant, is considered an internal event (except when the loss is caused by another evaluated hazard group, e.g., a tornado). Also by historical convention, internal flood and internal fire are separate hazard groups and thus not considered internal events.

External Events

An event originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or impact operator actions that may lead to core damage and potentially large early release. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from outside the plant are considered external events. (See also internal event). Fires from sources inside the plant are considered external events in the ASME/ANS Standard (Reference 8), but are evaluated separately from other external events. By historical convention, LOOP not caused by another external event is considered to be an internal event.

Human Error Probability

A measure of the likelihood that plant personnel will fail to initiate the correct, required, or specified action or response in a given situation, or by commission performs the wrong action. The HEP is the probability of the human failure event.

Human Reliability Analysis (HRA)

A structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment.

Modular Accident Analysis Program

MAAP is a computer program that has been widely used in the nuclear industry for the thermal hydraulic (T/H) analysis to simulate the plant response to various accidents and transients. Therefore calculating timing requirements for success criteria.

MAAP4 includes models for the important accident phenomena that might occur within the primary system, in the containment, and/or in the auxiliary/reactor building. For a specified reactor and containment system, MAAP4 calculates the progression of the postulated accident sequence, including the disposition of the fission products, from a set of initiating events to either a safe, stable state or to an impaired containment condition (by over-pressure or over-temperature) and the possible release of fission products to the environment.

Level 1 PRA

The Level 1 accident sequence analysis models (to the extent practical), the different possible progressions of events (i.e., accident sequences) that can occur from the start of the initiating event to either successful mitigation or core damage. The accident sequences account for the systems that are used (and available) and operator actions performed to mitigate the initiator based on the defined success criteria and plant operating procedures (e.g., plant emergency and abnormal operating procedures) and training. The availability of a system includes consideration of the functional, phenomenological, and operational dependencies and interfaces between the various systems and operator actions during the course of the accident progression.

Level 2 PRA

The Level 2 analysis interfaces with the Level 1 (CDF) accident sequence analysis through the appropriate definition of a set of core damage functional classes. These states are the endpoints of the sequences in the Level 1 portion of the event trees and the initiating events for the CETs. For changes to the licensing basis the figure of merit is LERF which is a subset of the Level 2 analysis. The end products of the LERF analysis include a set of release categories, which define the radionuclide releases into the environment, and a quantification of the frequency of each release category.

Risk Significance

The importance measure used in this analysis is Fussell-Vesely (FV), which is defined as follows:

> Fussell-Vesely - This importance measure gives the risk associated with a given component. That is how much the component is contributing to system failure. The Fussell-Vesely importance is expressed in relative terms and is defined as:

$$FV = (\text{Reference } P(top) - P(top/A=0)) / P(top)$$

Risk Thresholds

The primary quantitative figures of merit are CDF and LERF. There are, however, risk impacts that are not reflected (or are inadequately reflected) by changes to CDF and LERF. Therefore, the impacts of the proposed change on aspects of risk not captured (or inadequately captured) by these metrics should be addressed. For example, changes affecting long-term containment performance would impact radionuclide releases from containment occurring after evacuation and could result in substantial changes to offsite consequences such as latent cancer fatalities. Recognizing that the containment function is an important factor in maintaining the defense-in-depth philosophy, the impact of the proposed change on those aspects of containment function not addressed in the evaluation of LERF should be addressed in the licensee submittal documentation.

The risk-acceptance guidelines are based on the principles and expectations for risk-informed decisions on plant-specific changes to the licensing basis (RG 1.174, Reference 1) are structured as follows. Regions are established in the two planes generated by a measure of the baseline risk metric (CDF or LERF) along the x-axis, and the change in those metrics (ΔCDF or ΔLERF) along the y-axis (Figure 5 of Reference 1). Acceptance guidelines are established for each region as discussed below. These guidelines are intended for comparison with a full-scope (including internal and external hazards, at power, low power, and shutdown) assessment of the change in risk metric and, when necessary, as discussed below, the baseline value of the figure of merit (CDF or LERF). However, it is recognized that many PRAs are not full scope and PRA information of less than full scope may be acceptable.

1.3.2 Acronyms

The following acronyms are used in this Attachment. Also included is a listing of the designators for some of the more complex nodes that represent several systems, portions of systems, or operator actions.

ACU	–	Air Conditioning Unit
ADS	–	Automatic Depressurization System

ANS	–	American Nuclear Society
AOI	–	Abnormal Operating Instruction
AOP	–	Abnormal Operating Procedure
APRM	–	Average Power Rate Meter
ARP	–	Alarm Response Procedure
AS	–	Accident Sequences
ASDC	–	Alternate Shutdown Cooling
ASME	–	American Society of Mechanical Engineers
ATWS	–	Anticipated Transient Without Scram
AVI	–	Alternate Vessel Injection - RHRSW, RHR Unit 2 X-TIE
BFN	–	Browns Ferry Nuclear Plant
BOC	–	Break Outside Containment
BOP	-	Balance of Plant
BWROG	-	Boiling Water Reactor Owner's Group
CAP	-	Corrective Action Program
CCDP	–	Conditional Core Damage Probability
CDF	–	Core Damage Frequency
CLERP	–	Conditional Large Early Release Probability
CLTP	–	Current Licensed Thermal Power
CLTR	–	Constant Pressure Power Uprate LTR
CPPU	–	Constant Pressure Power Uprate
CR	–	Control Room
CRD	–	Control Rod Drive
CS	–	Core Spray
CsI	–	Cesium Iodide
CST	–	Condensate Storage Tank
DCN	–	Design Change Notice
DHR	–	Decay Heat Removal
DWS	–	Drywell Spray
EAL	-	Emergency Action Level
ECCS	–	Emergency Core Cooling System
ED	–	Emergency RPV Depressurization

EHC	-	Electro-Hydraulic Control
EHPM	–	Emergency High Pressure Make-Up
EOI	–	Emergency Operating Instruction
EOOS	–	Equipment Out Of Service
EPRI	-	Electric Power Research Institute
EPU	–	Extended Power Uprate
EPZ	–	Emergency Planning Zone
F&Os	–	Facts and Observations
FIVE	–	Fire Induced Vulnerability Evaluation
FAC	-	Flow Accelerated Corrosion
FPRA	–	Fire Probabilistic Risk Assessment
FV	–	Fussell-Vesely
FW	–	Feedwater
FWH	–	Feedwater Heater
GL	–	Generic Letter
HCTL	–	Heat Capacity Temperature Limit
HEP	–	Human Error Probability
HFE	–	Human Failure Events
HLR	–	High Level Requirements
HMI	-	Human Machine Interface
HPCI	–	High Pressure Coolant Injection
HRA	–	Human Reliability Analysis
HWWV	–	Hardened Wetwell Vent
HVAC	–	Heating Ventilation and Air Conditioning
IOOV	–	Inadvertent Opening of One SRV
IPE	–	Individual Plant Examination
IPEEE	–	Individual Plant Examination for External Events
ISLOCA	–	Interfacing Systems LOCA
JHEP	–	Joint Human Error Probability
LAR	–	License Amendment Request
LERF	–	Large Early Release Frequency
LOCA	–	Loss of Coolant Accident

LOOP	–	Loss of Offsite Power
LPCI	–	Low Pressure Coolant Injection (RHR only)
MAAP	–	Modular Accident Analysis Program
MELLLA	–	Maximum Extended Load Limit Line Analyses
MEOD	–	Maximum Extended Operating Domain
MOR	–	Model of Record
MSIV	–	Main Steam Isolation Valve
MWt	–	Megawatts Thermal
NEDP	-	Nuclear Engineering Department Procedure
NPSH	–	Net Positive Suction Head
NRC	–	Nuclear Regulatory Commission
NSSS	–	Nuclear Steam Supply System
OE	-	Operating Experience
OI	–	Operating Instruction
OLTP	–	Original Licensed Thermal Power
PCS	–	Power Conversion System
PDS	–	Plant Damage State
PER	-	Problem Evaluation Report
PPL	–	Preferred Pump Logic
PRA	–	Probabilistic Risk Assessment
PSP	–	Pressure Suppression Pressure limit
PUSAR	–	Power Uprate Safety Analysis Report
RAW	-	Risk Achievement Worth
RCIC	–	Reactor Core Isolation Cooling
RG	–	Regulatory Guide
RHR	–	Residual Heat Removal
RHRSW	–	RHR Service Water
RPS	–	Reactor Protection System
RPT	–	Recirculation Pump Trip
RPV	–	Reactor Pressure Vessel
RWCU	–	Reactor Water Clean Up
SAMG	–	Severe Accident Management Guidelines

SBO	–	Station Blackout
SDBD	-	Shutdown Board
SDC	–	Shutdown Cooling
SJAE	–	Steam Jet Air Ejector
SLC	–	Standby Liquid Control
SMA		Seismic Margins Assessment
SORV	–	Stuck Open Relief Valve
SP	–	Suppression Pool
SPC	–	Suppression Pool Cooling
SPDS	–	Safety Parameter Display System
SR	–	Supporting Requirements
SRP	–	Standard Review Plan
SRV	–	Safety Relief Valve
SSC	-	System, Structure or Component
FSS	–	Fire Safe Shutdown Instructions
T/H	–	Thermal Hydraulic
TAF	–	Top of Active Fuel
THERP	-	Technique for Human Error Rate Prediction
TVA	–	Tennessee Valley Authority
TBV	–	Turbine Bypass Valve
TS	-	Technical Specification
TSS	–	Transmission System Study
VSLOCA	–	Very Small LOCA
%EXFW	–	Excessive Feedwater Initiating Event Frequency
%FWBOC	–	Feedwater Break Outside Containment Initiating Event Frequency
%HIPT	–	RCS High Pressure Trip Initiating Event Frequency
%IMSIV	–	Inadvertent MSIV Closure Initiating Event Frequency
%ISLOCA	–	Interfacing Systems LOCA Initiating Event Frequency
%LCV	–	Loss of Condenser Initiating Event Frequency
%LLDA	–	Large LOCA Recirculation Discharge Line A Initiating Event Frequency
%LLDB	–	Large LOCA Recirculation Discharge Line B Initiating Event

		Frequency
%LLO	–	Other Large LOCA Initiating Event Frequency
%LLSA	–	Large LOCA Recirculation Suction Line A Initiating Event Frequency
%LLSB	–	Large LOCA Recirculation Suction Line B Initiating Event Frequency
%MLOCA	–	Medium LOCA Initiating Event Frequency
%MSBOC	–	Main Steam Break Outside Containment Initiating Event Frequency
%PLFW	–	Partial Loss of Feedwater Initiating Event Frequency
%TLFW	–	Total Loss of Feedwater Initiating Event Frequency
%TT	–	Turbine Trip Initiating Event Frequency

1.4 Assumptions, Modeling, Uncertainties, and Unit Differences

1.4.1 <u>Assumptions</u>

1. It is assumed that initial plant conditions consist of plant trip from extended operation at full power (120% of original licensed thermal power (OLTP), following implementation of EPU).
2. Un-isolated ISLOCAs or breaks outside containment (BOC) are assumed to always result in core damage and large early release. Their low probability justifies screening these events from further consideration with respect to a change from CLTP to EPU conditions.
3. Primary containment is vented with the Hardened Wetwell Vent (HWWV) System or the drywell purge valves when high containment pressures of approximately 50 to 60 psig are reached (Reference 2); and when containment is vented, pumps taking suction from the suppression pool fail in the internal events PRA model due to low NPSH. In the internal events model, no credit is taken for Alternate Shutdown Cooling (ASDC) being in service. Credit (i.e., modeling) for ASDC is taken in the FPRA (see Assumption 5).
4. Replacement of components with enhanced like components is assumed to not result in any significant increase in the long-term failure probability for the components. Equipment reliability can be 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); 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.
5. For the EPU FPRA model, it is assumed that the RHR pumps may be successfully used when ASDC is in service even when containment is not

isolated provided that the ASDC is started within 2 hours of reactor trip. Analyses confirming this assumption are documented in LAR Attachment 6, “Power Uprate Safety Analysis Report (PUSAR).”

6. The MAAP analyses assume that the properties of the steam dryer replacement are identical to the current design.
7. It is assumed that there are no changes in the timing for operator responses associated with LERF analysis. The timing for these operator actions is relatively long, and changes due to EPU power are unlikely to change the degree of recovery included in the individual HEPs.
8. Several HEPs are assumed to have a probability of 0.1 (screening value). The HRA dependency analysis was manipulated to ensure a minimum joint human error probability (JHEP) of 0.1 for any combination of screening events.
9. Changes in ignition sources and targets due to EPU modifications have not been incorporated into the FPRA. Instead, it is assumed that any increases in scenario frequencies due to EPU modifications will not significantly affect risk and that cable routing changes due to EPU modifications will not introduce significant changes in target sets for affected fire scenarios. The risk impact of this assumption is addressed by a sensitivity study (Section 5.7.2.2). The BFN engineering design change processes will assess any such changes to ignition sources and targets for impact to the FPRA and make the appropriate changes to the FPRA prior to implementation of the modification.
10. The BFN Fire Probabilistic Risk Assessment - Summary Document (Reference 5) provides details on open F&Os from the peer reviews of the BFN FPRA. These F&O are included in Appendix A, Table A.2. A subsequent focused scope peer review was conducted in May 2015. However, the responses to some existing F&Os have not been finalized. The focused peer review’s primary purpose was to formally close all outstanding F&O’s with the exception of those F&O’s pertaining to future procedure or plant modification development and implementation. The following F&Os refer to BFN procedures that have not yet been finalized: 2-38, 2-39, 2-41, 2-50, 4-12, 4-17, 4-21, and 9-4. Although procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on realistic proposed actions, including credit for logically routed redundant instrumentation trains. Therefore, the risk calculated by the FPRA models for EPU after the procedures have been finalized is expected to be very similar to the risk calculated for this LAR and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved, and adopted, verification must be made to ensure that the fire HRA still sufficiently matches the final procedures and that no new initiating events are associated with the new procedures. The BFN processes will assess any such changes to procedures for impact to the FPRA and make the appropriate changes to the FPRA prior to implementation of the modification.
11. A subsequent focused scope peer review was conducted in May 2015. PRM-B9-01 replaced existing FPRA F&O 9-2 (Appendix A, Table A.2), which refers to the

representation of the EHPM Pump in the FPRA. The FPRA includes logic to reflect the draft design criteria for the EHPM Pump. The model includes pump failure to start/run, failure of the motor operated injection valves and check valves, failure of the power supply, failure of the water supply, and unavailability due to test and maintenance. Any change in estimated risk due to refinement of the EHPM Pump model upon completion of design activities is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. The BFN engineering design change processes will assess any such changes to ignition sources and targets for impact to the FPRA and make the appropriate changes to the FPRA prior to implementation of the modification.

12. Failure to trip the recirculation pumps, or to inject with SLC during an ATWS, is always assumed to lead to core damage. The BFN models for both CLTP and EPU require both recirculation pumps to trip, and one SLC pump to inject boron to prevent core damage.

1.4.2 Modeling Considerations

1. The PRA success criteria are different than the success criteria used for design basis accident evaluations. It was assumed in the development of the PRA model that systems that can realistically perform a mitigation function (e.g., main condenser or containment venting for decay heat removal) can be credited. In addition, the PRA success criteria are based on the availability of a discrete number of systems or trains (e.g., number of pumps or trains for RPV makeup or decay heat removal).
2. HPCI and RCIC steam cross tie to the auxiliary boiler is not credited because it requires a piping spool piece to be installed and that is assumed to take too long.
3. The emergency high pressure makeup pump modification is only credited for the fire analysis. This pump is not credited in the internal events model, but it is credited in the FPRA models (both CLTP and EPU) as a source of low-pressure RPV inventory. This pump has not been installed yet, but is expected to be installed prior to EPU approval to mitigate core damage risk due to severe fires.

1.4.3 Unit Differences

These EPU Analyses were performed for all units using the updated version of the MOR Rev. 6. The models are similar across the three units; however, due to unit specific differences, the results differ for each unit. All unit differences are treated in the system fault trees and do not impact the accident sequence analysis. The list of unit differences are discussed in the BFN PRA Accident Sequence Analysis (Reference 37).

A brief summary of the most significant unit differences that cause differences in the cutsets include the following:

- RHR cross-tie: Unit 1 has the potential to intertie the division II loop to the adjacent division I loop at Unit 2. Unit 2 then has the ability to intertie with either Unit 1 through this alignment or to intertie the division II loop to the adjacent Unit

3 division I loop. Unit 3 then has the ability to intertie with Unit 2 through the adjacent loop.

- Unit 1 and Unit 2 each have a dedicated Control Rod Drive (CRD) pump and share a swing CRD pump. Unit 3 has two CRD pumps.
- A Unit 3 diesel generator can be used as an alternate power supply to Unit 1 and Unit 2 shutdown buses. However, Unit 1 and Unit 2 diesel generators are not credited to provide alternate power to Unit 3 shutdown buses because doing so has not been proceduralized.

2.0 Scope

The scope of this risk assessment for the EPU at BFN 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

* Internal Fires were originally evaluated in the IPEEE. However, FPRA models have been developed to support PRA applications (Reference 5).

Risk impacts due to internal events are assessed using the BFN Level 1 and Level 2 MOR Rev. 6. These models were modified for Pre-EPU and EPU conditions. External events are evaluated using the insights and results from the BFN IPEEE submittal (References 19, 20, 21, and 22) and more recent FPRA investigations. The impacts on shutdown risk contributions are evaluated on both qualitative and a quantitative bases.

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 BFN EPU modification list (LAR Attachment 47).

The BFN IPE (Reference 48) and IPEEE submittals were reviewed for identification of vulnerabilities, outliers, anomalies or weaknesses that would impact the BFN 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 (Reference 31) of Peach Bottom, which has a similar design to BFN. Additionally, the IPEEE did not identify any vulnerabilities associated with seismic, fire or other external events. Based on this review, there are no vulnerabilities, outliers, anomalies or weaknesses that would impact the results and conclusions of the BFN EPU risk assessment. In summary, all of the commitments resulting from the BFN IPE and IPEEE Programs have been adequately resolved.

3.0 <u>Methodology</u>

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 EPU 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

The General Approach taken to identify and quantify risk of an EPU is summarized below:

- Identify scope of CLTP
- Identify scope of EPU
- Review modifications, procedures and other EPU related changes to BFN
- Identify impact to key PRA elements (see Sections 4.1 through 4.8)
- Update PRA model and Quantify Results (Sections 5.1 through 5.5)
- Total Risk (Section 5.6)
- Identify other evaluations required (see Section 5.7)

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate" Topical Report, Class III, July 2003 (Reference 3), (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of CPPU. Section 10.4 of the CLTR addresses the effect of Constant Pressure Power Uprate on CLTR Individual Plant Evaluation (IPE).

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

3.1.1 <u>Identify PRA Elements</u>

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 (Reference 8). Section 3.2 summarizes the PRA elements assessed for the BFN EPU.

3.1.2 <u>Input to the Analysis</u>

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 EPU. This includes plant changes, 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 from further consideration those items that are estimated to have negligible or no impact on plant risk as modeled by the BFN PRA model.

3.1.4 Qualitative Results

A review of the PRA elements is performed to identify potential effects associated with the EPU. The result of this task is a summary which dispositions all the PRA elements regarding the effects of the EPU. 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

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

This section examines the EPU changes to identify PRA modeling changes needed to quantify the risk impact of the EPU. The impact of EPU changes to the following PRA elements listed is considered:

Identify the impact of EPU to PRA elements

- Initiating Events
    - Transients
    - Loss of Coolant Accidents
    - Support System Failures
    - Internal Floods
    - Interfacing Systems Loss of Coolant Accident (ISLOCA)
    - Internal Fires
- Functional success criteria
    - Timing
    - RPV Inventory Makeup Requirements
    - Heat Load to the Suppression Pool

- Blowdown Loads
- RPV Overpressure Margin
- SRV Actuations
- RPV Emergency Depressurization
- Structural Evaluation

- Accident Sequence Definition
  - Changes to Accident Sequences
  - Impact on Time available for LOOP recovery
- Systems Required to Meet Success Criteria
- Component Reliability
- Human Error Probability
- Containment
- 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 EPU.

3.3.1 Hardware Modifications

The hardware modifications associated with the EPU have been identified by TVA as input to this assessment and is included in LAR Attachment 47. All changes to the plant are evaluated and implemented via Design Change Notices (DCN). Many of the modifications with like components or upgraded existing components are being implemented to address the EPU operational parameter requirements such as higher flows, higher temperatures, or differential pressures, but these changes do not necessarily result in PRA logic changes based on the number of trains required (i.e., number of trains, support systems required, or component reliability). 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. In order to address the impact of these hardware changes, sensitivity evaluations are performed as noted below:

- The transient initiator frequencies are increased to conservatively bound the potential impact from various changes to the Balance of Plant (BOP).
- 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.

Section 5.7 of this document provides the results of all the sensitivity cases performed to address the potential impact of EPU.

It should be noted that many of the modifications have already been implemented, and EPU would only impact the operational margin available to mitigate plant upset conditions. These pre-EPU modifications are considered part of the current PRA baseline (i.e., pre-EPU). Other modifications are required to address the EPU conditions (i.e., current design would not accommodate operational parameters associated with EPU operation). The modifications that are yet to be implemented have been reviewed for PRA impact. The review did identify that the following set of changes could potentially have an impact on the PRA model.

- Design Change Notice (DCN) 51052 (Condensate Booster Pumps & Motors) requires manual actions for transferring alternate power to 4kV Unit Board 3C, whenever 480V SDBD 3B is in alternate alignment. However, this change is applicable only to unique power alignments for Unit 3 that may be needed for maintenance activities or due to a component failure.
- Emergency High Pressure Make Up Pumps will be credited in the FPRA for both for Pre-EPU and EPU conditions

A sensitivity evaluation (Appendix B) has been performed for DCN 51052 to identify the risk associated with this configuration. The sensitivity results indicate that this modification has an insignificant impact on the PRA results even when the manual alignment was assumed to fail (operator actions set to 1.0). Similarly, this modification does not impact the FPRA results since the 161kV system is not credited in the FPRA.

3.3.2 <u>Procedural Changes</u>

In order to ensure the plant is operated safely, adjustments to the BFN Emergency Operating Instructions/Severe Accident Management Guidelines (EOIs/SAMGs) will be made consistent with EPU operating conditions. In almost all respects, the EOIs/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. In addition, procedures need to be developed to align, start and operate the high pressure makeup pump, which is credited in the FPRA.

Based on generic EPU evaluations by the nuclear steam supply system (NSSS) vendor, General Electric, EOI 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 (PSP)

These and other 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 PSP relate to long-term scenarios and any perturbations in the scenario timings associated with EPU changes to these curves will be minor. As discussed in the EPU HRA Analysis, the timing was only updated for time critical operator actions since the independent HEP recovery is unlikely to be impacted for long-term scenarios (Section 4.1.6). The BFN model of record uses timing associated with EPU conditions for HRA evaluations.

As an integral part of the EPU project, the EOIs were reviewed and, where appropriate, will be updated to reflect uprated conditions. Changes made to the EOIs, as a result of the EPU effort, will be approved and implemented prior to raising unit thermal power above 3458 MWt (i.e., the CLTP) on the affected unit.

A list of the changes in Emergency and Abnormal Operating Instructions is included in Section 2.11.1.1 of the PUSAR. The changes in EOIs and the SAMGs reflect the change in power level but will not be changed in a manner that involves a change in accident mitigation philosophy. A list of EOIs that are planned to be revised as a result of EPU can be found in EPU LAR Attachment 6.

In addition, changes are also required in other operating procedures to address events such as loss of feedwater, loss of condenser, or grid instability. Abnormal Operating Procedures (AOPs) at BFN are defined as Abnormal Operating Instructions (AOIs), EOIs, Alarm Response Procedures (ARPs), and Fire Safe Shutdown (FSS) procedures, and select sections of Operating Instructions (OIs).

Finally, EOIs and AOIs will also be rescaled as required to reflect the power uprate.

The Human Performance Evaluation (Reference 32) concludes:

> "The changes to BFN operator actions, as a result of the EPU, are small in number. There are only two time sensitive operator actions. Both of these actions are simple tasks, require a small time duration to perform (< 10 minutes), are performed in the control room or auxiliary instrument room, and will easily be able to be successfully performed within the 1.5 hour and 2 hour required timeframes. The changes to operator actions will be reflected in the procedures and the operators will receive appropriate classroom and/or simulator training prior to EPU implementation. There are no new or revised operator workarounds as a result of EPU."

The Human Performance Evaluation (Reference 32) also addresses changes to the Control Room controls, displays and alarms. The evaluation concludes:

> "The changes to BFN (control Room) CR interfaces as a result of the EPU do not significantly affect operator human performance. Operator training for changes to CR interfaces, alarms, and indications will be accomplished in accordance with the plant training and simulator program as described in Section 2.11.1.5."

In addition, the Human Performance Evaluation (Reference 32) addresses changes in the Safety Parameter Display System (SPDS). The evaluation concludes:

> "The changes to BFN SPDS as a result of the EPU do not significantly affect operator actions and mitigation strategies. The changes will be made in accordance with the configuration change process and the operators will receive appropriate classroom and/or simulator training prior to implementation."

Finally, the Human Performance Evaluation (Reference 32) addresses changes to the Operator Training Program and the Control Room Simulator. Training of Operations personnel will occur on all EPU modifications necessary to support unit operation at EPU conditions. The evaluation states:

"Licensed and non-licensed operator training will be provided prior to the cycle implementing the changes and will focus on plant modifications, procedure changes, startup test procedures, and other aspects of EPU including changes to parameters, set points, scales, and systems."

3.3.3 Setpoint Changes

The RPV operating pressure and the operating temperature are not being changed as part of the EPU.

It should be noted that many of the modifications have already been implemented, and EPU only would impact the operational margin available to mitigate plant upset conditions. These modifications are considered in the current PRA (pre-EPU) baseline. Other modifications are required to address the EPU conditions (i.e., current design would not accommodate operational parameters associated with EPU operation) are considered in the EPU model. A brief summary of this categorization of modifications is included below:

List of Setpoint Change Modifications implemented prior to EPU approval

- Generator Uprate - Uprates the Main Generator to 1330 MVA on Unit 1 and 1332 MVA on Units 2 and 3. Remaining modifications on Unit 2 are limited to setpoint and nameplate changes.
- Steam Jet Air Ejector (SJAE) Condenser Condensate Pressure Switches - Lowers the setpoint of the SJAE Condenser Condensate Pressure Switches to prevent inadvertent SJAE isolation. (Margin Modification)
- BOP Instrument Respan - Respans instruments on the following systems: Hydrogen Water Chemistry, Extraction Steam, and Heater Drains and Vents. (Margin modification)

List of Setpoint Change Modifications after EPU approval

- Rod Worth Minimizer - Changes Rod Worth Minimizer setpoints to reflect operation at EPU conditions. The changes will be performed during the EPU outages.
- Main Steam Line High Flow Instruments - Revises setpoints and indication range for the Main Steam Line High Flow instruments
- Turbine First Stage Pressure Setpoints - Revises the Turbine First Stage Pressure scram bypass setpoints (P-1-81A & B and P-1-91A & B) for operation at EPU.
- Average Power Rate Meter (APRM) Flow Biased and Setdown Instruments - Revises setpoints and indication range for the APRM Flow Biased & Setdown Instruments
- Recirculation Pumps and Motor Uprate - Replaces Reactor Recirculation pump motors, modifies the VFD control system and modifies the Foxboro IA Reactor

Recirculation control system. Partially implemented. Remaining work is limited to Upper Power Runback setpoint changes.

- Feedwater Control System - Modifies the Foxboro Feedwater Control System software speed parameters and the Woodward Governor control parameters for each Reactor Feedwater Pump. Also modifies the Foxboro Human Machine Interface (HMI) displays. (Margin modification). Upon approval of EPU, setpoints will be changed to 120% EPU values.
- Electro-Hydraulic Control (EHC) Software - This modification revised EHC software to address changes in plant parameters required for EPU. The values are to be fine-tuned as part of testing and power ascension.

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 <u>Plant Operating Conditions</u>

The following operating conditions are a result of implementing EPU at BFN.

- Power increase from 3458 MWt to 3952 MWt and associated increase in decay heat
- Higher condensate and feedwater flows to accommodate EPU
- Higher main steam flows to accommodate EPU
- Equipment is exposed to higher differential pressures, and potential of vibration induced failures caused by higher flows in EPU.
- Increased cooling requirements due to larger heat loads generated by larger equipment (Heating Ventilation and Air Conditioning (HVAC) / Equipment Cooling / Isophase bus duct cooling)

It should be noted that RPV pressure will remain unchanged for the EPU.

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 <u>Hardware Changes</u>

The list of modifications is included in LAR Attachment 47. 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

The results of the qualitative review of plant modifications for potential impact to the PRA are included in Table C-1 of Reference 26.

3.4.2 Procedure Changes

Final changes to the EOIs/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 the PUSAR was reviewed for applicability to the PRA model. Based on this review, it is assumed that the procedural changes (e.g., modification to HCTL curve) have a very small 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 attachment). NEDP-26 requires the review of procedure changes for PRA model impact.

3.4.3 Setpoint Changes

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:

- SRV opening and closing setpoints
- RPV pressure setpoint (e.g., Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT) high pressure setpoint)
- The analyses discussed in the PUSAR show that the CLTP setpoints listed above 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. However, acknowledging that increased flow rates of the EPU can result in increased piping erosion/corrosion rates, risk sensitivities have been performed that increase the LOCA initiating event frequencies including main steam and feedwater line breaks (see Section 5.7.1.6).

4.0 PRA Changes Related to EPU Changes

Section 3 has examined the plant changes (hardware, procedural, setpoint, and operational) that are part of the 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 Fire Risk (4.4)
- Seismic Risk (4.4.2)
- Other External Hazards Risk (4.6)
- Shutdown Risk (4.7)
- Radionuclide Release (Level 2 PRA) (4.8)

4.1 Internal Events PRA Elements Potentially Affected by EPU

A review of the PRA elements for the Internal Events PRA has been performed to identify potential effects associated with the EPU. The result of this task is a summary which dispositions all the PRA elements regarding the effects of the EPU. 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 summarizes the results from this review and indicates only a small number of the PRA elements are found to be potentially influenced by the power uprate.

Table 4-1. Review of PRA Elements for Potential Risk Model Effects

PRA Elements	Disposition Category	Bases
Initiating Events	B	No new initiators or impact to the long term average of existing initiator frequencies is anticipated due to EPU. However, there may be an initial increase in the frequency of some internal events initiators due to changes in setpoints during a short term "break-in" period after EPU is implemented. See Section 4.1.1 Quantitative sensitivity cases that increase the transient and LOCA frequencies are performed as part of this analysis as discussed in Sections 5.7.1.4 through 5.7.1.6

Table 4-1. Review of PRA Elements for Potential Risk Model Effects		
PRA Elements	**Disposition Category**	**Bases**
Functional Success Criteria	B	There are a number of potential effects that could alter success criteria. These are discussed in Section 4.1.2. They include the following: • Timing • RPV Inventory Makeup • Heat Load to the Suppression Pool • Blowdown Loads • RPV Overpressure Margin (number of SRVs required) • SRV Actuations post-trip • RPV Depressurization (number of SRVs required) • Structural Evaluations
Accident Sequences	C	EPU results in slight changes to the timing in the accident progression. However, most of these impacts are addressed by the changes to the time available for operator actions. No changes to the plant configuration, or operation in a manner that impacts the internal events accident progression (event trees) are anticipated due to EPU. See Section 4.1.3
System Modeling	B	No new system failure modes or significant changes in system failure probabilities are anticipated due to the EPU. There are modifications that impact several systems, but these modifications do not impact the overall function or support systems included in the PRA model. See Section 4.1.4
Data Analysis (Component Reliability)	C	No change to component failure probabilities are anticipated due to EPU. Modifications are included to increase the robustness of some components that may be impacted by higher flow rates or loads. See Section 4.1.5
Human Reliability Analysis (Human Error Probabilities)	A	The change in initial power level results in decreases in the time available for operator response. The human error probabilities may be impacted due to the change in effective time available for recovery and consequently the dependency level used in the overall evaluation of the recovery included in the human error probability.

Table 4-1. Review of PRA Elements for Potential Risk Model Effects		
PRA Elements	**Disposition Category**	**Bases**
		See Section 4.1.6
Internal Flooding	C	No changes in the internal flooding initiating event frequencies or modeling of equipment is impacted by floods are anticipated based on EPU. Changes to the contribution from the flood initiators could occur due to other modeling changes such as timing for operator actions unrelated to the response of floods. A sensitivity case for flood initiating event frequencies is performed to address impacts to pipes from Flow Accelerated Corrosion (FAC) or vibration induced failures. See Sections 4.1.7 and 5.7.1.5
Level 2 Containment Analysis	B	Slight changes in accident progression timing result from the increased decay heat. This could result in slightly different release category magnitude and timing results. The release magnitude and timing category assignments are expected to be relatively unchanged because the PRA release category for LERF is defined based on the percentage of CsI released to the environment and early release is defined as a release occurring before the effective implementation of the off-site emergency response and protective actions. However, it should be noted that MAAP evaluations for the BFN MOR are based on 3952 MWt (EPU) for release category magnitude and timing. See Section 4.1.9
Quantification	C	No changes in the PRA quantification process (i.e., truncation limit, flag settings, etc.) due to EPU. See Sections 4.1.8

4.1.1 Initiating Events

The CLTR (Reference 3) states that the increase in power level results in the plant operating closer to limits (lower margins) 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 (long term average) of existing initiators are anticipated to result from the BFN EPU.

The BFN PRA Initiating events can be categorized as follows:

- Transients

- Inadvertent Opening of One SRV (IOOV)
- Loss of Coolant Accidents
- Support System Failures
- Internal Floods
- ISLOCA
- External Event Initiators

It should be noted that LOOP initiating events are included in the transient event tree. The interconnection system impact study process requires TVA to identify all adverse system impacts caused by a generation interconnection request, including specifying facility additions, modifications, and upgrades needed to maintain a reliable interconnection. Therefore, no changes to the LOOP initiating event frequency from EPU is expected.

The potential for ATWS has been considered for all initiating events. All transient initiating events can lead to ATWS so they are evaluated through the ATWS tree inside the fault tree model. Additionally, external event initiators are also discussed in this analysis for completeness.

<u>Transients</u>

All three units are similar in design (with respect to initiating events) and are operated with the same procedures and management philosophy as the other units. Units 1-3 have established a significant operational history to assist in the development of appropriate initiating event frequencies for use in the plant PRA models.

The evaluation of the EPU plant and plant 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 with respect to the number of normally operating pumps and equipment in BOP systems). The BFN 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 plant specific experience to include data from 2008 through December 31, 2011 (Reference 33). This method establishes an accepted basis for the applicability of the transient initiating event frequencies utilized in the Browns Ferry PRA model. The initiating events notebook (Reference 33) provides detailed description on the treatment of data for plant initiators that overlap with the data provided by the generic source data to avoid double counting when Bayesian update was performed.

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 scenarios (see Section 5.7.1.4)

<u>Inadvertent Opening of One SRV</u>

No change to the RPV operating pressure is planned in support of the EPU; as such, no impact on IOOV frequency due to the EPU can be postulated. However, as a result of the power increase from EPU, it is assumed that the SRVs may be demanded more frequently and this could result in an SRV failing to reclose. The analysis evaluates the probability of an SRV to close at EPU conditions as discussed in Section 4.1.2.6.

Loss of Coolant Accidents

No changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of the EPU (LAR Attachment 6); 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 (Section 5.7.1.6).

Support System Initiators

No significant changes to support systems (e.g., AC, DC, Control Air, 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 Floods

Since the methodology used in calculating the initiating event frequency for internal flooding is based on the length of piping (Reference 34) 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. 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 performed which conservatively increased all of the internal flood frequencies (see Section 5.7.1.5).

ISLOCA

The ISLOCA initiating events are identified through the development of a series of event trees that serve as the framework for the quantification of ISLOCA scenarios (Reference 35). By exposing piping with a low design pressure to the high pressures of the RCS, pipe failure may occur, generating a LOCA that bypasses the containment boundary establishing direct flow to the environment. No planned modifications as part of the BFN EPU (LAR Attachment 47) have been identified that expose low pressure piping to high pressure; as such, no impact on the ISLOCA frequencies due to EPU can be postulated.

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 frequency of external events is not affected by EPU. The potential impacts on their mitigation (i.e., fire, seismic, and other external events) are discussed in Sections 5.2, 5.4, and 5.5, respectively.

Internal Events Summary

No planned operational modifications as part of the BFN 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 (see Section 5.7).

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 (Reference 3) conclusions on this issue:

> "Based on PRA experience for uprated BWRs, CPPU 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 Functional Success Criteria

The success criteria for the MOR Rev. 6 used in both the pre-EPU and EPU assessments 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 BFN. 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. The analysis performed by General Electric and MAAP runs discussed in Reference 36 performed for the BFN EPU risk assessment were used to assess impacts on success criteria.

The BFN PRA model addressed the key safety functions that must be successful to bring the plant to a stable/safe core damage end state. The following key safety functions were selected to match the BFN EOIs, for which procedural guidance is available to assist the plant staff in mitigating accidents (Reference 37).

- Reactivity Control
- Reactor Pressure Control
- Inventory (RPV level) Control
- Decay Heat Removal
- Containment Pressure Control

The EOI's also address other safety functions associated with the primary containment that are not directly modeled in the Level 1 PRA including:

- Drywell Temperature
- Drywell Hydrogen Control
- Suppression Pool Level Control
- Secondary Containment Temperature
- Secondary Containment Level
- Secondary Containment Radiation

The final safety function to consider is associated with a Level 3 analysis is listed below. However, it should be noted that the BFN PRA models do not include a Level 3 model.

- Offsite Radiation Release

In order to address the impact to the functional success criteria derived from the safety functions listed above the areas listed below need closer examination. As noted in Table 4-1, the potential changes to the success criteria from EPU need to consider the effect of the following areas:

- Timing
- RPV Inventory Makeup Requirements
- Heat Load to the Suppression Pool
- Blowdown Loads
- RPV Overpressure Margin
- SRV Actions post-trip
- RPV Depressurization
- Structural Evaluations

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 Human Reliability Analysis (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., Feedwater (FW), High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC)) and low pressure (e.g., LPCI, CS, and condensate) injection systems have more than adequate flow margin for the post-uprate configuration. 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. The success criteria (Reference 36) are determined based on the accident sequence key timings provided in the T/H analysis in Reference 38. The success criteria remain unchanged since the current PRA model is based on the EPU power. However, the T/H was performed based on a MAAP version earlier than 4.07. Since the MAAP current version is 4.07, a benchmark was performed to compare with the original key timings generated by the thermal-hydraulics calculation (Reference 38). The MAAP version change is discussed in Attachment G of Reference 26.

4.1.2.3 Heat Load to the Suppression Pool

Energy to be absorbed by the pool during an isolation event or RPV depressurization increases for the EPU (3952 MWt) case relative to the original license basis power level (3293 MWt). 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 EPU. 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 Suppression Pool Cooling (SPC) systems are sufficient for containment heat removal for the EPU condition (LAR Attachment 6). With respect to containment venting, an evaluation has been performed (References 39 and 40) that shows that the emergency containment vent is clearly sufficient for the EPU conditions. It should be noted that BFN Units 1, 2, and 3 HWWV path consists of torus penetration X-205, the 20" pressure suppression chamber supply piping downstream of valve FCV-64-20, and a 14" line to a common 14" header. The 14" common header runs underground in the yard and then discharges in the stack above elevation 666.5' (Reference 41).

4.1.2.4 Blowdown Loads

Dynamic loads would increase slightly because of the increased stored thermal energy in the core. 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 (LAR Attachment 6).

4.1.2.5 RPV Overpressure Margin

The RPV dome operating pressure will not be increased as a result of the CPPU. However, the RPV pressure following a failure to scram is expected to increase slightly. For transient scenarios, Reference 27 indicates that there is sufficient overpressure protection for transient response. 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 negligible.

For ATWS scenarios, Reference 27 indicates that the plant-specific results of the analysis meet the ATWS acceptance criteria listed below. Therefore, the response to an ATWS event at EPU is acceptable. As such, there is no change warranted to the overpressure success criteria for ATWS scenarios.

a. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig;

b. Peak suppression pool temperature less than 281°F (Wetwell shell design temperature); and

c. Peak containment pressure less than 56 psig (Drywell design pressure).

The BFN T/H (Reference 38) presents an ATWS analysis. Results show that the peak dome pressure can be maintained within the Service Level C limit (1500 psig) if 11 of 13 SRV’s open. MAAP run, ATWS6, indicates that even if all SRV’s open, the Service

Level C limit will be exceeded within 16 seconds if recirculation pumps do not trip. The recirculation pumps are required to trip in ATWS scenarios for both CLTP and EPU.

It should be noted that the MOR Rev. 6 PRA does not require any SRVs for initial RPV overpressure control for transient or LOCA initiators. This success criterion also remains unchanged for the EPU. As such, no model changes regarding RPV Overpressure are required for this BFN PRA EPU risk assessment.

4.1.2.6 SRV Actions post-trip

The SRV setpoints have not been changed as a result of the BFN 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. It is assumed that the increase in the power level results in an increase in the number or duration of SRV actuations. This analyses assumes that this results in an increased likelihood of a SORV.

The PRA MOR Rev. 6 base case stuck open relief valve probability may be modified using different approaches to consider the effect of a postulated increase in valve cycles. As discussed, in similar analyses 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 1.143 in the case of the BFN EPU to represent the ratio of EPU/CLTP). 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., Modular Accident Analysis Program (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 case 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 the most conservative approach defined above and it is used in this study to modify the PRA stuck open relief valve probability. The SORV probability basic events in the BFN PRA are increased 14.3% for the EPU base case risk evaluation. This factor was used for one or more stuck open relief valves as shown in Table 4-2.

Table 4-2. Stuck Open Relief Valve Probabilities CLTP vs. EPU			
BE ID	**Description**	**CLTP Probability**	**EPU Probability**
SRVFC1PCV_0SORV	NO SRVS STICK OPEN	9.992E-01	9.9909E-01
SRVFC1PCV_1SORV	ONE SRV STICKS OPEN	7.695E-04	8.7943E-04
SRVFC1PCV_2SORV	TWO OR MORE SRVS STICK OPEN	2.555E-05	2.9200E-05
SRVFC2PCV_0SORV	NO SRVS STICK OPEN	9.992E-01	9.9909E-01
SRVFC2PCV_1SORV	ONE SRV STICKS OPEN	7.695E-04	8.7943E-04
SRVFC2PCV_2SORV	TWO OR MORE SRVS STICK OPEN	2.555E-05	2.9200E-05
SRVFC3PCV_0SORV	NO SRVS STICK OPEN	9.992E-01	9.9909E-01
SRVFC3PCV_1SORV	ONE SRV STICKS OPEN	7.695E-04	8.7943E-04
SRVFC3PCV_2SORV	TWO OR MORE SRVS STICK OPEN	2.555E-05	2.9200E-05
SRVFC1PCV_0SORVA	PROBABILITY OF NO STUCK OPEN SRVS WITH AN ATWS	9.992E-01	9.9909E-01
SRVFC1PCV_1SORVA	PROBABILITY OF STUCK OPEN SRV WITH AN ATWS	7.695E-04	8.7943E-04
SRVFC1PCV_2SORVA	ATWS, 2 OR MORE STUCK OPEN SRVS	2.555E-05	2.9200E-05
SRVFC2PCV_0SORVA	PROBABILITY OF NO STUCK OPEN SRVS WITH AN ATWS	9.992E-01	9.9909E-01
SRVFC2PCV_1SORVA	PROBABILITY OF STUCK OPEN SRV WITH AN ATWS	7.695E-04	8.7943E-04
SRVFC2PCV_2SORVA	ATWS, 2 OR MORE STUCK OPEN SRVS	2.555E-05	2.9200E-05
SRVFC3PCV_0SORVA	PROBABILITY OF NO STUCK OPEN SRVS WITH AN ATWS	9.992E-01	9.9909E-01
SRVFC3PCV_1SORVA	PROBABILITY OF STUCK OPEN SRV WITH AN ATWS	7.695E-04	8.7943E-04
SRVFC3PCV_2SORVA	ATWS, 2 OR MORE STUCK OPEN SRVS	2.555E-05	2.9200E-05

4.1.2.7 RPV Depressurization

The PRA assumes that two SRVs are required for emergency RPV depressurization (ED) at CLTP or EPU conditions. MAAP results confirm that two SRV are adequate for ED at EPU conditions as shown in Cases 2C and 2D (Reference 38(a)). More precisely, early core damage is avoided by depressurizing the reactor sufficiently to allow LPCI and LPCS injection prior to core melt. Reference 38(b) indicated that core melt can be prevented if two SRV's are opened as late as 30 minutes after transient initiation.

4.1.2.8 Structural Evaluations

The original Structural Analysis Notebook (Reference 42) was prepared to assemble the pertinent structural information associated with the BFN Units 2 & 3 PRA update in April 2000. It was revised in May, 2002 to address items specific to the 20% EPU of BFN Units 2 and 3 from the OLTP of 3293 MWt to 3952 MWt, assuming no change in maximum normal operating reactor dome pressure.

Sections 6.1 through 6.3.2 of Structural Analysis Notebook include structural information presented in the IPE. Section 6.3.3 provides the containment dynamic loading limits used in the probabilistic evaluation for the PRA model update of containment failure under postulated degraded conditions.

The Structural Analysis Notebook includes BFN Unit 1 plant specific information with the plant uprated to the same EPU conditions as BFN Units 2 and 3. Containment information initially developed for the BFN PRA update remains applicable to severe accident evaluations for BFN Unit 1 at EPU conditions.

Reference 27 addresses the effect of EPU on containment system performance.

In addition, the PUSAR documents several structural evaluations. Comprehensive review has assessed the effects of increase power conditions on the reactor vessel and its internals. These reviews and associated analyses show continued compliance with the original design and licensing criteria for the RPV and internals. The evaluations include:

- Reactor Vessel Structural Evaluation - (Reference 27)
- Internals Structural Evaluation - (Reference 27)

Finally, the task report on “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment” provides additional structural/mechanical evaluations (Reference 43)

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.

4.1.3 Accident Sequences

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 for the internal events PRA. This assessment for BFN is consistent with CLTR conclusions on this issue (Reference 3):

> "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."
>
> "For CPPU, operator responses to anticipated occurrences, accidents and special events are not significantly affected. Significant events result in automatic plant shutdown (scram). Some events result in automatic reactor coolant pressure boundary pressure relief, Automatic Depressurization System (ADS) actuation and/or automatic Emergency Core Cooling System (ECCS) actuation (for low water level events). All events included in the plant design basis result in safety related systems, structures and components remaining within their acceptance limits. CPPU does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow on operator actions for plant safety (e.g., maintaining safe shutdown, core cooling, containment cooling) do not change for CPPU, although required operator response time may change."

The BFN PRA uses a single general transient tree for most initiators with the exception of ATWS, IOOV, LOCAs, and ISLOCA. In general, all the event trees address loss of makeup to the RPV and decay heat removal. The trees for ATWS, IOOV, LOCAs and ISLOCA address some additional mitigation requirements. For instance, the ATWS tree also address functions that impact reactivity control (Reactor Protection System (RPS), Standby Liquid Control (SLC), Inhibit ADS, etc.). Similarly, the LOCA trees may include, depending on the size of the break, pressure suppression via the drywell vacuum breakers. Finally, the ISLOCA trees address operator action to isolate the break.

These functions and corresponding mitigation systems models are generally unchanged for EPU with the following exception. 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 only other change involves DCN 51052. This modification changes the electrical configuration for 4kV Unit Board 3C. The modification requires a manual transfer for the alternate power to the board during some unique power alignments created during maintenance activities. However, a sensitivity showed that this configuration had an insignificant impact on the PRA results (See Appendix B).

The Transmission System Study (TSS) evaluates the capability of the grid to support the safe shutdown in the event of a Design Basis Event. Based on the preliminary results of the Transmission System Study (also known as the Grid Stability Report), no new impacts are caused by implementation of EPU. This study compared the ability of the system to provide adequate shutdown power before and after the uprate. It was found that there was no significant impact to the ability of the grid to supply sufficient shutdown power due to the power uprate (Reference 53).

Finally, it should be noted that the BFN PRA LOOP recovery file was reviewed to determine whether it needed to be updated to address the timing differences between CLTP and EPU. However, BFN uses a relatively simplified approach for LOOP recovery based on early recovery or late recovery of offsite power (30 minutes or 4 hours). It was concluded that the change in power would not impact the LOOP recovery file.

4.1.4 System Models Required to Meet Success Criteria

For the most part, based on the review of modifications listed in LAR Attachment 47, the BFN plant changes associated with the EPU do not result in the need to change any system models credited in the Internal Events PRA. The only exception involves Condensate Booster Pumps and Motors (DCN 51052) for Unit 3. This modification requires manual actions for transferring alternate power to 4kV Unit Board 3C, whenever 480V SDBD 3B is in alternate alignment. However, implementation of this modification results in an insignificant change in risk.

Finally, it should be noted that the high pressure makeup pump is a new system being implement to reduced risks associated with fires. This system is not credited in the Internal Events PRA.

4.1.5 Component Reliability Probabilities

The CLTR (Reference 3) 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, or
- 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-hour PRA mission time due to the replacement/modification of plant components is anticipated, nor is such a quantification possible at this time. No planned operational modifications as part of the BFN 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

> "...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."

It should also be noted that several modifications have or are being implement to make components or structures more robust and minimize the effect of FAC or flow induced vibration. These modifications and any enhanced monitoring would help identify, detect and/or mitigate potential increases in the failure of components or structures that could results in plant trips or impact component reliability. In general, many of the modifications increase the robustness of components to help mitigate the impact of EPU conditions such as higher flows, differential pressures and increased temperatures. The following are examples of modifications in LAR Attachment 47 that assist in mitigating potential failures from EPU loads and conditions:

- Main Steam Supports
- Main Steam Acoustic Vibration Suppressors
- Main Steam Tie-Back Supports
- Replacement Steam Dryer
- #3 Feedwater Heater (FWH) Nozzle Relocation
- #3 FWH Upper Shell Replacement
- #4 FWH Tube Bundle Replacement
- FWH Pass Partition Plates
- FWH Nozzles/Shell Relief Valves
- Torus Attached Piping
- Isophase Bus Cooling
- Jet Pump Sensing Line Clamps

These modifications do not have any direct impact to components credited in the PRA. However, sensitivity analyses in Section 5.7.1.6 are used to gain insights about the potential impact on plant trips or LOCA frequencies.

Finally, it should be noted that the probability of an SORV was increased to address a higher likelihood of this failure mode in the EPU model.

4.1.6 Human Error Probabilities

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

The current BFN PRA MOR conservatively uses the EPU reactor power of 3952 MWt in the MAAP evaluations that are used to confirm the success criteria, and to determine the timing associated with operator actions. In this Attachment, MAAP evaluations are performed at the CLTP of 3458 MWt to allow establishing the change in timing used for operator actions.

All operator actions in the model were screened to determine the impact from EPU. However, the analysis focused additional scrutiny on several operator actions that were considered significant to the results. The operator actions identified for explicit review were selected based on the following criteria:

- Time critical evolutions (i.e., less than 45 minutes available) action

MAAP calculations for the BFN CLTP configuration were performed to determine how the operator action timelines were impacted. All time critical post-initiator HEPs in the model were then re-calculated using the same HRA methods used in the BFN HRA document. Table 4-4 provides the changes in operator action timings and associated HEPs due to the EPU.

The operator actions that are not part of this list of time critical evolutions were evaluated to determine if the recovery factor used for cognitive actions would change due to the change from CLTP to EPU. It was noted that for EPU conditions, the recovery factor for these actions was based on “Low Dependence,” therefore evaluation at CLTP conditions would yield similar dependence levels.

It should be noted that the MOR human reliability analyses generally does not credit “self-review” in the evaluation of cognitive errors. Similarly, this analysis does not credit “self-review” for cognitive errors at EPU conditions, but it credits “self-review” for the cognitive errors at the CLTP. This approach is conservative and the net impact is as follows:

- The approach maximizes the change in human error probabilities
- The approach maximizes the total risk estimate.

No significant changes are to be made to the Control Room for the EPU that would impact the existing actions included in the BFN 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 changing setpoints. None of these Control Room changes will have a measurable impact on the human reliability analysis for the BFN PRA. However, the changes that impacted the HEP values as identified in Table 4-4 are factored directly into the risk assessment and the changes to CDF and LERF are reported.

<u>Approach for Independent Post-Initiator HEPs</u>

A key area to determine the impact of EPU is the change in the time available for operator response. EPU is not expected to impact the human reliability analyses for pre-initiator human error probabilities. However, the change in power can impact the time assumed for post-initiators (See Appendix B). BFN uses the EPRI HRA calculator to estimate human error probabilities. Figure 8-20 of Reference 44 of shows the timing information considered in the analyses by the HRA calculator.

The terms associated with each timing element are defined mathematically next and then further described in the subsequent text:

T0 = start time = start of the event

Tdelay = time delay = duration of time it takes for an operator to acknowledge the cue

Tsw = system time window

Tavail = time available = time available for action = (Tsw - Tdelay)

Tcog = cognition time consisting of detection, diagnosis, and decision making

Texe = execution time including travel, collection of tools, donning of Personal Protective Equipment, and manipulation

of relevant equipment

Treqd = time required = response time to accomplish the action = (Tcog + Texe)

Structuring the timeline in this way allows the analyst to demonstrate, among other things, the feasibility of the action from the perspective of timing. An operator action is only feasible when the time required to complete the action is less than the time available. The time available (Tavail) consists of the system time window (Tsw) minus any time delays (Tdelay), for example, time delay until the relevant cue for the action is received. The time required (Treqd) consists of the time to recognize the needed action (Tcog) and the time to execute the action (Texe); this is also called the crew response time.

Based on the above data entries, the effective time available for recovery is calculated. Based on the time available for recovery, a minimum level of dependency applicable to recovery actions is suggested by the program. This is a dependency level based on timing alone, so the level of dependency should not be lower than this suggested level. This dependency level is propagated through the analysis as a “global variable” and shown as a note on the recovery windows for both cognitive and execution.

The first step to determine the impact on the human error probabilities caused by implementing EPU is use a thermal-hydraulics code to determine the relevant time information for each applicable operator action. For this analysis, the MAAP computer code is used to determine some of the times shown in Figure 8-20 of Reference 44, such as system time window (Tsw) and the time delay (Tdelay). The remaining relevant times shown on Figure 8-20 of Reference 44 are typically defined by operator interview, and/or simulator exercises. It should be noted that timing for some operator actions are impacted by changes in RPV power since these actions may be derived based on other parameters such as time for battery depletion, time for irreversible equipment failure, or depletion of inventory.

To determine the change in human error probabilities MAAP is used to determine the times at CLTP and EPU conditions. This information is input in the HRA for each of these two conditions, and the human error probabilities are recalculated. The updated human error probabilities are input in the fault tree model to estimate changes in CDF and LERF.

No additional changes are expected for the operator actions other than the time available for operator response. If the change in time between CLTP and EPU does not result in a different dependency level, the human error probability is not expected to change.

Approach for Joint Post-Initiator HEPs

Risk metrics can be significantly underestimated if dependencies between multiple human failure events are not addressed. The HRA Calculator is the tool used to help address dependencies.

The HRA Calculator dependency analysis considers the timing and other factors to establish sequence of events of a pair of operator actions. A level of dependency between these pairs of operator actions is assigned based on these factors and the joint human error probability (JHEP) is assigned using the formulas in Table 4-3.

Table 4-3. THERP Dependency Equations*	
Dependence Level	**Equation**
Zero	HEP
Low	(1+19xHEP)/20
Medium	(1+6xHEP)/7
High	(1+HEP)/2
Complete	1.0

***Once the level of dependency has been assigned, the HRA Calculator uses these equations to calculate JHEPs**

The JHEPs change if the independent probabilities change or the sequence of operator actions changes.

If the human error probability does not change, it may be desirable to perform sensitivity studies on a limited set of operator actions. The approaches for the sensitivity studies are slightly different since the change in time available for recovery does not result in a change in dependency level. In these cases, sensitivity studies may be performed parametrically or by assuming a higher stress level since this determines the multiplier used in the calculation of the "execution" HEP. Either of these approaches uses a multiplier in the HEP calculation. For sensitivity studies, selected HEPs can be updated to reflect changes the impact caused by EPU conditions.

Finally, it should be noted that there is some degree of uncertainty in results depending on the minimum joint HEP selected. The minimum joint HEP of 1.0E-7 was retained in this study for consistency in the comparison of results. This minimum joint HEP value is also used because of the large number of operator actions and combinations included in the model. This value is two orders of magnitude higher than the minimum joint HEP value recommended in Reference 45. However, several sensitivity evaluations were performed to address the impact of the minimum joint HEP value used in the dependency analysis.

Table 4-4 shows the independent operator actions and identifies whether they meet the criteria listed in the discussion above:

Table 4-4. BFN Independent Post-Initiator HEP Results Summary for Pre-EPU and EPU Conditions					
#	**BFN BE ID**	**Action Description**	**Pre-EPU HEP**	**EPU HEP**	**Key Timings: T(sw)_cltp / T(sw)_epu**
1	HFA_0_ADSINHIBIT	Failure to inhibit ADS during an ATWS event	7.53E-04	1.50E-03	522 s / 475 s
2	HFA_0_ATWSLEVEL	Operator Fails to Run Back RFPs and Maintain Level at TAF	2.89E-02	3.21E-02	885 s / 800 s
3	HFA_0001HPRVD1	Failure to initiate	3.99E-04	4.47E-04	32 m / 30 m

Table 4-4. BFN Independent Post-Initiator HEP Results Summary for Pre-EPU and EPU Conditions					
#	**BFN BE ID**	**Action Description**	**Pre-EPU HEP**	**EPU HEP**	**Key Timings: T(sw)_cltp / T(sw)_epu**
		reactor-vessel depressurization (transient or ATWS)			
4	HFA_0001MSIVATWS	Operator Fails to Bypass Low Level MSIV Closure Setpoint	1.84E-02	2.89E-02	32 m / 30 m
5	HFA_0002RPV_LVL	Operator Fails To Maintain RPV Level	4.78E-04	1.32E-03	42 m / 35 m
6	HFA_0003P_START_A	Operator Fails to Start Standby/Tripped RFW Pumps - ATWS	9.09E-03	2.72E-02	32 m / 30 m
7	HFA_0003PMP_START	Operator Fails To Restart RFW After Level 8 Trip	9.88E-03	1.14E-02	42 m / 35 m
8	HFA_0063SLCINJECT	Failure to SLC in response to an ATWS event	2.48E-04	4.95E-04	522 s / 475 s
9	HFA_0071L8RESTART	Operator fails to restart RCIC after Level 8 trip	6.97E-03	7.97E-03	42 m / 35 m
10	HFA_0071LVL8_TRIP	Failure to trip HPCI or RCIC upon reaching RPV level 8	2.16E-02	3.21E-02	885 s / 800 s
11	HFA_0071MANLEVEL	Operator fails to manually control level with RCIC	2.16E-02	3.21E-02	885 s / 800 s
12	HFA_0073L8RESTART	Operator fails to restart HPCI after Level 8 trip	4.39E-03	4.55E-03	42 m / 35 m
13	HFA_0073LVL8_TRIP	Failure to trip HPCI upon reaching RPV level 8	2.16E-02	3.21E-02	Conservatively used same HEP for HPCI or RCIC (see #10 HFA_0071LVL8_TRIP)
14	HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	2.16E-02	3.21E-02	885 s / 800 s
15	HFA_0268480CRSTIE	Failure to transfer de-energized 480v board to alternate supply	3.89E-02	4.00E-02	32 m / 30 m
16	HFA_0HCIINIT30	Operator Fails To Initiate HPI (30 Min)	1.50E-03	3.06E-03	42 m / 35 m

Table 4-4. BFN Independent Post-Initiator HEP Results Summary for Pre-EPU and EPU Conditions					
#	**BFN BE ID**	**Action Description**	**Pre-EPU HEP**	**EPU HEP**	**Key Timings: T(sw)_cltp / T(sw)_epu**
17	HFA_0LPIINIT10	Operator Fails To Manually Initiate Low Pressure Injection (10 Min)	5.42E-03	1.14E-02	42 m / 35 m

There are more than one thousand combinations of operator actions included in the cutset results for CLTP and EPU operator actions dependency analysis. Appendix B provides a list of changes for time delay overrides that is used to address inappropriate sequencing of operator actions. Table 4-5, Table 4-6, and Table 4-7 provide a listing of the significant JHEPs, their contribution and change in CDF for Units 1-3, respectively. The combinations listed contribute over ninety-five percent of the change in core damage frequency. A complete listing of all HEPs used in the PRA, and their corresponding descriptions are included in Appendix C.

Unit 1

Table 4-5. BFN Unit 1 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
COMBINATION_1092	9.45E-03	1.31E-02	6.27E-08	9.33E-08	3.07E-08	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1
COMBINATION_1148	3.84E-04	5.38E-04	2.55E-09	3.84E-09	1.29E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE
COMBINATION_1195	1.25E-03	2.37E-03	8.28E-09	1.69E-08	8.63E-09	HEP dependency factor for HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1199	8.62E-04	1.20E-03	5.71E-09	8.53E-09	2.82E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0085ALIGNCST
COMBINATION_1201	6.67E-04	1.27E-03	4.42E-09	9.06E-09	4.64E-09	HEP dependency factor for HFA_0HCIINIT30,HFA_0LPIINIT30
COMBINATION_1203	2.30E-03	4.42E-03	1.53E-08	3.15E-08	1.63E-08	HEP dependency factor for HFA_0HCIINIT30,HFA_0001HPRVD1
COMBINATION_1372	4.82E-03	4.62E-03	3.20E-08	3.30E-08	1.03E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0002RPV_LVL
COMBINATION_1709	1.09E-03	1.17E-03	7.25E-09	8.38E-09	1.13E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1715	1.22E-03	1.68E-03	8.06E-09	1.20E-08	3.95E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0LPIINIT30
COMBINATION_2424	1.68E-03	1.80E-03	1.11E-08	1.29E-08	1.77E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0001HPRVD1
COMBINATION_2679	2.41E-03	3.35E-03	1.60E-08	2.39E-08	7.93E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP
COMBINATION_276	9.20E-04	1.28E-03	6.10E-09	9.12E-09	3.02E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD
COMBINATION_3081	6.21E-04	8.60E-04	4.12E-09	6.14E-09	2.02E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0001HPRVD1
COMBINATION_3360	4.70E-03	6.51E-03	3.12E-08	4.65E-08	1.53E-08	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1_L
COMBINATION_342	4.76E-02	6.58E-02	3.16E-07	4.70E-07	1.54E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1
COMBINATION_366	3.40E-04	4.74E-04	2.26E-09	3.39E-09	1.13E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD

Table 4-5. BFN Unit 1 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
COMBINATION_45	8.73E-03	8.31E-03	5.78E-08	5.94E-08	1.51E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_455	1.76E-03	1.84E-03	1.17E-08	1.31E-08	1.42E-09	HEP dependency factor for HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_462	6.59E-04	9.11E-04	4.37E-09	6.50E-09	2.13E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0074ALIGN_DWS
COMBINATION_49	8.72E-03	8.31E-03	5.78E-08	5.93E-08	1.51E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_95	3.86E-02	5.34E-02	2.56E-07	3.81E-07	1.25E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_956	3.13E-03	3.37E-03	2.08E-08	2.41E-08	3.30E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0001HPRVD1_L
COMBINATION_98	1.57E-03	2.17E-03	1.04E-08	1.55E-08	5.09E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD,HFA_0074ALIGN_DWS
COMBINATION_990	5.09E-04	7.04E-04	3.38E-09	5.03E-09	1.65E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0001HPRVD1,HFA_0280ALNALTBBD
		Total	**3.34E-06**	**3.75E-06**	**4.13E-07**	

Unit 2

Table 4-6. BFN Unit 2 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
COMBINATION_1092	1.05E-02	1.44E-02	6.26E-08	9.33E-08	3.07E-08	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1
COMBINATION_1148	4.17E-04	5.76E-04	2.49E-09	3.73E-09	1.24E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE
COMBINATION_1195	1.39E-03	2.61E-03	8.27E-09	1.69E-08	8.63E-09	HEP dependency factor for HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1199	9.56E-04	1.32E-03	5.70E-09	8.52E-09	2.82E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0085ALIGNCST
COMBINATION_1201	7.49E-04	1.41E-03	4.46E-09	9.15E-09	4.69E-09	HEP dependency factor for

Table 4-6. BFN Unit 2 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	FV (CLTP)	FV (EPU)	CDF (CLTP)	CDF (EPU)	ΔCDF	Description
						HFA_0HCIINIT30,HFA_0LPIINIT30
COMBINATION_1203	2.57E-03	4.89E-03	1.53E-08	3.16E-08	1.63E-08	HEP dependency factor for HFA_0HCIINIT30,HFA_0001HPRVD1
COMBINATION_1709	1.21E-03	1.30E-03	7.24E-09	8.38E-09	1.14E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1715	1.40E-03	1.92E-03	8.32E-09	1.24E-08	4.09E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0LPIINIT30
COMBINATION_2424	1.86E-03	1.99E-03	1.11E-08	1.29E-08	1.78E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0001HPRVD1
COMBINATION_2679	2.64E-03	3.64E-03	1.57E-08	2.36E-08	7.81E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP
COMBINATION_276	9.97E-04	1.37E-03	5.94E-09	8.87E-09	2.93E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD
COMBINATION_3081	6.91E-04	9.50E-04	4.12E-09	6.14E-09	2.02E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0001HPRVD1
COMBINATION_3360	5.11E-03	7.04E-03	3.05E-08	4.56E-08	1.51E-08	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1_L
COMBINATION_342	5.30E-02	7.28E-02	3.16E-07	4.71E-07	1.55E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1
COMBINATION_45	9.70E-03	9.18E-03	5.78E-08	5.94E-08	1.57E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_455	1.96E-03	2.03E-03	1.17E-08	1.31E-08	1.44E-09	HEP dependency factor for HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_462	7.35E-04	1.01E-03	4.38E-09	6.53E-09	2.15E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0074ALIGN_DWS
COMBINATION_49	9.69E-03	9.17E-03	5.78E-08	5.93E-08	1.57E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_95	4.29E-02	5.88E-02	2.55E-07	3.81E-07	1.25E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_956	3.55E-03	3.80E-03	2.12E-08	2.46E-08	3.40E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0001HPRVD1_L
COMBINATION_98	1.74E-03	2.39E-03	1.04E-08	1.55E-08	5.09E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LV

Table 4-6. BFN Unit 2 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
						L8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD,HFA_0074ALIGN_DWS
COMBINATION_990	5.65E-04	7.76E-04	3.37E-09	5.02E-09	1.65E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0001HPRVD1,HFA_0280ALNALTBBD
		Total	**3.25E-06**	**3.66E-06**	**4.14E-07**	

Unit 3

Table 4-7. BFN Unit 3 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
COMBINATION_1092	9.53E-03	1.32E-02	6.26E-08	9.33E-08	3.07E-08	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1
COMBINATION_1148	3.80E-04	5.30E-04	2.50E-09	3.74E-09	1.24E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE
COMBINATION_1195	1.26E-03	2.39E-03	8.27E-09	1.69E-08	8.63E-09	HEP dependency factor for HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1199	8.63E-04	1.20E-03	5.67E-09	8.46E-09	2.79E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0085ALIGNCST
COMBINATION_1201	6.73E-04	1.28E-03	4.42E-09	9.06E-09	4.64E-09	HEP dependency factor for HFA_0HCIINIT30,HFA_0LPIINIT30
COMBINATION_1203	2.32E-03	4.47E-03	1.53E-08	3.15E-08	1.63E-08	HEP dependency factor for HFA_0HCIINIT30,HFA_0001HPRVD1
COMBINATION_1709	1.10E-03	1.19E-03	7.24E-09	8.38E-09	1.14E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0HCIINIT30,HFA_0002RPV_LVL
COMBINATION_1715	1.22E-03	1.69E-03	8.01E-09	1.19E-08	3.92E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0LPIINIT30
COMBINATION_2424	1.69E-03	1.83E-03	1.11E-08	1.29E-08	1.77E-09	HEP dependency factor for HFA_0003PMP_START,HFA_0001HPRVD1
COMBINATION_2679	2.43E-03	3.38E-03	1.60E-08	2.38E-08	7.89E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP
COMBINATION_276	1.62E-03	2.24E-03	1.06E-08	1.58E-08	5.20E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD
COMBINATION_3081	6.27E-04	8.69E-04	4.12E-09	6.13E-09	2.01E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,H

Table 4-7. BFN Unit 3 JHEP Post-Initiator HEP Results Summary For Pre-EPU And EPU Conditions						
JHEP ID	**FV (CLTP)**	**FV (EPU)**	**CDF (CLTP)**	**CDF (EPU)**	**ΔCDF**	**Description**
						FA_0001HPRVD1
COMBINATION_3360	7.45E-04	1.04E-03	4.89E-09	7.36E-09	2.47E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073L8RESTART,HFA_0001HPRVD1_L
COMBINATION_342	4.80E-02	6.65E-02	3.15E-07	4.70E-07	1.54E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1
COMBINATION_45	8.81E-03	8.42E-03	5.79E-08	5.94E-08	1.53E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_455	1.79E-03	1.87E-03	1.18E-08	1.32E-08	1.43E-09	HEP dependency factor for HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_462	6.89E-04	9.55E-04	4.52E-09	6.74E-09	2.22E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0074ALIGN_DWS
COMBINATION_49	8.82E-03	8.42E-03	5.79E-08	5.95E-08	1.54E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_527	6.49E-03	6.20E-03	4.26E-08	4.38E-08	1.15E-09	HEP dependency factor for HFA_0268480CRSTIE,HFA_0248ALNPWRSUP,HFA_0248ALNALTCHG,HFA_0280ALNALTBBD
COMBINATION_95	3.90E-02	5.40E-02	2.56E-07	3.81E-07	1.25E-07	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0001HPRVD1,HFA_0280ALNALTBBD
COMBINATION_98	1.55E-03	2.15E-03	1.02E-08	1.52E-08	4.98E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0074SPCLATE,HFA_0280ALNALTBBD,HFA_0074ALIGN_DWS
COMBINATION_990	5.09E-04	7.05E-04	3.34E-09	4.98E-09	1.63E-09	HEP dependency factor for HFA_0073MANLEVEL,HFA_0073LVL8_TRIP,HFA_0003PMP_START,HFA_0001HPRVD1,HFA_0280ALNALTBBD
		Total	**3.01E-06**	**3.41E-06**	**3.97E-07**	

4.1.7 Internal Flooding

No changes in the internal flooding modeling were incorporated based on changes to implement EPU. The initiating event frequencies and affected equipment from the flood event used in 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

In general, no significant changes in the BFN PRA quantification process due to the EPU have identified or implemented (i.e., the same, software, linked tree approach, configuration flags, HRA dependency process are used). Any changes in the quantification results (accident sequence frequencies) were realized as a result of the minor modeling changes, or changes in the human error probabilities as described above.

It should be noted that this study uses a 1.0E-12 truncation value in estimating both CDF and LERF. This change was necessary to address quantification of the results without impacting significantly the overall insights from the model. The code performance (speed) was impacted significantly due to the large number of joint HEP combinations analyzed in the BFN model.

Finally, it should also be noted that the minimum joint HEP value used in the model of record (1E-7) was retained for consistency in the comparison of results. There is significant debate in the industry regarding the best approach for addressing the impact of joint HEP thresholds. However, sensitivities are included in this study to determine the impact of varying the minimum joint HEP thresholds.

4.1.9 Containment Reliability (Level 2)

The deterministic evaluation of the containment performance and impact by the EPU has been performed by General Electric-Hitachi and addressed in Reference 27. For PRA modeling, the containment structural failure treatment has incorporated the detailed insights and analysis from the BFN specific drywell assessment with additional insights from the Chicago Bridge and Iron study on Peach Bottom coupled with the known differences in design, documented in the BFN Structural Analysis Notebook (Reference 42) to obtain an updated structural assessment of the containment capability under severe accident conditions.

The primary containment ultimate structural integrity is important in severe accident analysis due to its key role as a fission product barrier. The BFN Mark I containment has been analyzed to predict its ability to withstand severe accident conditions, i.e., pressures and temperatures imposed on containment prior to, during, and following core melt progression accidents.

The original Structural Analysis Notebook was prepared to assemble the pertinent structural information associated with the BFN Units 2 and 3 PRA update in April 2000. It was revised in May 2002 to address items specific to the 20% EPU of BFN Units 2 and 3 from the OLTP of 3293 MWt to 3952 MWt, assuming no change in maximum normal operating reactor dome pressure (Reference 42). Although the Level 2 MOR is based on the EPU thermal power (Reference 46), there should be insignificant change 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 14.3% 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 PRA LERF release category is defined based on the percentage (as a function of End of Cycle inventories) of Cesium Iodide (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. Reference 46 provided LERF analysis for EPU conditions. For pre-EPU, the variations in the absolute magnitude of the releases may occur and changes 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 will be higher from EPU power levels, the definition of LERF in the BFN PRA is based on fractional releases which do not change. The BFN PRA does not include a Level 3 model and is not explicitly required to be evaluated for EPU.

4.2 FPRA Elements Potentially Affected by EPU

A review of PRA elements specifically related to FPRA has been performed to identify potential effects associated with the EPU. The result of this task is a summary which dispositions the FPRA elements regarding the effects of the EPU. The disposition consists of three qualitative disposition categories.

- Category A: Potential FPRA change due to. power uprate, FPRA modification desirable or necessary
- Category B: Minor perturbation, negligible impact on FPRA, no FPRA changes required
- Category C: No change

Table 4-8 summarizes the results from this review. Based on Table 4-8, only a small number of the FPRA elements are found to be potentially influenced by the power uprate.

Table 4-8 Review of FPRA Elements for Potential Risk Model Effects

FPRA Element	Disposition Category	Bases
Fire Initiating Events	B	No new fire scenarios or significant impact to existing fire scenario frequencies is anticipated due to EPU. See Assumption 12 and Section 4.2.1

Table 4-8 Review of FPRA Elements for Potential Risk Model Effects		
FPRA Element	**Disposition Category**	**Bases**
Fire-Induced Accident Sequences	C	EPU results in slight changes to the timing in the accident progression. However, most of these impacts are addressed by the changes to the time available for operator actions. No changes to the plant configuration, or operation in a manner that impacts the fire-induced accident progression (event trees) are anticipated due to EPU. See Section 4.2.2
FPRA System Modeling	B	No new system failure modes or significant changes in system failure probabilities are anticipated due to the EPU. There are modifications that impact several systems, but these modifications do not impact the overall function or support systems included in the FPRA model. See Section 4.1.4
Post-Fire Human Reliability Analysis (Human Error Probabilities)	A	The change in initial power level results in decreases in the time available for operator response. The fire context human error probabilities may be impacted due to the change in effective time available for recovery and consequently the dependency level used in the overall evaluation of the recovery included in the human error probability. See Section 4.2.4
FPRA Quantification	C	No changes in FPRA quantification process (i.e., truncation limit, flag settings, etc.) due to EPU. See Section 4.2.5

4.2.1 Fire Initiating Events (Fire Scenarios)

It is expected that any changes in fire scenario frequencies as a result of EPU will have little effect on the risk results and no change is being made to the scenario frequencies in the base case analysis as a result of EPU (see Assumption 12). This expectation is addressed by a sensitivity study (see Section 5.7.2.2).

4.2.2 Fire-Induced Accident Sequences

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 for the internal events PRA. EPU results in slight changes to the timing in the accident progression. However, most of these impacts are addressed by the changes to the time available for operator actions (see Section 4.2.4). No changes to the plant configuration, or operation in a manner that impacts the fire-induced accident progression (event trees) are anticipated due to EPU. The discussion in Section 4.1.2 applies to fire-induced event sequences.

4.2.3 FPRA Systems Models Required to Meet Success Criteria

The discussion in Section 4.1.4 applies to the FPRA as well. As noted, the BFN plant changes associated with the EPU do not result in the need to change any system models credited in the PRA.

The high pressure makeup pump is a new system being implement to reduced risks associated with fires. This system is modeled in the same way in the CLTP and EPU FPRA analyses. See Assumption 8 for a discussion that indicates that the EHPM Pump is not expected to be needed for the alternate shutdown cooling mode under EPU conditions to inject additional inventory into the suppression pool, thereby providing additional net positive suction head (NPSH) margin for the RHR pumps. The risk impact of this assumption is investigated in a sensitivity study as noted in Section 5.7.2.3.

4.2.4 Post Fire Human Error Probabilities

All human failure events (HFEs) in the FPRA model were reviewed and considered for modification for the FPRA EPU impact study. HFEs that satisfy either of the following conditions were selected for detailed review:

- Fussell-Vesely (FV) importance greater than 5E-03, or
- Tsw less than 40 min.

FV importance is considered an appropriate discriminator because changing HEPs with low FV importances will have little effect on the risk measures and therefore on the conclusions of the analysis. In addition, because the post-transition FPRA currently uses EPU-specific HEPs, the HEP adjustments are to be made for the CLTP model. The required adjustments are reductions in HEP due to greater time available for the actions in the CLTP case. Therefore, the existing FV importances bound the corresponding importances for the CLTP model. A cutoff FV of 5E-03 was chosen because that is the value used to identify high-safety-significant equipment in the Maintenance Rule (that value was recommended in the NUMARC 93-01 guidance document (Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 4a, April 2011) and endorsed in Regulatory Guide 1.160 (Regulatory Guide 1.160, Rev. 3, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants)).

Table 4-9 lists the HFEs identified for detailed review based on FV > 5E-03 or Tsw < 40 min. Table 4-10 summarizes the HEPs that changed for CLTP and provides an indication of the importance of each HFE taken from the FPRA quantification notebook that supports the 805 LAR (Ref. 29 (NDN-000-999-2012-000012, Rev. 4, TVA FIRE PRA – TASK 7.14: Fire Risk Quantification)). Each action is characterized by either a MAAP case number, to be explained below, or the designation “Not Sensitive” where it was determined that the action timing was not affected by the EPU changes. If an HEP change was warranted, it was done in a special version of the HRA Calculator and the result is provided in the table. This altered HRA calculator file was used for all quantifications and dependency analyses associated with this Attachment.

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFEs with FV Greater than 5E-3				
HFFA_1SHV0760540_35 Local action - close 1-SHV-076-0540 (2- and 3- for Units 2 and 3) within 35 minutes	CASE1	3.78E-03	No change	HEP from calculator file action HFFA0ASD_1VLV35
HFFA_2SHV0760540_35 Local action - close 1-SHV-076-0540 (2- and 3- for Units 2 and 3) within 35 minutes	CASE1	3.78E-03	No change	HEP from calculator file action HFFA0ASD_1VLV35
HFFA_3SHV0760540_35 Local action - close 1-SHV-076-0540 (2- and 3- for Units 2 and 3) within 35 minutes	CASE1	3.78E-03	No change	HEP from calculator file action HFFA0ASD_1VLV35
HFFA_ASDCINIT OPERATOR FAILS TO INITIATE ALTERNATE SHUTDOWN COOLING	Not Sensitive	NA	NA	No comment
HFFA_BDISOL_2531_PANEL_35 Local action at 25-31 panel to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDISOL_480_DG_AUX_A_35 Local action at 480V Diesel Aux Board A to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDISOL_480_RMOV_1A_35 Local action at 480V RMOV Board 1A to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDISOL_480_RMOV_2A_35 Local action at 480V RMOV Board 2A to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDISOL_480_SDB_1B_35 Local action at 480V Shutdown Board 1B to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA_BDISOL_480_SDB_2A_35 Local action at 480V Shutdown Board 2A to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDISOL_480_SDB_3A_20 Local action at 480V Shutdown Board 3A to isolate circuits within 20 minutes	LOCA01C/03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_BDISOL_480_SDB_3A_35 Local action at 480V Shutdown Board 3A to isolate circuits within 35 minutes	CASE1	5.79E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_1B_35 Local action at 250V RMOV Board 1B (2B, 3B) to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_1C_35 Local action at 250V RMOV Board 1C to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_2B_35 Local action at 250V RMOV Board 1B (2B, 3B) to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_2C_35 Local action at 250V RMOV Board 2C to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_3B_35 Local action at 250V RMOV Board 1B (2B, 3B) to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_250_RMOV_3C_35 Local action at 250V RMOV Board 3C to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA_BDOPER_480_RMOV_1B_35 Local action at 480V RMOV Board 1B to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_480_RMOV_2B_35 Local action at 480V RMOV Board 2B to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPER_480_RMOV_3B_35 Local action at 480V RMOV Board 3B to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_480_RMOV_2A_35 Local action at 480V RMOV Board 2A to operate component – early HPI within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_480_RMOV_2D_20 Local action at 480V RMOV Board 2D to operate component – late DHR within 20 minutes	LOCA01C/03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_BDOPERLATE_480_RMOV_2D_35 Local action at 480V RMOV Board 2D to operate component – late DHR within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_480_RMOV_3D_20 Local action at 480V RMOV Board 3D to operate component – late DHR within 20 minutes	LOCA01C/03A	NA	NA	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_480_RMOV_3D_35 Local action at 480V RMOV Board 3D to operate component – late DHR within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_4KV_SDB_3EA_20 Local action at 4kV Shutdown Board 3EA to operated component – late DHR within 20 minutes	LOCA01C/03A	NA	NA	Action dependency level did not change with increased Tsw

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA_BDOPERLATE_4KV_SDB_3EA_35 Local action at 4kV Shutdown Board 3EA to operated component – late DHR within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_4KV_SDB_B_35 Local action at 4kV Shutdown Board B to operated component – late DHR within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_BDOPERLATE_4KV_SDB_C_35 Local action at 4kV Shutdown Board C to operated component – late DHR within 35 minutes	CASE1	5.64E-03	No change	Action dependency level did not change with increased Tsw
HFFA_SDBD_DG_4KV_3EA_20 Local action at 4kV Shutdown Board 3EA to isolate circuits within 20 minutes	LOCA01C/ 03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_SDBD_DG_4KV_3EA_35 Local action at 4kV Shutdown Board 3EA to isolate circuits within 35 minutes	CASE1	9.50E-03	No change	Action dependency level did not change with increased Tsw
HFFA_SDBD_DG_4KV_3EB_20 Local action at 4kV Shutdown Board 3EB to isolate circuits within 20 minutes	LOCA01C/ 03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_SDBD_DG_4KV_3EB_35 Local action at 4kV Shutdown Board 3EB to isolate circuits within 35 minutes	CASE1	9.50E-03	No change	Action dependency level did not change with increased Tsw
HFFA_SDBD_DG_4KV_A_20 Local action at 4kV Shutdown Board A to isolate circuits within 20 minutes	LOCA01C/ 03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_SDBD_DG_4KV_A_35 Local action at 4kV Shutdown Board A to isolate circuits within 35 minutes	CASE1	9.50E-03	No change	Action dependency level did not change with increased Tsw
HFFA_SDBD_DG_4KV_B_20 Local action at 4kV Shutdown Board B to isolate circuits within 20 minutes	LOCA01C/ 03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_SDBD_DG_4KV_B_35 Local action at 4kV Shutdown Board B to isolate circuits within 35 minutes	CASE1	9.50E-03	No change	Action dependency level did not change with increased Tsw

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA_SDBD_DG_4KV_C_20 Local action at 4kV Shutdown Board C to isolate circuits within 20 minutes	LOCA01C/ 03A	NA	NA	Current HEP conservative for EPU and non-EPU conditions
HFFA_SDBD_DG_4KV_C_35 Local action at 4kV Shutdown Board C to isolate circuits within 35 minutes	CASE1	9.50E-03	No change	Action dependency level did not change with increased Tsw
HFFA0001HPRVD1 FAILURE TO INITIATE REACTOR-VESSEL DEPRESSURIZATION (30 MIN INITIATION)	CASE6A	6.33E-04	No change	Current HEP conservative for EPU and non-EPU conditions
HFFA0001HPRVD3 FAILURE TO INITIATE REACTOR-VESSEL DEPRESSURIZATION (8 HOUR INITIATION)	Not Sensitive	NA	NA	No comment
HFFA0002RPV_LVL OPERATOR FAILS TO MAINTAIN RPV LEVEL (fire)	CASE1	4.75E-03	1.87E-03	No comment
HFFA0003RXLVLATWS OPERATOR FAILS TO MAINTAIN RPV LEVEL (NON-ATWS)(FIRE)	Not Sensitive	NA	NA	No comment
HFFA0023ALIGNEECW OPERATOR FAILS TO ALIGN BACKUP EECW PUMP (fire)	Not Sensitive	NA	NA	No comment
HFFA0023SBCI FAILURE TO INITIATE STANDBY COOLANT INJECTION (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0024RCW_START OPERATOR FAILS TO START BACKUP RCW PUMPS (fire)	Not Sensitive	NA	NA	No comment
HFFA0064HWWV FAILURE TO USE HARDENED WETWELL VENT FOR LONG-TERM DHR (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0064MANUAL OPERATOR FAILS TO MANUALLY OPEN HWWV	Not Sensitive	NA	NA	No comment
HFFA0074ALIGN_DWS FAILURE TO ALIGN DRYWELL SPRAY AND GAIN SPRAY VALVE CONTROL (FIRE)	Not Sensitive	NA	NA	No comment

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA0074HPSPC1 FAILURE TO ALIGN RHR FOR SUPPRESSION POOL COOLING (NON-ATWS/IORV)(FIRE)	Not Sensitive	NA	NA	No comment
HFFA0074RHR_CST OPERATOR FAILS TO ALIGN RHR PUMPS TO CST (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0074SDC_ALIGN OPERATORS FAILS TO ALIGN SDC (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0074SPCLATE FAILURE TO ALIGN RHR FOR SUPPRESSION POOL COOLING IN THE LONG TERM (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0075CSCST OPERATOR FAILS TO ALIGN CORE SPRAY PUMPS TO CST (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0085REFILCST OPERATOR FAILS TO ADD ADDITIONAL INVENTORY TO CST	Not Sensitive	NA	NA	No comment
HFFA02114KVCRSTIE FAILURE TO CROSSTIE DE-ENERGIZED 4KV SHUTDOWN BOARD TO ENERGIZED SHUTDOWN BOA	Not Sensitive	NA	NA	No comment
HFFA0231480SDBTIE FAILURE TO TRANSFER 480V SHUTDOWN BOARD TO ALTERNATE SOURCE (FIRE)	CASE1	2.14E-03	No change	Action dependency level did not change with increased Tsw
HFFA0248ALNALTCHG FAILURE TO ALIGN ALTERNATE BATTERY CHARGER (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0248ALNPWRSUP OPERATOR FAILS TO ALIGN ALTERNATE POWER SUPPLY (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0268480CRSTIE FAILURE TO TRANSFER DEENERGIZED 480V BOARD TO ALTERNATE SUPPLY (FIRE)	CASE1	1.28E-02	1.07E-02	No comment

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA0280ALNALTBBD OPERATOR FAILS TO ALIGN ALTERNATE FEEDER (FIRE)	Not Sensitive	NA	NA	No comment
HFFA043ISODGALIGN OPERATOR FAILS TO LOCALLY OPERATE 43 SWITCH TO MANUALLY ALIGN THE EDG	Not Sensitive	NA	NA	No comment
HFFA04KCDSBISO OPERATOR FAILS TO ISOLATE FIRE IMPACTED LOADS ON 4KV SB C/D	Not Sensitive	NA	NA	No comment
HFFA0ASD_MCRACOG OPERATOR FAILS TO ABANDON WHEN NECCESARY - 0 SORVs	Not Sensitive	NA	NA	No comment
HFFA0ASD_PTLRHR OPERATOR FAILS TO PLACE RHR PUMP IN PULL TO LOCK BEFORE ABANDONING MCR	Not Sensitive	NA	NA	No comment
HFFA0ASD_RCIC OPERATOR FAILS TO START RCIC	CASE1	3.45E-02	2.99E-02	No comment
HFFA0ASD_SRV20 OPERATOR FAILS TO EMERGENCY DE-PRESSURIZE THE RPV IN 20 MIN	Not Sensitive	NA	NA	No comment
HFFA0ASD_SRV35 OPERATOR FAILS TO EMERGENCY DE-PRESSURIZE THE RPV IN 35 MIN	CASE6A	NA	NA	Action dependency level did not change with increased Tsw
HFFA0ECCSBYP OPERATOR FAILS TO INHIBIT ECCCS SIGNALS AFTER SCRAM	Not Sensitive	NA	NA	No comment
HFFA0LPIINIT120 OPERATOR FAILS TO INITIATE LPCI IN 2 HOURS	Not Sensitive	NA	NA	No comment
HFFA0SUPHPIPWR OPERATOR FAILS TO TRANSFER TO ALTERNATE SUPP HPI POWER	Not Sensitive	NA	NA	No comment

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA0SUPPHPI1 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 20 MIN	LOCA01C/ 03A	9.92E-03	No change	Current HEP conservative for EPU and non-EPU conditions
HFFA0SUPPHPI2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN	CASE1	2.92E-03	1.10E-03	No comment
HFFA0SUPPHPI3 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 2 hr	Not Sensitive	NA	NA	No comment
HFFA4KVISO_LPI OPERATOR FAILS TO MANUALLY INITIATE INJECTION INTO DRYWELL AFTER CORE DAMAGE (FIRE)	CASE1	6.48E-03	No change	Action dependency level did not change with increased Tsw
HFFAMSO2PSCRCIC-F OPERATOR FAILS TO STOP RCIC AND PREVENT OVERFILL (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0001HPRVD2 OPERATOR FAILS TO INITIATE DEPRESSURIZATION (LERF)	Not Sensitive	NA	NA	No comment
HFFA0IR1_HPI OPERATOR FAILS TO MANUALLY INITIATE INJECTION FOR IN-VESSEL RECOVERY (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0TD2_HPI OPERATOR FAILS TO MANUALLY INITIATE INJECTION INTO DRYWELL AFTER CORE DAMAGE (FI	Not Sensitive	NA	NA	No comment
HFFA0064PCICLOSE OPERATOR FAILS TO MANUALLY CLOSE PRIMARY CONTAINMENT ISOLATION VALVES (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0IR2_LPI OPERATOR FAILS TO MANUALLY INITIATE INJECTION FOR IN-VESSEL RECOVERY (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0ASD_MSIV OPERATOR FAILS TO CLOSE MSIVs	Not Sensitive	NA	NA	No comment

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA0ASD_CILCLOSE OPERATOR FAILS TO FAIL AIR TO CONTAINMENT ISOL VLVS	Not Sensitive	NA	NA	No comment
HFFA0ADSINHIB OPERATOR FAILS TO INHIBIT ADS	Not Sensitive	NA	NA	No comment
HFFA0TD2_LPI OPERATOR FAILS TO MANUALLY INITIATE INJECTION INTO DRYWELL AFTER CORE DAMAGE (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0RHRPMPTRIP OPERATOR FAILS TO TRIP DEADHEADED RHR PUMP	Not Sensitive	NA	NA	No comment
HFFA0MSDRAINISO OPERATOR FAILS TO ISOLATE THE MS DRAINS AFTER FIRE IMPACT	Not Sensitive	NA	NA	No comment
HFFA0OVERLOADSDB OPERATOR FAILS TO LOCALLY ISOLATE LOADS FROM SD BOARD AND REPOWER FROM OSP	Not Sensitive	NA	NA	No comment
HFFA0N2DIVISO OPERATOR FAILS TO ISOLATE FLOW DIVERSION FROM DW N2	Not Sensitive	NA	NA	No comment
HFFA0OPENSPCVLV OPERATOR FAILS TO LOCALLY OPEN THE RHRHX DISCHARGE VALVE	Not Sensitive	NA	NA	No comment
HFEs with Tsw Less than 40 minutes (and FV Less than 5E-3)				
HFFA0EDGTRIP OPERATOR FAILS TO TRIP EDG IF NO COOLING AVAILABLE	Not Sensitive	NA	NA	No comment
HFFA0ASD_MCR OPERATOR FAILS TO PERFORM IMMEDIATE MCR ACTIONS	Not Sensitive	NA	NA	No comment
HFFAMSO2PSKLM-F OPERATOR FAILS TO CLOSE DISCHARGE VALVE AFTER CCW PUMP TRIP	Not Sensitive	NA	NA	No comment

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFAMSO2UHPCI-F OPERATOR FAILS TO STOP HPCI AND PREVENT OVERFILL (FIRE)	Not Sensitive	NA	NA	No comment
HFFA0RHRCS_LPP OPER FAILS TO BYPASS ECCS LOW PRESSURE PERMISSIVE	CASE1	1.94E-02	5.80E-03	No comment
HFFA_RCWLOAD OPERATOR FAILS TO RECOVER RCW BY CLOSING LOAD VALVES	Not Sensitive	NA	NA	No comment
HFFA0LPCIINJAUTO OPERATOR FAILS TO MANUALLY OPEN LPCI INJECTION VALVE	CASE1	9.54E-03	2.82E-03	No comment
HFFA0SBISO OPERATOR FAILS TO ISOLATE AND REPOWER 4kv SDB	CASE1	2.22E-03	No change	Action dependency level did not change with increased Tsw
HFFA0ASD_HPMU2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN AFTER MCR ABANDONMEN	CASE1	6.22E-04	No change	Action dependency level did not change with increased Tsw
HFFA0LPIINIT30 FAILURE TO ESTABLISH LOW-PRESSURE INJECTION GIVEN LOSS OF HIGH PRESSURE INJECTIO	CASE1	3.62E-03	1.41E-03	No comment
HFFA0RESTARTLPI OPERATOR FAILS TO RESTART LPI AFTER PAS OR AUTOSTART FAIL (30 min)	Not Sensitive	NA	NA	No comment
HFFA0071CTLPOWER OPERATOR FAILS TO TRANSFER TO BACKUP POWER (FIRE)	CASE1	2.81E-02	9.24E-03	No comment
HFFA0071L8RESTART OPERATOR FAILS TO RESTART RCIC AFTER LEVEL 8 TRIP (FIRE)	CASE1	8.07E-03	No change	Action dependency level did not change with increased Tsw
HFFA0073L8RESTART OPERATOR FAILS TO RESTART HPCI AFTER LEVEL 8 TRIP (FIRE)	CASE1	5.70E-03	No change	Action dependency level did not change with increased Tsw

Table 4-9. FPRA Human Actions Reviewed				
HFE Name / Description	**MAAP Case**	**EPU HEP**	**CLTP HEP**	**Note**
HFFA0HCIINIT30 OPERATOR FAILS TO INITIATE HPI (30 MIN)(FIRE)	CASE1	1.50E-03	No change	Action dependency level did not change with increased Tsw

HFE Name / Description	MAAP Case	EPU HEP	CLTP HEP	805 LAR FV Importance for CDF (Ref. 29)
Table 4-10. Summary of FPRA Human Actions Changed for CLTP				
HFFA0002RPV_LVL OPERATOR FAILS TO MAINTAIN RPV LEVEL (fire)	CASE1	4.75E-03	1.87E-03	Unit 1: 1.72E-02 Unit 2: 3.47-02 Unit 3: 9.57E-03
HFFA0268480CRSTIE FAILURE TO TRANSFER DEENERGIZED 480V BOARD TO ALTERNATE SUPPLY (FIRE)	CASE1	1.28E-02	1.07E-02	Unit 1: 1.81E-02 Unit 2: 2.31E-02 Unit 3: 2.02E-02
HFFA0ASD_RCIC OPERATOR FAILS TO START RCIC	CASE1	3.45E-02	2.99E-02	Unit 1: 5.91E-02 Unit 2: 5.17E-02 Unit 3: 5.15E-02
HFFA0SUPPHPI2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN	CASE1	2.92E-03	1.10E-03	Unit 1: 3.47E-02 Unit 2: 6.25E-02 Unit 3: 2.71E-02
HFFA0RHRCS_LPP OPER FAILS TO BYPASS ECCS LOW PRESSURE PERMISSIVE	CASE1	1.94E-02	5.80E-03	Unit 1: 6.80E-04 Unit 2: 3.90E-04 Unit 3: 2.25E-03
HFFA0LPCIINJAUTO OPERATOR FAILS TO MANUALLY OPEN LPCI INJECTION VALVE	CASE1	9.54E-03	2.82E-03	Unit 1: 1.00E-05 Unit 2: 1.00E-05 Unit 3: No importance provided
HFFA0LPIINIT30 FAILURE TO ESTABLISH LOW-PRESSURE INJECTION GIVEN LOSS OF HIGH PRESSURE INJECTIO	CASE1	3.62E-03	1.41E-03	Unit 1: 2.00E-05 Unit 2: 2.00E-05 Unit 3: 2.90E-04
HFFA0071CTLPOWER OPERATOR FAILS TO TRANSFER TO BACKUP POWER (FIRE)	CASE1	2.81E-02	9.24E-03	No importance provided

4.2.4.1 MAAP Cases

The metrics for two sets of MAAP analyses were reviewed to assess their impact on the modeled FPRA post-accident human actions. These two sets were:

1. MAAP 4.0.7 run of equivalent cases from Reference 38 (407_EPU) (see KeyTimings-BC.xlsx in Attachment 2)
2. MAAP 4.0.7 run of equivalent cases from Reference 38 with current licensing thermal power (CLTP) (407_CLTP) (see KeyTimings-BC.xlsx in Attachment 2).

From this review it was determined that only one metric, time to core melt, had any significance for the evaluation of the fire human failure events (HFEs). The review determined the following MAAP cases were important for determining the impact of the EPU changes on the HFEs. The cases and their impact are discussed below. The metric presented is time to core damage, which is represented by Tsw in the HRA.

Case 1 is a general transient scram with no injection.

Time to core damage (Tsw):

- 407_EPU: Tsw = 0.59 hrs or 35.4 min
- 407_CLTP: Tsw = 0.70 hrs or 42 min

A Tsw of 35 minutes typically used in the EPU Fire HRA for actions to initiate injection if all injection is lost at the scram and there are no stuck open relief valves. The CLTP modeling uses 42 minutes for these same HRAs in all general transient (GTRAN) sequences. If an HFE uses a Tsw different than 35 minutes that is still dependent on time to core damage, then Tsw will be increased by 7 minutes for the CLTP modeling. In most cases, this timing change will result in a dependency level change and thus a change in HEP. Even with additional time available for the CLTP case, extra crew is not credited for the FPRA because the presence of the fire may prevent additional involvement by other crew members for these short term actions.

Case 5 is total loss of FW with no control rod drive (CRD) injection. One safety relief valve (SRV) is open at time t = 0.

Time to core damage (Tsw):

- 407_EPU: Tsw = 0.51 hrs
- 407_CLTP: Tsw = 26.46 hrs

For Case 5 under EPU conditions, the vessel is not depressurized quickly enough to allow low pressure injection to restore and maintain level prior to core damage. For Case 5 under CLTP conditions, the vessel is depressurized and low pressure injection restores level prior to core damage. Core damage for Case 5 under CLTP conditions does not occur until after containment fails.

Case 5A is the same as Case 5, except that the SRV opening is delayed 5 to 39 minutes.

Time to core damage (Tsw):

- 407_EPU: Tsw = 0.58 hrs
- 407_CLTP: Tsw = 0.69 hrs

In the CLTP analysis of Case 5, the open SRV is allowed just enough time to establish injection with a low-pressure system, whereas in the EPU analysis of Case 5, there was not enough time and core damage occurred before the reactor pressure was low enough to establish effective low-pressure injection. This scenario is the same as the IOOV sequences where one SRV is open at t=0. The EPU IOOV model requires manual depressurization within 30 minutes for successful injection. The CLTP results indicate that manual depressurization is not needed (assuming the SRV is open at t=0); however, if the SRV opening is delayed (fire opens it later) then a manual depressurization is still needed. This condition is more represented by Case 5A which assumes a delay in depressurization between 5 and 35 minutes. The credited depressurization action in the IOOV model is a 30 minutes action. Because the CLTP analysis of Case 5 is obviously right on the edge between long-term success and early core damage, and because there is no guarantee in a fire that the SRV will be open at time t=0, Case 5A is the most applicable case and the IOOV modeling is correct as is. In other words, an operator action to manually depressurize the RPV within 30 minutes is needed for sequence success. No HRA changes are needed for this change in case results.

Because the EPU and CLTP timing for Case 5A is almost identical to that of Case 1, the time available for initiation of injection will change as recommended in Case 1 for all IOOV sequences.

<u>Case 6A is total loss of FW with no CRD injection.</u>

This case shows 30 minutes is available to depressurize with two SRVs for GTRAN sequences. Time to core damage (Tsw):

- 407_EPU: Tsw = 21.93 hrs
- 407_CLTP: Tsw = 26.48 hrs

These results are immaterial to the PRA because the PRA assumes 30 minutes is available to depressurize and this is applicable to both EPU and CLTP MAAP analyses. Both scenarios assume no DHR and core damage occurs after containment failure at around 17 to 18 hours. The FPRA is not sensitive to this timing.

<u>Cases LOCA 01C and LOCA 03A</u>

LOCA 01C is a 0.14 sq ft break and LOCA 03A is a 4.2 sq ft break. If these two data sets are compared with cases involving multiple stuck open SRVs, each SRV having a throat diameter of approximately 5 inches, it can be concluded that the current Tsw of 20 minutes used for sequences with multiple stuck open SRVs is conservative for both CLTP and EPU conditions. It was estimated that the non-EPU timing to core damage for CLTP conditions is about 3 minutes longer than for EPU conditions. This small difference in timing was not considered long enough to warrant any changes to the fire HRA, especially in light of the fact that the current timing of 20 minutes is conservative for all cases.

4.2.4.2 HRA Dependency Analysis

An HRA dependency analysis is an integral part of all HRA evaluations. Multiple operator actions found in a single cutset are of special interest to PRA results, because they may require consideration to avoid nonconservative treatment of the operator actions. Dependencies between operator actions are treated during the quantification post-processing stage where specific sets of multiple operator actions are recovered by inserting into the affected cutsets the ratio between the calculated dependent joint probability and the independent joint probability as a multiplier.

The control-room abandonment and nonabandonment models are treated separately because the fault trees themselves are distinct. The HFE combinations are selected from cutset files obtained by quantifying the models with the fire HEPs set to a value of

0.1. In order to remain within the capability of the available HRA tools and techniques, the number of cutsets retained is limited to less than 500,000 for all three units. The process uses the cutset probabilities as a basis for retaining the most significant cutsets and HFE combinations. This is done as follows:

- When compiling the HFE combination cutsets, they are sorted by probability.
- The total number of cutsets retained is limited to 500,000 by adjusting the truncation limit. The top 500,000 cutsets are kept. A consistent truncation limit across the analysis is applied.
- The HRA calculator is used to extract the unique combinations from the retained cutsets and compute the dependency multipliers. This process generates recovery rule files to be used in the final model quantifications.
- User defined dependencies are applied to the HRA calculator model as required. These user defined dependencies are defined in the HRA calculator DAF file.

The HRA calculator automatically assigns dependencies between HFE pairs for each combination. These dependencies and the order of the HFEs established for each combination are based on the value of the parameter t_{delay}. The HRA calculator orders the HFEs for each combination according to the HFE's t_{delay} value for all nonzero instances of t_{delay}. It also assigns dependencies between HFEs based on the t_{delay} time between the actions. This process does not always work well with t_{delay} values normally assigned to HFEs. In order to establish the correct order of HFEs and the correct dependencies between HFEs, dummy t_{delay} values were assigned to each HFE to ensure its correct ordering and dependencies for each combination.

Additional information on the procedure for determining and implementing the HRA dependencies is provided in *TVA FIRE PRA –Task 7.12 Post-Fire Human Reliability Analysis* (Ref. 72). The recovery rules that result from the dependency analysis are contained in the rule files associated with each quantification. The dependency analysis is used to support the comparison of the two plant conditions, CLTP and EPU. To ensure a proper comparison, the same HRA Calculator file is used to evaluate both conditions for a particular case. An HRA Calculator file used for a particular condition does, however, have timing parameters unique to that condition. The cases evaluated are:

- Abandonment CDF
- Abandonment LERF
- Nonabandonment CDF
- Nonabandonment LERF

All of the above use a different fault tree for quantification.

The nonabandonment models were treated in the same manner as delineated in NDN-000-999-2012-000011, Rev. 4, TVA FIRE PRA – Task 7.12 Post-Fire Human Reliability Analysis (Ref. 72). The user-defined dependency tool in the HRA Calculator is used to fine tune the dependency analysis and reduce dependency conservatisms that affect significant cutsets. Due to the large number of combinations (six to seven thousand in the nonabandonment models), the fine tuning is limited to the top cutsets.

The abandonment models are treated in a different manner. The main control room fire abandonment process, in its current state of development, uses a single master procedure for all three units. This procedure specifies the actions required of all of the unit shift supervisors and unit operators in a series of either sections in the main procedure or attached subordinate procedures. Each attachment is given to an individual plant operator to perform. Because of this tightly controlled process, there is a great deal of independence between modeled actions. An individual human failure event (HFE) may be defined to include, for example, performance of several steps in the procedure attachment that isolates a pump's breaker at the board and then places the breaker in the desired position. Another operator may meanwhile perform a similar task at another board for another piece of equipment. The two HFEs representing these actions would have a very low dependency. The T_{delay} parameter in the HRA calculator was used to establish, at least to a practical extent, the ordering of the actions. The dependency levels were then set, using the user defined dependency tool, by an analyst with knowledge of the BFN FPRA, the BFN fire HRA, the BFN fire abandonment model, the draft BFN fire abandonment procedure, and the BFN plant itself.

4.2.5 FPRA Quantification

In general, no significant changes in the BFN FPRA quantification process due to the EPU have been identified or implemented (i.e., the same software, linked tree approach, configuration flags, HRA dependency process are used). Any changes in the quantification results (fire scenario risk metrics) were realized primarily as a result of the changes in the human error probabilities as described above.

For the FPRA, consistent with the FPRA model of record, this study uses 1E-12 as the truncation threshold for estimating CDF and 1E-13 as a truncation threshold for estimating LERF.

The minimum joint HEP values used in the FPRA model of record were retained for consistency in the comparison of results. Joint probability floors are established for the combinations. Normally this floor is 1E-5; however, using a 1E-5 floor for all HFE combinations does not give proper credit to long term decay heat removal (DHR) HFE dependencies in the Level 1 model. Combinations containing those long term DHR HFEs were assigned a 1E-6 floor for the following reasons. Long term DHR HFEs are those actions associated with using the containment vent to remove decay heat including post vent injection HFEs. Also included are HFEs associated with establishing late suppression pool cooling for DHR. These HFEs occur approximately 10 hours to 12 hours after the reactor scram, whereas most of the other HFEs occur within approximately six hours of the reactor scram. By the time these long term DHR actions are needed, a shift turnover would have occurred and an emergency response organization would have been implemented. A low dependency exists between these long term DHR HFEs and the earlier actions.

4.3 EPU Impact to Internal Events PRA Summary

Level 1 PRA

Sections 4.1.2 through 4.1.9 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 these sections.

Loss of Inventory Makeup Transients

In general, 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, LPCI, CS, and condensate 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 BFN power (3458 MWt) for events initiated from full power. However, CRD is considered successful in the BFN PRA for late RPV injection given initial RPV injection from another source.
- RHR Service Water (RHRSW) 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.
- In the MOR Rev. 6 PRA for the number of SRVs required to open to assure RPV emergency depressurization is two (2). This criterion is applicable both at CLTP and EPU conditions. Timing differences associated with operator actions to depressurize have been factored into the HEP analysis (Table 4-4)

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

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 BFN configuration at CLTP (assuming similar failures).

The number of SRVs that must fail to open during an isolation ATWS in order to overpressurize the RPV is three (3) for the EPU case (i.e., 11 of 13 SRV must open). This is consistent with the CLTP model such that no change to the common cause failure contribution for this event is required for use in the BFN 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 4-4 for changes in ATWS related HEPs. Appendix B provides a listing of timing associated with time critical operator actions. These ATWS HEP changes have an impact to the existing risk profile associated with ATWS accidents result.

It should be noted that failure to trip the recirculation pumps, or to inject with SLC is always assumed to lead to core damage. The BFN models for both CLTP and EPU require both recirculation pumps to trip, and one SLC pump to inject boron to prevent core damage.

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 to ensure NPSH is satisfied for the pumps taking suction from the torus is not credited in the PRA. 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 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.

Station Blackout

Station Blackout (SBO) 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. 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 the success criteria for the key systems (Residual Heat Removal (RHR), main condenser, and torus hard-piped vent) in the loss of containment heat removal accident sequences are not affected.

The time available to initiate containment heat removal is approximately 16 hours in the PRA (Reference 38b). 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.

4.4 EPU Impact to FPRA

The impact of EPU to the FPRA is driven by HEP changes. All human failure events (HFEs) in the FPRA model were reviewed and considered for modification for the FPRA EPU impact study as discussed in Section 4.2.4.

Table 4-11 summarizes the HEPs that changed for CLTP.

Table 4-11. Estimate of EPU Impact on Fire Human Error Probabilities		
HFE Name / Description	**CLTP HEP**	**EPU HEP**
HFFA0002RPV_LVL OPERATOR FAILS TO MAINTAIN RPV LEVEL (fire)	1.87E-03	4.75E-03
HFFA0268480CRSTIE FAILURE TO TRANSFER DEENERGIZED 480V BOARD TO ALTERNATE SUPPLY (FIRE)	1.07E-02	1.28E-02
HFFA0ASD_RCIC OPERATOR FAILS TO START RCIC	2.99E-02	3.45E-02
HFFA0SUPPHPI2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN	1.10E-03	2.92E-03
HFFA0RHRCS_LPP OPER FAILS TO BYPASS ECCS LOW PRESSURE PERMISSIVE	5.80E-03	1.94E-02
HFFA0LPCIINJAUTO OPERATOR FAILS TO MANUALLY OPEN LPCI INJECTION VALVE	2.82E-03	9.54E-03
HFFA0LPIINIT30 FAILURE TO ESTABLISH LOW-PRESSURE INJECTION GIVEN LOSS OF HIGH PRESSURE INJECTION	1.41E-03	3.62E-03
HFFA0071CTLPOWER OPERATOR FAILS TO TRANSFER TO BACKUP POWER (FIRE)	9.24E-03	2.81E-02

An HRA dependency analysis is an integral part of all HRA evaluations. Multiple operator actions found in a single cutset are of special interest to PRA results, because they may require consideration to avoid nonconservative treatment of the operator actions. Dependencies between operator actions are treated during the quantification post-processing stage where specific sets of multiple operator actions are recovered by inserting into the affected cutsets the ratio between the calculated dependent joint probability and the independent joint probability as a multiplier.

The control-room abandonment and nonabandonment models are treated separately because the fault trees themselves are distinct. The HFE combinations are selected from cutset files obtained by quantifying the models with the fire HEPs set to a value of 0.1. In order to remain within the capability of the available HRA tools and techniques, the number of cutsets retained is limited to less than 500,000 for all three units. The process uses the cutset probabilities as a basis for retaining the most significant cutsets and HFE combinations. This is done as follows:

- When compiling the HFE combination cutsets, they are sorted by probability.
- The total number of cutsets retained is limited to 500,000 by adjusting the truncation limit. The top 500,000 cutsets are kept. A consistent truncation limit across the analysis is applied.
- The HRA calculator is used to extract the unique combinations from the retained cutsets and compute the dependency multipliers. This process generates recovery rule files to be used in the final model quantifications.
- User defined dependencies are applied to the HRA calculator model as required. These user defined dependencies are defined in the HRA calculator DAF file.

The HRA calculator automatically assigns dependencies between HFE pairs for each combination. These dependencies and the order of the HFEs established for each combination are based on the value of the parameter t_{delay}. The HRA calculator orders the HFEs for each combination according to the HFE's t_{delay} value for all nonzero instances of t_{delay}. It also assigns dependencies between HFEs based on the t_{delay} time between the actions. This process does not always work well with t_{delay} values normally assigned to HFEs. In order to establish the correct order of HFEs and the correct dependencies between HFEs, dummy t_{delay} values were assigned to each HFE to ensure its correct ordering and dependencies for each combination.

Additional information on the procedure for determining and implementing the HRA dependencies is provided in *TVA FIRE PRA –Task 7.12 Post-Fire Human Reliability Analysis* (Reference 55). The recovery rules that result from the dependency analysis are contained in the rule files associated with each quantification. The dependency analysis is used to support the comparison of the two plant conditions, CLTP and EPU. To ensure a proper comparison, the same HRA Calculator file is used to evaluate both conditions for a particular case. An HRA Calculator file used for a particular condition does, however, have timing parameters unique to that condition. The cases evaluated are:

- Abandonment CDF
- Abandonment LERF
- Nonabandonment CDF
- Nonabandonment LERF

All of the above use a different fault tree for quantification.

The nonabandonment models were treated in the same manner as delineated in the BFN Fire HRA Calculation (Ref. 55). The user-defined dependency tool in the HRA Calculator is used to fine tune the dependency analysis and reduce dependency conservatisms that affect significant cutsets. Due to the large number of combinations (six to seven thousand in the nonabandonment models), the fine tuning is limited to the top cutsets.

The abandonment models are treated in a different manner. The main control room fire abandonment process, in its current state of development, uses a single master procedure for all three units. This procedure specifies the actions required of all of the unit shift supervisors and unit operators in a series of either sections in the main procedure or attached subordinate procedures. Each attachment is given to an individual plant operator to perform. Because of this tightly controlled process, there is a great deal of independence between modeled actions. An individual human failure event (HFE) may be defined to include, for example, performance of several steps in the procedure attachment that isolates a pump's breaker at the board and then places the breaker in the desired position. Another operator may meanwhile perform a similar task at another board for another piece of equipment. The two HFEs representing these actions would have a very low dependency. The T_{delay} parameter in the HRA calculator was used to establish, at least to a practical extent, the ordering of the actions. The dependency levels were then set, using the user defined dependency tool, by an analyst with knowledge of the BFN FPRA, the BFN fire HRA, the BFN fire abandonment model, the draft BFN fire abandonment procedure, and the BFN plant itself.

4.4.1 Equipment Reliability Changes for EPU

The probabilities of stuck open relief valves (SORVs) may be greater under EPU conditions due to increased frequency of cycling. The random probabilities of stuck open relief valves are the same in the FPRA as for internal events and are discussed in Section 4.1.2.6.

4.4.2 FPRA Logic Changes

No FPRA logic changes are required to model EPU conditions for the baseline case. Logic changes were required for one of the sensitivity studies as discussed in Section 5.7.2.3.

4.5 EPU Impact to Seismic PRA

The frequency of earthquakes is not dependent on reactor power or operation; therefore no impact on the seismic initiating event frequency is expected.

The BFN seismic risk analysis was performed as part of the IPEEE. TVA performed a SMA following the guidance of EPRI NP-6041 (Reference 24). Since the SMA is a deterministic evaluation process no core damage frequencies were quantified as part of the seismic risk evaluation. The SMA was not re-performed as part of this assessment.

The Unit 2 and 3 seismic IPEEE submittal identified items for seismic margin improvement. The Unit 1 seismic IPEEE submittal identified outliers associated with the BFN Unit 1 USI A-46 screening evaluations. These identified vulnerabilities and outliers have all been addressed and corrected through the corrective action program. These changes further 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 proposed EPU modifications have been reviewed qualitatively and no potential seismic impact was identified. Therefore, it is judged that the SMA adequately reflects the as-built, as-operated plant.

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-1150 study (Reference 26)) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures. Dominant seismic initiators included LOSP and LOCA scenarios. Operator actions associated with these initiators are not significantly impacted by EPU.

Based on the above discussion, the increase in the BFN seismic risk due to the EPU is much less than that calculated for internal events. An estimate of the seismic risk is provided in Section 5.4 of this Attachment as well as Section 7.5 and Attachment E of Reference 26.

4.6 EPU Impact to Shutdown Risk

Shutdown Risk Management

The impact of EPU on shutdown risk is similar to that of the at-power PRA analysis. Based on insights of the at-power Level 1 PRA impact assessment, the following areas were reviewed for shutdown risk:

- 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 (Mode 3) 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.

Outage Risk Management

BFN does not have a quantitative shutdown risk model. BFN utilizes shutdown risk management guidelines and the Equipment out of Service (EOOS) computer program (Reference 51), which qualitatively performs risk assessment, as tools for controlling outage risk. This software takes the status of key plant equipment and produces an output of the relative level of safety/ defense-in-depth of key shutdown functions from Section 4 of NUMARC 91-06 (Reference 47):

- Decay Heat Removal Capability
- Inventory Control

- Power Availability
- Reactivity Control
- Containment (primary/secondary)

The models which are built within EOOS include fault trees to identify specific components utilized in maintaining a key safety function. The fault trees are used to determine the number of required systems/components that are required to get a predetermined output for a given plant operational state/condition. EOOS can be utilized to determine optimum places to schedule specific work activities. It is primarily a tool for outage scheduling and management personnel to help plan and maintain defense-in-depth in an outage. It does not take the place of technical specifications, procedures, or the schedule in maintaining defense-in-depth. During outage execution, a defense-in-depth checklist must be filled out each shift by operations and be provided to the shift outage manager for review.

The process by which outage defense-in-depth risk is assessed will not be impacted by EPU. Procedural controls are in place at BFN to ensure the risk impacts of EPU on shutdown operations are not significant. Shutdown Risk Management at BFN is described in procedure NPG-SPP-07.2.11, "Shutdown Risk Management" (Reference 49) and NPG-SPP-07.2, "Outage Management" (Reference 50). These procedures specify how outage risk is assessed in order to ensure that the assessments reflect the as-built/as-operated plant. The guidelines of NUMARC 91-06 are implemented to assure nuclear risk is assessed and that structures, systems, and components that perform key safety functions are available when needed. A defense-in-depth strategy is implemented to enforce minimum equipment unavailability for maintaining the key shutdown functions. The goal of this strategy is to maintain at least the minimum systems and equipment required by technical specifications plus an additional component or system for each critical safety function. Procedures cover outage management, level of activities, defense-in-depth, contingency planning, training, safety review, and effective communications.

Procedure NPG-SPP-09.11.3 “Shutdown Equipment Out of Service Management” (Reference 51) specifies the process by which the shutdown defense-in-depth software models are implemented and revised such that they reflect the as-built, as-operated plant configuration. The process ensures that equipment and procedure changes identified by the site are incorporated into the software logic. The software is controlled under the TVA software control program (Reference 52). Furthermore, this procedure provides guidance for using the EOOS software to represent a given shutdown configuration to assess risk.

<u>Shutdown Initiating Events</u>

Shutdown initiating events include the following major categories:

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

No new initiating events or increased potential for existing initiating events during shutdown were identified due to the EPU.

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 at CLTP is 9.9 hours (8.8 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. This information is taken from Reference 27, Attachment F, Tables 3 and 4.

The increased power level decreases the time for boil down, which decreases the time available for operator actions to recover. The higher decay heat level delays the time at which lower capacity decay heat removal systems may be used as alternatives to shutdown cooling; however the highest shutdown risk is early in the outage when decay heat levels are high and lower-capacity decay heat removal systems are not sufficient to provide adequate core cooling, so the overall impact of this effect is small.

Other success criteria are marginally impacted by the EPU. The EPU has a minor impact on RPV inventory makeup during loss of decay heat removal scenarios during shutdown operations. This is 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 plant operational states with the RPV intact) is also lower because of the relatively low decay heat level such that the margins for SPC 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 relatively low decay heat level during shutdown.

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 Attachment F, Tables 2 through 4 of Reference 27, the reduction in times to core damage (i.e., CLTP case compared to EPU case) is on the order of 11-13%. Such small changes in already lengthy allowable operator response times result in negligible changes in calculated human error probabilities.

The allowable operator action times to respond to loss of decay heat removal scenarios during shutdown operations are greater than 5 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 6 hours to 5.3 hours (refer to "1 Day After Shutdown" case in Attachment F, Table 2 of Reference 27 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 (when the boil down times are much longer) would result in indiscernible changes in HEPs using current human reliability analysis methods.

In addition to traditional human error probabilities, the offsite AC recovery failure probabilities can be influenced by changes in allowable timings. A calculation is performed in Attachment F of Reference 27 to estimate the impact on shutdown risk due to changes in the offsite AC recovery failure probability.

Shutdown Risk Summary

Based on a review of the potential impacts on shutdown initiating events, success criteria, and HRA, the EPU is assessed to have a non-significant impact (delta CDF of approximately 1% percent). This assessment is consistent with CLTR conclusions on this issue (Reference 3), “The shutdown risks for BWR plants are generally low and the impact of CPPU (Reference constant pressure power uprate) on the CDF and LERF during shutdown is expected to be negligible.” See Attachment F of Reference 27 on shutdown risk.

4.7 EPU Impact to Other External Events Risk

In addition to internal fires and seismic events, the BFN IPEEE submittal analyzed the following external hazards:

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

The frequency of other external events is not dependent on reactor power or operation, therefore no impact on initiating event frequencies are expected.

The BFN IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, and nearby facility accidents was accomplished by reviewing the plant environs against regulatory requirements (1975 Standard Review Plan criteria, Reference 25) regarding these hazards. The screening approach used in analysis of external floods, and nearby facilities/transportation accidents demonstrates that they meet NRC SRP 1975 criteria and have adequate defense against these threats. Since BFN does not meet the SRP 1975 criteria for high winds, a bounding analysis was performed. This analysis showed the contribution to core damage frequency due to high winds to be less than the IPEEE screening criteria of 1E-6 (Reference 19)

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 BFN other external events risk due to the EPU is much less than that calculated for internal events. The proposed EPU modifications have been reviewed qualitatively and no potential vulnerabilities related to “other” external events risk were identified.

4.8 Large Early Release Frequency (Level 2 PRA)

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. Changes in plant response due to EPU represent relatively small changes to the overall challenge to containment under severe accident conditions.

No new Level 2 MAAP runs were performed for the EPU evaluation. The Level 2 MAAP runs performed for CLTP as part of the PRA for determining accident scenario timings were deemed adequate for evaluating EPU Level 2 timings. This is considered appropriate because:

- The PRA MOR success criteria, accident progression and Thermal Hydraulic Analysis is conservatively based on 3952 MWt (EPU).
- A large release is defined as an environmental release of CsI exceeding 10% of the initial core inventory (Reference 46).
- The release magnitudes are not based on absolute terms, and BFN only evaluates LERF.

Also, both EPU and CLTP were run using MAAP version 4.07. 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

Level 1 Input

The LERF analysis interfaces with the Level 1 accident sequence analysis through the appropriate definition of a set of core damage functional classes. These states are the endpoints of the sequences in the Level 1 portion of the event trees and the initiating events for the CETs. The end products of the LERF analysis include a set of release categories, which define the radionuclide releases into the environment, and a quantification of the frequency of each release category. The LERF analysis is also quantified for EPU as part of this 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 changes to the ultimate containment strength. No Level 2 safety function success criteria (e.g., coolant flowrate 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 BFN 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 BFN PRA employs a fully integrated Level 1 transfer to the Level 2 PRA model, changes to HEP values (refer to Section 6.9) 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 (See 4.1.9). Therefore, no changes to Level 2 modeling with respect to success criteria are made as part of this analysis.

Containment Capability

The containment structural failure treatment has incorporated the detailed insights and analysis from the BFN specific drywell assessment with additional insights from a Peach Bottom analysis. This has been coupled with the known differences in design, documented in the BFN Structural Analysis Notebook (Reference 42), to obtain an updated structural assessment of the containment capability under severe accident conditions. The primary containment ultimate structural integrity is important in severe accident analysis due to its key role as a fission product barrier. The BFN Mark I containment has been analyzed to predict its ability to withstand severe accident conditions, i.e., pressures and temperatures imposed on containment prior to, during, and following core melt progression accidents. The increase in decay heat levels upon the implementation of the EPU will result in slight reduction in the keys timings, more precisely the time to reach loading challenges; however, the time frames are relatively long and the accident timing reduction is small due to the EPU and will have a small impact on the 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 BFN Level 2 PRA.

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

The BFN release magnitude classification is based on the percentage (as a function of the initial EOC radionuclide inventory in the core) of CsI released to the environment (Reference 46); 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 BFN Level 2 PRA are not necessary; the MOR MAAP analyses is based on 3952 MWt.

Level 2 Impact Summary

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

5.0 Conclusions

The PRA models were used to assess the impact to CDF and LERF from implementation of EPU. The list of modifications in LAR Attachment 47 were reviewed to determine the impact to the key PRA elements. In addition, MAAP sensitivities were performed to address impact on timing that could affect time critical operator actions. Based on these reviews, the primary impacts to the internal events were related to timing available to perform operator actions. The shorter delay times and system windows for operator actions impact the time available to recover by the operating crew and effects the cognitive human error probability. This analysis evaluated time critical operator actions to determine changes to the overall human error probabilities. In addition, it is postulated that higher probability of SRV failure to close after it is demanded occurs due to EPU conditions. The increase in the likelihood of SORV is reasonable because the larger decay heat associated with EPU may require an increase number of demands to maintain and/or reduce reactor pressure during reactor cooldown after events that cause a plant trip.

Based on the results of this analysis, it is determined that major contributor to total CDF risk (approximately 80%) for both CLTP and EPU come from fire scenarios. The major contributor to the change in CDF (approximately 82-91%) for both CLTP and EPU also comes from fire scenarios.

Based on the results of this analysis, it is determined that the major contributor to total LERF risk (approximately 75%) for both CLTP and EPU comes from fire scenarios. The major contributor to the change in LERF (approximately 75%) for both CLTP and EPU also comes from fire scenarios.

The risks from internal events, external events and fires must be added to arrive at the aggregated risk for EPU conditions.

Sections 5.1 through 5.5 summarize 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

Section 5.6 provides the total risks associated with EPU. The aggregated risks are shown relative to the acceptance guidelines given in Regulatory Guide 1.174 (Reference 1). As can be seen in Table 5-15, the total risks versus corresponding change in risks are within Region II of Regulatory Guide 1.174, and is characterized as a small change.

Section 5.7 provides a sensitivity analysis and discussion of uncertainties.

5.1 Internal Event Results

5.1.1 Core Damage and Large Early Release Frequencies

A summary of the internal events PRA changes in core damage frequency, and changes in large early release frequency for each unit is shown below in Table 5-1.

Table 5-1. Internal Events PRA Change in Risk

Unit	CDF (CLTP)	CDF (EPU)	ΔCDF	LERF (CLTP)	LERF (EPU)	ΔLERF
1	6.63E-06	7.14E-06	5.10E-07	1.44E-06	1.61E-06	1.70E-07
2	5.96E-06	6.47E-06	5.10E-07	1.39E-06	1.55E-06	1.60E-07
3	6.57E-06	7.06E-06	4.90E-07	1.38E-06	1.54E-06	1.60E-07

5.1.2 Dominant Initiators

Table 5-2 through Table 5-7 provide a comparison of BFN CLTP CDF (and LERF) versus EPU CDF (and LERF) for BFN Units 1, 2, and 3. As expected, the general transients are the most significant contributors to the EPU increase in CDF and LERF.

Table 5-2. Comparison of BFN CLTP CDF vs. EPU CDF by Initiator (Unit 1)					
Initiator	Description	U1 CDF (CLTP)	U1 CDF (EPU)	% Increase by Initiator	Relative % of CDF
%1LCV	LOSS OF CONDENSER VACUUM	8.65E-07	9.70E-07	12.1%	1.6%
%1TLFW	TOTAL LOSS OF FEEDWATER	7.87E-07	8.83E-07	12.2%	1.4%
%1IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	6.06E-07	6.80E-07	12.2%	1.1%
%1EXFW	EXCESSIVE FEEDWATER FLOW	2.80E-07	3.33E-07	18.9%	0.8%
%1INTAKE	INTAKE PLUGGING INITIATOR	1.99E-07	2.18E-07	9.5%	0.3%
%1SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	3.92E-07	4.10E-07	4.6%	0.3%
%1TBU	TURBINE BYPASS UNAVAILABLE	1.27E-07	1.43E-07	12.6%	0.2%
%1LSBA	480V SHUTDOWN BOARD A INITIATING EVENT	5.31E-07	5.46E-07	2.8%	0.2%
%1LRMOVA	480V RMOV BOARD A INITIATING EVENT	2.55E-07	2.70E-07	5.9%	0.2%
%1LRMOVB	480V RMOV BOARD B INITIATING EVENT	2.68E-07	2.80E-07	4.5%	0.2%
%1LSBB	480V SHUTDOWN BOARD B INITIATING EVENT	2.11E-07	2.23E-07	5.7%	0.2%
%1IOOV	INADVERTANT OPEN RELIEF VALVE	1.28E-07	1.39E-07	8.6%	0.2%
%1MSBOC	Unit 1 - Main Steam Break Outside Containment	2.72E-07	2.82E-07	3.7%	0.2%
	All Other Initiators	1.66E-06	1.70E-06	2.5%	0.6%

1. The relative % of CDF for each initiator (i) is calculated as follows:100% × (CDF_EPUi - CDF_CLTPi)/CDF_CLTP(U1) where:

CDF_EPUi = CDF for that initiator under EPU conditions

CDF_CLTPi = CDF for that initiator under CLTP conditions

CDF_CLTP(U1) = total Unit 1 CDF (Table 5-1)

Table 5-3. Comparison of BFN CLTP CDF vs. EPU CDF by Initiator (Unit 2)					
Initiator	**Description**	**U2 CDF (CLTP)**	**U2 CDF (EPU)**	**% Increase by Initiator**	**Relative % of CDF1**
%2LCV	LOSS OF CONDENSER VACUUM	8.61E-07	9.67E-07	12.3%	1.8%
%2TLFW	TOTAL LOSS OF FEEDWATER	7.84E-07	8.80E-07	12.2%	1.6%
%2IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	6.03E-07	6.78E-07	12.4%	1.3%
%2EXFW	EXCESSIVE FEEDWATER FLOW	2.79E-07	3.33E-07	19.4%	0.9%
%2INTAKE	INTAKE PLUGGING INITIATOR	1.97E-07	2.16E-07	9.6%	0.3%
%2SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	3.56E-07	3.74E-07	5.1%	0.3%
%2LSBA	480V SHUTDOWN BOARD 2A INITIATING EVENT	4.37E-07	4.52E-07	3.4%	0.3%
%2TBU	TURBINE BYPASS UNAVAILABLE	1.27E-07	1.42E-07	11.8%	0.3%
%2LRMOVA	480V RMOV BOARD 2A INITIATING EVENT	1.77E-07	1.90E-07	7.3%	0.2%
%2LSBB	480V SHUTDOWN BOARD 2B INITIATING EVENT	2.32E-07	2.44E-07	5.2%	0.2%
%2MSBOC	Unit 2 - Main Steam Break Outside Containment	2.57E-07	2.68E-07	4.3%	0.2%
%2IOOV	INADVERTANT OPEN RELIEF VALVE	1.10E-07	1.21E-07	10.0%	0.2%
%2LRMOVB	480V RMOV BOARD 2B INITIATING EVENT	2.03E-07	2.13E-07	4.9%	0.2%
	All Other Initiators	1.29E-06	1.33E-06	3.6%	0.8%

1. The relative % of CDF for each initiator (i) is calculated as follows:100% × (CDF_EPUi - CDF_CLTPi)/CDF_CLTP(U2) where:

CDF_EPUi = CDF for that initiator under EPU conditions

CDF_CLTPi = CDF for that initiator under CLTP conditions

CDF_CLTP(U2) = total Unit 2 CDF (Table 5-1)

Table 5-4. Comparison of BFN CLTP CDF vs. EPU CDF by Initiator (Unit 3)					
Initiator	**Description**	**U3 CDF (CLTP)**	**U3 CDF (EPU)**	**% Increase by Initiator**	**Relative % of CDF**
%3LCV	LOSS OF CONDENSER VACUUM	7.33E-07	8.35E-07	13.9%	1.6%
%3TLFW	TOTAL LOSS OF FEEDWATER	6.68E-07	7.61E-07	13.9%	1.4%
%3IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	5.14E-07	5.85E-07	13.8%	1.1%
%3EXFW	EXCESSIVE FEEDWATER FLOW	2.46E-07	2.97E-07	20.7%	0.8%
%3LSBA	480V SHUTDOWN BOARD 3A INITIATING EVENT	6.36E-07	6.56E-07	3.1%	0.3%
%3INTAKE	INTAKE PLUGGING INITIATOR	1.97E-07	2.16E-07	9.6%	0.3%
%3SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	2.99E-07	3.17E-07	6.0%	0.3%
%3TBU	TURBINE BYPASS UNAVAILABLE	1.08E-07	1.23E-07	13.9%	0.2%
%3LRMOVA	480V RMOV BOARD 3A INITIATING EVENT	2.81E-07	2.94E-07	4.6%	0.2%
%3IOOV	INADVERTANT OPEN RELIEF VALVE	1.04E-07	1.14E-07	9.6%	0.2%
%3LRMOVB	480V RMOV BOARD 3B INITIATING EVENT	2.00E-07	2.09E-07	4.5%	0.1%
%3LSBB	480V SHUTDOWN BOARD 3B INITIATING EVENT	1.27E-07	1.36E-07	7.1%	0.1%
%3MSBOC	Unit 3 - Main Steam Break Outside Containment	2.48E-07	2.57E-07	3.6%	0.1%
	All Other Initiators	2.16E-06	2.20E-06	2.0%	0.7%

1. The relative % of CDF for each initiator (i) is calculated as follows:100% × (CDF_EPUi - CDF_CLTPi)/CDF_CLTP(U3) where:

CDF_EPUi = CDF for that initiator under EPU conditions

CDF_CLTPi = CDF for that initiator under CLTP conditions

CDF_CLTP(U3) = total Unit 3 CDF (Table 5-1)

Table 5-5. Comparison of BFN CLTP LERF vs. EPU LERF by Initiator (Unit 1)					
Initiator	**Description**	U1 LERF (CLTP)	U1 LERF (EPU)	% Increase by Initiator	Relative % of LERF
%1LCV	LOSS OF CONDENSER VACUUM	1.92E-07	2.27E-07	18.2%	2.4%
%1TLFW	TOTAL LOSS OF FEEDWATER	1.75E-07	2.07E-07	18.3%	2.2%
%1IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	1.35E-07	1.59E-07	17.8%	1.7%
%1EXFW	EXCESSIVE FEEDWATER FLOW	7.85E-08	9.89E-08	26.0%	1.4%
%1SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	7.01E-08	7.73E-08	10.3%	0.5%
%1INTAKE	INTAKE PLUGGING INITIATOR	2.70E-08	3.25E-08	20.4%	0.4%
%1TBU	TURBINE BYPASS UNAVAILABLE	2.80E-08	3.32E-08	18.6%	0.4%
%1MSBOC	Unit 1 - Main Steam Break Outside Containment	1.49E-07	1.53E-07	2.7%	0.3%
%1LSBA	480V SHUTDOWN BOARD A INITIATING EVENT	8.05E-08	8.44E-08	4.8%	0.3%
%1IOOV	INADVERTANT OPEN RELIEF VALVE	1.74E-08	2.11E-08	21.3%	0.3%
%1LRMOVA	480V RMOV BOARD A INITIATING EVENT	5.03E-08	5.40E-08	7.4%	0.3%
%1LRMOVB	480V RMOV BOARD B INITIATING EVENT	4.27E-08	4.62E-08	8.2%	0.2%
%1LSBB	480V SHUTDOWN BOARD B INITIATING EVENT	3.09E-08	3.42E-08	10.7%	0.2%
	All Other Initiators	3.48E-07	3.64E-07	4.7%	1.1%

1. The relative % of LERF for each initiator (i) is calculated as follows:100% × (LERF_EPUi - LERF_CLTPi)/LERF_CLTP(U1) where:

LERF_EPUi = LERF for that initiator under EPU conditions

LERF_CLTPi = LERF for that initiator under CLTP conditions

LERF_CLTP(U1) = total Unit 1 LERF (Table 5-1)

Table 5-6. Comparison of BFN CLTP LERF vs. EPU LERF by Initiator (Unit 2)					
Initiator	**Description**	**U2 LERF (CLTP)**	**U2 LERF (EPU)**	**% Increase by Initiator**	**Relative % of LERF**
%2LCV	LOSS OF CONDENSER VACUUM	1.93E-07	2.27E-07	17.6%	2.4%
%2TLFW	TOTAL LOSS OF FEEDWATER	1.76E-07	2.06E-07	17.0%	2.2%
%2IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	1.35E-07	1.59E-07	17.8%	1.7%
%2EXFW	EXCESSIVE FEEDWATER FLOW	7.87E-08	9.86E-08	25.3%	1.4%
%2SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	6.46E-08	7.12E-08	10.2%	0.5%
%2INTAKE	INTAKE PLUGGING INITIATOR	2.7E-08	3.23E-08	19.6%	0.4%
%2TBU	TURBINE BYPASS UNAVAILABLE	2.81E-08	3.31E-08	17.8%	0.4%
%2IOOV	INADVERTANT OPEN RELIEF VALVE	1.66E-08	2.02E-08	21.7%	0.3%
%2LRMOVA	480V RMOV BOARD 2A INITIATING EVENT	3.67E-08	3.99E-08	8.7%	0.2%
%2LRMOVB	480V RMOV BOARD 2B INITIATING EVENT	3.92E-08	4.23E-08	7.9%	0.2%
%2LSBB	480V SHUTDOWN BOARD 2B INITIATING EVENT	3.01E-08	3.32E-08	10.3%	0.2%
%2LSBA	480V SHUTDOWN BOARD 2A INITIATING EVENT	6.96E-08	7.27E-08	4.5%	0.2%
%2MSBOC	Unit 2 - Main Steam Break Outside Containment	1.48E-07	1.51E-07	2.0%	0.2%
	All Other Initiators	3.32E-07	3.45E-07	4.1%	1.0%

1. The relative % of LERF for each initiator (i) is calculated as follows:100% × (LERF_EPUi - LERF_CLTPi)/LERF_CLTP(U2) where:

LERF_EPUi = LERF for that initiator under EPU conditions

LERF_CLTPi = LERF for that initiator under CLTP conditions

LERF_CLTP(U2) = total Unit 2 LERF (Table 5-1)

Table 5-7. Comparison of BFN CLTP LERF vs. EPU LERF by Initiator (Unit 3)					
Initiator	**Description**	**U3 LERF (CLTP)**	**U3 LERF (EPU)**	**% Increase by Initiator**	**Relative % of LERF**
%3LCV	LOSS OF CONDENSER VACUUM	1.73E-07	2.06E-07	19.1%	2.4%
%3TLFW	TOTAL LOSS OF FEEDWATER	1.57E-07	1.88E-07	19.7%	2.2%
%3IMSIV	INDVERTANT MSIV CLOSURE INITIATING EVENT	1.21E-07	1.44E-07	19.0%	1.7%
%3EXFW	EXCESSIVE FEEDWATER FLOW	7.53E-08	9.52E-08	26.4%	1.4%
%3SCRAM	MANUAL SHUTDOWN, MANUAL SCRAM, INADVERTENT SCRAM	5.57E-08	6.25E-08	12.2%	0.5%
%3INTAKE	INTAKE PLUGGING INITIATOR	2.71E-08	3.24E-08	19.6%	0.4%
%3TBU	TURBINE BYPASS UNAVAILABLE	2.52E-08	3.02E-08	19.8%	0.4%
%3LSBA	480V SHUTDOWN BOARD 3A INITIATING EVENT	7.52E-08	7.88E-08	4.8%	0.3%
%3IOOV	INADVERTANT OPEN RELIEF VALVE	1.49E-08	1.84E-08	23.5%	0.3%
%3LRMOVA	480V RMOV BOARD 3A INITIATING EVENT	3.65E-08	3.96E-08	8.5%	0.2%
%3MSBOC	Unit 3 - Main Steam Break Outside Containment	1.50E-07	1.53E-07	2.0%	0.2%
%3LSBB	480V SHUTDOWN BOARD 3B INITIATING EVENT	2.36E-08	2.65E-08	12.3%	0.2%
%3L480UBA	480V UNIT BOARD 3A INITIATING EVENT	1.40E-08	1.68E-08	20.0%	0.2%
	All Other Initiators	4.20E-07	4.34E-07	3.2%	1.0%

1. The relative % of LERF for each initiator (i) is calculated as follows:100% × (LERF_EPUi - LERF_CLTPi)/LERF_CLTP(U3) where:

LERF_EPUi = LERF for that initiator under EPU conditions

LERF_CLTPi = LERF for that initiator under CLTP conditions

LERF_CLTP(U3) = total Unit 3 LERF (Table 5-1)

5.1.3 Dominant Accident Sequences

The Internals Events PRA results were reviewed to determine the more significant accident sequence contributors to CDF and LERF as a result of EPU. The dominant sequences are discussed below.

For Unit 1, the following sequences contribute more than 95% of the total change in CDF, and more than 1% of the total change individually.

1. GTRAN-012 - 79.3%
2. GTRAN-011 - 6.50%
3. GTRAN-006 - 4.00%
4. GTRAN-008 - 2.77%
5. IOOV-007 - 2.65%
6. GTRAN-005 - 1.27%

For Unit 2, the following sequences contribute more than 95% of the total change in CDF, and more than 1% of the total change individually.

1. GTRAN-012 - 79.6%
2. GTRAN-011 - 6.50%
3. GTRAN-006 - 4.19%
4. GTRAN-008 - 2.98%
5. IOOV-007 - 2.66%

For Unit 3, the following sequences contribute more than 95% of the total change in CDF, and more than 1% of the total change individually.

1. GTRAN-012 - 82.6%
2. GTRAN-011 - 6.72%
3. GTRAN-008 - 3.20%
4. IOOV-007 - 2.76%

For Unit 1, the following sequences contribute more than 95% of the total change in LERF, and more than 1% of the total change individually.

1. GTRAN-012 - 83.7%
2. GTRAN-011 - 9.43%
3. IOOV-007 - 2.73%
4. ATWS-025 - 1.51%

For Unit 2, the following sequences contribute more than 95% of the total change in LERF, and more than 1% of the total change individually.

1. GTRAN-012 - 84.1%
2. GTRAN-011 - 9.23%

3. IOOV-007 - 2.75%
4. ATWS-025 - 1.52%

For Unit 3, the following sequences contribute more than 95% of the total change in LERF, and more than 1% of the total change individually.

1. GTRAN-012 - 84.3%
2. GTRAN-011 - 9.24%
3. IOOV-007 - 2.74%
4. ATWS-025 - 1.52%

Descriptions of each of the dominant internal events sequences are provided below:

<u>ATWS-025</u>

This sequence includes all initiators that can lead to an Anticipated Transient Without Scram. After the failure of the control rods to insert, the recirculation pumps are successfully tripped. The reactor vessel overpressurization is initially prevented by the opening of at least 11 of the 13 safety relief valves (SRVs). The operator action of manually inhibiting ADS to prevent unwanted RPV depressurization is failed. With the RPV depressurized, an uncontrolled rate of makeup to the vessel with cold un-borated water occurs. The RPV fails due to overpressurization.

<u>GTRAN-005</u>

The scram successfully occurs due to a transient, the Power Conversion System (PCS) fails, and there are no breaks outside containment or stuck open relief valves. HPCI or RCIC is successful for at least 4 hours. Early SPC is not successful or initiated in time to prevent exceeding 190°F in the suppression pool. Therefore, long term HPCI or RCIC is not successful. The CRD fails to provide injection and manual depressurization is challenged and initiated successfully with 2 SRVs about 4 hours after the scram. After depressurization, low pressure injection by RHR in the Low Pressure Coolant Injection (LPCI) mode and Core Spray (CS) is failed. Alternate Vessel Injection from RHR Service Water is unsuccessful. Core damage is caused by loss of injection and occurs at about 1.5 hours after accident initiation at low RPV pressure.

<u>GTRAN-006</u>

The scram successfully occurs due to a transient, the PCS fails, and there are no breaks outside containment or stuck open relief valves. HPCI or RCIC is successful for at least 4 hours. Early SPC is not successful or initiated in time to prevent exceeding 190°F in the suppression pool. Therefore, long term HPCI or RCIC is not successful. The CRD fails to provide injection and manual depressurization fails. Core damage is caused by loss of injection and occurs at about 1.5 hours after accident initiation at high RPV pressure.

<u>GTRAN-008</u>

The scram successfully occurs due to a transient, the PCS fails, and there are no breaks outside containment or stuck open relief valves. Early HPCI or RCIC are unsuccessful as an initial injection source. CRD may be available but is not challenged because it lacks sufficient capacity to be used as an initial injection source. When RPV level drops to Top of Active Fuel (TAF), manual depressurization is successful. Low pressure injection by RHR in the LPCI mode or CS is successful as an initial injection source but has to be initiated within 30 minutes. ASDC is unsuccessful. Late SPC and drywell spray are unsuccessful. The HWWV and Drywell Vent are unsuccessful for the purposes of long term decay heat removal (DHR). This causes RPV pressurization due to SRV closure and failing low pressure injection. Post venting CRD is successful. CRD maintains injection after the RPV re-pressurizes and the primary containment fails before core damage.

<u>GTRAN-011</u>

The scram successfully occurs due to a transient, the PCS fails, and there are no breaks outside containment or stuck open relief valves. Early HPCI or RCIC are unsuccessful as an initial injection source. CRD may be available but is not challenged because it lacks sufficient capacity to be used as an initial injection source. When RPV level drops to TAF, manual depressurization is successful. All low pressure injection sources fail. Core damage occurs in about 30 to 40 minutes with the RPV at low pressure.

<u>GTRAN-012</u>

The scram successfully occurs due to a transient, the PCS fails, and there are no breaks outside containment or stuck open relief valves. Early HPCI or RCIC are unsuccessful as an initial injection source. CRD may be available but is not challenged because it lacks sufficient capacity to be used as an initial injection source. When RPV level drops to TAF, manual depressurization fails. Without depressurization there are no other available injection sources and core damage occurs. Core damage occurs in about 30 to 40 minutes with the RPV at high pressure.

<u>IOOV-007</u>

This sequence includes all initiators that can lead to an individual stuck open relief valve. Vapor suppression in the torus is successful and the reactor is successfully scrammed. High pressure injection fails. Emergency depressurization fails. Without emergency depressurization, the stuck open relief valve will depressurize the RPV to approximately 400 psig in 48 minutes. Low pressure injection is not available at this pressure. Core damage occurs at 50 minutes.

5.2 Fire Risk Results

The fire risk results, showing the change in CDF and LERF due to the EPU, are presented in Table 5-8.

Table 5-8. Fire Risk Results for CLTP and EPU						
	Core Damage Frequency (per reactor yr)			Large Early Release Frequency (per reactor yr)		
Unit	CLTP	EPU	ΔCDF	CLTP	EPU	ΔLERF
1	4.78E-05	4.90E-05	1.18E-06	6.05E-06	6.65E-06	6.04E-07
2	4.72E-05	4.85E-05	1.23E-06	5.96E-06	6.46E-06	5.03E-07
3	5.17E-05	5.29E-05	1.18E-06	5.16E-06	5.54E-06	3.85E-07

The top ten fire scenarios contributing to delta risk for CDF and LERF for Units 1, 2, and 3 are identified in Table 5-9 through Table 5-14. Because the model logic is the same for CLTP and EPU, the changes in risk are attributable to basic event probability changes. CLTP and EPU cutsets from the fire scenarios listed in Table 5-9 through Table 5-14 were reviewed and dominant accident sequences are described in each table. Major HFE contributors to delta risk were noted and are listed in the table footnotes. The HFEs that were found by the cutset review described to be major contributors are the following:

- HFFA0002RPV_LVL OPERATOR FAILS TO MAINTAIN RPV LEVEL (FIRE)
- HFFA0ASD_RCIC OPERATOR FAILS TO START RCIC
- HFFA0268480CRSTIE FAILURE TO TRANSFER DEENERGIZED 480V BOARD TO ALTERNATE SUPPLY (FIRE)
- HFFA0SUPPHPI2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN
- HFFA0RHRCS_LPP OPER FAILS TO BYPASS ECCS LOW PRESSURE PERMISSIVE

Table 5-9. Top Ten ΔCDF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔCDF (/ reactor year)**	**Unit 1 ΔCDF (%)**
05	05.5001-0-BDAA-211-000A	Fire originating in fixed ignition source 0-BDAA-211-000A	This fire scenario results in two or more spuriously open SRVs, with failure of low pressure injection (LPI) from either RHR or Core Spray (CS) due to fire impacts to Loop I RHR injection and minimum flow valves and fire impacts to Loop I CS pumps and valves. Loop II RHR and CS are lost due to fire impacts to electrical power including the 480V load shed logic and the operators fail to bypass the ECCS signal after trip, resulting in an overload of 4 kV Shutdown Board C. Condensate flood-up succeeds and the operators successfully start the EHPM Pump, but fail to refill the CST for long term operation resulting in a loss of injection.	1.70E-07	14.2%
16-A	16-A.Habitability	Fire causing MCR Habitability Abandonment	This fire scenario results in a loss of habitability which leads to abandonment of the MCR. This fire scenario risk is dominated by operator action failures to start and align RCIC or the EHPM Pump within the time available prior to core damage.	1.06E-07	9.0%
05	05.5000-0-BDAA-211AHEAF	Fire due to high energy arcing fault (HEAF) in fixed ignition source 0-BDAA-211-000A	This fire scenario has the same risk insights as 05.5001-0-BDAA-211-000A.	8.09E-08	6.9%
04	04.001-CAB	Fire originating in cabinet 0-BDAA-211-000B	This fire scenario results in two or more spuriously open SRVs, with failure of LPI from either RHR or CS due to fire impacts to Loop I RHR minimum flow valve and fire impacts to Loop I CS pumps. Loop II RHR is lost due to fire impacts to the Loop II RHR injection valve. Loop II CS is lost due to failure of electrical power. Condensate flood-up is failed due to a post-trip conditional loss of offsite power, which results in a total loss of injection.	7.01E-08	6.0%

Table 5-9. Top Ten ΔCDF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔCDF (/ reactor year)**	**Unit 1 ΔCDF (%)**
16-A	16-A.Habitability U3 SRV	Fire causing MCR Habitability Abandonment with Unit 3 SRV Failures	This fire scenario results in a loss of habitability which leads to the need to abandon the MCR. The fire scenario risk is dominated by operator action failures to abandon the control room or to start and align RCIC within the time available prior to core damage. After successful reactor SCRAM, MSIV and turbine bypass valve (TBV) isolation, and no spurious open SRVs, the operators may fail to correctly decide to abandon the main control room, leading to core damage. Or, if operators correctly decide to abandon the control room, early high pressure injection via RCIC and the EHPM Pumps may fail, leading to core damage.	6.27E-08	5.3%
16-K	16-K.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0033	This fire scenario results in two or more SRVs stuck open by spurious Automatic Depressurization System (ADS) actuation that is not inhibited by operators. Depressurization is initiated with the SORVs, but CS and LPCI fail as low pressure injection sources due to fire damage to the LPCI injection paths and CS Loops I and II. Standby coolant injection, condensate injection, and EHPM are unsuccessful, or condensate flood up is unsuccessful and injection is lost.	5.18E-08	4.4%
16-K	16-K.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0032	This fire scenario has the same risk insights as 16-K.024-CAB-SUP.	5.10E-08	4.3%
04	04.011-HEAF	Fire due to HEAF in fixed ignition source 0-BDAA-211-000B	This fire scenario has the same risk insights as 04.001-CAB.	3.11E-08	2.6%

Table 5-9. Top Ten ΔCDF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔCDF (/ reactor year)**	**Unit 1 ΔCDF (%)**
01-04	01-04.3000-T-1-I-1	Transient combustible fire with target set 01-04.3000-T-1-I-1	Early HPCI or RCIC and longer term HPCI are unsuccessful as an initial injection source due to one of several random equipment or human failures that include failure of operator action to initiate supplemental injection, supplemental injection pump in maintenance, failure to start of supplemental injection pump, or failure of operator action to align raw cooling water (RCW) or extend CST inventory. Manual depressurization is successful. Early low pressure injection fails due to failure of Loop I and II injection paths or pumps and failure of CS Loop I and II pumps and paths. Condensate injection fails long term due to failure of standby coolant injection due to fire-induced valve failures including spurious operations, and failures of CRD injection and hotwell level control.	2.44E-08	2.1%
01-04	01-04.4001-C	Self-ignited cable and junction box fire 01-04.4001-C	This fire scenario results in two or more spuriously open SRVs with, in one of the dominant accident sequences, operator failure to inhibit automatic depressurization. Early low pressure injection fails due to failures of CS and RHR including fire-induced failures of automatic LPI CS Loop I and Loop II. CS Loop II is unavailable due to loss of power. Condensate as a long term injection source fails due to failures of operator actions to initiate supplemental high pressure injection after condensate flood-up and to initiate low pressure injection. In several similar accident sequences, important failed operator actions include failure to locally isolate loads from shutdown board and repower from off-site power, failure to start backup raw cooling water pumps, failure to initiate alternate shutdown cooling, failure to manually open hardened wetwell vent, and failure to cross tie de-energized 4KV shutdown board to energized shutdown board.	2.08E-08	1.8%
SUM				6.65E-07	56.6%
NOTE:	Important contributing HFEs include HFFA0002RPV_LVL, HFFA0ASD_RCIC, HFFA0268480CRSTIE, and HFFA0SUPPHPI2.				

Table 5-10. Top Ten ΔLERF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔLERF (/ reactor year)**	**Unit 1 ΔLERF (%)**
05	05.5001-0-BDAA-211-000A	Fire originating in fixed source 0-BDAA-211-000A	This fire scenario results in an interfacing systems loss of coolant accident (ISLOCA) due to a combination of fire-induced and random failures of the RHR shutdown cooling suction valves. The fire event results in spuriously opening 1-FCV-074-0048 and a random failure of 1-FCV-074-0047 which results in the Interfacing Systems LOCA LERF sequence.	5.21E-08	8.6%
04	04.001-CAB	Fire originating in cabinet 0-BDAA-211-000B	The fire scenario results in a stuck open relief valve sequence. Early high pressure injection using HPCI is successful. Low pressure injection using RHR LPCI fails due to fire impacts to the Loop I RHR minimum flow valve and due to fire impacts to the Loop II injection valves. Loop I CS fails due to fire impacts to the CS pumps and valves, and Loop II is failed due to failure of electrical power. Condensate in level-control mode succeeds, but the operators fail to control vessel level, then fail to start the EHPM Pump to recover level prior to core damage. After core damage, the operators fail to depressurize the vessel and fail to start an injection source to arrest the core damage. The reactor and containment fail at high pressure, resulting in the large early release.	5.14E-08	8.5%
04	04.011-HEAF	Fire due to HEAF in fixed ignition source 0-BDAA-211-000B	This fire scenario has the same risk insights as 04.001-CAB.	2.38E-08	3.9%
16-K	16-K.TS03	Transient combustible fire TS#3	This fire scenario results in failure of early high pressure injection from HPCI and RCIC. HPCI is failed due to fire impacts to the steam supply valves, discharge valves, and turbine controls. RCIC is failed due to fire impacts to the steam supply valves and discharge valves. The EHPM Pump fails due to random failure or the operators fail to inject. Depressurization fails due to fire impacts to the SRVs, resulting in core damage. After core damage, depressurization fails due to the fire impacts to the SRVs, and injection in the drywell fails. The vessel and containment fail at high pressure, resulting in the large early release.	2.34E-08	3.9%

Table 5-10. Top Ten ΔLERF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔLERF (/ reactor year)**	**Unit 1 ΔLERF (%)**
16-K	16-K.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0033	This fire scenario results in a reactor trip with early successful high pressure injection from HPCI or RCIC. RHR suppression pool cooling is failed early due to fire impacts to Loop I RHR pumps, and Loop II fails due to overload failures of 4 kV Shutdown Boards C and D due to a spurious common accident signal, resulting in the loss of suppression pool as a suction source for HPCI and RCIC. Depressurization succeeds, but low pressure injection using RHR, CS, standby coolant injection, and condensate in level control mode fails due to fire impacts, and the EHPM Pump randomly fails, resulting in a loss of long term low pressure injection. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The vessel and containment fail at low pressure.	2.24E-08	3.7%
16-K	16-K.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0032	This fire scenario has the same risk insights as16-K.024-CAB-SUP.	2.24E-08	3.7%
05	05.5000-0-BDAA-211AHEAF	Fire due to HEAF in fixed ignition source 0-BDAA-211-000A	This fire scenario has the same risk insights as 05.5001-0-BDAA-211-000A.	2.20E-08	3.6%

Table 5-10. Top Ten ΔLERF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔLERF (/ reactor year)**	**Unit 1 ΔLERF (%)**
16-M	16-M.022-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0032	This fire scenario results in a reactor trip with early successful high pressure injection from HPCI or RCIC. RHR suppression pool cooling is failed early due to fire impacts to Loop I RHR pumps and valves, and Loop II fails due to overload failures of 4 kV Shutdown Boards C and D due to a spurious common accident signal, resulting in the loss of suppression pool as a suction source for HPCI and RCIC. Depressurization succeeds, but low pressure injection using RHR, CS, standby coolant injection, and condensate in level control mode fails due to fire impacts, and the EHPM Pump randomly fails, resulting in a loss of long term low pressure injection. After core damage, the operators can successfully depressurize but all low pressure injection sources are failed which precludes arrest of the core damage or injection into the drywell. The vessel and containment fail at low pressure, resulting in the large early release.	1.92E-08	3.2%
16-M	16-M.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0033	This fire scenario has the same risk insights as 16-M.022-CAB-SUP.	1.92E-08	3.2%

Table 5-10. Top Ten ΔLERF Scenarios for Unit 1					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 1 ΔLERF (/ reactor year)**	**Unit 1 ΔLERF (%)**
16-O	16-O.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0032	In this scenario, HPCI or RCIC is successful for at least four hours. Early suppression pool cooling is not successful or initiated in time to prevent exceeding heat capacity temperature limit (HCTL) and 190°F in the suppression pool. Therefore, long term HPCI or RCIC is not successful. Due to failure of operator action to initiate supplemental injection during the fire, EHPM systems fail, so manual depressurization is challenged and may or may not be successful. If operators fail to depressurize there are no other available injection sources and core damage occurs. In vessel recovery fails due to failure to initiate supplemental injection and failure of suppression pool cooling due to fire damage. RPV breaches at pressure and Reactor Building effectiveness fails given no depressurization. In another dominant accident sequence, manual depressurization succeeds. Low pressure injection by RHR in the LPCI mode or CS also succeeds. Alternate Shutdown Cooling is unsuccessful. Late suppression pool cooling is unsuccessful. The hardened wetwell vent (HWWV) and drywell vent (DWV) are unsuccessful in providing DHR. Injection fails due to reclosure of the SRVs and no high pressure injection source is available. CS and drywell sprays fail due to fire damage. Low pressure injection for in-vessel recovery fails. Injection into the RPV or drywell after core damage fails due to failure of CS hardware and the LPCI injection path. Reactor Building effectiveness fails given a depressurized vessel with no injection and no in-vessel recovery.	1.77E-08	2.9%
SUM				2.74E-07	45.3%
NOTE:	Important contributing HFEs include HFFA0SUPPHPI2, HFFA0002RPV_LVL, HFFA0ASD_RCIC, and HFFA0268480CRSTIE.				

Table 5-11. Top Ten ΔCDF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔCDF (/ reactor year)**	**Unit 2 ΔCDF (%)**
16-A	16-A.Habitability	Fire causing MCR Habitability Abandonment	This fire scenario results in a loss of habitability which leads to abandonment of the MCR. This fire scenario risk is dominated by operator action failures to start and align RCIC or the EHPM Pump within the time available prior to core damage.	1.06E-07	8.6%
16-A	16-A.Habitability U3 SRV	Fire causing MCR Habitability Abandonment with Unit 3 SRV Failures	This fire scenario results in a loss of habitability which leads to the need to abandon the MCR. RPV depressurization fails due to power failure to 9 of 10 SRVs associated with fire-induced failures to open and unavailability of battery Main No. 1 due to test and maintenance. RCIC and supplemental injection fail due to power failure. Additional accident sequences that may result from this fire scenario are dominated by failures of operator actions such as failure to abandon the MCR when appropriate, failure to initiate ASDC, failure to start RCIC, failure to manually open the HWWV.	6.27E-08	5.1%
16-M	16-M.022-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0032	In this scenario, two or more SRVs spuriously open. Depressurization is initiated with the SORVs. Low pressure injection by RHR in the LPCI mode and CS are unsuccessful as an injection source due to fire damage affecting both loops of LPCI injection and CS. Condensate injection to isolated RPV fails due to random equipment or human failure events. EHPM or condensate flood up is unsuccessful due to random equipment and human failure events. Thus, all injection is lost and core damage results.	5.18E-08	4.2%
16-M	16-M.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0033	This fire scenario has the same risk insights as 16-M.022-CAB-SUP.	5.10E-08	4.2%

Table 5-11. Top Ten ΔCDF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔCDF (/ reactor year)**	**Unit 2 ΔCDF (%)**
09	09.5002-0-BDAA-211-000C-FRB	Fire causing full compartment damage originating in fixed ignition source 0-BDAA-211-000C	For many of the dominant potential accident sequences in this scenario, a single SRV spuriously opens due to fire-induced hot short. In the most likely of such cases, the EHPM Pump is unsuccessful due to failed operator action, injection from HPCI and RCIC is unsuccessful due to fire damage and failed operator action to initiate supplemental injection. Manual depressurization is successful. Low pressure injection by RHR in the LPCI mode or CS is successful. Alternate Shutdown Cooling (ASDC) is unsuccessful. In other dominant accident sequences, when no SRVs spuriously open, early HPCI is unsuccessful as an initial injection source due to operator failure to initiate supplemental injection. When RPV level drops to TAF, manual depressurization is successful. Low pressure injection by RHR in the LPCI mode and CS are unsuccessful as an initial injection source due to fire damage. Alternate vessel injection in the form of standby coolant injection coupled with condensate flood-up (allows time to align SBCI), or condensate injection is unsuccessful due to fire damage and failed operator action to control RPV level.	5.07E-08	4.1%

Table 5-11. Top Ten ΔCDF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔCDF (/ reactor year)**	**Unit 2 ΔCDF (%)**
02-03	02-03.3000-T-2-M-T3-1	Transient combustible fire with target set 02-03.3000-T-2-M-T3-1	This fire scenario results in a stuck open relief valve. Early high pressure injection from HPCI and RCIC fails due to fire impacts. HPCI and RCIC fail due to fire impacts to the steam supply valves, suction valve from CST, discharge valves, and turbine controls. The operators fail to initiate the EHPM Pump. Depressurization succeeds but RHR LPCI and CS are failed due to fire impacts. Loop I RHR is failed due to fire impacts to a LPCI injection valve and the pump suction valves. Loop II RHR is failed due to fire impacts to RHR Pump suction valves and loss of 4 kV Shutdown Boards C and D. CS Loop I is failed due to fire impacts to the loop I discharge valves, and Loop II is failed due to fire impacts to the Loop II CS pumps, valves and loss of 4 kV Shutdown Boards C and D power. Condensate in level control is failed due to fire impacts to the level control circuit and to the control air supply. Core damage results from a loss of all injection.	4.72E-08	3.8%
02-03	02-03.010-CAB	Cabinet fire originating in fixed ignition source 2-JBOX-252-11952	Early HPCI and RCIC are unsuccessful as initial injection sources due to fire damage. A cool down is initiated but EHPM is unsuccessful due to random equipment or human failure events, such as failure to initiate supplemental injection or supplemental injection pump fails to start or is in test and maintenance. When RPV level drops to TAF, manual depressurization is successful. Low pressure injection by RHR in the LPCI mode and CS are unsuccessful as an initial injection source due to fire-induced equipment failures of CS loops and LPCI injection paths. Alternate vessel injection from standby coolant injection and condensate injection fails due to fire damage.	4.46E-08	3.6%
02-01	02-01.027-CAB	Cabinet fire originating in fixed ignition source 2-LPNL-925-0022	This fire scenario has the same risk insights as 02-03.010-CAB.	4.27E-08	3.5%

Table 5-11. Top Ten ΔCDF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔCDF (/ reactor year)**	**Unit 2 ΔCDF (%)**
02-03	02-03.003-CAB	Cabinet fire originating in fixed ignition source 2-LAC-247-0204	One SRV spuriously opens due to fire-induced hot short. The EHPM Pump is unsuccessful due to failed operator actions or random equipment failures. Injection from HPCI and RCIC is unsuccessful due to fire damage and manual depressurization is successful. Low pressure injection by RHR in the LPCI mode and CS is unsuccessful due to fire damage. Injection from standby coolant injection, or condensate injection is unsuccessful due to fire damage affecting both loops of LPCI injection and both loops of CS.	4.26E-08	3.5%
02-03	02-03.3000-T-2-L-T3-1	Transient combustible fire with target set 02-03.3000-T-2-L-T3-1	In one dominant accident sequence, one SRV spuriously opens due to fire-induced hot short. The EHPM Pump is unsuccessful due to failed operator actions or random equipment failures. Injection from HPCI and RCIC is unsuccessful due to fire damage and manual depressurization is successful. Low pressure injection by RHR in the LPCI mode and CS is unsuccessful due to fire damage and failed operator actions. Injection from standby coolant injection, or condensate injection is unsuccessful due to failed operator actions, random equipment failures, and fire damage. In another dominant accident sequence, early HPCI or RCIC are unsuccessful as an initial injection source due to fire damage. A cool down is initiated but EHPM is unsuccessful due to random equipment or human failure events. When RPV level drops to TAF, manual depressurization is successful. Low pressure injection by RHR in the LPCI mode and CS are unsuccessful as an initial injection source due to fire damage affecting both loops of CS and due to fire damage affecting Loop I of the LPCI injection path combined with random equipment or human failure events. Alternate vessel injection fails due fire damage affecting standby coolant injection and condensate injection.	3.34E-08	2.7%
SUM				5.32E-07	43.2%
NOTE:	Important contributing HFEs include HFFA0ASD_RCIC, HFFA0SUPPHPI2, HFFA0002RPV_LVL, and HFFA0RHRCS_LPP.				

Table 5-12. Top Ten ΔLERF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔLERF (/ reactor year)**	**Unit 2 ΔLERF (%)**
16-M	16-M.022-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0032	This fire scenario results in two SORVs due to fire-induced ADS signals. Condensate flood up is successful but initial RHR and CS fail due to fire-induced loss of power. The supplemental EHPM system fails due to operator failure to initiate it. Long term condensate injection fails due to operator failure to recover RCW pump failures. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach. The vessel fails at low pressure.	2.24E-08	4.4%
16-M	16-M.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 2-PNLA-009-0033	This fire scenario has the same risk insights as 16-M.023-CAB-SUP.	2.23E-08	4.4%
16-K	16-K.023-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0032	This fire scenario results in a reactor trip with early successful high pressure injection from HPCI or RCIC. RHR suppression pool cooling is failed early due to fire impacts to Loop I RHR pumps, and Loop II fails due to overload failures of 4 kV Shutdown Boards C and D due to a spurious common accident signal, resulting in the loss of suppression pool as a suction source for HPCI and RCIC. Depressurization succeeds, but low pressure injection using RHR, CS, standby coolant injection, and condensate in level control mode fails due to fire impacts, and the EHPM Pump randomly fails, resulting in a loss of long term low pressure injection. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The vessel and containment fail at low pressure.	1.90E-08	3.8%
16-K	16-K.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 1-PNLA-009-0033	This fire scenario has the same risk insights as 16-K.023-CAB-SUP.	1.90E-08	3.8%

Table 5-12. Top Ten ΔLERF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔLERF (/ reactor year)**	**Unit 2 ΔLERF (%)**
16-O	16-O.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0032	This fire scenario results in a reactor trip with early successful high pressure injection from HPCI or RCIC. RHR suppression pool cooling is failed early due to fire impacts to Loop II RHR pumps, and Loop I fails due to overload failures of 4 kV Shutdown Boards C and D due to a spurious common accident signal, resulting in the loss of suppression pool as a suction source for HPCI and RCIC. Depressurization succeeds, but low pressure injection using RHR Loop II, CS, and standby coolant injection fails due to fire impacts. RHR Loop I fails due to failure of the auto start and failure of the operator to manually start the loop. The EHPM Pump fails due to operator failure to initiate it and long term condensate fails due to operator action to restore RCW cooling to the condensate pumps. This results in a loss of long term low pressure injection. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach.	1.80E-08	3.6%
16-O	16-O.025-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0033	This fire scenario results in a reactor trip with early successful high pressure injection from HPCI or RCIC. RHR suppression pool cooling is failed early due to fire impacts to Loop I RHR pumps, and Loop II fails due to overload failures of 4 kV Shutdown Boards C and D due to a spurious common accident signal, resulting in the loss of suppression pool as a suction source for HPCI and RCIC. Depressurization succeeds, but low pressure injection using RHR, CS, and standby coolant injection fails due to fire impacts. The EHPM Pump fails due to operator failure to initiate it and long term condensate fails due to operator action to restore RCW cooling to the condensate pumps. This results in a loss of long term low pressure injection. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach and the vessel fails while at high pressure.	1.77E-08	3.5%

Table 5-12. Top Ten ΔLERF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔLERF (/ reactor year)**	**Unit 2 ΔLERF (%)**
09	09.5002-0-BDAA-211-000C-FRB	Fire causing full compartment damage originating in fixed ignition source 0-BDAA-211-000C -FRB	This fire scenario results in a reactor trip with a single fire-induced SORV. Early injection from HPCI and RCIC fails due to fire-induced control power failures. The operator fails to initiate the EHPM Pump. Emergency depressurization and condensate flood up are successful but low pressure injection from RHR, CS, and standby coolant injection fail due to fire impacts. Long term condensate fails due to operator failure to maintain level. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The vessel and containment fail at low pressure.	1.52E-08	3.0%
16-M	16-M.TS06	Transient combustible fire TS#6	This fire scenario results in multiple stuck open relief valves. Low pressure injection fails due to fire impacts. RHR LPCI loop I fails due to fire impacts to the loop injection valves, minimum flow valves, pumps, and electrical power. Loop II RHR is failed due to fire impacts to 4 kV Shutdown Boards C and D operation. CS Loop I and Loop II are failed similarly, with direct fire impacts to the Loop I pumps, valves, and electrical power and Loop II 4 kV Shutdown Boards C and D operation. The EHPM Pump fails randomly, which results in core damage. After core damage, the operators fail to establish high pressure injection, fail to ensure the vessel is depressurized after core damage due to fire impacts to SRVs, and all low pressure injection is failed, precluding the ability to arrest core damage or inject into the drywell. The vessel and containment fail at high pressure.	1.51E-08	3.0%

Table 5-12. Top Ten ΔLERF Scenarios for Unit 2					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 2 ΔLERF (/ reactor year)**	**Unit 2 ΔLERF (%)**
08	08.001-CAB	Fire originating in cabinet 0-BDAA-211-000D	This fire scenario results in a loss of early high pressure injection from HPCI and RCIC due to fire impacts. HPCI is lost due to fire impacts to the steam supply isolation signal, and RCIC is failed due to fire impacts to the steam supply valves. The operators fail to start the EHPM Pump. Depressurization is successful, but low pressure injection using RHR and CS fails due to fire impacts. Loop I RHR and CS are failed due to impacts to 4 kV Shutdown Board B offsite power feeder breaker and the operators fail to isolate the breaker to align the EDG to support the RHR and CS pump and valves. Loop II RHR LPCI is failed due to fire impacts to a LPCI injection valve and minimum flow valve, and fire impacts to electrical power. Loop II CS is lost due to fire impacts to the CS pumps and fire impacts to electrical power. Condensate level control succeeds but the operators fail to maintain level, resulting in core damage. After core damage, the low pressure injection is failed, precluding arrest of core damage or injection into the drywell. The vessel and containment fail at low pressure.	1.35E-08	2.7%
08	08.3000-T-4	Transient combustible fire with target set 08.3000-T-4	This fire scenario results in a reactor trip with a single fire-induced SORV. Early injection from HPCI fails due to a spurious isolation signal and RCIC fails due to fire-induced control power failures. The operator fails to initiate the EHPM Pump. The operator fails to emergency depressurize in time to prevent core damage. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The vessel and containment fail at low pressure due to the open relief valve.	1.05E-08	2.1%
SUM				1.73E-07	34.3%
NOTE:	Important contributing HFEs include HFFA0ASD_RCIC, HFFA0SUPPHPI2, HFFA0002RPV_LVL, and HFFA0RHRCS_LPP.				

Table 5-13. Top Ten ΔCDF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔCDF (/ reactor year)**	**Unit 3 ΔCDF (%)**
16-A	FS-EC-107	Fire originating in fixed source 3-PNLA-009-0023BA	This fire scenario risk is dominated by a general transient core damage sequence that is characterized by intact or functioning SRVs, failure of late high pressure injection (HPCI/RCIC after 4 hours), failure of late low pressure injection, and failure of the power conversion system. The fire event results in failure of power to 4kV shutdown boards 3EA and 3EB. The dominant cutset involves the random failures of diesel generator 3D and the EHPM Pump causing the loss of all remaining injection sources.	1.97E-07	16.6%
16-A	16-A.Habitability	Fire causing MCR Habitability Abandonment	This fire scenario results in a loss of habitability which leads to abandonment of the MCR. This fire scenario risk is dominated by operator action failures to start and align RCIC or the EHPM Pump within the time available prior to core damage.	1.06E-07	8.9%
16-O	16-O.025-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0033	This fire scenario risk is dominated by a general transient core damage sequence that is characterized by two or more SRVs spuriously opening, which causes the high-pressure injection system to be unavailable. In addition, the plant trip causes a loss of offsite power situation that leads to the loss of condensate flood-up. The EHPM system is also lost. The fire event results in failure of power to 4kV shutdown boards 3EA, 3EB, 3EC, and 3ED, due to MSO 5PSa. This precludes operation of the low-pressure injection system, leading to core damage.	5.26E-08	4.5%
16-O	16-O.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0032	This fire scenario has the same risk insights as 16-O.025-CAB-SUP.	5.26E-08	4.5%

Table 5-13. Top Ten ΔCDF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔCDF (/ reactor year)**	**Unit 3 ΔCDF (%)**
12	12.1002-CAB	Fire originating in cabinet 250V RMOV Board 3B	This fire scenario risk is dominated by the unavailability of high-pressure injection systems (HPCI/RCIC). These systems are either lost due to fire-induced failures, or by fire-induced spurious opening of two or more SRVs. The high pressure make up system fails to be actuated by the operators. Inventory control is further lost because the operators fail to depressurize in time, or they successfully depressurize but fail to initiate low pressure injection. In addition, alternate vessel injection (standby coolant injection) is lost to the fire. This leads to core damage.	2.76E-08	2.4%
13	13.004-CAB	Fire originating in cabinet 3-BDDD-281-0003A	This fire scenario risk is dominated by the unavailability of high-pressure injection systems (HPCI/RCIC). These systems are lost due to fire-induced failures. The high pressure make up system fails to be properly operated by the operators. Inventory control is further lost because the operators fail to depressurize in time, or they successfully depressurize but fail to operate low pressure injection. In addition, alternate vessel injection (standby coolant injection) is lost due to the fire.	2.60E-08	2.2%
03-03	03-03.3000-T-3-D-1	Transient combustible fire with target set 03-03.3000-T-3-D-1	This fire scenario risk is dominated by the fire-induced failures of both the high-pressure injection systems (HPCI/RCIC), and the low-pressure injection systems (due to loss of 4kV shutdown boards 3EA, 3EB, 3EC, and 3ED). Alternate vessel injection (standby coolant injection) is lost to the fire. This leaves the high pressure make up system as the only long-term source of inventory control, but it is unavailable (due to maintenance, by random failure, or human failure to properly operate that system), which leads to core damage.	2.47E-08	2.1%

Table 5-13. Top Ten ΔCDF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔCDF (/ reactor year)**	**Unit 3 ΔCDF (%)**
16-N	16-N.017-CAB-RFNMT	Fire originating in the Security Multiplexer Cabinet	This fire scenario risk is dominated by the fire-induced failures of the high-pressure injection systems (HPCI/RCIC). The high pressure make up system is also lost to the fire. Depressurization is not successful, or, if it is, the low pressure injection systems, including alternate vessel injection (standby coolant injection) are unavailable due to a combination of fire-induced failures and random failures of their power supply.	2.35E-08	2.0%
16-N	16-N.021-CAB-RFNMT	Fire originating in the COMM Relay Cabinets (74, 75, 76, 78, 79, 82, 83, 84)	This fire scenario risk is dominated by the fire-induced failures of the high-pressure injection systems (HPCI/RCIC). The high pressure make up system is also lost due to the fire. Depressurization is not successful, or, if it is, the low pressure injection systems, including alternate vessel injection (standby coolant injection) are unavailable due to a combination of fire-induced failures and random failures of their power supply.	2.35E-08	2.0%
03-03	03-03.3000-T-3-N-T1-2	Transient combustible fire with target set 03-03.3000-T-3-N-T1-2	This fire scenario risk is dominated by the fire-induced failures of both the high-pressure injection systems (HPCI/RCIC), and the low-pressure injection systems (due to loss of 4kV shutdown boards 3EA, 3EB, 3EC, and 3ED). Alternate vessel injection (standby coolant injection) is lost to the fire. This leaves the high pressure make up system as the only long-term source of inventory control, but it is unavailable (due to maintenance, by random failure, or human failure to properly operate that system), which leads to core damage.	2.25E-08	1.9%
SUM				5.55E-07	48.0%
NOTE:	Important contributing HFEs include HFFA0268480CRSTIE, HFFA0ASD_RCIC, HFFA0SUPPHPI2, HFFA0RHRCS_LPP, and HFFA0002RPV_LVL.				

Table 5-14. Top Ten ΔLERF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔLERF (/ reactor year)**	**Unit 3 ΔLERF (%)**
16-A	FS-EC-107	Fire originating in fixed source 3-PNLA-009-0023BA	This fire scenario risk is dominated by a general transient core damage sequence 5. This accident sequence is characterized by intact or functioning SRVs, failure of late high pressure injection (HPCI/RCIC after 4 hours), failure of late low pressure injection, and failure of the power conversion system. The fire event results in failure of power to 4kV shutdown boards 3EA and 3EB. The dominant cutset involves the random failures of diesel generator 3C and the EHPM Pump causing the loss of all remaining injection sources. After core damage, the operators can successfully depressurize but all low pressure injection sources are failed which precludes arrest of the core damage or injection into the drywell. The vessel and containment fail at low pressure, resulting in the large early release.	6.03E-08	15.7%
16-O	16-O.025-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0033	This fire scenario risk is dominated by a general transient core damage sequence 17. This accident sequence is characterized by two or more SRVs spuriously opening, which causes the high-pressure injection system to be unavailable. In addition, the plant trip causes a loss of offsite power situation that leads to the loss of condensate flood-up. The EHPM system is also lost. The fire event results in failure of power to 4kV shutdown boards 3EA, 3EB, 3EC, and 3ED, due to MSO 5PSa. This precludes operation of the low-pressure injection systems, leading to core damage. After core damage, the loss of low-pressure injection prevents in-vessel recovery, leading to reactor vessel breach. Containment is intact, but the absence of injection into the drywell precludes an effective retention of radionuclides.	2.25E-08	5.8%

Table 5-14. Top Ten ΔLERF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔLERF (/ reactor year)**	**Unit 3 ΔLERF (%)**
16-O	16-O.024-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0032	This fire scenario risk is dominated by a general transient core damage sequence 17. This accident sequence is characterized by two or more SRVs spuriously opening, which causes the high-pressure injection systems to be unavailable. In addition, the plant trip causes a loss of offsite power situation that leads to the loss of condensate flood-up. The EHPM system is also lost. The fire event results in failure of power to 4kV shutdown boards 3EA, 3EB, 3EC, and 3ED, due to MSO 5PSa. This precludes operation of the low-pressure injection systems, leading to core damage. After core damage, the loss of low-pressure injection prevents in-vessel recovery, leading to reactor vessel breach. Containment is intact, but the absence of injection into the drywell precludes an effective retention of radionuclides. Another dominant accident progression associated with this scenario involves an unisolated ISLOCA, caused by a fire-induced spurious opening of valves in the RHR suction path. This results in a radionuclide release outside containment	2.25E-08	5.8%

Table 5-14. Top Ten ΔLERF Scenarios for Unit 3					
Compart-ment	Fire Scenario (See NOTE)	Fire Scenario Description	Risk Insights	Unit 3 ΔLERF (/ reactor year)	Unit 3 ΔLERF (%)
16-O	16-O.TS05	Transient combustible fire TS#5	This fire scenario risk is dominated by a general transient core damage sequence 12. This accident sequence is characterized by intact or functioning SRVs, failure of early high pressure injection (HPCI/RCIC after 4 hours), failure to depressurize, failure of the EHPM Pump, and failure of the power conversion system. The fire event results in failure of all the 4kV shutdown board power. The only remaining injection source is the emergency EHPM Pump. The dominant cutset involves the testing and maintenance unavailability of the emergency EHPM Pump. Other dominant cutsets involve other random failures of the emergency EHPM Pump and failure to align additional inventory for the CST. Post-core damage, depressurization fails due to the fire impacts to the SRVs. The absence of injection prevents in-vessel recovery, leading to reactor vessel breach. Containment is intact, but the absence of injection into the drywell eventually precludes an effective retention of radionuclides.	2.15E-08	5.6%
16-A	FS-EC-136	Fire originating in fixed source 3-PNLA-009-0023CD	This fire scenario risk is dominated by a general transient core damage sequence 5. This accident sequence is characterized by intact or functioning SRVs, failure of late high pressure injection (HPCI/RCIC after 4 hours), failure of late low pressure injection, and failure of the power conversion system. The fire event results in failure of all the 4kV shutdown board power thus resulting in failure of late high pressure injection and late low pressure injection. The only remaining injection source is the emergency EHPM Pump. The dominant cutset involves the testing and maintenance unavailability of the emergency EHPM Pump. Other dominant cutsets involve other random failures of the emergency EHPM Pump and failure to align additional inventory for the CST. After core damage, the operators can successfully depressurize but all low pressure injection sources are failed which precludes arrest of the core damage or injection into the drywell. The vessel and containment fail at low pressure, resulting in the large early release.	1.42E-08	3.7%

Table 5-14. Top Ten ΔLERF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔLERF (/ reactor year)**	**Unit 3 ΔLERF (%)**
16-A	FS-EC-137	Fire originating in fixed source 3-PNLA-009-0023BA	This fire scenario has the same risk insights as scenario FS-EC-136.	1.42E-08	3.7%
12	12.1002-CAB	Fire originating in cabinet 250V RMOV Board 3B	This fire scenario results in a single stuck open relief valve. Early HPCI fails due to a fire-induced steam line isolation and early RCIC fails due to fire-induced loss of control power. Emergency depressurization is successful. Auto start of RHR LPCI Loop I fails due to fire impacts and the operator fails to manually start it. RHR Loop II fails due to fire-induced loss of power. CS is failed in a similar manner. The operator fails to initiate the EHPM Pump. Standby coolant injection fails due to fire-induced power failures. The operator fails to initiate long term condensate injection. After core damage, injection to the drywell and vessel fails. The containment is isolated and intact at the time of vessel failure.	1.05E-08	2.7%
16-O	16-O.003-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0081	This fire scenario results in two SORVs due to fire-induced ADS signals. Initial RHR and CS fail due to fire-induced loss of power. The supplemental EHPM system fails due to operator failure to initiate it. Long term condensate injection fails due to operator failure to recover RCW pump failures. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach. The vessel fails at low pressure.	7.43E-09	1.9%
16-O	16-O.004-CAB-SUP	Fire with credited automatic suppression originating in cabinet 3-PNLA-009-0082	This fire scenario results in two SORVs due to fire-induced ADS signals. Initial RHR and CS fail due to fire-induced loss of power. The supplemental EHPM system fails due to operator failure to initiate it. Long term condensate injection fails due to operator failure to recover RCW pump failures. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach. The vessel fails at low pressure.	7.43E-09	1.9%

Table 5-14. Top Ten ΔLERF Scenarios for Unit 3					
Compart-ment	**Fire Scenario (See NOTE)**	**Fire Scenario Description**	**Risk Insights**	**Unit 3 ΔLERF (/ reactor year)**	**Unit 3 ΔLERF (%)**
21-E	21-E.1002-BCHG	Fire originating in battery charger 3-CHGB-254-0000DA	This fire scenario has no open relief valves. Initial HPCI and RCIC are successful, but long term HPCI and RCIC fail due to lack of suppression pool cooling caused by fire-induced electrical failures. The supplemental EHPM system fails due to operator failure to initiate it. Emergency depressurization is successful. Late RHR and CS fail due to fire induced loss of power. The operator fails to initiate standby coolant injection. Long term condensate injection fails due to the loss of control building HVAC which fails the startup bypass valve. Control building HVAC is lost due to fire impacts and operator failure to recover. After core damage, all injection is failed, precluding arrest of the core damage and injection into the drywell. The containment is intact and isolated at vessel breach. The vessel fails at low pressure.	7.10E-09	1.8%
SUM				1.88E-07	48.7%
NOTE:	Important contributing HFEs include HFFA0268480CRSTIE, HFFA0SUPPHPI2, HFFA0ASD_RCIC, HFFA0RHRCS_LPP, and HFFA0002RPV_LVL.				

5.3 Shutdown Risk

BFN does not have a shutdown PRA analysis. As such, a numerical estimate of outage risk does not exist. Section 4.6 and Attachment F of Reference 26 provides a bounding estimate of approximately 1% of shutdown CDF for the risk increase due to EPU, due to an increase in the assumed non-recovery probabilities associated with LOOP.

5.4 Seismic Risk

BFN does not have a seismic PRA analysis. The IPEEE used a Seismic Margins Analysis methodology that does not predict a numerical risk result. A conservative bounding estimate using more recent 2008 USGS seismic hazard curves predicts an upper bounding estimate of 3.7E-6/yr (Unit 1), 5.4E-6/yr (Unit 2), and 5.4E-6/yr (Unit 3) for seismic CDF.

5.5 Other External Event Risks

BFN does not have a PRA analysis for "other" external events (high winds /tornados/transportation and nearby facility accidents and external floods). As such an exact estimate of other External Events risk does not exist. The screening approach used in the BFN Units 1, 2, and 3 IPEEE analysis of high winds/tornados, external floods, and nearby facilities transportation accidents demonstrates adequate defense against these threats. No weaknesses or plant modifications were identified as a result of this analysis. The IPEEE concluded based on bounding analyses that the risk due to these events is less than 1E-6/yr and therefore a change in risk would also be <1E-6/yr.

5.6 Total Risk

The total core damage frequency is calculated by adding the CDF contributions from internal events, fire, seismic, and other external events. The same approach is used to estimate the total large early release frequency and to estimate the change in CDF or LERF. The CDF contribution for seismic, and other external events is approximated as discussed in Sections 5.4 and 5.5. The LERF contribution for seismic and other external events is assumed to be an order of magnitude lower than the corresponding CDF. The total risk and change in risk for each unit is shown in Table 5-15.

Table 5-15. Total Risk

Unit	CDF (CLTP)	CDF (EPU)	ΔCDF	LERF (CLTP)	LERF (EPU)	ΔLERF
1	5.91E-05	6.08E-05	1.69E-06	7.96E-06	8.73E-06	7.74E-07
2	5.96E-05	6.14E-05	1.74E-06	7.99E-06	8.65E-06	6.63E-07
3	6.47E-05	6.64E-05	1.67E-06	7.18E-06	7.72E-06	5.45E-07

5.7 Sensitivity Analysis and Uncertainties

5.7.1 Internal Events PRA Sensitivity Analyses

The following Internal Events PRA sensitivity analyses were performed to address the impact of EPU conditions.

5.7.1.1 Startup Testing

An 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. The purpose of this section is to show that performing these two (2) start-up tests (CCDP & CLERP) results in unnecessary risk. This information is provided below in Table 5-16. The CCDP and the CLERP are the Birnbaum risk importances taken from the base case EPU analysis.

Table 5-16. Startup Testing CCDP and CLERP Results

Unit	Initiating Event	Conditional Core Damage Probability (CCDP)	Conditional Large Early Release Probability (CLERP)
Unit 1	Turbine Trip	2.10E-07	3.88E-08
Unit 1	MSIV Closure	9.42E-06	2.20E-06
Unit 2	Turbine Trip	1.92E-07	3.59E-08
Unit 2	MSIV Closure	9.40E-06	2.20E-06
Unit 3	Turbine Trip	1.61E-07	3.15E-08
Unit 3	MSIV Closure	8.11E-06	2.00E-06

These CCDPs and CLERPs represent the incurred risk caused by performing the proposed tests (i.e., the initiating events occur). If both tests are performed, the total conditional probabilities would be for Unit 1: 9.63E-6 (CCDP) and 2.24E-06 (CLERP), for Unit 2: 9.59E-6 (CCDP) and 2.24E-6 (CLERP), and for Unit 3: 8.27E-6 (CCDP) and 2.03E-06 (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.

5.7.1.2 Sensitivity of Setting Screening Human Error Probabilities to 1.0

The following list of operator action events (definitions provided in Appendix C) use a screening HEP value (0.1) in the HRA analysis. This sensitivity evaluation sets these events to 1.0 to see what the impact is from these events in the model.

- HFA_0_LCISTARTATWS
- HFA_0_SPRAYIOOV
- HFA_0_SPRAYLLOCA
- HFA_0_VSSDEP
- HFA_0032LEAK_ISO

- HFA_0074UNITXTIE
- HFA_0085ALIGNFCV
- HFA_0085REFILLCST
- HFA_0099MGRESET
- HFA_0HCIINIT10
- HFA_0HCIINIT15
- HFA_0LPIINIT06
- HFA_0LPIINIT15
- HFA_PARALLEL_DG

Table 5-17 shows the results of this sensitivity analysis. As can be seen in the table, setting the events that use screening values (0.1) to a value of 1.0 does not significantly change the results.

Table 5-17. Screening HEPs set to 1.0

	CDF	LERF	ΔCDF	ΔLERF
Base Case EPU PRA - Unit 1	7.14E-06	1.64E-06	3.52E-8	1.02E-8
Sensitivity - Unit 1	7.17E-06	1.65E-06		
Base Case EPU PRA - Unit 2	6.47E-06	1.55E-06	3.80E-8	4.04E-8
Sensitivity - Unit 2	6.51E-06	1.59E-06		
Base Case EPU PRA - Unit 3	7.06E-06	1.54E-06	2.44E-8	1.03E-8
Sensitivity - Unit 3	7.08E-06	1.55E-06		

5.7.1.3 Sensitivity for RPV Overpressure

The probabilities for the following basic events were adjusted by a factor 2 to address the potential impact to RPV overpressure.

U1_RV_13FO (3 OF 13 RELIEF VALVES FAIL TO OPEN (COMMON CAUSE FAILURE)):
Failure Probability = 4.81E-05 x 2 = 9.62E-05

U3_RV_13FO (3 OF 13 RELIEF VALVES FAIL TO OPEN (COMMON CAUSE FAILURE):
Failure Probability = 4.81E-05 x 2 = 9.62E-05

U2_RV_13FO (3 OF 13 RELIEF VALVES FAIL TO OPEN (COMMON CAUSE FAILURE)):
Failure Probability = 4.81E-05 x 2 = 9.62E-05

In addition, the pre-initiator human error basic event, HFL_0001RV_CAL, Miscalibration of three or more relief valves, was increased by a factor of 2.

HFL_0001RV_CAL: 3.0E-03 x 2 = 6.0E-03

Table 5-18 shows the results of this sensitivity analysis. As can be seen in the table, doubling the probability of RPV overpressure does not significantly change the results.

Table 5-18. RPV Overpressure					
RVP Overpressure	**Initiating Event Frequency (/yr)**	**CDF**	**LERF**	**ΔCDF**	**ΔLERF**
Base Case EPU PRA - Unit 1	4.81E-05	7.14E-06	1.64E-06	1.80E-8	0.00
Sensitivity - Unit 1	9.62E-05	7.16E-06	1.64E-06		
Base Case EPU PRA - Unit 2	4.81E-05	6.47E-06	1.55E-06	1.70E-8	0.00
Sensitivity - Unit 2	9.62E-05	6.49E-06	1.55E-06		
Base Case EPU PRA - Unit 3	4.81E-05	7.06E-06	1.54E-06	1.80E-8	0.00
Sensitivity - Unit 3	9.62E-05	7.08E-06	1.54E-06		

5.7.1.4 Sensitivity for Impact to Transient Initiators

Sensitivity - Turbine Trip

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 (%TT) 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 0.463/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 a total 1.5 additional trips in the first and second years as described above, the revised Turbine Trip frequency for this sensitivity case is calculated as:

$$\%TT = \frac{(10 \times 0.463) + 1.5}{10} = 0.613$$

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

Table 5-19 shows the results of this sensitivity analysis. As can be seen in the table, increasing the turbine trip frequency does not significantly change the results.

Table 5-19. Sensitivity for Impact to Transient Initiators - Turbine Trip

%TT - Turbine Trip	Initiating Event Frequency	CDF	LERF	ΔCDF	ΔLERF
Base Case EPU PRA - Unit 1	4.63E-01	7.14E-06	1.64E-06	2.73E-08	6.20E-9
Sensitivity - Unit 1	6.13E-01	7.17E-06	1.65E-06		
Base Case EPU PRA - Unit 2	4.63E-01	6.47E-06	1.55E-06	2.79E-08	3.59E-8
Sensitivity - Unit 2	6.13E-01	6.50E-06	1.59E-06		
Base Case EPU PRA - Unit 3	4.63E-01	7.06E-06	1.54E-06	9.30E-09	4.90E-9
Sensitivity - Unit 3	6.13E-01	7.07E-06	1.54E-06		

<u>Sensitivity - Loss of Feedwater</u>

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 sensitivity of a potential increase was evaluated.

An assessment was performed assuming that the EPU changes would cause an increase to the loss of feedwater (total and partial) initiating event frequency (%TLFW and %PLFW). The change in the loss of feedwater initiating event frequency is calculated as follows for this sensitivity case

Base long-term total loss of feedwater frequency is 9.38E-2/yr
Base long-term partial loss of feedwater frequency is 9.21E-2/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 a total of 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:

$$\%TLFW = \frac{(10 \times 9.38E\text{-}2) + 1.5}{10} = 2.44E\text{-}01/yr$$

$$\%PLFW = \frac{(10 \times 9.21E\text{-}2) + 1.5}{10} = 2.42E\text{-}01/yr$$

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

Table 5-20 shows the results of this sensitivity analysis. As can be seen in the table, increasing the loss of feedwater frequency does not significantly change the results. Note that initiating event frequencies for total and partial loss of feedwater are shown in the table, but only the total (summation from both initiators) CDF and LERF are shown for the base case and the sensitivity case for each unit.

Table 5-20. Sensitivity for Impact to Transient Initiators - Loss of Feedwater

Loss of Feedwater	Initiating Event Frequency	CDF	LERF	ΔCDF	ΔLERF
Base Case EPU PRA - Unit 1	%TLFW 9.38E-02/yr %PLFW 9.21E-02/yr	7.14E-06	1.64E-06	1.45E-06	3.41E-7
Sensitivity - Unit 1	%TLFW 2.44E-01/yr %PLFW 2.42E-01/yr	8.58E-06	1.98E-06		
Base Case EPU PRA - Unit 2	%TLFW 9.38E-02/yr %PLFW 9.21E-02/yr	6.47E-06	1.55E-06	1.44E-06	7.37E-7
Sensitivity - Unit 2	%TLFW 2.44E-01/yr %PLFW 2.42E-01/yr	7.91E-06	2.29E-06		
Base Case EPU PRA - Unit 3	%TLFW 9.38E-02/yr %PLFW 9.21E-02/yr	7.06E-06	1.54E-06	1.23E-6	3.08E-7
Sensitivity - Unit 3	%TLFW 2.44E-01/yr %PLFW 2.42E-01/yr	8.29E-06	1.85E-06		

<u>Sensitivity - Loss of Condenser Vacuum, Total and Partial Loss of Condensate</u>

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 the loss of condenser initiator based on EPU; however, the sensitivity of a potential increase was evaluated by assuming that the EPU changes would cause an increase to the loss of condenser vacuum, total and partial loss of condensate initiating event frequencies. The change in initiating event frequencies are calculated as follows for this sensitivity case:

- Base long-term loss of condenser vacuum frequency is 1.03E-1/yr

- Base long-term Total loss of condensate frequency is 7.48E-3/yr
- Base long-term Partial loss of condensate frequency is 1.74E-2/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 a total of 1.5 additional loss of condenser vacuum, total and partial loss of condensate events in the first and second years, the revised Loss of Condenser, Total Loss of Condensate, and Partial Loss of Condensate frequencies for this sensitivity case are calculated as:

$$\%LCV = \frac{(10 \times 1.03E\text{-}1) + 1.5}{10} = 2.53E\text{-}1/yr$$

$$\%TLCF = \frac{(10 \times 7.48E\text{-}3) + 1.5}{10} = 1.58E\text{-}1/yr$$

$$\%PLCF = \frac{(10 \times 1.74E\text{-}2) + 1.5}{10} = 1.67E\text{-}1/yr$$

All other parameters are maintained the same as the EPU base case. Table 5-21 shows the results of this sensitivity analysis. As can be seen in the table, increasing the loss of condenser vacuum, frequency does not significantly change the results. Note that initiating event frequencies for loss of condenser vacuum, total and partial loss of condensate events are shown in the table, but only the total (summation from all three initiators) CDF and LERF are shown for the base case and the sensitivity case for each unit.

Table 5-21. Sensitivity - Loss of Condenser Vacuum, Total and Partial Loss of Condensate					
Loss of Condenser Vacuum, Total and Partial Loss of Condensate	**Initiating Event Frequency**	**CDF**	**LERF**	**ΔCDF**	**ΔLERF**
Base Case EPU PRA - Unit 1	%LCV 1.03E-1/yr %TLCF 7.48E-3/yr %PLCF 1.74E-2/yr	7.14E-06	1.64E-06	3.89E-6	8.36E-7
Sensitivity - Unit 1	%LCV 2.53E-1/yr %TLCF 1.58E-1/yr %PLCF 1.67E-1/yr	1.10E-05	2.48E-06		
Base Case EPU PRA - Unit 2	%LCV 1.03E-1/yr %TLCF 7.48E-3/yr %PLCF 1.74E-2/yr	6.47E-06	1.55E-06	1.82E-6	9.05E-7
Sensitivity - Unit 2	%LCV 2.53E-1/yr %TLCF 1.58E-1/yr %PLCF 1.67E-1/yr	1.03E-05	2.46E-06		
Base Case EPU PRA - Unit 3	%LCV 1.03E-1/yr %TLCF 7.48E-3/yr %PLCF 1.74E-2/yr	7.06E-06	1.54E-06	1.63E-6	7.91E-7
Sensitivity - Unit 3	%LCV 2.53E-1/yr %TLCF 1.58E-1/yr %PLCF 1.67E-1/yr	1.04E-05	2.33E-06		

Sensitivity - RCS High Pressure Trip

Because of the flow increases for EPU, the frequency of the RCS high pressure trip could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of RCS high pressure trips based on EPU; however, the potential sensitivity of an increase was evaluated.

The revision to the RCS high pressure trip initiating event frequency (%HIPT) uses an approach that assumes an additional RCS high pressure 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 high pressure trip initiating event frequency is calculated as follows for this sensitivity case:

Base long-term RCS high pressure trip frequency is 7.48E-3/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 a total of 1.5 additional trips in the first and second years as described above, the revised RCS High Pressure Trip frequency for this sensitivity case is calculated as:

$$\%HIPT = \frac{(10 \times 7.48E\text{-}3) + 1.5}{10} = 1.58E\text{-}1$$

All other parameters are maintained the same as the EPU base case. Table 5-22 shows the results of this sensitivity analysis. As can be seen in the table, increasing the RCS high pressure trip frequency does not significantly change the results.

Table 5-22. Sensitivity for Impact to Transient Initiators - RCS High Pressure Trip

%HIPT - RCS High Pressure Trip	Initiating Event Frequency	CDF	LERF	ΔCDF	ΔLERF
Base Case EPU PRA - Unit 1	7.48E-03	7.14E-06	1.64E-06	2.51E-8	5.90E-9
Sensitivity - Unit 1	1.58E-01	7.16E-06	1.65E-06		
Base Case EPU PRA - Unit 2	7.48E-03	6.47E-06	1.55E-06	2.61E-8	3.56E-8
Sensitivity - Unit 2	1.58E-01	6.49E-06	1.59E-06		
Base Case EPU PRA - Unit 3	7.48E-03	7.06E-06	1.54E-06	7.20E-9	4.50E-9
Sensitivity - Unit 3	1.58E-01	7.06E-06	1.54E-06		

Sensitivity - Excessive Feedwater Flow

Because of increased flow rates with EPU the frequency of an excessive feedwater flow initiating event could increase slightly. The initiating event frequency task for the PRA update will not increase the frequency of excessive feedwater flow based on EPU; however, the potential sensitivity of an increase was evaluated.

A similar assessment was performed assuming that the EPU changes would cause an increase to the excessive feedwater initiating event frequency (%EXFW). The change in the initiating event frequency is calculated as follows for this sensitivity case

- Base long-term Excessive Feedwater Flow frequency is 4.74E-2/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 a total of 1.5 additional excessive feedwater events in the first and second years as described above, the revised Excessive Feedwater Flow frequency for this sensitivity case is calculated as:

$$\%EXFW = \frac{(10 \times 4.74E\text{-}2) + 1.5}{10} = 0.1974$$

All other parameters are maintained the same as the EPU base case. Table 5-23 shows the results of this sensitivity analysis. As can be seen in the table, increasing the RCS excessive feedwater frequency does not significantly change the results.

Table 5-23. Sensitivity for Impact to Transient Initiators - Excessive Feedwater Flow

%EXFW - Excessive Feedwater Flow	Initiating Event Frequency	CDF	LERF	ΔCDF	ΔLERF
Base Case EPU PRA - Unit 1	4.74E-02	7.14E-06	1.64E-06	1.06E-6	3.17E-7
Sensitivity - Unit 1	1.97E-01	8.20E-06	1.96E-06		
Base Case EPU PRA - Unit 2	4.74E-02	6.47E-06	1.55E-06	1.06E-6	3.47E-7
Sensitivity - Unit 2	1.97E-01	7.53E-06	1.90E-06		
Base Case EPU PRA - Unit 3	4.74E-02	7.06E-06	1.54E-06	9.30E-7	3.03E-7
Sensitivity - Unit 3	1.97E-01	7.99E-06	1.84E-06		

Sensitivity - Inadvertent MSIV Closure

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

The revision to the inadvertent MSIV closure initiating event frequency (%IMSIV) uses an approach that assumes an additional inadvertent MSIV closure 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 inadvertent MSIV closure initiating event frequency is calculated as follows for this sensitivity case:

Base long-term Inadvertent MSIV Closure frequency is 7.21E-2/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 a total of 1.5 additional events in the first and second years as described above, the revised Inadvertent MSIV Closure frequency for this sensitivity case is calculated as:

$$\%IMSIV = \frac{(10 \times 7.21E\text{-}2) + 1.5}{10} = 0.2221$$

All other parameters are maintained the same as the EPU base case. Table 5-24 shows the results of this sensitivity analysis. As can be seen in the table, increasing the inadvertent MSIV closure frequency does not significantly change the results.

Table 5-24. Sensitivity for Impact to Transient Initiators - Inadvertent MSIV Closure

%IMSIV - Inadvertent MSIV Closure	Initiating Event Frequency	CDF	LERF	Δ CDF	Δ LERF
Base Case EPU PRA - Unit 1	7.21E-02	7.14E-06	1.64E-06	1.41E-6	3.35E-7
Sensitivity - Unit 1	2.22E-01	8.55E-06	1.97E-06		
Base Case EPU PRA - Unit 2	7.21E-02	6.47E-06	1.55E-06	1.414E-6	3.65E-7
Sensitivity - Unit 2	2.22E-01	7.88E-06	1.92E-06		
Base Case EPU PRA - Unit 3	7.21E-02	7.06E-06	1.54E-06	1.21E-6	3.03E-7
Sensitivity - Unit 3	2.22E-01	8.26E-06	1.84E-06		

5.7.1.5 Sensitivity for Impact to Internal Flooding frequencies

One additional evaluation was considered to address the potential for increased internal flood initiating event frequencies. This is not included in the set of sensitivity cases provided because the majority of the internal flood initiators are from systems that are not subjected to 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 the Section 5.7.1.6. To determine the potential impacts from an increase to the internal flood frequencies, it is noted that the total internal flood contribution to the internal events CDF and LERF for all units are both less than 1%. 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.

5.7.1.6 Sensitivity for Impact to LOCA Frequencies

Increased flow rates and 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 Loss of Coolant Accident (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 (this is broken up into smaller initiators as shown below):

- Core Spray LOOP 1: %1LLCA = 3.99E-7 x 2 = 7.98E-7/yr
- Core Spray LOOP 2: %1LLCB = 3.99E-7 x 2 = 7.98E-7/yr
- Recirc Suction Line A: %1LLSA = 1.99E-7 x 2 = 3.98E-7/yr
- Recirc Suction Line B: %1LLSB = 1.99E-7 x 2 = 3.98E-7/yr
- Recirc Discharge Line A: %1LLDA = 2.79E-6 x 2 = 5.58E-6/yr
- Recirc Discharge Line B: %1LLDB = 2.79E-6 x 2 = 5.58E-6/yr
- Other Large LOCA: %1LLO = 1.99E-7 x 2 = 3.98E-7/yr
- Medium LOCA: %1MLOCA = 1.03E-4 x 2 = 2.06E-4/yr
- Small LOCA: %1SLOCA = 5.00E-4 x 2 = 1.0E-3/yr
- FW Line Break: %1FWBOC =3.29E-3 x 2 = 6.58E-3/yr
- MS Line Break: %1MSBOC = 9.88E-3 x 2 = 1.98E-2/yr

All other parameters are maintained the same as the EPU base case. Table 5-25 shows the results of this sensitivity analysis. As can be seen in the table, doubling the LOCA frequency does not significantly change the results. Note that initiating event frequencies for all LOCAs are shown above, but only the total (summation from all initiators) CDF and LERF are shown for the base case and the sensitivity case for each unit.

<table>
<tr><th colspan="6">Table 5-25. Impact to LOCA Frequencies</th></tr>
<tr><th>LOCA - Loss of Coolant Accident</th><th>Initiating Event Frequency</th><th>CDF</th><th>LERF</th><th>Δ CDF</th><th>Δ LERF</th></tr>
<tr><td>Base Case EPU PRA - Unit 1</td><td rowspan="6">See Above</td><td>7.14E-06</td><td>1.64E-06</td><td rowspan="2">2.58E-7</td><td rowspan="2">2.15E-7</td></tr>
<tr><td>Sensitivity - Unit 1</td><td>7.40E-06</td><td>1.86E-06</td></tr>
<tr><td>Base Case EPU PRA - Unit 2</td><td>6.47E-06</td><td>1.55E-06</td><td rowspan="2">3.70E-7</td><td rowspan="2">2.45E-7</td></tr>
<tr><td>Sensitivity - Unit 2</td><td>6.84E-06</td><td>1.80E-06</td></tr>
<tr><td>Base Case EPU PRA - Unit 3</td><td>7.06E-06</td><td>1.54E-06</td><td rowspan="2">2.00E-7</td><td rowspan="2">2.13E-7</td></tr>
<tr><td>Sensitivity - Unit 3</td><td>7.27E-06</td><td>1.75E-06</td></tr>
</table>

5.7.1.7 Sensitivity for Minimum JHEP ("Floor Values")

When the Base EPU model results were quantified an HRA floor value of 1E-7 was chosen. This sensitivity was done to see how sensitive the CDF and LERF are to removing the floor value (i.e. set to zero). As can be seen in the table, removal of the floor value does not significantly change the results.

Table 5-26. Minimum JHEP is not set					
	No HRA Floor Value	**CDF**	**LERF**	**Δ CDF**	**Δ LERF**
Base Case EPU PRA - Unit 1	No Floor	7.14E-06	1.64E-06	-4.34E-7	-1.14E-7
Sensitivity - Unit 1		6.70E-06	1.53E-06		
Base Case EPU PRA - Unit 2		6.47E-06	1.55E-06	-4.05E-7	-8.1E-8
Sensitivity - Unit 2		6.06E-06	1.47E-06		
Base Case EPU PRA - Unit 3		7.06E-06	1.54E-06	-3.35E-7	-8.90E-8
Sensitivity - Unit 3		6.73E-06	1.45E-06		

<u>Sensitivity Using 1E-6 Floor Value</u>

When the Base EPU model results were quantified an HRA floor value of 1E-7 was chosen. This sensitivity was done to see how sensitive CDF and LERF are to adjusting the floor value to 1E-6. As can be seen in the table, changing the floor value to 1E-6 does not significantly change the results.

Table 5-27. Minimum JHEP set to 1.0E-06					
	1E-7 HRA Floor Value	**CDF**	**LERF**	**Δ CDF**	**Δ LERF**
Base Case EPU PRA - Unit 1	1E-6 Floor	7.14E-06	1.64E-06	4.39E-6	1.14E-6
Sensitivity - Unit 1		1.15E-05	2.78E-06		
Base Case EPU PRA - Unit 2		6.47E-06	1.55E-06	4.10E-6	1.14E-6
Sensitivity - Unit 2		1.06E-05	2.69E-06		
Base Case EPU PRA - Unit 3		7.06E-06	1.54E-06	3.36E-6	8.85E-7
Sensitivity - Unit 3		1.04E-05	2.42E-06		

5.7.2 FPRA Sensitivity Studies

The following FPRA sensitivities were performed to address the impact of EPU conditions.

5.7.2.1 Detailed HEPs Reduced

Human failure events (HFEs) in the FPRA model were reviewed and considered for modification for the EPU FPRA. The descriptions of each of these HFEs is provided in Table 4-9. The HPEs that demonstrated a sensitivity at EPU are shown in Table 5-28. In this fire sensitivity study, the sensitivity HEPs at CLTP are set to a factor of 5 less than the EPU HEPs as shown in Table 5-28. The altered HEPs are those that are based on detailed HRA analysis under EPU conditions. The physical execution of the actions addressed in the HFE is the same at EPU and CLTP. The actions are described, or will be described, in plant procedures. Operator training programs ensure operators possess the knowledge, skills, and abilities to perform the actions. The actual highest single change for an HEP increased by a factor of 3.38. Applying the factor of 5 to the HEPs in Table 5-28 regardless of the actual time consideration, as determined by MAAP, creates an unrealistic significance of the EPU HEPs on the FPRA result.

Table 5-28. Revised CLTP Probabilities for the HEP Sensitivity Study

HFE Name	Nominal CLTP HEP	Sensitivity CLTP HEP	EPU HEP
HFFA0002RPV_LVL	1.87E-03	9.50E-04	4.75E-03
HFFA0ASD_RCIC	2.99E-02	6.90E-03	3.45E-02
HFFA0SUPPHPI2	1.10E-03	5.84E-04	2.92E-03
HFFA4KVISO_LPI	6.48E-03	1.30E-03	6.48E-03
HFFA0RHRCS_LPP	5.80E-03	3.88E-03	1.94E-02
HFFA0LPCIINJAUTO	2.82E-03	1.91E-03	9.54E-03
HFFA0LPIINIT30	1.41E-03	7.24E-04	3.62E-03
HFFA0071CTLPOWER	9.24E-03	5.62E-03	2.81E-02
HFFA0268480CRSTIE	1.07E-02	2.56E-03	1.28E-02
HFFA_1SHV0760540_35	3.78E-03	7.56E-04	3.78E-03
HFFA_2SHV0760540_35	3.78E-03	7.56E-04	3.78E-03
HFFA_3SHV0760540_35	3.78E-03	7.56E-04	3.78E-03
HFFA_BDISOL_2531_PANEL_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_DG_AUX_A_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_RMOV_1A_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_RMOV_2A_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_SDB_1B_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_SDB_2A_35	5.79E-03	1.16E-03	5.79E-03
HFFA_BDISOL_480_SDB_3A_35	5.79E-03	1.16E-03	5.79E-03

Table 5-28. Revised CLTP Probabilities for the HEP Sensitivity Study			
HFE Name	Nominal CLTP HEP	Sensitivity CLTP HEP	EPU HEP
HFFA_BDOPER_250_RMOV_1B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_250_RMOV_1C_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_250_RMOV_2B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_250_RMOV_2C_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_250_RMOV_3B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_250_RMOV_3C_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_480_RMOV_1B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_480_RMOV_2B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPER_480_RMOV_3B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_480_RMOV_2A_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_480_RMOV_2D_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_480_RMOV_3D_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_4KV_SDB_3EA_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_4KV_SDB_B_35	5.64E-03	1.13E-03	5.64E-03
HFFA_BDOPERLATE_4KV_SDB_C_35	5.64E-03	1.13E-03	5.64E-03
HFFA_SDBD_DG_4KV_3EA_35	9.50E-03	1.90E-03	9.50E-03
HFFA_SDBD_DG_4KV_3EB_35	9.50E-03	1.90E-03	9.50E-03
HFFA_SDBD_DG_4KV_A_35	9.50E-03	1.90E-03	9.50E-03
HFFA_SDBD_DG_4KV_B_35	9.50E-03	1.90E-03	9.50E-03
HFFA_SDBD_DG_4KV_C_35	9.50E-03	1.90E-03	9.50E-03
HFFA0001HPRVD1	6.33E-04	1.27E-04	6.33E-04
HFFA0231480SDBTIE	2.14E-03	4.28E-04	2.14E-03
HFFA0SUPPHPI1	9.92E-03	1.98E-03	9.92E-03
HFFA0SBISO	2.22E-03	4.44E-04	2.22E-03
HFFA0ASD_HPMU2	6.22E-04	1.24E-04	6.22E-04
HFFA0071L8RESTART	8.07E-03	1.61E-03	8.07E-03
HFFA0073L8RESTART	5.70E-03	1.14E-03	5.70E-03
HFFA0HCIINIT30	1.50E-03	3.00E-04	1.50E-03

The FPRA model was requantified using the sensitivity HEPs and the results are provided in Table 5-29 as described above. The results show the key insight that the HEPs are driving the delta risk and demonstrates the importance of realism in determining which HEPs would be affected by the less demanding time windows available under CLTP conditions and in assessing HEP changes related to the timing changes.

Table 5-29. Results of the FPRA Sensitivity Study on HEPs							
		CDF per reactor yr			LERF per reactor yr		
Unit	**Sensitivity Case**	**CLTP**	**EPU**	**ΔCDF**	**CLTP**	**EPU**	**ΔLERF**
1	Detailed HEPs Reduced	4.23E-05	4.90E-05	6.66E-06	5.14E-06	6.65E-06	1.51E-06
2	Detailed HEPs Reduced	4.02E-05	4.85E-05	8.27E-06	4.56E-06	6.46E-06	1.90E-06
3	Detailed HEPs Reduced	4.66E-05	5.29E-05	6.31E-06	4.57E-06	5.54E-06	9.70E-07

5.7.2.2 Fire Scenario Frequencies Increased

Some large pumps may be drawing more power after the EPU. Potentially, larger lubricating oil reservoirs could sustain larger fires and thereby lead to increases in fire scenario frequencies as reflected in increased severity factors. For this sensitivity study, severity factors are increased to reflect increased scenario frequencies for large oil fires associated with the pumps that have increased capacities for EPU (Table 5-30).

Increasing the severity factor of a fire scenario, which is the fraction of fires that cause the target damage associated with the scenario, increases the scenario frequency proportionally. Scenario frequencies for the pump oil scenarios may be expected to increase somewhat as a result of somewhat larger pumps with proportionally larger lubricating oil reservoirs. Increasing scenario frequencies by a factor of 5 is expected to bound by a large margin any actual increases that may occur with somewhat larger pump oil reservoirs. An increase in severity factors by a factor of 5 did not increase the CDF or LERF in the EPU results beyond the reporting precision (three significant figures).

Table 5-30. Baseline Severity Factors for Fire Scenario Frequency Sensitivity				
Ignition Source Description	**Component ID**	**Large Oil Fire Scenario ID**	**Baseline Severity Factor**	**Sensitivity Severity Factor**
Condensate Pump 1A	1-PMP-002-0026	26-A.2737-PMP	2.30E-02	1.15E-01
Condensate Pump 1B	1-PMP-002-0021	26-A.2736-PMP	2.30E-02	1.15E-01
Condensate Pump 1C	1-PMP-002-0015	26-A.2734-PMP	2.30E-02	1.15E-01
Condensate Pump 2A	2-PMP-002-0026	26-A.2513-PMP	2.30E-02	1.15E-01
Condensate Pump 2B	2-PMP-002-0021	26-A.2512-PMP	2.30E-02	1.15E-01
Condensate Pump 2C	2-PMP-002-0015	26-A.2511-PMP	2.30E-02	1.15E-01
Condensate Pump 3A	3-PMP-002-0026	26-A.3049-PMP	2.30E-02	1.15E-01
Condensate Pump 3B	3-PMP-002-0021	26-A.3048-PMP	2.30E-02	1.15E-01

Condensate Pump 3C	3-PMP-002-0015	26-A.3047-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 1A	1-PMP-002-0056	26-A.2711-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 1B	1-PMP-002-0062	26-A.2712-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 1C	1-PMP-002-0068	26-A.2713-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 2A	2-PMP-002-0056	26-A.2878-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 2B	2-PMP-002-0062	26-A.2879-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 2C	2-PMP-002-0068	26-A.2880-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 3A	3-PMP-002-0056	26-A.3050-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 3B	3-PMP-002-0062	26-A.3051-PMP	2.30E-02	1.15E-01
Condensate Booster Pump 3C	3-PMP-002-0068	26-A.3052-PMP	2.30E-02	1.15E-01
Turbine Driven Feedwater Pump 1A	1-PMP-003-0015	26-A.2530-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 1B	1-PMP-003-0008	26-A.2529-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 1C	1-PMP-003-0001	26-A.2528-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 2A	2-PMP-003-0015	26-A.2533-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 2B	2-PMP-003-0008	26-A.2532-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 2C	2-PMP-003-0001	26-A.2531-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 3A	3-PMP-003-0015	26-A.2536-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 3B	3-PMP-003-0008	26-A.2535-MFP	3.03E-03	1.52E-02
Turbine Driven Feedwater Pump 3C	3-PMP-003-0001	26-A.2534-MFP	3.03E-03	1.52E-02
Reactor Recirculation Pump 1A	1-PMP-068-0060A	No fire scenarios	NA	NA
Reactor Recirculation Pump 1B	1-PMP-068-0060B	No fire scenarios	NA	NA
Reactor Recirculation Pump 2A	2-PMP-068-0060A	No fire scenarios	NA	NA
Reactor Recirculation Pump 2B	2-PMP-068-0060B	No fire scenarios	NA	NA
Reactor Recirculation Pump 3A	3-PMP-068-0060A	No fire scenarios	NA	NA
Reactor Recirculation Pump 3B	3-PMP-068-0060B	No fire scenarios	NA	NA

5.7.2.3 EHPM Pump Needed for ASDC under EPU Conditions

This sensitivity study begins with the assumption that the ASDC configuration under EPU conditions would fail due to reduced NPSH if containment isolation is lost. In this sensitivity study, the EHPM Pump is credited under EPU conditions to provide inventory to the suppression pool to overcome reduced NPSH when containment isolation is lost in the ASDC configuration. Thus, for the purposes of this sensitivity study, logic is added in the EPU fault tree that requires a loss of containment isolation and failure of the EHPM Pump to fail ASDC. The CLTP model does not need to be requantified for this sensitivity because the CLTP model is unchanged from the baseline. The results of this sensitivity study are presented in Table 5-31.

Table 5-31. Results of the FPRA Sensitivity Studies (EHPM Pump for ASDC)							
		CDF per reactor yr			LERF per reactor yr		
Unit	**Sensitivity Case**	**CLTP**	**EPU**	**ΔCDF**	**CLTP**	**EPU**	**ΔLERF**
1	EHPM Pump for ASDC	4.78E-05	5.02E-05	2.37E-06	6.05E-06	6.70E-06	6.52E-07
2	EHPM Pump for ASDC	4.72E-05	5.21E-05	4.83E-06	5.96E-06	6.48E-06	5.17E-07
3	EHPM Pump for ASDC	5.17E-05	5.66E-05	4.88E-06	5.16E-06	5.54E-06	3.85E-07

5.7.2.4 Revised Circuit Failure Probabilities

This sensitivity study is performed to investigate the effect of corrected circuit failure probabilities on risk results. F&O CF-A1-01 is a new finding from the May 2015 focused peer review. Revision 8 of calculation EDQ0009992012000110, Circuit Failure Mode Likelihood Analysis [Ref. 56] was developed to correct the modeling of the impacts of DCN 71214 and address the panel wiring and modification alignment issues identified in the finding. Updated circuit failure probabilities were developed accordingly. DCN 71214 electrically reconfigures the control circuits for RHR System valves to limit the likelihood of spurious operation. The new cables that will be installed per DCN 71214 will only contain a target conductor and will have a braided shield up to the cable termination points, which prevents the valves from spuriously opening. Only at the termination points where the braided shield does not provide protection can these new cables cause a spurious opening. The corrected circuit failure probabilities account for the fact that the braids are not effective at cable endpoints. The sensitivity study incorporates the corrected circuit failure probabilities from Ref. 56. The results of this sensitivity study are presented in Table 5-32.

Table 5-32. Results of the FPRA Sensitivity Studies (Revised Circuit Failure Probabilities)							
		CDF per reactor yr			LERF per reactor yr		
Unit	**Sensitivity Case**	**CLTP**	**EPU**	**ΔCDF**	**CLTP**	**EPU**	**ΔLERF**
1	Revised CF Probabilities	4.98E-05	5.11E-05	1.23E-06	6.08E-06	6.68E-06	6.05E-07
2	Revised CF Probabilities	5.62E-05	5.76E-05	1.39E-06	6.11E-06	6.62E-06	5.06E-07
3	Revised CF Probabilities	5.20E-05	5.32E-05	1.22E-06	5.16E-06	5.55E-06	3.85E-07

5.7.3 Key Uncertainties

Sources of uncertainty for the accident sequence (AS) model were considered and addressed in the development of the BFN PRA Model. The uncertainties identified in the EPRI report "Guideline for the Treatment of Uncertainties in Risk-Informed Applications" (Reference 29). A list of modeling uncertainties for the AS element and how they are addressed in the BFN PRA is included in Attachment I of Reference 19.

5.7.4 Uncertainties

The following areas of uncertainty can impact the CDF and LERF results. The impact is assessed by performing sensitivity analyses. Areas of uncertainty considered for this study follow:

1. The likelihood of some initiators may be impacted in the initial implementation of EPU break-in period.
2. The risk results are sensitive to the number of JHEPs and the minimum JHEP assumed. The industry group working on an approach to address the minimum JHEP assumed in the HRA dependency analysis. BFN currently uses a minimum joint HEP of 1E-7 since the BFN model includes a large number of operator actions and approximately 9000 JHEPs combinations.
3. The HRA analyses for BFN include several operator errors that use a screening value for the HEP. The use of screening HEP values may impact the results.
4. EPU conditions such as higher condensate, feedwater and steam flows may impact the likelihood of LOCAs.
5. EPU conditions such as higher flows may impact the likelihood of internal floods.
6. Un-isolated interfacing LOCAs (ISLOCAs) or breaks outside containment (BOCs) always result in core damage. (Assumption 2)
7. The dependency of equipment and systems on room cooling is believed to be conservative. It is considered an uncertainty; however, it is treated similar to other BWRs without detailed room heat-up and/or equipment failure analysis. Sensitivity analyses were not necessary to addresses the dependency on HVAC because room heat-up calculations have been performed to determine plant areas where HVAC is not required, and to reduce over-conservatisms.

This analysis includes several sensitivity evaluations to address these sources of uncertainty. The sensitivity analyses are discussed in Sections 5.7.1 and 5.7.2.

5.8 Summary of Conclusions

The internal events and FPRA models were used to assess the impact to CDF and LERF from implementation of EPU. The list of modifications in LAR Attachment 47 were reviewed to determine the impact to the key PRA elements. In addition, sensitivity studies using MAAP were performed to address impact on timing that could affect time critical operator actions. Based on these reviews, the primary impacts to the internal events were related to timing available to perform operator actions. The shorter delay times and system windows for operator actions impact the time available to recover by the operating crew and therefore increase the probabilities of cognitive human errors.

This analysis evaluated time critical operator actions to determine any changes to the human error probabilities. In addition, it is postulated that higher probability of SRV failure to close after it is demanded occurs due to EPU conditions. The increase in the likelihood of SORV is reasonable because the larger decay heat associated with EPU may require an increased number of demands to maintain and/or reduce reactor pressure during reactor cooldown after events that cause a plant trip.

5.8.1 Dominant Contributors to Change in Risk

Based on the results of this analysis, it is determined that the major contributor to total CDF risk (approximately 80%) for both CLTP and EPU come from fire scenarios. The major contributor to the change in CDF (approximately 82-91%) for both CLTP and EPU also comes from fire scenarios.

Based on the results of this analysis, it is determined that the major contributor to total LERF risk (approximately 75%) for both CLTP and EPU come from fire scenarios. The major contributor to the change in LERF (approximately 75%) for both CLTP and EPU also comes from fire scenarios.

The total change in risk for CDF and LERF is almost entirely due to the decreased time available to perform operator actions, which results in higher HEPs and JHEPs. The remainder of the risk increase is due to the higher probabilities of one or more stuck open relief valves. The increased probability of one or more stuck open relief valves is because it is projected that the existing valves will require additional cycling to remove the additional post-trip energy resulting in more opportunities to stick open (i.e., the valves are not less reliable due to the transition to EPU).

The risks from internal events, external events and fires must be added to arrive at the aggregated risk for EPU conditions. Section 5.6 provides the total risks associated with EPU. The aggregated risks are shown relative to the acceptance guidelines given in Regulatory Guide 1.174 (Reference 1) and the change in risk due to EPU relative to the aggregate baseline risk for all three BFN units is small (Region II).

5.8.2 Key Insights

The following sections discuss the key insights from the individual contributors to risk.

Internal Events

A review of the results indicates that accident sequences involving loss of inventory makeup in which the reactor pressure remains high are the major contributors to risk. The operator action combinations in these cutsets involve failure to control level with high pressure systems combined with failure to depressurize.

Fire Risk

The key insights for the top ten fire scenarios contributing to delta risk for CDF and LERF for Units 1, 2, and 3 are identified in Table 5-9 through Table 5-14. Because the model logic is the same for CLTP and EPU, the changes in risk are attributable to basic event probability changes. CLTP and EPU cutsets from the fire scenarios listed in Table 8 through Table 13 were reviewed and dominant accident sequences are described in each table. Among those HEPs that change going from CLTP to EPU, major HFE contributors to delta risk were noted and are listed in the table footnotes.

The HFEs that were found by the cutset review described to be major contributors are the following:

- HFFA0002RPV_LVL OPERATOR FAILS TO MAINTAIN RPV LEVEL (fire)
- HFFA0ASD_RCIC OPERATOR FAILS TO START RCIC
- HFFA0268480CRSTIE FAILURE TO TRANSFER DEENERGIZED 480V BOARD TO ALTERNATE SUPPLY (FIRE)
- HFFA0SUPPHPI2 OPERATOR FAILS TO INITIATE SUPPLEMENTAL INJECTION IN 35 MIN
- HFFA0RHRCS_LPP OPER FAILS TO BYPASS ECCS LOW PRESSURE PERMISSIVE

Consideration of the actions listed above can provide insight into overall fire risk with regard to EPU changes and what might be done to reduce the delta risk associated with those changes. The two actions from the list above (1) HFFA0002RPV_LVL and (4) HFFASUPPHPI2 represent the establishment and maintenance of RPV water level for nonabandonment scenarios. The action HFFA0002RPV_LVL applies only to maintaining level using the condensate system and the action HFFASUPPHPI2 is the action to initiate the proposed emergency high pressure makeup (EHPM) pump. In general, the importance of these actions demonstrates the importance of establishing an initial injection source that is not dependent on systems in the reactor building. The systems meeting this criterion and credited in the fire PRA are the condensate system and the EHPM system. The action to initiate the EHPM pump is a simple action on a proposed system. The HRA for this action has been developed in as much detail as possible with existing information. The use of condensate as an initial injection source is a current focus of a fire related plant modification. Existing EOIs preclude the use of condensate as an initial injection source due to the long time required to establish it. The fire PRA credits condensate in a flood-up mode where condensate would automatically inject whenever the RPV is depressurized. Currently, EOI or fire procedure changes are being considered that would allow condensate to be quickly initialized in a manner similar to flood-up but with RPV level control. It is anticipated that these changes, when finalized, will further reduce delta risk for this application.

The two actions (3) HFFA0268480CRSTIE and (5) HFFA0RHRCS_LPP represent the establishment and maintenance of the LPCI injection path. The action HFFA0268480CRSTIE isolates the 480 V AC board from fire impacts and repowers it from an alternate source. The time available for this action depends on when initial injection must be established to prevent core damage. This is determined from MAAP analysis and is consistent with all other actions necessary for initial injection. The execution time is taken from Appendix R training data and represents measured operator performance. The calculated HEP is reasonable for this type of action. The action HFFA0RHRCS_LPP bypasses the LPCI injection valve low pressure permissive interlock if the low pressure permissive is failed due to fire impact. This is a simple action in the main control room and the execution timing is relatively simple and short. The HEP for this action is driven by the short time available to perform it. While operators have about 35 minutes to establish initial injection with no open SRVs, they will not know that the permissive is failed until after they depressurize and attempt to open the injection valve. At that time they have only about 15 minutes before core damage occurs. The calculated HEP is reasonable for this type of action.

Finally, the action to start RCIC, (2) HFFA0ASD_RCIC, is an action performed after MCR abandonment. The action is important because RCIC would usually be the initial injection source after abandonment. It is governed by the abandonment fire procedure and its execution involves multiple well defined steps. The timing and HEP is reasonable for this type of action under these conditions.

Seismic Risk

Based on a review of the BFN IPEEE, the conclusions of the seismic margins assessment (SMA) will be unaffected by the EPU. EPU has little or no impact on the seismic qualifications of the SSCs. Specifically, EPU 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.

Other External Hazards

Based on review of the BFN IPEEE, EPU has no significant impact on the plant risk profile associated with tornadoes, external floods, transportation accidents, and other external hazards. Refer to Section 4.7 of this Attachment for further discussion.

Shutdown Risk

The impact of the 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., relatively 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 of this report for further discussion which indicate that the EPU is assessed to have a non-significant impact (delta CDF of approximately 1% of shutdown CDF).

6.0 References

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Appendix A – PRA Technical Adequacy

The guidance provided in Regulatory Guide 1.200, Revision 2 (Reference 9), "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 update summary and to address the requirements identified in Item 4 above.

A.1 Internal Events PRA Technical Adequacy

The BFN Internal Events PRA was subjected to three peer reviews – a full scope review, a focused scope follow on peer review for internal flooding, and a focused scope peer review to evaluate specific aspects of the Internal Events PRA and assess existing F&O dispositions. All peer reviews used the process defined in NEI 05-04, Revision 1 (Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard), ASME/ANS RA-Sa-2009, and Regulatory Guide 1.200, Revision 2. The initial Peer Review for BFN Units 1, 2, and 3 Internal Events PRA was performed in May,

2009. A separate review was performed for the Internal Flooding portion of the BFN PRA in September, 2009. The 2015 focused scope peer review evaluated specific changes made to the Internal Events PRA (excluding internal flooding) and assessed some F&O's from the previous peer review that were considered closed by TVA self review. Thirty-seven of the existing F&Os were not addressed in the focused scope peer review.

A team of independent PRA experts from nuclear utility groups and PRA consulting organizations carried out these Peer Review Certifications.

The purpose of these reviews was to provide a method for establishing the technical adequacy of the BFN PRA for the spectrum of potential risk-informed plant licensing applications for which the BFN PRA may be used. These reviews provided a full-scope review of the Technical Elements of the internal events and internal flooding for at-power conditions. There have been no changes made to the internal events model following these peer reviews that would constitute an upgrade and, thus, does not require another focused scope peer review.

The Peer Review Certification of the BFN PRA model performed in May 2009, September 2009, and July 2015 resulted in a total of 78 open findings for the three unit model for internal events and internal flooding.

All findings from these assessments have been dispositioned. The certification team determined that with these proposed changes incorporated, the quality of all elements of the BFN PRA model is sufficient to support "risk significant evaluations with deterministic input." As a result of the effort to incorporate the latest industry insights into the BFN PRA model upgrades and certification peer reviews, TVA has concluded that the results of the risk evaluation are technically sound and consistent with the expectations for PRA quality set forth in Regulatory Guide 1.174.

Complete results of the peer reviews and the applicability to the EPU application in Table A-1.

<u>Summary of Updates to the Internal Events PRA Model Since the Peer Reviews</u>

1. Rev. 0 - Initial CAFTA model issued after August 2009 peer review
2. Rev. 1 - Initiating events were updated to include current generic data, recent plant events and multi-unit initiators. Fire initiators that fail offsite power were added to the model to assess the Diesel Generator Allowed Outage Time Extension. Some logic errors and type code errors were also corrected that were identified from the Revision 0 to Revision 1 model.
3. Rev. 2 - Initiators %VR and %VS were added for all units. The human error probability for HFA_0085ALIGNCST was re-evaluated based on an additional MAAP run performed. A design change was incorporated into the model that requires three air compressors to supply the entire plant instead of all four. Some logic errors were also corrected that were identified from the Revision 1 to Revision 2 model.
4. Rev. 3 - The Fire initiators used to assess the Diesel Generator Allowed Outage Time Extension were removed from Revision 3 of the model. Some logic errors and type code errors were also corrected that were identified from the Revision 2 to Revision 3 model.

5. Rev. 4 - Changes were made from the Revision 3 to Revision 4 model to support increased unavailability for infrequent maintenance performed on the Emergency Diesel Generators and corrections to logic errors and type code errors found.
6. Rev. 5 - The major change in this update was to revise the data, and mutually exclusive logic. Changes were made in the Revision 5 model to correct errors in the logic noted during review following the issuing of the Revision 4 documentation and to support increased unavailability for infrequent maintenance being performed on the Emergency Diesel Generators. The data in the PRA model was updated for plant specific failures and successes through January 1, 2012. There were no changes in the Accident Analysis, Success Criteria, Internal Flooding, or LERF Analysis, from Revision 4 to Revision 5. (Reference 4)
    - The initiating event analysis has been updated to include initiating event data through January 1, 2012 to include current industry generic data, recent plant events and multi-unit initiators.
    - Changes were made in the Revision 5 model to correct errors in the logic noted during review following the issuing of the Revision 4 documentation.
    - The unreliability, unavailability, and common cause data analyses were updated. The unreliability (or failure rate) data are based on generic industry data that has undergone Bayesian updating with plant specific data. Plant specific data for the period 1/1/2003 to 1/1/2012 was evaluated and used as input to the Bayesian analysis. Plant maintenance unavailability data is based on the same time period as the failure data, 1/1/2003 to 1/1/2012. Generic industry data from NUREG/CR-6928 was used for components for which no plant specific data was available.
7. Rev. 6 - A model update was performed to merge the Internal Events PRA and the FPRA into a single model, to improve the event tree logic, to resolve issues for AC and DC power. A brief overview of these changes is included in the bullets shown below:
    - Event Tree changes to credit RCIC for IOOV scenarios
    - Event Tree changes to separate the DHR functional top logic in a more logical manner (HWWV and DWV, Drywell Sprays)
    - Event Tree changes to incorporate ASDC for the FPRA
    - Event Tree changes to incorporate the EHPM for the FPRA
    - Logic fault tree changes to address NPSH w/o containment accident pressure
    - Correct the logic for DC chargers (OR gate is now used between the batteries and chargers to address charger trips due to voltage swings caused by inrush current of large loads.
    - Logic fault tree changes to address overload and load shed logic
    - Logic changes to address preferred pump logic (PPL)
    - Logic changes to address diesel paralleling logic
    - Logic changes to address conditional LOOP logic for multi-unit initiators
    - Limited enhancement for LOOP recovery
    - Develop recoveries for MSL BOC instrumentation
    - Updated RCW logic

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
AS-B3 SY-B14	Closed	1-6	The sequence descriptions generally include a description of the sequences but the phenomenological conditions created are not specifically identified. Some references to phenomenology are provided but not consistently (e.g., ATWS sequence descriptions conclude with the statement "There no phenomenological conditions identified.")	**Basis for Significance:** The SR calls for identification of the phenomenological conditions for each sequence. **Possible Resolution:** Include a listing of phenomenological conditions that result for each sequence.	The phenomenology is discussed in the ATWS sequence descriptions. The statement “There no phenomenological conditions identified” was removed from the TVA Calculation, NDN00099920070036 Revision 0, “AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis.” In addition, other phenomena are discussed as noted below: Loss of suction due to venting is discussed in TVA Calculation, NDN00099920070036, “AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis”, Section 6.2.2. Harsh environment is discussed in TVA Calculation, NDN00099920070036 Revision 0, “AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis”, Section 6.2.4.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SC-B1 SY-B6 SY-B7	Closed	1-12	Several examples found for lack of engineering analyses regarding HVAC System that could be justified by calcs. Condensate System Notebook (SY.01) assumes active ventilation is not required due to plant experience. Core Spray System Notebook (SY.04) assumes keep-fill system is not required. HPCI System Notebook (SY.07) assumes dependence on quad cooling for the remaining 20 hours of post accident operation.	**Basis for Significance**: The SR expects that engineering analyses will be performed to determine whether these statements are correct. **Possible Resolution**: Perform analyses to validate these statements.	Keep fill systems are monitored daily by operations. They are alarmed so failures of these systems are detected and corrected in a timely manner. Based on this, an assumption is made that these systems are properly charged with water at the time of an initiator. Based on operator interviews, no system has leakage great enough to create a water hammer condition should its keep fill system fail after the scram. The only exception to this is the potential drain down of the RHR loop if it is being used for SPC and LOOP occurs. This condition is modeled and discussed in calculation NDN-000-074-2007-0025 Revision 4, "SY.19 – BFN Probabilistic Risk Assessment – Residual Heat Removal." Calculations are not needed for these systems. The assumptions section for each applicable SY notebook reflects the discussion above. A consensus model is not available to guide the HVAC System dependency issues. The intent of SY-B6 is to make sure adequate analysis exists to support removing modeled dependencies from systems. It is not the intent of SY-B6, or the ASME standard for that matter, to establish what analysis is needed to support plant operations and design. In the case of HVAC System, adequate plant specific analysis is not available to remove room cooling dependencies from most equipment. Room heat-up calculations may be available, but realistic (non-EQ) equipment failure temperatures are not available. This situation is shared by many plants in the industry. The BFN model took the conservative approach and required an HVAC System dependency for all equipment that could not be reasonably argued to not have the dependency. Since that time, room heatup calculations have been performed which resulted in the removal of many of the HVAC System dependencies. There are still HVAC systems required for the RHR and Core Spray pump rooms and the main control room. The condensate and condensate booster pumps are not located in a room. They are in a long corridor that is continually open to the turbine building environment. These pumps have cooling air from fans ducted directly onto the pumps. The system engineers and	No Impact. The CLTP and EPU PRA includes the same HVAC System dependencies as the internal events model. With the PSC Head Tank volume available, Calculation MDN099920110021 Rev. 0 concludes that the PSC system pressure would remain adequate to prevent ECCS line voiding for at least 6 hours after PSC pump failure on Units 1 and 3. Unit 2 can only be shown to last about 2½ hours after PSC pump failure. The operators are trained to check header pressure before manually starting an ECCS pump. Therefore, it is assumed that, after 2 1/2 hours, the operators will have established positive control over the ECCS pumps and will have properly started them. Hence, spurious Primary Containment Isolation System (PCIS) signals isolating the PSC Head Tank pump system is not modeled.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
					operators were interviewed and stated plant operational experience showed these pumps would operate for an extended length of time without that forced cooling. (NDN-000-002-2007-0008 Revision 2, “SY.01 – BFN Probabilistic Risk Assessment – Condensate System”). They concluded the pumps would survive for 24 hours without forced cooling. This conclusion was based on a qualitative analysis that included plant walkdowns, expert opinion from both operators and engineers, and past plant operating experience.	
DA-C6	Closed	1-17	Reviewed DA.01. The source of demands is not discussed. Based upon discussions with the PRA staff, exposure is collected directly from plant data systems and is therefore actual component exposure. However, post-maintenance testing demands are also included in these numbers and are not removed.	**Basis for Significance**: Post-maintenance testing must be excluded from the exposure data per the SR. **Possible Resolution**: Develop a means of identifying the post-maintenance related exposure and remove them from the data calculations.	As it stands the ability to remove post maintenance testing (PMT) from the database would require a massive re-tool of the database to allow for discrete removal of specific times. The ability to perform these actions is limited due to the lack of interface between the Operations Logs and the Plant Equipment Display System (PEDS). To quantify the amount of effect removal of potential PMT would have on the results, seven scenarios were analyzed with the CDF & LERF for each unit and compared This review is documented by PRA Evaluation BFN-0-15-079. The results show that even with an extremely unrealistic number of PMTs the data is not significantly skewed by the inclusion of the PMT data.	No impact. The CLTP and EPU PRA uses the same data for random failures. However, the SORV probability has been increased in the EPU model to address a higher number of demands caused by higher decay heat.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
DA-C10	Closed	1-22	There is no discussion of the process to be applied in the use of surveillance test data. The use of this data is required for situations in which there is no MR data available (for example), so a process for its use should be in place.	**Basis for Significance**: All levels of capability in this SR indicate that the process for use of surveillance data needs to possess specific attributes. There is no process defined. **Possible Resolution**: Provide a process for use of surveillance data that incorporates the requirements of this SR.	A description of the process to be applied in the use of surveillance test data has been incorporated in calculation NDN-000-999-2007-0033 Revision 7, "DA.01 – BFN Probabilistic Risk Assessment - Data Analysis."	No impact. The CLTP and EPU PRA uses the same data for random failures..
LE-F2	Closed	1-33	There is no discussion of the review of the LERF contributors (ASME/ANS RA-Sa-2009 Table 2-2.8-9) for reasonableness per the review of the QU Notebook and LE.01.	LE-F2 is related to this F&O. The SR is NOT met. **Basis for Significance**: A review of the reasonableness of the results of the analysis of the contributors to LERF is required per the SR. **Possible Resolution**: Perform and document a review of the reasonableness of the contributors to LERF.	The review of the LERF contributors (ASME/ANS RA-Sa-2009 Table 2-2.8-9) for reasonableness was performed as discussed in calculation NDN-000-999-2007-0041 Revision 6, "QU – BFN Probabilistic Risk Assessment – Quantification."	No impact. The CLTP and EPU PRA uses the same LERF model.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
QU-D2	Closed	1-34	Additional attention should be applied to significant cutsets to determine that the bases for the cutsets are consistent with modeling and operating philosophies.	**Basis for Significance**: The top accident sequence cutset for both CDF and LERF deals with clogging of the intake and includes events that are very uncertain. The attention given this cutset to minimize the uncertainty associated with the contributing basic events has not been sufficient. The approach to dealing with such important cutsets should assure that the contributors are understood and are supported by appropriate rigorous analyses and/or assessment. **Possible Resolution**: Make sure that the top cutsets (reviewed per the PRA Procedures) are discussed and evaluated. During the quantification process make sure that an evaluation is performed in addition to capturing the results.	The BFN Internal Events PRA has undergone six revisions since the 2009 peer review. It has been subjected to several cutset reviews. The intake structure model has been modified and clogging of the intake structure is no longer a significant contributor to risk. The review of the significant accident sequences is documented in calculation NDN-000-999-2007-0041 Revision 6, “BFN Probabilistic Risk Assessment – Quantification.”	No impact. The MOR has been updated and results reviewed and documented based on model control process (NEDP-26). The EPU project uses MOR Rev. 6 as the starting point for the analyses.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
HR-D5 HR-C3	Closed	2-14	HFL_1003_CCFT0056 is Common cause miscalibration of all 4 level transmitters, inspection of the fault tree shows that specific pairs of failures (AC, BD) would also cause a failure to initiate the logic. These CCF pairs should be added to the model. This will apply to other miscalibration CCFs also.	**Basis for Significance**: The pair CCFs will have a higher value than the 4 of 4 event thus impact the results. **Possible Resolution**: Calculate the pair CCFs and add to the fault tree	The F&O relates to all of the pre-initiators that accounted for common miscalibration errors. Fault trees have been updated and calculation NDN-000-999-2007-0032 Rev. 4, "HR – BFN Probabilistic Risk Assessment - Human Reliability Analysis" has been revised to reflect this change. HFL_1003_LT56A, HFL_1003_LT56B, HFL_1003_LT56C, and HFL_1003_LT56D have been added to the model.	No impact. The pre-initiators are modeled in the same manner for the CLTP and EPU
SY-A3	Closed	2-23	In section 3.2.6.1 of the HVAC-system notebook, it states that the running ACU for Unit 3 electric boards must be tripped before the standby unit can be started. Failure of this trip to occur is not reflected in the fault tree.	Given Priority 2 because model change may be required. **Basis for Significance** A breaker failing to provide tripped indication for a start permissive can happen and this failure mode should be included. **Possible Resolution**: Include running ACU fail to trip (indicate as tripped) as a start failure for the standby ACU.	Failure of the operating ACU to trip has been added to the model as a failure mode of the standby ACU. Since the original resolution of this F&O was developed, room heatup calculations have been developed which shows the HVAC System is not required for the Unit 3 Electric Board Rooms. The model was modified to add a switch that removes the HVAC System dependency from the Unit 3 4kV electric boards.	No impact. SR SY-A3 was assessed as MET for Category 2 in the May 2009 Peer Review. The HVAC System dependencies are modeled the same way in both the CLTP and EPU PRA models.
SY-A5 SY-A13	Closed	2-31	For SPC and LPCI, the LPCI injection valves and SPC return valves are required to reposition when swapping RHR modes, but this is not included in the model. The RHR system notebook indicates that these valves need to close for the opposite function. However in one location in the notebook it is indicated that flow	Priority 2 is given because of the potential for model changes. **Basis for Significance**: All active components should be included in the failure modes of a system.	The injection valves need to change position for split LPCI/SPC flow; two valves would have to fail to cycle or close in either path to fail either system. An operator interview was conducted to discuss the method of modeling LPCI and SPC in the transient event tree and a concern that either LPCI injection valves or SPC torus supply valves would have to close if LPCI and SPC both had to be successful on the same RHR loop. The following was concluded from	No Impact. SRs SY-A5 and SY-A13 were assessed as MET for Category 2 in the May 2009 Peer Review. The BFN CLTP and EPU models require an initial MOV opening for success of both RHR injection and RHR

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			can be split between LPCI and SPC.	**Possible Resolution**: Add failure mode to the fault trees and clarify documentation	this conversation: 1. If either LPCI or SPC did have to be isolated, then two MOV's on either system would have to fail to close in order to fail to isolate the system. Each valve has a different power supply. 2. The operators may either initiate LPCI to fill up the vessel and then shut the pump off, or they may use just enough LPCI injection to maintain level and divert the rest back to the suppression pool. The only difference in the latter mode and the mode in question is the use of the RHR heat exchangers. 3. The normally open LPCI injection valve FCV-74-52(66) would have to modulate in order to allow adequate SPC flow to prevent pump run out, or the normally closed LPCI injection valve FCV-74-53(67) would have to fail to modulate. The common cause failure probability of two MOV's to close is less than 1E-5. The RHR pump start failure probability is approximately 1.4E-3. The failure of two MOV's to close is less than 2 orders of magnitude lower than another failure that would fail the system in a similar manner. Therefore, failure to close (or cycle) either the LPCI or SPC injection path can be neglected. Calculation NDN-000-074-2007-0025 Revision 3, "SY.19 – BFN Probabilistic Risk Assessment – Residual Heat Removal" was modified to reflect this and the operator interview was added.	operation for SPC. As stated in NUREG 6823, the standby failure model concludes that demands soon after a successful test (demand) have smaller probability of failure than those demands that occur during normal operations. The NUREG also suggests that the stand by failure rate model is most appropriate for MOV failures. Since the BFN probability of a MOV failing to open is about 2.4E-4/demand, a smaller probability for subsequent valve openings would not have a significant impact on the PRA results.
LE-D1	Closed	2-35	The containment structural analysis does not address the Unit 3 primary containment ultimate capacity in section 6.3.	**Basis for Significance**: All three unit containments must be addressed. **Possible Resolution**: Address the Unit 3 containment ultimate capability.	Calculation NDN-000-999-2007-0037 Revision 3, "LE.01 – BFN Probabilistic Risk Assessment - LERF Analysis" calculations are applicable to all three BFN units. However, much of the previous work, including industry studies, has been based on BFN Unit 1. Thus, the plant description in the LERF analysis NDN-000-999- that specifically applies to Unit 1 is supplemented with a discussion of unit differences. The unit differences are examined from the perspective of LERF and it is concluded that the minor differences between the units do not impact the LERF quantification.	No impact. SR LE-D1 was assessed as MET for Category 2 in the May 2009 Peer Review. The CLTP and EPU PRA use the same LERF models.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-A7	Closed	3-7	Scheduled manual shutdowns (especially for refueling outages) should not be included in the statistical basis for the scram initiator. This can lead to an overly conservative scram initiator frequency. Note that CNRM interpretation for FAQ 06-1060 (should non-forced manual trips which are part of the normal shutdown procedure be counted) states that 'a normal controlled shutdown would not present the same challenges as a trip from full power if the manual trip was prompted by conditions other than the normal shutdown procedure which could occur at full power, it should be counted.	IE-A7 is related to this F&O. SR is met. **Basis for Significance** CRNM ASME Standard Interpretation #5 (for FAQ 06-1060) states that normal controlled shutdowns should not be included when counting initiating events. The current practice at Browns Ferry regarding this item, therefore, does not meet the requirements of the standard. **Possible Resolution** Remove planned shutdowns from the SCRAM initiator data set.	Calculation NDN-000-999-2007-0030 Revision 2, “IE.01 - BFN Probabilistic Risk Assessment - Initiating Event Analysis,” describes treatment of manual shutdowns which have been conservatively consolidated with automatic reactor scrams. There are no identifiable plant response differences between automatic and manual shutdown above low power situations. Low power manual shutdowns will be included in the Low Power/Shutdown PRA at a future date.	No impact. SR IE-A7 was assessed as MET for Category 2 in the May 2009 Peer Review.
HR-I3 IE-D3 LE-F3 SY-C3 SC-C3 QU-E2 QU-E4 QU-F4 QU-E1 DA-E3	Closed	3-10	Modeling uncertainty comes from two general types of issues, plant specific and generic. Plant specific uncertainties and assumptions should be identified and documented during the model development. The generic sources of uncertainty are listed in EPRI Report 1016737 Table A-1. Both types of uncertainties must be addressed for the base model. Examples of plant specific uncertainties include: (1) ISLOCA valve failing to close after testing is not listed in the sources of uncertainty, nor is the	**Basis for Significance** Sources of uncertainty must be identified and documented. **Possible Resolution** NUREG-1855 and EPRI 1016737 provide an acceptable approach to identifying, documenting and characterizing sources of uncertainty. Use this method or a similar method.	Identified sources of uncertainty are documented in Table A8-1 of calculation NDN-000-999-2007-0041 Revision 6, “QU – BFN Probabilistic Risk Assessment - Quantification” per SR QU-E1 and QU-E2 of ASME RA-Sa 2009 Addendum B. Key modeling uncertainties (e.g., HVAC System dependencies and intake structure plugging) are addressed in the Quantification calculation in the various accident sequence contribution discussions. The requirements and procedures for characterizing generic and plant-specific modeling uncertainties are specified in SR QU-E4 of ASME-ANS RA-Sa 2009, RG 1.200, Revision 2, NUREG 1855, and EPRI-1016737. These requirements and procedures were formalized shortly before the 2009 peer review for BFN. The additional requirements for ASME-ANS RA-S 2009 are	No Impact. Parametric, model, and completeness uncertainties are addressed in Section A.3.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			conditional probability that the break is greater than 93 or 600 gpm. (2) For Initiating Events, the factors affecting INTAKE initiating event is not included in the assumptions section, nor are any of the other assumptions in the analysis. (3) Specific assumptions for the detailed HFEs is not discussed, including assumptions made for timing of operator responses (versus analyzed or those observed on a simulator).		documented in revision 6 of the BFN PRA model.	
SC-A5	Closed	3-12	There is no evidence of an analysis for sequences that go beyond the 24 hour period to evaluate the appropriate treatment relative to the CC II/III requirements for SC-A5.	**Basis for Significance**: A CC II/III for SC-A5 requires that options other than assuming sequences in which a stable state has not been reached in 24 hours goes to core damage. **Possible Resolution:** Perform and document an analysis of sequences that do not achieve a stable state in 24 hours to determine which of the options presented in the SR would be a most appropriate disposition for that sequence. Then change the PRA model accordingly.	General Transient sequence GTRAN_S002 is a non-Inadvertent opening of a relief valve/stuck open relief valve (IORV/SORV) success sequence with successful SPC and long term HPCI or RCIC. MAAP (Modular Accident Analysis Program) analysis show HPCI and RCIC can be successful for greater than 24 hours with effective SPC. Drywell temperature, however, increases throughout this sequence due to heat transfer from the vessel and drywell piping (drywell fan coil units are not credited). MAAP analysis shows drywell temperature increases to, but does not surpass, 300 °F within a 36 hour analysis time duration. The EOI's require the operators to emergency depressurize when drywell temperature reaches 281 °F. This will fail HPCI and RCIC and prevent further high pressure injection. This sequence was analyzed by interviews with operators and review of other non-MAAP analysis to determine 1) if the operators would emergency depressurize if there were no low pressure injection sources available, and 2) if the MAAP analysis was reasonable. Operator interviews determined that the operators would emergency depressurize when instructed by the EOI's even if no low pressure injection systems were available. A review of General Electric calculation W79 040331 003 confirmed the conclusions drawn from the	No Impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
					MAAP results. As a result of the above analysis, the sequence was changed to require successful low pressure injection for sequence success.	
QU-F2	Closed	3-28	A detailed discussion of the quantification asymmetries (with respect to different units, system alignments, etc.) is not presented.	**Basis for Significance**: This is an important part of the quantification documentation process. **Possible Resolution**: A detailed discussion of the quantification asymmetries (with respect to different units, system alignments, etc.) should be presented in the Quantification Notebook.	Calculation NDN-000-999-2007-0041 Revision 6, “QU – BFN Probabilistic Risk Assessment - Quantification” documents unit differences that impact the quantification results. This calculation documents the quantification of all three BFN units. Unit differences are explicitly addressed in the system fault tree models. Some unit differences have a significant impact on the quantification results as follows: 1. The HVAC dependencies on electrical boards have a significant impact on the results. The Units 1 and 2 electrical boards are cooled by air conditioning units that depend on chillers. The Units 3 electrical boards are cooled by air conditioning units that depend on EECW. 2. The USSTs supply power to the 4-kV shutdown boards via the unit boards and shutdown buses (Units 1 and 2 only; there is no shutdown bus for Unit 3). This allows Unit 1 or 2 to experience a single unit LOOP and still utilize the shutdown buses to power the respective 4kV shutdown boards.	No impact. The asymmetries are modeled in the same manner for the CLTP and EPU.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
QU-D6 QU-F3 QU-F6 LE-G6	Closed	3-31	The definitions for significant when presenting lists of important equipment, operator actions, etc. do not always conform to the strict ASME standard definition of significant. Justifications for the alternatives used are not presented.	**Basis for Significance**: This issue causes the supporting requirement QU-F6 not to be met. **Possible Resolution**: When presenting lists of significant equipment strictly adhere to the ASME standard definition or present a rationale for using an alternative.	Calculation NDN-000-999-2007-0041 Revision 6, "QU - BFN Probabilistic Risk Assessment – Quantification" documents the significance criteria for important equipment and operator actions.	No impact. Documentation Issue only.
HR-G2	Closed	4-18	Some operator actions assume that the execution failure probability (Pe) is 0.0. Example 1: Several operator actions for ATWS scenarios (e.g., HFA_1063SLCINJECT: Failure to SLC in response to an ATWS event) assume the execution failure probability (Pe) is 0.0. Example 2: Operator action HFA_0024RCWINTAKE (Failure to clear debris at intake before reactor scram) assumes an execution error of 0.0 based on the following: 'Cleaning traveling screens does not relate to a series of manual actions, but to an effort among several operators. It is assumed that, if the action is initiated within 1 hr, it will be successful.' The same rationale is provided for no execution error in HFA_0027INTAKE.	**Basis for Significance** Execution failure is a required part of the HEP calculation, and the argument for ignoring execution failure is not necessarily compelling, especially for maintaining level (HFA_0_ATWSLEVEL). Some of the actions for which Pe is not considered are important to the overall results. Note 1: The explanation given for no execution failure for HFA_0_ATWSLEVEL describes the actions required for starting SLC (HFA_1063SLCINJECT) Note 2: Cleaning debris from traveling screens is not a simple action,	In general, errors of omission in execution were not modeled when execution entails a single action. Skipping the step in the procedure is accounted for in the cognitive portion; for a single execution step, it is non-sensical to say that the step is not skipped, but that the execution is not performed. There still could be a commission error in execution (that is, trying to implement the action but doing it wrong), even if there is a single execution step. With respect to the events listed by the reviewer, they are documented in the HRA Calculator files, with exception to HFA_0024IFISOL which had been updated to include execution errors.	No impact. SR HR-G2 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
				an assumption, that if the actions are started they are guaranteed to be completed in 1 hour, is not justified. **Possible Resolution** Include Pe in the quantification of HFA_1063SLCINJECT, HFA_0_ADSINHIBIT, HFA_0_ATWSLEVEL, HFA_0024RCWINTAKE and HFA_0027INTAKE. Insure that execution errors are considered appropriately in other HEPs, as well.		
QU-C2 HR-I3 HR-G7	Closed	4-21	The joint HEP for several combined operator actions are too low and cannot be justified. Specifically, three combined actions have joint HEPs of less than 1E-7, and eight are less than 1E-6. Note that the HRA acknowledges these low combined HEPs, but does not enforce any lower bound. Further, it states that a sensitivity will be performed in the Quantification Notebook, but none is performed.	**Basis for Significance**: If the joint HEP for combined events is too low, sequence and overall results may be artificially lowered, and the importance of the operator actions may be understated. **Possible Resolution**: Establish a reasonable lower bound for combined HFE probabilities. Perform sensitivities to determine the significance of this lower bound.	Section 5.3.3.6 of NUREG -1792 states that the total combined probability of all the HFEs in the same accident sequence/cutset should not be less than a justified value. A suggested floor value of 1.0E-05 is provided based on potential dependent failure modes that are not typically treated. The HRA Calculator provides the capability to explicitly calculate the joint probability of dependent and independent post-initiator HFEs in the same accident sequence/cutset: This methodology improvement reduces the need for a threshold value. Overly conservative threshold values have the potential for skewing the results. The MOR uses a floor value of 1.0E-07 because of the large number of independent operator errors and associated combinations skew the results in a conservative direction.	No impact. The MOR uses a minimum joint HEP threshold of 1E-7 because the large number of independent operator errors and associated combinations skew the results conservatively. Sensitivity evaluations using different thresholds are included in Section 5.7 of this attachment.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
HR-F2 HR-G4 HR-G5	Closed	4-25	There are many operator actions that use screening values; see Table 8 of the HRA. None of these actions appear to use any information to base the time available and the times to operator cues and perform the actions are not documented.	**Basis for Significance** Without any real timing information, it is not possible to estimate, even at a screening level, the probability of operator failure or success. **Possible Resolution**: Provide timing information for all operator actions, including those HEPs estimated by using screening values.	HFEs have been reviewed and detailed analyses have been performed for many HFEs that previously used screening values. All significant HFEs have detailed analyses. In addition, timing analyses have been reviewed. Timing is based primarily on plant specific MAAP calculations, timing from BFN simulator exercises, or estimates from BFN operator interviews. In response to this comment, updated timing analyses have been re-reviewed by BFN operations staff and additional changes have been incorporated. All model changes are included in calculation NDN-000-999-NDN-000-999-2007-0032 Rev. 4, “HR – BFN Probabilistic Risk Assessment - Human Reliability Analysis.”	No impact. The CLTP and EPU PRA were updated to credit no more than a single screening value in a combination.
HR-C1	Closed	4-28	Non-screened miscalibration events are not provided with designators in Appendix A of the HRA. Thus HFEs associated with these miscalibration events cannot be readily determined.	**Basis for Significance**: The requirements of HR-C1 cannot be verified due to lack of traceability from HRA Appendix A table to the rest of the pre-initiator analysis. **Possible Resolution**: For miscalibration events, provide traceability from Table A of the HRA to the remainder of the pre-initiator analysis and the PRA model.	Calculation NDN-000-999-2007-0032 Revision 4, “BFN Probabilistic Risk Assessment - Human Reliability Analysis,” Appendix A “Screening of Routine Procedures for Relevance to Pre-Initiator Human Failure Events” had been updated, subsequent to the 2009 peer-review to include the designators for the non-screened miscalibration events to provide traceability.	No impact. SR HR-C1 was assessed as MET for Category 2 in the July 2015 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
HR-A1 HR-A2	Closed	4-29	The list of activities reviewed in the HRA Appendix A table is primarily focused on Unit 2 or Unit 0 SRs and SIs. There are a few Unit 1 procedures listed, but it is not clear why certain procedures from Unit 1 are reviewed but not others. More importantly, there do not appear to be any Unit 3 procedures reviewed. A sample review of one procedure between all three units (3.5.1.5(CS I)) found that the Units 1/2 tests affected two relays that are not tested in the Unit 3 procedure.	**Basis for Significance**: The review of procedures should not be limited to one unit. Differences between units may present additional pre-initiator actions. Although the one example found would not likely result in a pre-initiator, the point is that there are differences between the units' procedures. **Possible Resolution**: A more complete review of the procedures for all three units is warranted. There should at least be a focus on procedures for systems that may be different between the units.	A focus on review of procedures that may be different between the units was performed. No changes were made to the pre-initiators as a result of this review.	No impact. SRs HR-A1 and HR-A1 were assessed as MET for Category 2 in the May 2009 Peer Review. The CLTP and EPU PRAs use the same pre-initiators. EPU conditions impact some post-initiator HEPs.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SY-A8 SY-B9	Closed	4-32	Several electrical system boards are modeled to receive power from multiple sources (e.g., normal and alternate buses, and/or EDGs) without considering the need for undervoltage detection and operation circuitry for breakers and EDGs.	Priority 1 because model change is required. **Basis for Significance**: Component boundaries for breakers do not include such circuitry, based on NUREG/CR-6928. Note that local circuitry and protection devices are included. **Possible Resolution**: Review component boundaries and modeled events for automatic electrical bus transfers.	The EDG logic to start and load (close output breaker) is currently modeled. The component description for the circuit breaker component in Appendix A of NUREG/CR-6928 states: “The circuit breaker (CBK) is defined as the breaker itself and local instrumentation and control circuitry. External equipment used to monitor under voltage, ground faults, differential faults, and other protection schemes for individual breakers are considered part of the breaker”. External equipment used to monitor under voltage is considered part of the breaker. The modeling of automatic bus transfer in the BFN model contains both the normal supply breaker failure to open (FTO), and the alternate supply breaker failure to close (FTC). Since both failure modes are included, and the data from NUREG/CR-6928 includes under voltage detection in the breaker boundary, the current modeling methodology is appropriate. No model change was required.	No impact. SRs SY-A8 and SY-B9 were assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SY-A19	Closed	4-33	The unavailability or failure of a bus is not considered in the logic used to provide alternate electrical power supplies to other buses and boards. Example: U1_SDREC_A is used to re-energize 4kV SD Board A from 4kV SD Board 3A. However, the unavailability or failure of 4kV SD Board 3A does not fail the function (it should).	Priority 2 because Model change is required. **Basis for Significance**: Unavailability or failure of the alternate power supply would prevent being able to credit it as an alternate source. Although the failure probability of a bus is much less than the failure probability of other equipment that could affect the power transfer (e.g., breaker demand failure), the unavailability could be substantial, especially during an outage **Possible Resolution**: Include unavailability and/or bus failures as appropriate, or justify not modeling due to low failure probability.	The failure of the bus has been added to the BFN PRA model. The applicable 4 kV shutdown board failure has been added to gates U1_SDREC_A, U2_SDREC_A, U3_SDREC_A, U1_SDREC_B, U2_SDREC_B, U3_SDREC_B, U1_SDREC_C, U2_SDREC_C, U3_SDREC_C, U1_SDREC_D, U2_SDREC_D, and U3_SDREC_D.	No impact. SR SY-A19 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
QU-D2 QU-D7 QU-F2	Closed	4-36	The assumption that A HVAC is normally running and B HVAC is in standby leads to skewed basic event importance's and non-sensical cutsets. For example, with A HVAC always running: (1) The Loss of RMOV Board A importance is much higher than RMOV Board B (10% vs. 2.5%) (2) Non-sensical cutsets exist, such as where RMOV Board A is in maintenance and B HVAC fails to start (due to operator or hardware failure).	**Basis for Significance**: The assumption that one train is always normally running (the HVAC is only an example) does not reflect the plant operation, and can result in skewed importance results or missing cutsets/sequences (i.e., how would the results be different if the other train were assumed to be running?). **Possible Resolution**: Potential resolution is to remove flag settings for what train is normally running, and use flag events to represent the fraction of time that a given train is running and standby (e.g., 0.5).	The running and standby flags for the HVAC System trains have been changed to 0.5 to represent equal running times for all trains. To prevent non-sensical cutsets, the mutually exclusive (MUTEX) logic was expanded to include all events under the unit start gates (any failure event that only occurs during a unit start). In order to ensure proper application of the failure of a unit to start, the AHU fails to start after a LOOP event was made unique by adding a "_LOOP" to the event name.	No impact. The majority of the HVAC System dependencies have been removed from both the internal events PRA models. The stated resolution still applies to the remaining HVAC System dependencies.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
HR-H3 QU-D5	Closed	4-40	A review of non-significant cutsets found many LOOP cutsets that have combinations of two independent HFEs which should have some level of dependency: HFA_02114KVCRSTIE (Failure to cross-tie 4kV SD Board) AND HFA_0231480SDBTIE (Failure to provide alternate power to 480V SD Board).	**Basis for Significance:** This is an example of non-significant cutsets that, had they been reviewed, would have uncovered the need to perform additional operator dependency analyses. **Possible Resolution:** (1) Re-perform operator action dependency analysis. (2) Re-perform review of non-significant cutsets prior to finalizing and documenting results.	Dependency analysis has been re-performed and results are documented in the TVA Calculation, NDN00099920070032 Rev. 3, "HR – BFN Probabilistic Risk Assessment - Human Reliability Analysis." A review of non-significant cutsets prior to finalizing and documenting results was performed and was documented in the TVA Calculation, NDN00099920070041 Revision 3, "QU - BFN Probabilistic Risk Assessment – Quantification."	No impact. The HRA dependency analysis was updated in the CLTP and EPU evaluations
QU-D3	Closed	4-41	Offsite power recovery is applied in cutsets where it might not be possible. See U1 CDF cutset at 1.151E-08: LOOP with common cause failure of shutdown board normal feeder breakers to open.	**Basis for Significance**: Recoveries should only be applied to scenarios or cutsets where the recovery can be expected to be successful. **Possible Resolution**: Review recovery logic/rules to ensure that recoveries are not applied to non-recoverable failures.	The recovery logic/rules have been reviewed to ensure that recoveries are not applied to non-recoverable failures The example cited in the F&O is incorrect. If the breakers failed to open, they would still be closed and available for offsite power recovery.	No impact. SR QU-D3 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SY-A8	Closed	4-42	Table 3 of the data notebook says that EDG boundaries included the output breakers, but the EDG system notebook and the model have them as separate events. NUREG/CR-6928 lists breakers as WITHIN the boundary of the EDG.	**Basis for Significance**: Apparent inconsistency in data and component boundary definitions. **Possible Resolution**: Resolve discrepancy.	The output breakers (1818, 1822, 1812, 1816, 1838, 1842, 1832, and 1836) are no longer explicitly modeled, but within the boundary of the EDG. Calculations NDN-000-082-2007-0012 Revision 4, "SY.05 - BFN Probabilistic Risk Assessment - Emergency Diesel Generator System" and NDN-000-999-2007-0033 Revision 7, "DA.01 - BFN Probabilistic Risk Assessment - Data Analysis" have been updated to reflect this change.	No impact. SR SY-A8 was assessed as MET for Category 2 in the May 2009 Peer Review. The EDG logic to start and load (close output breaker) are modeled the same way in both the CLTP and EPU internal events models.
LE-C11 LE-C12	Closed	4-48	No credit is taken for equipment survivability or human actions following containment failure.	**Basis for Significance**: LE-C11 implies credit be taken for equipment survivability following containment failure, for Cat II/III. **Possible Resolution**: REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF.	LE-C11 states: JUSTIFY any credit given for equipment survivability or continued operation of equipment and operator actions that could be impacted by equipment failure. Section 3.1.3 of calculation NDN-000-999-2007-0037 Revision 3, "LE.01 - BFN Probabilistic Risk Assessment - LERF Analysis" contains the following: "The equipment survivability assessment, based on a review of the IDCOR Technical Report 17 is documented in the Structural Analysis Notebook [NDN-000-999-2007-0038 Revision 36, "LE.02 - BFN Probabilistic Risk Assessment - Structural Analysis" for BFN Unit 1. NDN-000-999- As long as the drywell and torus are intact, it is assumed that the environment in the reactor and turbine buildings will not prevent the use of equipment in those buildings. However, at the time of drywell failure, it is assumed in the Level 2 assessment that any active equipment in the torus room, adjacent corner rooms, and anywhere else in the reactor building will not be available due to elevated temperature, humidity, and radiation environments. Qualitatively, this equipment survivability assessment does not take any undue credit for the operation of equipment that is exposed to an extreme environment resulting from core damage and subsequent containment breach".	No impact. No undue credit taken for the operation of equipment that is exposed to an extreme environment resulting from core damage and subsequent containment breach.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
					Therefore, credit given for equipment survivability or continued operation of equipment and operator actions that could be impacted by equipment failure is justified since it is only credited if the containment is still intact. The significant accident progression sequences resulting in a large early release were reviewed to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. None were identified. This is documented in calculation NDN-000-999-2007-0037 Revision 3, "LE.01 - BFN Probabilistic Risk Assessment - LERF Analysis." Section 7.5, "CET Development" and Appendix A, "Containment Event Tree Nodal Overview."	
LE-C10	Closed	4-50	Although equipment survivability beyond equipment qualification limits is credited, there is no indication that significant accident progression sequences were reviewed to determine if continued equipment operation could be credited to REDUCE LERF.	**Basis for Significance**: LE-C10 Cat II/III requirements are to REVIEW significant sequences to determine if engineering analyses can be used to take credit for additional equipment operation beyond normal qualification limits to reduce LERF. **Possible Resolution**: Review significant large early release sequences to determine where additional equipment credit may be taken.	Significant large early release sequences have been reviewed to determine where additional equipment credit may be taken. The significant sequences are ISLOCA sequences, Main Steam Breaks Outside of Containment (MSBOC) Sequences and Feedwater Breaks Outside of Containment (FWBOC) sequences. Section 6.3.1.3 of calculation NDN-000-999-2007-0036 Revision 2, "AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis" discusses the equipment credited to prevent LERF for MSBOC and FWBOC sequences. Section 6.3.4.5 of calculation NDN-000-999-2007-0039 Revision 0, "IE.02 - BFN Probabilistic Risk Assessment - Interfacing Systems LOCA Analysis" discusses credit for isolating the LOCA before the ECCS pumps are flooded. This is intended to reduce LERF. Credit is based on a review of the ISLOCA cutsets that indicate sufficient time to depressurize the ISLOCA path to allow isolation. Depressurization is required to facilitate operation of isolation valves at lower differential pressure.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
LE-C1 LE-C8	Closed	4-51	Class3A (B,C)-006 LERF sequences are non-sensical. In these sequences, TD2 succeeds (i.e., DW Spray hardware is available and operator initiates injection per Table A.5.7-1) but DWS fails later in the CET (DWS_ALL_SUP branch is questioned).	**Basis for Significance**: Sequence is invalid since DWS is assumed to work but at the same time be unavailable. **Possible Resolution**: Review and correct CET.	TD2 is successful if Low Pressure Injection (LPI), Core Spray (CS), Alternate Vessel Injection (AVI) or Drywell Spray (DWS) is available. It is not guaranteed that DWS is the available system. From this perspective, a subsequent failure of DWS may still be valid. The Boolean logic works itself out when the failure branch fault tree models are linked in the accident sequence quantification. A review of the old CETs indicates that the DWS top is really DWI which does not involve failure of DW sprays. It should only be asked if TD (Injection established to RVP or containment) fails.	No impact. SRs LE-C1 and LE-C8 were assessed as MET for Category 2 in the May 2009 Peer Review. The CLTP and EPU PRAs use the same Level 2 model.
LE-G2	Closed	4-54	The method used to quantify split fractions was very difficult to review and appears to be based on an old LERF model that is not consistent with the current Level 1 model. The split fraction fault trees were not provided. Further, many of the split fraction descriptions provided in Appendix A of LE.01 do not appear to be current or are no longer used in the LERF model.	**Basis for Significance**: Split fraction values could not be determined by the reviewer, and descriptions for many split fractions do not appear to be valid any more. **Possible Resolution**: Review and update LE.01 Appendix A, especially to remove discussions or explanations that no longer apply to the LERF model.	Calculation NDN-000-999-2007-0037 Revision 3, "BFN Probabilistic Risk Assessment - Large Early Release Frequency Analysis" Attachment A "Containment Event Tree Nodal Overview" has been has been revised to address this comment. Note that "Appendix" is now "Attachment" in the Revision 3 calculation. Fault tree events specific to the LERF analysis are discussed and methodology to obtain split fractions has been revised.	No impact. SR LE-G2 was assessed as MET for Category 2 in the May 2009 Peer Review.
DA-B2	Closed	5-3	The data analysis does not appear to consider outlier components.	**Basis for Significance**: The inclusion of outlier components can incorrectly impact the failure rate assigned to a component group. Such outlier components should be placed into a separate	The plant-specific raw data was reviewed to identify any outlier components; none were found. Discussion of the lack of outliers is documented in calculation NDN-000-999-2007-0033 Revision 7, "DA.01 - BFN Probabilistic Risk Assessment - Data Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
				suitable component group. **Possible Resolution**: Add to Section 6.1.4 of DA.01 a discussion of how outlier components were analyzed. If outlier components were not analyzed, then add such a discussion and perform the required analysis.		
AS-A7	Closed	5-5	Section 6.3.2.4.1 of the Accident Sequence Analysis states that if Alternate Rod Insertion succeeds and either the recirculation pumps fail to trip or the SRVs fail to open, then a non-ATWS LOCA occurs which is not modeled in the PRA. While this new LOCA might be quantitatively insignificant, no qualitative argument is made to justify its omission.	**Basis for Significance** The omission of this sequence could result in an incorrectly-low CDF or cause the analyst to miss important insight about the event. **Possible Resolution**: Either model the sequence explicitly or qualitatively justify its omission in the Accident Sequence Analysis.	A qualitative justification was added to calculation NDN-000-999-2007-0036 Revision 2, “AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis.” It states that the frequency of an ATWS induced non-ATWS LOCA is less than the ASME standard recommended limit of 1E-7 /rx-yr; therefore is screens from further consideration because of its’ low probability of occurrence	No Impact. SR AS-A7 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SY-A11 AS-B6 SY-B9	Closed	5-7	Control power for the RHRSW and RCW pumps is currently modeled such that failure of control power will result in failure of the pumps to continue running. Typically, control power is only needed for starting the pump.	Priority 1 because model change is required. **Basis for Significance**: Currently the model overestimates the dependency on control power. **Possible Resolution**: Move the DC control logic under the gate associated with RHRSW and RCW pump start. Review this also for other normally running pump fault trees.	Control power was placed under pump start gates in the BFN PRA Model for all pumps and air compressors where it was determined that control power was not necessary to maintain a running pump or compressor. This model treatment documented in the respective Calculations, NDN-000-023-2007-0026 Revision 4, "SY.20 - BFN Probabilistic Risk Assessment - Residual Heat Removal Service Water System", NDN-000-024-2007-0019 Revision 2, "SY.13 - BFN Probabilistic Risk Assessment - Raw Cooling Water System" and Plant Control Air.	No impact. SRs SY-A11, AS-B6, and SY-B9 were assessed as MET for Category 2 in the May 2009 Peer Review.
IE-A5 IE-C6	Closed	6-2	Loss of HVAC System as an initiating event is screened, based on the 1995 PRA of the event. It appears the model and the assumptions for loss of HVAC System have changed, and loss of HVAC System as an initiating event should not be screened.	**Basis for Significance** Modeling changes have resulted in HVAC System becoming one of the top 5 systems in the present PRA. Based on this, a loss of HVAC System initiating event is likely to be significant as a contributor to core damage, and should not be screened. **Possible Resolution** Add Loss of HVAC System initiating events to the analyzed events for the PRA.	Screening of the loss of HVAC System initiating event is based upon the current calculation NDN-000-999-2007-0040 Revision 5, "SY.08 - BFN Probabilistic Risk Assessment - Heating, Ventilation, and Air Conditioning." Discussion of the 1995 PRA model was included to add additional insight into the impact of loss of HVAC System. Discussion of the 1995 PRA model has been removed from calculation NDN-000-999-2007-0030 Revision 1, "IE.01 - BFN Probabilistic Risk Assessment - Initiating Events Analysis" to avoid confusion in the future. Calculation NDN-000-999-2007-0040 Revision 5, "SY.08 - BFN Probabilistic Risk Assessment - Heating, Ventilation, and Air Conditioning" states "It is not expected that failure of any of these systems will cause a scram due to the long time available to repair them, provide a backup, or provide alternate room cooling. Additionally, many of the systems cool areas that do not have high heat loads during normal power operations or do not have equipment necessary for	No Impact. BFN plant specific HVAC System calculations have been developed to provide the basis for the elimination of many of the HVAC System dependencies previously in the PRA model. The HVAC System is no longer a top 5 system in the present PRA.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
					normal operation." Calculation NDN-000-999-2007-0030 Revision 1, "IE.01 - BFN Probabilistic Risk Assessment - Initiating Events Analysis" has been updated to state "The loss of important HVAC System is well annunciated, and heat up calculations show that there is ample time for the operators to restore HVAC System or take procedurally guided steps to prevent unnecessary isolation or SCRAM. Additionally, many of the systems cool areas that do not have high heat loads during normal power operations or do not have equipment necessary for normal operation. For additional discussion see calculation NDN-000-999-2007-0040 Revision 5, "SY.08 - BFN Probabilistic Risk Assessment - Heating, Ventilation, and Air Conditioning." This meets ASME standard IE-C4 part c screening criteria which states "the resulting reactor shutdown is not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically)."	

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-B4 IE-C6 IE-A5	Closed	6-5	The calculation of HPCI Steam Lines breaks (IE Section 6.2.3.8) does not appear to be reasonable, using older EPRI data and Wash-1400 data. The resulting steam line break calculated is 4.55E-10/year, which does not compare with results from other plants. Using newer data, the pipe break frequencies would likely be 2-orders of magnitude higher. Additionally, although the isolation valves may be available to eventually isolate the break, the impact of the break may have already occurred prior to isolation. Also, the generic MOV FTC value (from NUREG/CR-6928) in Data Table 4 is 1.07E-03/demand. Finally, the CCF probability used should be changed to the HPCI MOV FTC, with Alpha = 1.41E-02.	**Basis for Significance** Pipe break in the HPCI line can affect RCIC and many other components, due to the HPCI pump being open to other areas. The modeling as documented does not provide basis for screening, and if reperformed, the analysis will likely result in orders of magnitude increases here. **Possible Resolution** Consider including a HELB for HPCI in the PRA. Also, look at the impact of the HPCI analysis with respect to the RCIC.	DCD BFN-80-707 R19 states: "Temperature detectors shall be located in the HPCI equipment area and shall initiate isolation before ambient room temperature reaches the Environmental Qualification (EQ) temperature limits for safety related devices located in this area. This statement with a reference to the Design Criteria Document has been added to calculation NDN-000-999-2007-0030 Revision 1, "IE.01 - BFN Probabilistic Risk Assessment - Initiating Events Analysis." The generic MOV FTC value of 1.07E-03/demand from NUREG/CR-6928, is now utilized. The HPCI MOV FTC CCF probability has been updated to the value of 1.41E-02 which is consistent with NUREG CR/5497 (2007 Version). The updated HPCI Steam Line Break value is 1.93E-09/year. However, this does not change the conclusion of calculation NDN-000-999-2007-0030 Revision 1, "IE.01 - BFN Probabilistic Risk Assessment - Initiating Events Analysis" to not include this IE in the BFN PRA model.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-C8	Closed	6-8	RCW initiating event appears to be incorrectly reduced by factor RCWMTCF for combinations where the reduction factor does not appear to be valid. In particular, the event is applied to cutsets containing common transformer events. Also, reduction factor appears to be calculated incorrectly (1/365)**2.	**Basis for Significance**: Loss of RCW initiating event appears to be reduced by a factor of 1E-02 from the actual **Possible Resolution** Correct the fault tree initiating event for Loss of RCW to get correct results.	The RCW initiator model is described in calculation NDN-000-024-2007-0019 Revision 2, "SY.13 - BFN Probabilistic Risk Assessment - Raw Cooling Water System." The RCW success criteria states that a net loss of three pumps must occur before RCW fails. When this happens, it is assumed to fail for all thee units and all supported equipment. The failure of three RCW pumps is considered a loss of RCW. The frequency is calculated by summing all combinations of a failure of three pumps. All combinations of the failure of a single running pump (frequency per yr) and the failure of two additional pumps (probabilities) are included in the system initiating event model. The Loss of Raw Cooling Water (LRCW) is described in calculation NDN-000-999-2007-0030 Revision 2, "IE.04 - BFN Probabilistic Risk Assessment - Initiating Events Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-C8	Closed	6-10	CCF for Battery Chargers is not included in the Initiating Event Fault Tree for loss of 2 DC buses, other than for the standby chargers (not in the yearly failure rate logic).	**Basis for Significance** Can affect the loss of DC initiating events by a factor of 10, depending on how CCF is calculated. **Possible Resolution** Include CCF under the yearly failure rate logic or as a top event for all loss of DC initiating events.	Common Cause Factors (CCFs) were not included in fault tree initiating events with year-long mission times. As stated in Support System Initiating Events: Identification and Quantification Guideline. EPRI, Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Washington, D.C.: 2008. 1016741: "Current models and data for common cause failure (CCF) of operating components are often based on minimal data that have been evaluated and developed for use in a post-initiator, 24-hour mission time model (which typically involves some conservatism). While the conservatism may be acceptable for a 24-hour mission time, extrapolation of this data to model common cause failure frequencies for the year-long mission time used in initiating event modeling often results in frequencies exceeding those observed in industry experience". Based on the above recommendation, CCF of battery chargers has not been added to the yearly failure rate logic in the Loss of 2 DC bus initiating events fault tree. The data used for modeling the individual buses is so conservative that it would be overly conservative to include common cause failure. No changes to the model or the documentation are required.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-C14	Closed	6-13	The impact of Surveillance Procedures is not included in the ISLOCA Calculation. For example, for Core Spray, Surveillances in the CS Notebook indicate an MOV opening every 92 days. The likelihood of an ISLOCA during this MOV test is not calculated in the ISLOCA IE Fault Tree, including the sequence where the check valve would have previously failed prior to the surveillance.	**Basis for Significance**: Unknown impact on the ISLOCA Frequency, without analyzing the specifics of the site procedure. If the procedure has the operator check downstream pressure (etc.) prior to opening the MOV, likely there is minimal impact. However, given the ISLOCA has a large impact on LERF, the impact could be significant. **Possible Resolution**: Include the impact of Surveillance Procedures in the ISLOCA Analysis.	The impact of surveillance procedures for the CS and RHR injection paths are addressed in the third through fifth paragraphs of Section 6.3.1.7 of calculation NDN-000-999-2007-0039 Revision 0, "IE.02 - BFN Probabilistic Risk Assessment - Interfacing Systems LOCA Analysis." The fourth paragraph and remaining paragraphs of this section addresses the methodology used to address the quantification of the surveillance test impact.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
SY-A2	Closed	6-17	System models do not appear to incorporate operating experience in developing the fault tree logic. RHR Service Water operating experience does not appear to be complete or reviewed. HVAC System Notebook says LERs and OER was reviewed, but none are listed (no evidence of the review). Similarly for 120 VAC and others. CRD Notebook includes only a discussion of the BFN Fire, but no review of OE is presented.	**Basis for Significance**: Review of experience from BFN and other plants does not appear to be used in developing the fault tree system logic or data. In some cases, review of BFN OE is not included in the notebooks. **Possible Resolution**: Expand operating experience review and account for any lessons learned in the PRA model.	The write-up in the system notebooks discussing the level of SER, OER and LER reviews has been enhanced. There is no requirement in the ASME standard that requires a detailed listing or discussion of the generic or plant specific experience reviewed. Therefore, no detailed listing or discussion of the generic or plant specific experience reviewed needs to be included in the documentation.	No impact. SR SY-A2 was assessed as MET for Category 2 in the May 2009 Peer Review.
SY-A14	Closed	6-20	Event STRPL1STN_0750664, CS Suction Strainer Plugging, is only assumed for Large LOCA in the Model. The phenomenon causing plugging is not limited to large LOCA only, and is possible on Medium LOCA, SRV opening, etc. A question was asked to the analyst on this, and the reference to the absence of permanently installed air filters or other sources in the drywell. However, the debris, if present, would be swept into the suction strainer by any LOCA.	**Basis for Significance**: Affects multiple Initiating Events. Pre-existing material in the Torus can also affect the strainer plugging likelihood. **Possible Resolution**: Include CS Suction Strainer failure for all applicable LOCA events, including SRV lift events. It is possible to use different plugging likelihood values for each LOCA size.	The core spray suction strainer plugging event was added for Medium Loss of Coolant Accident (MLOCA).All SRVs discharge directly to the suppression pool, so a stuck open SRV could not dislodge material from the drywell. Calculation NDN-000-075-2007-0010 Revision 4, “SY.04 - BFN Probabilistic Risk Assessment - Core Spray System” documents the discussion for this scenario.	No impact. SR SY-A14 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
HR-H1	Closed	6-25	Event HFA_3003P_START_A does not appear to be applied correctly in the model. A question was asked of the analysts on the logic, and the response referred to gate U3_FWH_INIT for events were FW recovery is not credited. However, the logic under gate U3_FWH_G50 limits the operator failure event to only excessive FW events; resulting in no failures coming through for other events were FW is credited.	**Basis for Significance:** Significance is unknown, since model modification is required in order to determine the impact. **Possible Resolution:** Remove the requirement for excessive FW events only when applying the HFE.	The human action HFA_3003P_START_A has been changed to the non unitized name HFA_0003P_START_A. It is used in every situation where a feedwater pump has to be started (for all three units). One of those cases is where the pump is running and is tripped due to excessive feedwater flow. It is assumed the pump can still be operated but must be restarted. This gate is OR'd with a gate where the feedwater pump is not running and either has to be started or is in T&M. This human action is used in that tree also. There is no incorrect logic with this human action. No changes are necessary.	No impact.
HR-G5	Closed	6-28	Basis for operator action time (30 min) for HFA_0085ALIGNCST appears to be roughly estimated, as is the time available (7 hours).	**Basis for Significance**: Event provides over 5% of CDF. **Possible Resolution**: Provide more a more accurate assessment for the timing for HFA_0085ALIGNCST.	HFA_0085ALIGNCST is used in fault trees for sequences where the source of inventory from the CST is required for 24 hours. A MAAP case documented in calculation NDN-000-999-2008-0006 Revision 3, "SC.02 - BFN Probabilistic Risk Assessment - PRA MAAP Thermal Hydraulics Calculation" shows that a single CST will provide adequate inventory for 10 hours. Case 3F used an initial level in the Condensate Storage Tank (CST) of 15 feet (180,000 gallons or 24,060 ft^3). The purpose of this case was to allow for a more realistic analysis of the time to core damage following a loss of feedwater with one stuck open safety relief valve. Plant data indicates that the level of the CSTs for all three units is an average of approximately 19 feet and operator interviews reveal that it is plant practice to keep the levels of the CSTs above 15 feet during corresponding unit operation. The HRA for HFA_0085ALIGNCST has been revised using the 10 hour time period.	No impact.
HR-G7 QU-C2	Closed	6-30	Dependencies between operator actions appear to be non-conservatively applied. Mainly, the Zero Dependence (ZD) between actions is commonly	**Basis for Significance**: Systematic error affecting around 1/2 of the combo events, including combo 18.	In general, dependencies between operator actions have been derived within the rules outlined in the HRA Calculator. In one case, the dependency rules have been over-ridden by a user defined rule. In this particular case, a note was added stating the reason	No impact. The HRA dependency analysis has been updated for the CLTP

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			applied, simply when one of the actions takes longer than 60 minutes. What appears to be the mistake is applying the last event tree node in the Dependency Event Tree. In this tree, if the stress of either HFE is moderate or high, the upper leg of the event tree is used. SO for combo 2, the HRA assumes ZD, while the event tree would designate Low Dependency.	**Possible Resolution**: Correct dependency analysis in the HRA.	for the over-ride, which is documented in calculation NDN-000-999-NDN-000-999-20070032 Rev. 4, "HR – BFN Probabilistic Risk Assessment - Human Reliability Analysis." "Need to depressurize would arise no less than 2 hr after ability to initiate SPC would no longer permit use of HPCI/RCIC after CST depletion." This statement is under the dependency event tree and occurs for combinations of HFA_0074HPSPC1, Failure to align RHR for SPC (non-ATWS/IORV) and HFA_0001HPRVD1, Failure to initiate reactor-vessel depressurization (transient or ATWS). The timing for the cues implies that there should be a complete dependence; however the timing for HFA_0074HPSPC1 occurs over 5.4 hours and therefore there is no time dependence. The cue comes in, but the required action has such a long time in which to be accomplished, there is no dependence, hence zero dependence was manually chosen. The note in the calculator is sufficient to address the issue and the discussion in the calculation.	and EPU PRAs.
IE-C8	Closed	6-36	The ISLOCA Conditional Pipe Break Frequencies calculated for the analysis appear to be too low, in comparison with other pants. From NUREG/CR-5102, Appendix F, Table 2, the RHR and CS piping would generally get a failure probability of 2.65E-02 and 2.54E-03 respectively. Other reference documents used should get similar results. The BFN analysis is supported by and Excel Spreadsheet for the overpressure estimate, and this analysis is not included in the system notebook. In the excel spreadsheet it appears the temperature assumed for the CS and RHR analysis assumes room temperature, where as full RCS	**Basis for Significance:** ISLOCA is a significant contributor to LERF **Possible Resolution:** Revise the conditional pipe break frequencies to match industry accepted values, based on use of RCS temperature in the CS and RHR piping. Benchmarking of other plant methods and values may be useful here. Include the overpressure/pipe break analysis (excel spreadsheet) as a part	Section 6.3.4.2 and Table 6.21 of the TVA Calculation NDN00099920070039 Revision 0, "IE.02 - BFN Probabilistic Risk Assessment - Interfacing Systems LOCA Analysis" were revised to include calculation details for the ISLOCA break frequencies assuming a temperature of 600°F.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			temperature is more appropriate.	of the reviewed system notebook.		
SY-B11	Closed	6-41	Fuel oil transfer pumps to refill the day tank are not part of the EDG boundary in NUREG/CR-6928.	Priority 1 is given because a model is required. **Basis for Significance**: Issue with EDG Component Boundary. **Possible Resolution**: Add separate failure of fuel oil transfer to the EDG Fault Tree Model.	NUREG/CR-6928 states that the EDG boundary is the following: "The EDG boundary includes the diesel engine with all components in the exhaust path, electrical generator, generator exciter, output breaker, combustion air, lube oil systems, fuel oil system, and starting compressed air system, and local instrumentation and control circuitry. However, the sequencer is not included. For the service water system providing cooling to the EDGs, only the devices providing control of cooling flow to the EDG heat exchangers are included. Room heating and ventilating is not included." Calculation NDN-000-082-2007-0012 Revision 3, "SY.05 - BFN Probabilistic Risk Assessment - Emergency Diesel Generator System" documents the modeling for the emergency diesel generator system which defines the "fuel oil system" as up to the fuel oil day tank including the safety-related fuel oil transfer pumps. Each EDG at BFN has a 550-gallon day tank that provides enough fuel to operate for 2.5 hours at full load. Fuel is then transferred from the 40,000-gallon 7-day diesel storage tank with one of two diesel fuel oil safety-related transfer pump to continue operation. There is one 40,000-gallon 7-day diesel storage tank for each diesel generator and it is included in the diesel generator boundary. The non-safety related pumps that transfer fuel from the 71,000 gallon yard storage tank to the 40,000-gallon 7-day diesel storage tanks are outside the boundary and are not considered in the model.	No impact. SR SY-B11 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
AS-A5	Closed	6-49	The %1INTAKE initiating event is modeled in a simplistic manner, and does not appear to represent the expected plant and operating response. On the conservative side, the plant in many instances can reduce power to extend the time to clean the screens. On the non-conservative side, there are possible events that operator actions (cleaning the screens) will not prevent plugging, given a very large amount of material plugging the intake. Additionally, some events could break through the screens causing plugging of the system (Hx, strainers, or pumps). The above events have actually occurred at other plants.	**Basis for Significance** %1INTAKE is the number 1 CDF and LERF contributor. **Possible Resolution**: Modify the model to include the factors the affect risk, including power reductions, screen breakthroughs, operator actions causing screen breakthroughs, and the likelihood that an event would occur where cleaning activities will not prevent plugging. Other plants have typically assumed a single CCF event (much lower in frequency) for plugging of all intakes, where operator response for cleaning is not possible, but with other sequences where partial plugging occurs.	An intake plugging initiator that scrams all three units and fails RCW was developed from plant specific data. This initiator replaces the current initiator estimate and operator actions in the model. A conditional probability event of the RHRSW/EECW system failure due to intake plugging was developed that replaces the human action in the model. The model, along with the calculations listed below have been revised to reflect the refined modeling. NDN-000-999-2007-0036 Revision 2, “AS – BFN Probabilistic Risk Assessment – Accident Sequence Analysis”, NDN-000-999-2007-0030 Revision 2, "IE.01 - BFN Probabilistic Risk Assessment - Initiating Events Analysis", NDN-000-024-2007-0019 Revision 2, "SY.13 - BFN Probabilistic Risk Assessment - Raw Cooling Water System", NDN-000-023-2007-0026 Revision 4, "SY.20 - BFN Probabilistic Risk Assessment - Residual Heat Removal Service Water System", and NDN00006720070013 Revision 4, "SY.06 - BFN Probabilistic Risk Assessment - Emergency Equipment Cooling Water System."	No impact. SR AS-A5 was assessed as MET for Category 2 in the May 2009 Peer Review.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IE-C11 SY-A22	Closed	6-50	Some of the MOVs credited in the ISLOCA Fault Tree are not tested to close against full DP. These MOVs are not originally included in the design as RCS isolation valves. Examples include 74-55 and 74-66 (note: this is not a complete list, but 2 of 4 valves reviewed were not in the MOVATs 89-10 program).	**Basis for Significance**: MOVs closing for ISLOCA are risk significant, with a RAW of greater than 2. **Possible Resolution**: Do not credit MOVs in the ISLOCA without verification the valves will close against full DP of RCS pressure.	Some MOVs credited for closure for isolation during an ISLOCA cannot be shown to close against full reactor pressure because they are not in the MOVATs 89-10 program. Therefore, credit for MOV closure for isolation during an ISLOCA event is based on alarmed procedural actions to reduce RCS pressure as RCS inventory is discharged through the break. Reduced differential pressure across the MOVs allows for ISLOCA isolation prior to flooding the reactor building quads where the ECCS pumps are located. This clarification was added to the second paragraph of Section 6.3.4.5 of calculation NDN-000-999-2007-0039 Revision 0, "IE.02 - BFN Probabilistic Risk Assessment - Interfacing Systems LOCA Analysis."	No Impact. SRs IE-C11 and SY-A22 were assessed as MET for Category 2 in the May 2009 Peer Review.
LE-B1	Closed	7-6	Section 7.1 of BFN Probabilistic Risk Assessment - LERF Analysis LE.01 directly addresses those contributors from the table, but plant specific issues do not appear to be addressed.	**Basis for Significance**: The SR requires the consideration of unique plant issues. **Possible Resolution**: Include discussion of plant specific issues that may contribute to LERF.	ASME/ANS RA-Sa-2009 "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plants," Table 2-2.8.9 "LERF Contributors to be Considered" identifies potential contributors to LERF for BWR plants with Mark I containment designs. Each of these LERF contributors is considered in the Browns Ferry PRA through various CET top events as described in calculation NDN-000-999-2007-0037 Revision 3, "LE.01 - BFN Probabilistic Risk Assessment - LERF Analysis," Attachment A "Containment Event Tree Nodal Overview." There were no plant specific contributors to LERF identified through the LERF Analysis.	No impact.
LE-B1	Closed	7-7	The definition of Early appears to be inconsistent and may eliminate some scenarios from consideration for LERF.	**Basis for Significance**: Definition of the timing of accident sequences determines whether a sequence can contribute to LERF. Timing based from accident initiation will be	Calculation NDN-000-999-2007-0037 Revision 3, "LE.01 - BFN Probabilistic Risk Assessment - LERF Analysis" provides clarity with respect to the timing definition used and includes information that shows the timing used for each scenario or group of scenarios based on the MAAP calculations.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
				different than timing from declaration of General Emergency. **Possible Resolution** Clarify the timing definition used and include information that shows the timing used for each scenario or group of scenarios based on the MAAP calculations.		
IFEV-A5	Closed	IFEV-A5-03	For flooding events that cannot result in the "major flood" scenario due to limit in the flood source system inventory, the portion of the piping system failure frequencies for "major flood" should be combined with the "flood" scenario. In this case, only the "flood" impact should be modeled. For example, the total frequency for the RBCCW flood on El. 593' or El. 565' of Reactor Building (derived from the total piping system failure frequency) was split into three portions based on the possible spill rate: major flood (> 2,000 gpm), flood (between 100 gpm and 2,000 gpm), and spray (up to 100 gpm). Even though the RBCCW could not cause the impact of a "major flood" because of the limited system inventory, the total flood frequency resulting from failure of the RBCCW piping system should be accounted for in modeling the RBCCW-induced flooding scenario (by combining both the "major flood" frequency and the	N/A	The spray and flood frequencies for all applicable floods were combined. This is documented in the calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			"flood" frequency for the "flood" scenario) since the RBCCW pipe dimension permits a spill rate in excess of 2,000 gpm.			
IFEV-A6	Closed	IFEV-A6-01	Only generic data is used in the estimation of pipe failure and flooding frequencies including pressure boundary rupture and human-induced breach of boundary. No plant-specific operating experience is accounted for.	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis," Table C-1 "Summary of Flood Experience in the U.S." provides a summary of the internal flooding events that have occurred at-power in nuclear power plants within the United States. There is one event in the table that was recorded at Browns Ferry; however, it was classified as failure of HPCI turbine and was inconsequential as flood. There is no other plant data available to incorporate into the BFN PRA, so only generic data was used.	No impact. A sensitivity was included in the analyses for EPU to address higher flows in some systems (5.7.1.5)

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFEV-A6	Closed	IFEV-A6-02	It appears that the data used for the Circulating Water expansion joint may not be consistent with the latest version of EPRI data as documented in EPRI report 1013141 (Reference 6). Additionally, it is not clear why the analysis did not consider the possibility of "flood" scenario (i.e., leak rate between 100 gpm and 2,000 gpm) for expansion joint failure (no justification was given in the IFPRA notebook). The most recent version of EPRI data represents the "major flood" resulting from expansion joint failure by two separate scenarios: one between 2,000 gpm and 10,000 gpm, and another one greater than 10,000 gpm. However, the BFN IFPRA only has one scenario for "major flood" representing a flood spill rate of more than 2,000 gpm.	N/A	Browns Ferry is unique in that it has a very large lower area in the turbine building that has to be flooded. This is because the lower areas of all three units' turbine areas are interconnected. Most plants only have to fill the area under a single turbine unit before significant damage is encountered. The time available to detect and mitigate this accident is much greater for Browns Ferry. This same condition also significantly reduces the difference between a "small" major flood and a "large" major flood.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFEV-A7	Closed	IFEV-A7-01	Generic data was used to estimate the frequency of human-induced flooding scenarios associated with maintenance on the EECW/RCW system (Reference Section 6.5 indicates 2 events for EECW in the Reactor Building (not accounted for in the BFN IFPRA result), while Appendix G indicates 1 event for RCW in the Turbine Building (not clearly documented in Section 6.5)). Systematic evaluation of all of the systems potentially susceptible to this type of flooding scenarios was not consistently provided. Maintenance-related human-induced flooding scenarios are highly plant-specific and system-specific. Using only sparse generic data cannot systematically identify vulnerable areas for human-induced flooding scenarios that may result during power operation; e.g., maintenance of the condenser water boxes (opening of the manways for tube plugging), RBCCW heat exchanger maintenance (opening of the heat exchanger), maintenance of the fire water pre-action/clapper valves, frequent maintenance on the chillers, etc. The description of analysis for operation/maintenance-related flood associated with condenser waterboxes given in the IFPRA notebook indicates that human-induced flood is extremely unlikely because of the local operator monitoring, etc. However, with the same types of	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis," Section 6.5 "Characterization of Flood Scenarios" includes a subsection on maintenance-induced flooding. This section systematically evaluates all of the systems potentially susceptible to maintenance-induced flooding scenarios. The conclusions from this section have been verified to be consistent with Appendix G "Initiating Event Frequency Calculations"	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			protection, human-induced flooding events resulting from condenser waterbox maintenance has actually occurred in the past at other plant. The description of analysis for operation/maintenance-related flood associated with EECW and A/C equipment indicates that human-induced flood is very unlikely because the system is rarely opened for maintenance and local operator monitoring of the proper isolation of chillers. However, chiller maintenance is actually a quite frequent event. More thorough and better justifications should be considered, including the size of the possible human-induced leak/flood, etc.			
IFEV-B2	Closed	IFEV -B2-01	It appears that not all of the assumptions used in the analysis were documented; e.g., the assumption that the pipe diameters and pipe lengths for the same systems at the same locations are approximately identical among the 3 units was used for some areas, but was not documented.	N/A	All assumptions have been documented in the Assumptions section of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFEV-B3	Closed	IFEV-B3-01	Sources of uncertainty and relevant assumptions associated with potential flood initiating events were not identified consistently. Table 4-1 did not identify sources of uncertainty relative to the flood-induced risk contributors (e.g., frequencies of failure/leakage/rupture from the various flood sources, and other mitigation factors such as door failure likelihood, etc.).	N/A	All assumptions are listed in the assumptions section of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis." Table 4-1 "Identification of Key Sources of Uncertainty" in calculation NDN-000-999-2007-0031 was revised to include additional discussion on potential uncertainties.	No impact. .

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFQU-A5	Closed	IFQU-A5-01	Operator actions for flood mitigation analyzed are not listed in Table F-1 as stated in Section 6.8. Table 4 in Appendix H provides the description of two actions (i.e., Reactor Building major flood isolation, HFA_0_RXMAJORFLOOD; and isolation of major RCW flood in Turbine Building, HFA_024RCW-M with a HEP value of 1.0). The same HEP for HFA_0_RXMAJORFLOOD is used for all scenarios where this action is applied. However, no analysis details (e.g., performance shaping factors such as timing, accessibility, etc.) were documented in the IFPRA notebook for either HFE. Based on a word search, HFA_0_RXMAJORFLOOD was not found in any of the HRA notebooks. It is not clear what instrumentation was relied on for the detection of a flood event and for the identification of the flood source and the location of the breach which are required to determine the specific isolation action to perform (e.g., the specific valves to close for the isolation of the breach).	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis," Section 6.8 "Evaluate Flood Mitigation Strategies," describes the methodology used to evaluate the flood mitigation strategies. The HRA performed for the Internal Flooding analysis is documented in Appendix K of calculation NDN-000-999-2007-0031, "Human Reliability Analysis." Appendix K includes references to alarm response procedures for each modeled human action related to flood mitigation. These procedures identify the instrumentation that is relied upon for the detection of a flood event, the identification of the flood source, and the location of the breach. These instruments are required to determine the specific operator actions to perform to mitigate the breach (e.g., identification of specific valves to close for isolation).	No impact. .

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFQU-A6	Closed	IFQU -A6-01	The effects of flood on the human actions modeled in the internal events PRA that are not directly related to flood mitigation (i.e., isolation of the flood) may not have been considered consistently. Only one human action event (HFA_0074UNITXTIE) is listed in Table 4 of Appendix H. It is not clear if this is the only non-flood human action in the PRA model for which no credit is taken due to the effects of the flood. Typically, the effects of flood on these human actions may result in either an increase in the HEP (e.g., due to increase in stress, workload, etc.) or failure of the human action (i.e., no credit can be taken for the human action if it is an ex-control room action performed in an area affected by the flooding effects). Additionally, manual isolation action to terminate the flooding scenario may not have been applied to all applicable scenarios where appropriate.	N/A	The internal flooding documentation was revised to address this F&O. Section 6.7 of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" provides the methodology used for the flood consequence analysis. Appendix H of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" contains the listing of unscreened initiators, the list of affected components, and the list of impacted Human Actions for each flood scenario.	No impact.
IFQU-A7	Closed	IFQU -A7-01	The flood-induced CDF and LERF for selected spray scenarios (e.g., such high CDF/LERF contribution scenarios as %IFS1RB565-ECS, %IFS1RB565-RCW, %IFS3RB565-ECS, %IFS3RB565-RCW, etc.) are probably conservative without considering some of the unique	N/A	The following provides information that was documented in calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" to address this F&O. In formulating the potential impacts of specific flood sources on PRA-relevant equipment, spray effects were not explicitly modeled if the impact of the spray was to cause failure of only one component (e.g., one pump motor). The intent of the flood analysis is to search for potential common causes of failure; failure	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			characteristics of water spray; e.g., portion of the piping system considered in the calculation of the spray frequency may be outside the spray impact range, equipment within the spray impact range (360°) may not be damaged simultaneously in the same spray scenario due to the directional nature of spray, equipment being sprayed on may not necessarily fail even if the component is not designed for water intrusion proof, etc.		of a single component due to spray effects is assumed to be captured adequately by the failure rate for the component. The risk contribution from sprays in the turbine building is negligible. Sprays would only affect the power conversion system (PCS). The estimated failure frequency of the PCS due to sprays is at least two orders of magnitude lower than other non flood initiators that assume PCS is failed. Sprays are considered for a grouping of electrical equipment in one TB corridor. Sprays of jacketed or insulated pipes were not considered since the spray stream would mitigated by the jacket/insulation. Only flood damage due to inundation was considered for these pipes. Failure due to sprays was not considered for PRA components outside the spray range (10 feet used in this analysis). Due to the insignificant contribution of internal flooding to overall CDF and LERF (less than 1% in the Revision 5 Model), it is apparent that the spray contributions are not overly conservative.	
IFQU-B1	Closed	IFQU-B1-01	The derivation of the XINIT input file and the XINIT input information should be presented in the Internal Flood PRA notebook. Table 4 in Appendix H lists the impact of the flood scenarios (i.e., components failed and human failure events). However, the specific model elements affected by these flood impacts and incorporated into the PRA model are not documented in the IFPRA report (e.g., how the effects of the initiating event is modeled in the PRA).	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis," Appendix H "Flood Initiating Events and Impacts" includes the listing of unscreened initiators, the list of affected components, and the list of human actions. The failed basic events are included in Table H-1 of the calculation.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFQU-B2	Closed	IFQU-B2-01	Description should be provided for each of the top (based on CDF/LERF contribution) flooding scenarios presented in the results section.	N/A	Descriptions of flooding scenarios are provided in calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis," Table 7 "Quantification Results for Internal Flooding," along with CDF and LERF contributions. Discussion of the top CDF/LERF flooding scenarios has been added to the results section (Section 7.0).	No impact.
IFQU-B3	Closed	IFQU-B3-01	Sources of uncertainty and relevant assumptions associated with potential flood initiating events were not identified consistently. Table 4-1 did not identify sources of uncertainty relative to the flood-induced risk contributors (e.g., Failure probabilities of operator flood mitigation actions, impact of flooding scenarios on the HEPs associated with the non-flood operator actions included in the internal events PRA model, effects of the initiating event group selection for modeling the flooding scenarios in the PRA model, etc.).	N/A	All assumptions are now listed in the assumptions section of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis." Table 4-1 "Identification of Key Sources of Uncertainty," was revised to include more discussion on potential uncertainties.	No impact.
IFSN-A10	Closed	IFSN-A10-01	Flood scenarios resulting from failure of the CST suction lines causing failure of RCIC or HPCI were not enumerated in Tables 6-4, F-1, and Appendix H. Even if the water inventory in each CST is insufficient to cause PRA equipment damage in the Reactor Building basement due to water submergence, some PRA components could still be damaged by spray effects.	N/A	Analysis shows that at least 500,000 gal is required to flood the RB519 level to a point where equipment is failed by submergence. The CST maximum volume is only 375,000 gal; therefore, this flood cannot fail components due to submergence. Walk downs have confirmed that all of the PRA components in the reactor building basement quadrants (i.e., all four corner rooms in each unit) are protected from sprays. The CST flooding scenario is therefore screened. This discussion has been added to Section 6.5 of the Internal Flooding calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A10	Closed	IFSN-A10-02	The use of the pre-action fire water system reduces the likelihood of flooding resulting from failure of the dry pipe segments and spurious actuation. However, failure of the wet pipe segments (i.e., upstream of the pre-action/clapper valves) in the buildings evaluated could still lead to the water spray and submergence effects considering the "unlimited" supply of fire water. The wet pipe segments should be present in the Reactor Building, Turbine Building, and the Control Bay Corridor. No flood submergence scenarios resulting from Fire Water piping system failure are shown in Table 7, Appendix G, and Appendix H. Only spray scenarios resulting from the Fire Water piping system failure in the Turbine Building are considered in Table 7, Appendix G, and Appendix H.	N/A	Discussions with the BFN fire protection engineer determined that all of the preaction clapper valves for the control bay are in the turbine building. Walk downs provided the pipe lengths and locations for these sections of fire protection piping in the reactor building. Initiators for these RB flood sources have been included. Turbine building elevation 565' is the only area that has the water charged sections of fire protection piping. Scenarios involving fire water piping are now included in the PRA model.	No impact.
IFSN-A10	Closed	IFSN-A10-03	Consideration, analysis, or documentation of the flood scenarios do not appear to be consistent between the 3 units. For example, The initiating event frequency calculations in Appendix G only include flooding scenarios for Unit 1 and Unit 2 Raw Cooling Water on El. 593' in Reactor Building, while the walkdown sheet in Appendix A documents the Raw Cooling Water lines on El. 593' in the Unit 3 Reactor Building. However,	N/A	Walk downs were conducted for all three units. Initiators were developed for all three units for both spray and submergence. Credible spray scenarios were not screened out. This was reflected in the body of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" as well as in the appendices in a consistent manner.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
			Table 4 in Appendix H includes “major flood” scenarios resulting from Raw Cooling Water piping system failure on El. 593’ in the Reactor Building for all 3 units. Additionally, the spray effects were not considered for any of these “spray”, “flood”, and “major flood” scenarios. “Spray” and “flood” scenarios were screened out even though PRA equipment could be damaged by the spray effects (Reference no probabilistic basis provided to satisfy standard requirement IFEV-A8(b)). Treatment of the spray effects for EECW line failure on El. 565’ in the Unit 1 Reactor Building and for piping system failures in the Reactor Building suppression pool area is similar (i.e., “spray” and “flood” scenarios were screened out).			

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A10	Closed	IFSN -A10-04	Inconsistency exists between Table F-1, Appendix G and Appendix H for failure of the Raw Cooling Water piping system in shutdown board room B on El. 593' in Reactor Building. Table F-1 indicates that both "flood" and "spray" scenarios for the RCW line in the shutdown board room B on El. 593' in Unit 1 Reactor Building should be analyzed. However, Appendix H only includes the frequencies for the "major flood" and "flood" scenarios for the RCW line in the shutdown board room B on El. 593' in Unit 1 Reactor Building. Also, Table F-1 indicates that the "spray" and "major flood" scenarios resulting from failure of the RCW piping system in shutdown board room A on El. 621 in Unit 1 Reactor Building are not screened and should be analyzed. However, neither Appendix G nor Appendix H included the analysis of flooding scenarios in shutdown board room A on El. 621 in Unit 1 Reactor Building.	N/A	The piping in the shutdown board room was found to be drain piping from the roof. There is no RCW piping in this room. The shutdown board rooms in the reactor building have no sources including drains that might allow propagation into the rooms. Documentation has been changed to reflect this.	No impact. .
IFSN-A12	Closed	IFSN -A12-01	Some of the rooms/zones were qualitatively screened out (in Table 6-4 and F-1) solely based on the consideration of flood submergence (i.e., insufficient flood volume); i.e., without considering the possible damage potential by the spray effects.	N/A	After the peer review, additional internal flooding analysis was performed on those areas that were previously qualitatively screened out (in Table 6-4 and F-1) solely based on the consideration of flood submergence (i.e., insufficient flood volume); i.e., without considering the possible damage potential by the spray effects. Spray sources were located, components identified, and sprays assessed in all flood areas of the reactor buildings, control bay, diesel	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
					generator buildings, and intake pumping station. Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" was revised to reflect this. turbine building spray was handled differently as discussed in the original flooding report.	
IFSN-A12	Closed	IFSN -A12-02	DG building was screened out because flood damage to the EDG equipment would not lead to an automatic reactor scram or immediate plant shutdown (Section 6.4). This does not meet the requirement for IFSN-A12 in which an area is only screened out if flooding of the area would not cause an initiating event and would not cause damage to mitigating equipment. To screen out the EDG flood areas in this case, justification should be provided to satisfy PRA standard requirement IFEV-A8(b). Damage to a major component (e.g., EDG) due to spray resulting from failure of other equipment (piping associated with other systems such as EECW) is typically not accounted for in the generic and plant-specific random failure rates of the affected component (Assumption 2 in Section 4.1).	N/A	Flooding in the EDG building was evaluated in a manner consistent with the other plant areas. Initiators were included even if they did not cause a reactor scram. The analysis is documented in calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A12	Closed	IFSN-A12-03	RHRSW/EECW pump bays in the Pumping Station were screened out because it was determined that there is no PRA impact (Section 6.4 and Tables 6-4 and F-1). However, 3 of these pumps could be damaged if one bay is flooded. In accordance with PRA standard requirements IFSN-A12 and IFSN-A13, this flood area should be retained. Note that PRA standard requirement IFEV-A8(b) may not be applicable since multiple components are involved.	N/A	Flooding in the pumping station was evaluated in a manner consistent with the other plant areas. Initiators were included even if they did not cause a plant reactor scram. The analysis is documented in calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact.
IFSN-A12	Closed	IFSN-A12-04	Some of the flood sources in the Reactor Building were screened out (e.g., rupture of EECW piping) because only limited PRA equipment is damaged (e.g., one loop of Core Spray, one loop of RHR, or RCIC) requiring no immediate plant shutdown (and would not cause an automatic scram). See Tables 6-4, F-1, and Appendix H. This does not satisfy the PRA standard requirements IFSN-A12 and IFSN-A13. To allow screening of these flood areas, justification should be provided to satisfy PRA standard requirement IFEV-A8(b).	N/A	Evaluations were performed on the flood sources identified in the finding, which were previously screened out. Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" was updated to reflect the analysis of the previously screened flood sources. Some of the previously screened out flood sources still screen out, but for other reasons (e.g., all piping in the area is insulated or sheathed so spray of PRA components is not a concern).	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A15	Closed	IFSN -A15-01	Spray scenario resulting from failure of the RBCCW line was screened out based on the consideration that break is not large enough to cause failure of the RBCCW system and thus will not cause a reactor scram (Tables 6-4 and F-1). This is questionable because RBCCW is a closed loop system with no automatic makeup. Loss of inventory will result in failure of the RBCCW and thus a scram eventually due to impact to its loads.	N/A	The RBCCW line failures were evaluated further and not screened just because they may not cause a scram. Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" was updated to reflect the analysis of the potential flood and spray scenarios associated with the RBCCW piping. The flooding due submergence or spray from RBCCW piping is not a significant contributor to risk since the piping is sheathed and the system does not a volume large enough to submerge components in the basement of the reactor buildings.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A5	Closed	IFSN-A5-01	Table 6-1 in Section 6.1 is intended to also identify SSCs for each flood area. However, no SSCs are listed in this table. The only section that includes the SSCs by location is in Table 6-3C, Appendix A.2, and Appendix H. However, the flood damage susceptible components listed in Table 6-3C are high level, descriptive (does not distinguish between MOVs/AOVs, etc. and does not include component IDs). Both Table 6-3C and Appendix A.2 only include SSCs for locations that were walked down. Similarly, Appendix H does not include all flood areas either. The information related to SSCs should include the full component IDs (tag numbers), not just the train designation and descriptive name. Selected information collected during plant walkdowns should be documented in Appendix A walkdown sheets (e.g., spray shield, whether the component is located within the spray impact range, etc.).	N/A	Due to the number of PRA components (components modeled in the PRA that have the potential to affect the mitigation of core damage or large early release) in the flood areas, they are now delineated in Appendix A “Walkdown Notes” and Appendix H “Flooding Initiating Events and Impacts” of the analysis. They include the component ID numbers. A component location table, Appendix I “Moderate Energy Line Break Analysis (Unit 1), has also been included that delineates, in addition to the component ID numbers, the component locations. The main body of the report was changed to reflect this. Appendix H “Flooding Initiating Events and Impacts” of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis” has been updated to include all initiating events for which effects need to be included in the PRA. Flood areas that are screened are not included in this appendix. Appendix A ”Walkdown Notes” of Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis” has been updated to include Information collected during plant walkdowns (e.g., spray shield, whether the component is located within the spray impact range, etc.).	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A6	Closed	IFSN-A6-01	The effects of high energy line breaks for Main Steam, Feedwater, RWCU, HPCI steam supply line, and RCIC steam supply line (e.g., jet impingement, high temperature/humidity, pipe whip, etc.) are not fully addressed and accounted for in the flood scenario analysis (see Section 6.5 under Initiating Events). The detrimental effects of the high energy line break could cause damage to cables and other equipment that would not otherwise be failed by water submergence and spray. Although this is a Capability Category III issue, it needs to be considered for such application as Risk-Informed Inservice Inspection of Piping. It is possible that the effects of high energy line breaks were already evaluated in the previous RI-ISI program completed for BFN.	N/A	The High Energy Line Break analysis was performed earlier for BFN for the power uprate. The HELB report has been identified in the reference section for this flooding report. That analysis was limited to break scenarios that were successfully isolated (RWCU, HPCI, and RCIC successfully isolated). Main steam line and feedwater line breaks that are not successfully isolated are treated in the non-flood PRA model with break outside containment events that consider the initiator frequencies based on line lengths.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	Status	F&O ID	Finding	F&O Recommendations	Resolution	Impact to EPU
IFSN-A6	Closed	IFSN-A6-02	The water spray effects may not have been modeled consistently for all flooding scenarios considered. In many instances, the decisions to not quantitatively evaluate the flooding scenarios were based on the consideration of PRA equipment damage due to water submergence only (i.e., without considering the damage effects of water spray). For example, only two flooding scenarios were quantitatively considered for the Control Bay, while there may be other spray damage scenarios that should have been quantitatively evaluated.	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" Revision 0," has been revised and all spray scenarios from all sources have been considered and either modeled or justification provided for not modeling the spray source.	No impact.
IFSN-A8	Closed	IFSN-A8-01	No actual consideration was given in the evaluation for inter-area propagation through drain lines or back flow through drain lines due to failed back flow prevention devices (e.g., check valves or other isolation valves).	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" has been revised to include the following information. The diesel generator flood areas are served by large (24") drains to the outside so there are no propagation paths through drains. The intake pumping station rooms are not interconnected by drains so there are no propagation paths through drains. All of the reactor building drains go to the RB sumps on the 519 level. Most of these drains interconnect on their way to the sumps; however, the same areas have large open hatches or stairwells that go to the 519 level so the drains are immaterial. The only way the drains could cause a problem is if they backed up into a shutdown board room, and the shutdown board rooms do not have any floor drains. The turbine building drains are immaterial due to the way the flooding analysis is performed in that area.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A9	Closed	IFSN-A9-01	A screening value of 0.1 is used for the failure of the door to the air conditioning equipment room at El. 606 in Control Bay (IF-CB593-DOOR for %IFM1CB606-AC). Flooding in this room (resulting from failure of the EECW piping system) could potentially cause water accumulation to a height in excess of several feet according to the flood height analysis performed for 1CB606-ACM (Appendix E). Since this door opens outward from the room, the door could potentially fail with an internal flood height in excess of 1' to 4' (per EPRI draft final guideline for IFPRA). As such, the use of a screening value of 0.1 (without actual structural analysis of the door capability) for scenario %IFM1CB606-AC is probably optimistic. For %IFL1CB606-AC, the flood accumulation in the room could potentially reach to more than 2', which in principle could also cause failure of this door to withstand the static pressure from the flood.	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" has been revised to include the following information. Flooding scenarios within the Control Bay show propagation from the 606' elevation to the stairwell and subsequently to the 593' corridor. At this level, the continued accumulation of flood water will release to the outside through the double door emergency exit doors at the Unit 3 end of the corridor. However, a 0.1 factor was applied to the failure of this emergency door to release flood waters and to cause the propagation to the battery rooms and battery board rooms for the units. This factor of 0.1 is conservative given the glass double door emergency exit opens easily to the outside and the single doors to the adjacent rooms open outward (into the CB corridor).	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-A9	Closed	IFSN-A9-02	Flood height calculations for selected Control Bay scenarios were provided in Appendix E. For Reactor Building and Turbine Building, however, no calculations are provided to demonstrate that selected flood sources would not cause damage to PRA equipment due to flood immersion in the basement. For example, it is indicated in the IFPRA notebook that neither CST has sufficient inventory to result in a flood height severe enough to cause failure of the PRA equipment located at the lowest level in the Reactor Building, but no actual analysis is provided to substantiate that conclusion.	N/A	Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" has been revised to include the following information. Two RB calculations were performed to obtain timing for 2,000 gpm floods (upper limit for Flood) and 24,000 gpm floods (upper limit for Major Floods). These are the only two calculations needed since all reactor building breaks flow to the 519' level without submerging any other area that contains PRA equipment that could be failed by submergence.	No impact.

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSN-B2	Closed	IFSN -B2-03	Information collected during the walkdown should be documented more fully and consistently in the walkdown sheets (Reference e.g., the type of doors (normally open/closed egress door, fire door, door with card key entry, water tight submarine door, etc.), floor/wall/ceiling openings, sumps and sump capacity, sump level instrumentation, number, size, and condition of drains, equipment occupancy fraction, etc.). There are some inconsistencies in the information related to these items presented between different sections of the report. For example, the walkdown sheets show no drain in the corridor area on El. 593' in the Control Bay. However, the flood height evaluation in Appendix E shows 2 drains in this area.	N/A	Additional walk downs were conducted and documented. Plant studies and drawings were examined to locate all of the PRA components in flood areas. . Appendix A "Walkdown Notes" of calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" contains the information collected during the walkdown (Reference e.g., the type of doors (normally open/closed egress door, fire door, door with card key entry, water tight submarine door, etc.), floor/wall/ceiling openings, sumps and sump capacity, sump level instrumentation, number, size, and condition of drains, equipment occupancy fraction, etc.). Appendix A has been reviewed for consistency with the remainder of the document and corrections have been made in the revision 0.	No impact. .

Table A-1 Internal Events PRA F&O Resolution						
SR	**Status**	**F&O ID**	**Finding**	**F&O Recommendations**	**Resolution**	**Impact to EPU**
IFSO-A1	Closed	IFSO -A1-01	Tables 6-1 and 6-2 provide a list of the potential flooding sources. However, some of the plant water and steam systems (e.g., domestic water/potable water/ sanitary water system, chilled water system, hot water system, main steam, etc.) appear to be absent from the evaluation considered in these tables. In addition, there is no documentation of the complete flood sources for locations that were not walked down (the flood sources documentation is geared to the walk down). Flood sources need to be identified by location as the basis for developing flooding scenarios.	N/A	A complete list of flood sources for each flood area has been included in calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis" for the reactor building, control bay, diesel generator buildings and the intake pumping station. The Turbine building is being handled in a manner that does not require detailed listing of flood sources. Tables 6-1 and 6-2 have been updated. The sources have also been listed in Appendix A of Calculation NDN-000-999-2007-0031 Revision 0, "IF - BFN Probabilistic Risk Assessment - Internal Flooding Analysis."	No impact. The differences between the EPU and CLTP results are related to operator timing and probability of a SORV.

A.2 FPRA Technical Adequacy

The BFN FPRA was subjected to three peer reviews – a full scope review, a focused scope follow on peer review, and a focused scope peer review to evaluate specific aspects of the FPRA and assess existing F&O dispositions. The full scope peer review was performed January 23 to 27, 2012. The focused scope follow on peer review was conducted June 25 to 27, 2012. The final focused scope peer review was conducted May 12 to 15, 2015. All peer reviews used the NEI 07-12 process, ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2. The purpose of these reviews was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The full scope peer review examined all of the technical elements of the BFN FPRA against all technical elements in Part 4 of ASME/ANS RA-Sa-2009, including the referenced internal events Supporting Requirements (SRs) in Part 2. The focused scope follow on peer review performed a review against a list of High Level Requirements (HLRs) and SRs that were selected based on the FPRA model changes implemented subsequent to the full scope peer review. The 2015 focused scope peer review evaluated specific changes made to the FPRA and assessed the F&O's from the previous two peer reviews that were considered closed by TVA self review. The final conclusion of the peer reviews was that the BFN FPRA meets Capability Category II following final resolution and closure of all of the F&Os. Most of the F&Os from the full-scope peer review were resolved in the follow-on peer review. The F&Os from the follow-on peer review, some of which remain open, are listed and discussed in Table A-2.

The certification team determined that with these proposed changes incorporated, the quality of all elements of the BFN PRA model is sufficient to support "risk significant evaluations with deterministic input." As a result of the effort to incorporate the latest industry insights into the BFN PRA model upgrades and certification peer reviews, TVA has concluded that the results of the risk evaluation are technically sound and consistent with the expectations for PRA quality set forth in Regulatory Guide 1.174.

The remaining open findings and their applicability to the EPU application are listed in Table A-2.

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
2-38	Open	Finding	The existing BFN procedures are based on the existing SISBO strategy and the current fire model is based on an as yet to be defined non-SISBO strategy for which there are no procedures yet in place. (This F&O originated from SR HRA-A2)	**Basis for Significance** Analysis does not reflect the as-built as-operated plant **Possible Resolution** When non-SISBO procedures are available, incorporate any new fire-specific safe shutdown actions called out in the plant fire response procedures.	Since the peer review, the BFN FPRA team has worked closely with the 805 transition team to match the FPRA recovery actions with those actions proposed and credited by the 805 transition team for the 805 RISK analysis. The FPRA team is only crediting those recovery actions that have been shown to sufficiently reduce CDF. A feasibility study, (Calculation MDQ-000-999-2012-000108 Revision 5, “NFPA 805 Operator Action Feasibility Analysis”) has been performed to demonstrate that the credited actions can be performed in the available time. Calculation NDN-000-999-2012-000011, “TVA FPRA –Task 7.12 Post-Fire Human Reliability Analysis”, Revision 4 documents the BFN FPRA human reliability analysis. Human failure event timing information was obtained from two sources. The total time available was obtained from MAAP analysis by a practitioner who had adequate knowledge of the BFN accident sequences. The cognitive and execution times were obtained both from a PRA practitioner who had previous knowledge of the IE HRA, and from operator interviews (Attachment B of NDN-000-999-2012-000011). Timing information is documented in the HRA calculator files and in the operator interview forms. The operator interview forms instructed the operators to consider the assumed worst case conditions for performing the action with regard to work load, additional procedures, response time during fire conditions, travel time impacted	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on realistic proposed actions, including credit for logical AND of routed redundant instrumentation trains. Therefore, any change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved and adopted, verification must be made to ensure the fire HRA still sufficiently matches the final procedures and that no new

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
					by fire conditions, etc. The FPRA credited actions have been developed to the extent possible to make these HRA's represent those proposed actions. The final fire procedures are not available to complete and verify the fire HRA's. The FPRA model therefore assumes that these actions will be in the final procedures as currently proposed. Before the FPRA recovery actions can be considered complete, they will have to be re-evaluated when the fire procedures are approved and ready to be implemented in the post 805 transition. The new recovery actions are included in the HRA calculator database that is delineated in Attachment D of the fire HRA calculation. This F&O is considered open until the procedures are finalized. (Refer to Attachment S, Table S-3, Item 33).	initiating events are associated with the new procedures.
2-50	Open	Finding	The modeling of the human actions in the FPRA includes the consideration of instruments that are credited as cues. There are several instances that were noted where the listing of possible instrument cues includes many individual devices. This modeling is treated as multiple inputs to a single AND gate, as an example. As modeled, the availability of any single instrument even if the majority of the other instruments are failed, would not disable the human action. This treatment is made without any consideration of or confirmation that operator guidance is available to allow them to discern which instrument is the known valid (not failed) instrument. (This F&O originated from SR HRA-C1)	**Basis for Significance** The treatment as modeled could conceal instances where instrumentation failures have a material impact on the HEP. Failure to address this situation could cause the analysis to apply invalid credit. **Possible Resolution** A justification for the current modeling treatment needs to be provided. Such a justification would need to address the manner by which an operator would be able to discern which instrument should be used and/or how they would recognize the need for action even if the majority of the available instrument might indication that no action is required. In the absence of such a justification, a modification to the logic structure would be required.	Calculation NDN-000-999-2012-000011, "TVA FPRA –Task 7.12 Post-Fire Human Reliability Analysis", Revision 4 documents the BFN FPRA human reliability analysis. The Fire HRA Notebook calculation includes a discussion on the treatment of the instrumentation. Every routed instrument train that was credited by an HFE was included in the modeling. The redundant instruments are still AND'ed together and an assumption is made that the fire procedures will include the impacted instrumentation for fires in the respective area. Therefore, as long as one instrument is available and the operators know, from the applicable fire procedure, which instrument that is, that instrument can be credited even though the redundant instruments are impacted by the fire.	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on consideration of realistic proposed actions and the instrumentation that is needed by operators performing the actions. Therefore, any

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
					This F&O is resolved to the extent possible with the current state of the 805 project. The instrumentation cannot be listed in the fire procedures until the procedures are developed. Once the fire procedures are complete, approved and accepted, verification must be made to ensure the operator can determine from the fire procedure which instruments are free of fire damage for the applicable fire scenarios and those instruments are properly credited in the FPRA model. This F&O is considered open until the procedures are finalized. (Refer to Attachment S, Table S-3, Item 33).	change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved and adopted, verification must be made to ensure the fire HRA still sufficiently matches the final procedures and to ensure that operators can determine from the fire procedures which instruments are free of fire damage for the applicable fire scenarios and those instruments are properly credited in the FPRA model.
4-12	Open	Finding	As documented in Section 6.2 of Component Selection report, a review of the fire emergency procedures (FEPs) or similar fire-related instructions was not conducted since the BFN fire safe shutdown strategies will updated as part of the NSCA. The FPRA therefore does not consider modifications of existing internal events accident sequences that will require modification based on unique aspects of the plant fire response procedures. This approach does not reflect the as-built as operated plant. (This F&O	**Basis for Significance** Step not performed **Possible Resolution** Consider modifications of existing internal events accident sequences that will require modification based on unique aspects of the plant fire response procedures when it is available.	Calculation NDN-000-999-2012-000011, "TVA FPRA –Task 7.12 Post-Fire Human Reliability Analysis", Revision 4 documents the BFN FPRA human reliability analysis. A review of the EOIs for all three units was performed and documented in the analysis. The fire procedures when complete will be reviewed for infeasible operator actions. If undesired operator actions are identified, either the procedure will be modified to eliminate the potential action or the potential	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	Status	Type	Fact/Observation	Basis for Significance/Possible Resolution	Disposition	Impact to EPU
			originated from SR PRM-B5)		action will be modeled and its risk significance determined. This review will include the main control room abandonment procedures. If modifications to the existing internal events accident sequences require modification based on unique aspects of the plant fire response procedures after the procedures are approved, calculation NDN-000-999-2012-000015, "TVA FPRA – Subtask 7.2.1 FPRA Component Selection," Revision 1, will be updated to reflect the required changes. This F&O is considered open until the procedures are finalized. The fire HRA will be updated upon completion of procedure updates, modifications and training. (Refer to Attachment S, Table S-3, Item 33).	FPRA based on realistic proposed actions, including consideration of the potential for undesired operator actions that could lead to new accident sequences. Therefore, any change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved and adopted, verification must be made to ensure the fire HRA still sufficiently matches the final procedures and that no new initiating events or accident sequences are associated with the new procedures. If modifications to the existing internal events accident sequences are found based on unique aspects of the plant fire response procedures after the procedures are

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
						approved, the HRA will be updated accordingly.
4-17	Open	Finding	In considering whether there are possible new scenarios not addressed in the Internal Events PRA that should be considered for the FPRA resulting in additional equipment that needs to be included in the FPRA, Section 6.2 states that the following was performed with the observations documented. (1) Considered sequences screened out of the Internal Events PRA that may become relevant to the FPRA and need to be implemented in the FPRA Model. A review was conducted for such scenarios, originally eliminated from the Internal Events PRA, to determine if the analyst needs to add components to the FPRA Component List, as well as, model those components (and failure modes) in new sequences in the FPRA Model; (2) Considered the possible effects of spurious operations that may result in new accident sequences and associated components of interest that should be addressed in the FPRA and go beyond considerations in the Internal Events PRA. Typically, these new sequences arise as a result of spurious events that cause a LOCA: e.g., spurious opening of safety relief valves, Adversely affect plant pressure control: e.g., safety relief valve events, Allow overfill situations: e.g., reactor vessel overfill that if unmitigated could subsequently fail credited safe shutdown equipment such as HPCI or RCIC pumps, or Introduce other “new” scenarios that may not be addressed in the Internal Events PRA; and (3) A review of the fire emergency procedures (FEPs) or similar fire-related instructions was not conducted since the BFN fire safe shutdown strategies will be updated as part of the NSCA. To the extent that the associated human actions and their effects will be explicitly included in the FPRA Model, new sequences and	**Basis for Significance** Insufficient documentation **Possible Resolution** Document a review of any new accident sequences, including timing considerations not in the internal events, including a review of fire emergency procedures.	A review was conducted of 1) screened initiating events from the internal events PRA model documentation, and 2) MSO impacts on plant safe shutdown and on the potential for new initiating events. The results of this review are documented in calculation NDN-000-999-2012-000015, “TVA FPRA –Subtask 7.2.1 FPRA Component Selection," Revision 1, the Component Selection report, subtask 7.2.1, section 6.2, and Table 16. The review included an evaluation of generic and plant specific MSO scenarios to identify the potential for any unique failure impacts. No new sequences were identified which were not already included in the FPRA model, or adequately addressed by system logic models as modified for the FPRA. A review of fire emergency procedures will be performed after procedure development is complete.	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on realistic proposed actions. A review that was conducted to identify any new initiating events or accident sequences did not identify any new initiating events or accident sequences associated with the proposed actions. Therefore, any change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved and adopted,

Table A-2. BFN FPRA Open Peer Review Findings

F&O No.	Status	Type	Fact/Observation	Basis for Significance/Possible Resolution	Disposition	Impact to EPU
			corresponding components may need to be included in the FPRA. It should be recognized that some of the human actions from these potentially new sequences may have to be addressed in the FPRA. Examples are: The Internal Events PRA likely will not have addressed main control room abandonment scenarios where fire-specific operator actions and equipment sets are relied upon; Fire-specific manual actions designed to preclude or overcome spurious operations will likely not have been addressed in the Internal Events PRA. Other procedural actions may address a degraded barrier, or deal with a breaker coordination problem, among others; Fire specific manual actions may cause intentional failure of a safe shutdown function or a subset of that functional response. For example, a proceduralized action may be to trip a power supply thereby disabling ("failing") certain equipment in the plant. The effect of this action should be implemented in the FPRA Model by acknowledging the affected components in the FPRA Component List and noting the success of the proceduralized human action as a "failure mode" of that component in the FPRA Model (including any new resulting accident sequences as appropriate). Table 9 of the CS notebook provides this review for new accident sequences. However, Table 9 does not provide much information. It lists the following considerations: Spurious opening of one or more safety relief valves, Spurious closure of all MSIVs, Loss of Condenser Vacuum, Loss of Feedwater, and Turbine Bypass Unavailable. The expectation would be to document the entire review to accomplish the above steps, such as (examples only) 1) examining all MSO scenarios for potentially new accident sequences (e.g., overfill as an initiating event); 2) fire-induced floods, from causes such as: a. system relief valves opening due to system overpressurization			verification must be made to ensure the fire HRA still sufficiently matches the final procedures and that no new initiating events or new accident sequences are associated with the new procedures. If modifications to the existing internal events accident sequences are found based on unique aspects of the plant fire response procedures after the procedures are approved, the HRA will be updated accordingly.

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
			that result from spurious operations (not the SRVs, but relief valves designed to protecting from system overpressure), b. spurious opening of system drain valves, or c. water hammer; examples are: i) fire water system actuates and isolation valve spuriously closes, ii) keep fill pump for injection system fails, pump outlet piping drains and pump starts, iii) drain valve spuriously opens on pump outlet piping, draining the piping and pump receives signal to start, etc. d. fire-specific ISLOCA leakage sources; 3) Loss of power to the control room annunciator tile boards. (This F&O originated from SR PRM-B5)			
4-21	Open	Finding	The review of EOIs and annunciator response procedures for instruments applicable to undesired operator actions is documented in Section 5.6.1 and 5.6.2 of HRA notebook, and Attachment F. However, fire emergency procedures and control room abandonment procedures were not reviewed, since these procedures employ the SISBO approach. Therefore, review is not for the as-built as operated plant. (This F&O originated from SR ES-C2)	**Basis for Significance** Incomplete analysis **Possible Resolution**	Calculation NDN-000-999-2012-000011, "TVA FPRA –Task 7.12 Post-Fire Human Reliability Analysis", Revision 4 documents the BFN FPRA human reliability analysis. A review of the EOIs for all three units was performed and documented in the analysis. The fire procedures when complete will be reviewed for infeasible operator actions. If undesired operator actions are identified, either the procedure will be modified to eliminate the potential action or the potential action will be modeled and its risk significance determined. This review will include the main control room abandonment procedures. This F&O is considered open until the procedures are finalized. The fire HRA will be updated upon completion of procedure updates, modifications and training. (Refer to Attachment S, Table S-3, Item 33).	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on realistic proposed actions, including consideration of the potential for undesired operator actions. Therefore, any change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
						the new procedures have been completed, approved and adopted, verification must be made to ensure the fire HRA still sufficiently matches the final procedures and that no undesired operator actions are associated with the new procedures. If undesired operator actions are identified, either the procedure will be modified to eliminate the potential action or the potential action will be modeled and its risk significance determined. This review will include the main control room abandonment procedures.
9-4	Open	Finding	All of the recovery actions were not included in the dependency analysis. (This F&O originated from SR HR-H3, FQ-A3, FQ-D1, HRA-D2)	**Basis for Significance** Incomplete dependency analysis. **Possible Resolution** Ensure all recovery actions are included in the final dependency analysis.	All of the recovery actions used by the FPRA have been included in the dependency analysis, The dependency analysis is documented in Calculation NDN-000-999-2012-000011, “TVA FPRA –Task 7.12 Post-Fire Human Reliability Analysis”, Revision 4. Since the peer review, the BFN FPRA team has worked closely with the 805 transition team to match the FPRA recovery actions with those actions proposed and credited by the 805 transition team for the 805 Risk	Minimal impact. Although BFN procedures that will be put in place upon transition to NFPA 805 are not yet finalized, corresponding HFEs and HEPs have been developed for the FPRA based on realistic proposed

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
					analysis. The FPRA team is only crediting those recovery actions that have been shown to sufficiently reduce CDF. The FPRA credited actions have been developed to the extent possible to make these HRA's represent those proposed actions. The final fire procedures are not available to complete and verify the fire HRAs. The FPRA model therefore assumes that these actions will be in the final procedures as currently proposed. Before the FPRA recovery actions can be considered complete, they will have to be re-evaluated when the fire procedures are approved and ready to be implemented in the post 805 transition. (Refer to Attachment S, Table S-3, Item 33). Sections 5.1.3 and 5.1.4 of the fire HRA calculation include discussion on how this F&O was addressed. Data used in the FPRA is provided in Table 5 and Table 5a. Additional discussion was placed in section 5.7. The recovery actions are also included in the HRA calculator database that is delineated in Attachment D of the fire HRA calculation. This F&O is considered open until the procedures are finalized. The fire HRA will be updated upon completion of procedure updates, modifications and training. (Refer to Attachment S, Table S-3, Item 33).	actions, including modeling of dependencies between HFEs. Therefore, any change in risk estimated by the FPRA models for EPU is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation. When the new procedures have been completed, approved and adopted, verification must be made to ensure the fire HRA still sufficiently matches the final procedures and that HFE dependencies introduced by the new procedures are properly accounted for.
CF-A1-01	Open	Finding	The revised methodology for CFMLA, documented in Section 6.0 of EDQ0009992012000110, is consistent with the latest industry guidance in NUREG/CR-7150, Vol. 2. A review of likelihood values assigned in Appendix C of EDQ0009992012000110 identified some anomalies for specific circuit configurations related to Design Change Notice (DCN 71214), which is described in EDQ0009992012000110,	**Basis for Significance:** Failure to accurately represent the modification could potentially result in mischaracterization of the fire risk associated with spurious operation of the valves and the modification. Use of the inter-cable spurious operation values instead of the aggregate values per the guidance in NUREG/CR-7150 could under-	Calculation EDQ0009992012000110, "Circuit Failure Mode Likelihood Analysis" Revision 8 was developed to correct the modeling of the impacts of DCN 71214 and address the panel wiring and modification alignment issues identified in the finding. Updated circuit failure probabilities were developed accordingly.	This is a new finding from the 2015 focused peer review. A sensitivity study was performed that incorporated the corrected circuit failure probabilities. This sensitivity study

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
			Attachment 2. This DCN electrically reconfigures the control circuit for Unit 1, 2, and 3 RHR System valves to resolve spurious operation of these valves. 1-FCV-074-0060, 1-FCV-074-0071, 1-FCV-074-0074, 2-FCV-074-0057, 2-FCV-074-0060, 2-FCV-074-0071, 2-FCV-074-0074, 3-FCV-074-0057, 3-FCV-074-0060, 3-FCV-074-0074 Based on Attachment 2 of EDQ0009992012000110, DCN 71214 is intended to prevent spurious operation of the components except at cable endpoints, which are identified in Attachment 2. The spurious operation probabilities assigned in Appendix C of EDQ0009992012000110 for these valves are assigned a value of zero except for areas of the endpoints, where a spurious operation value is provided. Several issues were identified: 1) Panel Wiring - The spurious operation value of 1.30E-02 is provided, which corresponds to single break circuits, ungrounded ac, thermoplastic insulated cable spurious operation probability for inter-cable hot shorts (Table 8-1 NUREG/CR-7150, Vol. 2). However, the failure mode of concern does not appear to account for the electrical panels where the cables terminate. NUREG/CR-7150 Volume 2, Section 7.4 provides the following guidance for panel wiring. Panel Wiring There are no test data for evaluating the likelihood of hot short-induced spurious operations for panel wiring. A hot short in the panel wiring's conductor bundles within a cabinet could behave similarly to any of the failure modes of an electrical cable (i.e., intra-cable, inter-cable, or GFEHS) depending upon the proximity of the conductors to the fire and the tightness of their bundles. The conditional probability of hot short-induced spurious operation most likely is affected by the configuration and tightness of the	estimate the risk of spurious operation of the valves for fires that impact the cable endpoints. Update the treatment of spurious operation of the these valves to align with the modification scope **Possible Resolution:** Review the treatment of spurious operations of panel wiring and update the treatment per NUREG/CR-7150, Volume 2, if the risk significance of the scenarios warrants this treatment (per SR CF-A1). Note that these modifications have not yet been implemented in the Fire PRA results per Note 2 of Appendix E of EDQ0009992012000110. Alternatively, additional design measures could be implemented to reduce the vulnerability of the panel wiring to hot short induced spurious operation.		found that the delta risks meet the numerical guidelines of Regulatory Guide 1.174.

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	Status	Type	Fact/Observation	Basis for Significance/Possible Resolution	Disposition	Impact to EPU
			conductor bundles, along with the proximity of source and target cables. Considering the lack of applicable test data and the potential risk importance of panel wiring, the PRA panel recommends using aggregate values in the tables in Sections 4 and 5. The aggregate value for Table 8-1 of NUREG/CR-7150, Vol. 2 for Ungrounded ac, thermoplastic insulated cable spurious operation probability for MOVs is 0.39. Therefore, by applying value for inter-cable hot shorts exclusively in a fire area, it may not account for the higher spurious operation probability for the panel wiring. 2) Alignment with Modification - In addition, based upon discussion with TVA and PRA project staff, it appears that the values proposed in Appendix C of EDQ0009992012000110, Revision 7, were based on routing of the spurious operation target cable in conduit, (based on modification descriptions provided in Attachment 2 of EDQ0009992012000110). The proposed treatment was that the cable in conduit routed in endpoint fire areas would be subject to spurious operation (i.e., not provided with shielded, braided protection). Based on a 5/14/15 conference call, the modification was going to provide shielded braided cable for the entire route of the target cable, except at the specific terminal locations and not be subject to spurious operation, except at the terminal endpoints (i.e., the potential spurious operation of the target cable in conduit was not part of the proposed design). Therefore, the CFMLA modeling described in Appendix C of EDQ0009992012000110 did not appear to match the proposed modification. While it is expected that checks and balances in the plant modification process could refine the CFMLA following completion of the modification, the current treatment of the proposed			

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
			modification does not appear to align with the modification or industry guidance.			
PRM-B9-01	Open	Finding	This F&O supersedes and incorporates the issues identified in F&O 9-2 from Table A7 of the BFN Fire Probabilistic Risk Assessment Summary Document, calculation # NDN-000-999-2012000096. The wording from 9-2 is: "The new Safe Shutdown Injection Pump is currently modeled as a single event with a probability of 0.1. This system is still in the conceptual phase. This F&O is written as a placeholder to model the system in detail at some future date." The purpose of this F&O is to ensure that the as-built of the Emergency High Pressure Makeup (EHPM) System clearly documents the following in order to fully comply with standard ASME-RA-S-2009 (latest revision). Key safety functions satisfied by the system - Success criteria for the system Ensure the as-built modification does not introduce any new initiating events during any plant mode of operation Phenomenological effects on the system System dependencies such as cooling, electrical, etc. Time dependencies such as water volume depletion Key assumptions Any unit cross connects and their effect on success criteria , accident progression and new initiating events Review any relevant plant experience to ensure potential for new initiators is addressed and any precursor for an initiating event is addressed Potential for common cause failures	**Basis for Significance:** Ensure complete documentation so that the as modeled fire PRA aligns with the as-built and as-operated plant and is compliant with the ASME standard. **Possible Resolution:** Follow established processes for a new system and ensure the system notebook documents items listed above as well as listing components in the component selection notebook and follow appropriate protocols.	Calculation NDN-000-NA-2012-000090 "SY.27 - BFN Probabilistic Risk Assessment - Fire Safe Shutdown Injection Pump" has been initiated to document the alternate high-pressure injection pump. This notebook will remain in draft form until the system is installed and operational. This is a new system (non-safety related) for Browns Ferry (an independent system for each unit) and will provide a means for providing reactor inventory over a wide range of pressures at a flow-rate at or above the capability of RCIC. The following attributes are built into the S&L (Sargent & Lundy) design modification.100% Capacity electric motor-driven pump per unit Piping, Valves and Instrumentation for the following Makeup supply from the condensate storage tank (CST) header through a duplex strainer Backup Capability to Refill the CST with Raw Water Pump discharge to the Feedwater System injection line Full-flow return test line to the CST Electrical Distribution Supply Control Power On-Site 4.16 kV Power Supply (Unit Board) Fukushima Diesel-Generator 4.16 kV Power Supply Switchgear (Distribution Board)	This is a new finding from the 2015 focused peer review. Minimal impact. The FPRA includes logic to reflect the draft design criteria for the EHPM pump. The model includes pump failure to start/run, failure of the injection motor operated and check valves, failure of the power supply, failure of the water supply, and unavailability due to test and maintenance. Any change in calculated risk due to refinement of the EHPM Pump model upon completion of design activities is expected to be small and is not expected to change the conclusions of the EPU FPRA calculation.

Table A-2. BFN FPRA Open Peer Review Findings						
F&O No.	**Status**	**Type**	**Fact/Observation**	**Basis for Significance/Possible Resolution**	**Disposition**	**Impact to EPU**
					Transformer - 480 VAC, 208V/120V Remote (MCR) and Local Operation This new system will be installed for each unit. Logic has been built to reflect the draft design criteria. The model includes pump failure to start/run, failure of the injection motor operated and check valves, failure of the power supply, failure of the water supply, and unavailability due to test and maintenance. The notebook will be finalized after the system is operational. In addition, potential adverse impacts of the system (new initiating events, impact on system dependencies, etc.) will be evaluated as part of the evaluation of the compliance of the system modeling with the ASME-RA-S-2009 Standard.	

A.3 Parametric, Model, and Completeness Uncertainties

Parametric Uncertainty

To verify that the use of point estimates of the internal events PRA results is acceptable for determining the change in risk for EPU, detailed Monte Carlo calculations using EPRI R&R workstation UNCERT software were performed to compare the mean value determined from the Monte Carlo simulation as compared to the point estimate. The parametric means were calculated using 30,000 samples. The results of the parametric uncertainty assessments are provided in Table A-3.

Table A-3. Results of Parametric Uncertainty Study for Internal Events

	CDF (CLTP)	CDF (EPU)	ΔCDF	LERF (CLTP)	LERF (EPU)	ΔLERF
Point Estimate Mean Values						
Unit 1	6.63E-06	7.14E-06	5.10E-07	1.44E-06	1.61E-06	1.70E-07
Unit 2	5.96E-06	6.47E-06	5.10E-07	1.39E-06	1.55E-06	1.60E-07
Unit 3	6.57E-06	7.06E-06	4.90E-07	1.38E-06	1.54E-06	1.60E-07
Propagated Mean Values						
Unit 1	6.63E-06	7.25E-06	6.20E-07	1.43E-06	1.67E-06	2.34E-07
Unit 2	6.12E-06	6.52E-06	4.02E-07	1.45E-06	1.49E-06	4.22E-08
Unit 3	6.66E-06	7.30E-06	6.45E-07	1.34E-06	1.48E-06	1.40E-07

A similar calculation was performed to determine if the use of the point estimate is acceptable for representing the results of the FPRA (40,000 samples were used in the FPRA analysis). The results of the propagated mean sensitivity study for the FPRA are presented in Table A-4.

Table A-4. Results of Parametric Uncertainty Study for Fire

	CDF (CLTP)	CDF (EPU)	ΔCDF	LERF (CLTP)	LERF (EPU)	ΔLERF
Point Estimate Mean Values						
Unit 1	4.78E-05	4.90E-05	1.18E-06	6.05E-06	6.65E-06	6.04E-07
Unit 2	4.72E-05	4.85E-05	1.23E-06	5.96E-06	6.46E-06	5.03E-07
Unit 3	5.17E-05	5.29E-05	1.18E-06	5.16E-06	5.54E-06	3.85E-07
Propagated Mean Values						
Unit 1	4.78E-05	4.86E-05	8.30E-07	5.90E-06	6.71E-06	8.15E-07
Unit 2	4.71E-05	4.83E-05	1.12E-06	5.89E-06	6.51E-06	6.25E-07
Unit 3	5.11E-05	5.35E-05	2.35E-06	5.06E-06	5.55E-06	4.86E-07

The parametric means are very close to the point estimate values and in some cases slightly lower for the Internal Events PRA and the FPRA analyses. This is reasonable because internal events CDF and LERF are driven by common cause failures and dependent human actions and FPRA CDF and LERF are driven by components that are failed as a result of the fire. Failures of correlated events that are not within common cause groups or dependent human action groups have very little effect on the results for either model.

The results of these analyses show that the mean results and point estimates are approximately equal and the bulk of the contributing cutsets do not involve multiple events that rely on the same parameter for their quantification. Therefore, the point estimates are suitable for calculating delta risks. The conclusions drawn when comparing the point estimates and the propagated means with the RG 1.174 acceptance guidelines would not be different.

Model Uncertainty

This analysis perform several sensitivities to address sources of uncertainty associated with model changes from CLTP conditions to EPU conditions.

As discussed in Section 5.7, sensitivities are performed to address the impact:

- The increased likelihood of some initiators in the initial implementation of EPU break-in period.
- Different minimum joint human error probabilities
- The increased likelihood of LOCAs due to higher condensate, feedwater and steam flows and loads associated with EPU
- The increased likelihood of floods due to higher condensate, feedwater and steam flows and loads associated with EPU
- Use of screening HEP values
- Increased probability of RPV overpressure due to higher power

The results of the sensitivity evaluations included in main body of this report in Sections 5.7.1 and 5.7.2.

Completeness Uncertainty

The Internal Events and FPRA models were utilized to obtain quantitative risk metric results as described in Sections 5.1 and 5.2. Section 5.3 and presents a consideration of shutdown risk from the EPU changes. Seismic events were demonstrated to be an insignificant contributor to risk based in Section 5.4. Other external events are addressed in Section 5.5. Attachment E of Reference 26 shows the approach used to estimate, the contribution of seismic events to the total CDF. Both the CLTP and EPU risk assessment used the same bounding estimates for seismic events. Additionally, there are no open items from the recent industry peer review related to model completeness associated with the Internal Events or FPRA models.

Appendix B – Impact of EPU on Operator Action Response Times

HRA Calculator Timing Changes

The following table lists all the events where changes in timing due to EPU were incorporated. The system window (T_{SW}) and delay time (T_d) are based on MAAP calculations (Attachment G of Reference 26).

Table B- 1. CLTP vs. EPU timing changes

BE ID	T(sw)_cltp	T(sw)_epu	Time Unit	T(d)_cltp	T(d)_epu	Time Unit	T(rec)	T(rec)	Description
HFA_0_ADSINHIBIT	522	475	Sec.	200	200	Sec.	4.37	3.58	Failure to inhibit ADS during an ATWS event
HFA_0_ATWSLEVEL	885	800	Sec.	108	194	Sec.	11.95	9.10	Operator Fails to Run Back RFPs and Maintain Level at TAF
HFA_0_SPRAYMLOCA	13.8	13.5	Min.	54	54	Sec.	8.90	8.60	Operator fails to spray drywell during MLOCA w failed Suppression Pool (SP) **
HFA_0001HPRVD1	32	30	Min.	22.8	21	Min.	7.70	7.50	Failure to initiate reactor-vessel depressurization (transient or ATWS)
HFA_0001MSIVATWS	32	30	Min.	0	0	Min.	7.00	5.00	Operator Fails to Bypass Low Level MSIV Closure Setpoint
HFA_0002RPV_LVL	42	35	Min.	2	3.6	Min.	37.00	28.40	Operator Fails To Maintain RPV Level
HFA_0003P_START_A	32	30	Min.	0	0	Min.	26.50	24.50	Operator Fails to Start Standby/Tripped RFW Pumps - ATWS
HFA_0003PMP_START	42	35	Min.	2	3.6	Min.	38.50	29.90	Operator Fails To Restart RFW After Level 8 Trip
HFA_0003RXLVLATWS	164	165	Sec.	51	54	Sec.	0.88	0.85	Operator fails to maintain RPV level (non-ATWS) **
HFA_0032CMP_START	32	30	Min.	0	0	Min.	27.00	25.00	Operator Fails To Manually Start Compressor **
HFA_0063SLCINJECT	522	475	Sec.	0	0	Sec.	4.87	4.08	Failure to SLC in response to an

Table B- 1. CLTP vs. EPU timing changes									
BE ID	**T(sw)_cltp**	**T(sw)_epu**	**Time Unit**	**T(d)_cltp**	**T(d)_epu**	**Time Unit**	**T(rec)**	**T(rec)**	**Description**
									ATWS event
HFA_0071L8RESTART	42	35	Min.	2	3.6	Min.	38.50	29.90	Operator fails to restart RCIC after Level 8 trip
HFA_0071LVL8_TRIP	885	800	Sec.	108	194	Sec.	11.95	9.10	Failure to trip HPCI or RCIC upon reaching RPV level 8
HFA_0071MANLEVEL	885	800	Sec.	108	194	Sec.	11.95	9.10	Operator fails to manually control level with RCIC
HFA_0073L8RESTART	42	35	Min.	2	3.6	Min.	38.50	29.90	Operator fails to restart HPCI after Level 8 trip
HFA_0073MANLEVEL	885	800	Sec.	108	194	Sec.	11.95	9.10	Operator fails to manually control level with HPCI
HFA_02114KVCRSTIE	32	30	Min.	0	0	Min.	22.00	20.00	Failure to crosstie de-energized 4kV shutdown board to energized shutdown board **
HFA_0231480SDBTIE	32	30	Min.	0	0	Min.	26.00	24.00	Failure to transfer 480V shutdown board to alternate source **
HFA_0268480CRSTIE	32	30	Min.	0	0	Min.	17.00	15.00	Failure to transfer de-energized 480v board to alternate supply
HFA_0HCIINIT30	42	35	Min.	2	3.6	Min.	38.00	29.40	Operator Fails To Initiate HPI (30 Min)
HFA_0LPIINIT10	42	35	Min.	0	0	Min.	22.00	15.00	Operator Fails To Manually Initiate Low Pressure Injection (10 Min)
HFA_0LPIINIT30	32	30	Min.	22.8	21	Min.	7.70	7.50	Failure to establish low-pressure injection given loss of high pressure injection **

Note: The Tsw or Td changed in the 22 basic events in the table above, and the HRA calculator yielded 16 operator actions where the HEP changed (i.e. the timing change was not significant for six events denoted by ** in the description column). However, 17 HEPs were changed in the PRA model. Event HFA_0073LVL8_TRIP (not in the list above) was manually changed. This event used the same HEPs for both HPCI and RCIC.

Time Delay Override Changes

List of adjusted time delay overrides to address inappropriate sequencing of CLTP HEPs due to conservative MAAP timing.

- A time delay override was used to sequence event HFA_0071LVL8_TRIP after HFA_0071MANLEVEL since failure to manually control level precedes failure to trip the pump prior to reaching L8.
- A time delay override was used to sequence event HFA_0073LVL8_TRIP after HFA_0073MANLEVEL since failure to manually control level precedes failure to trip the pump prior to reaching L8.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0HCIINIT30 since failure to initiate high pressure injection precedes failure to control level with the lower pressure condensate system.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0071L8RESTART since failure to restart a high pressure injection (071) precedes failure to control level with the lower pressure condensate system.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0073L8RESTART since failure to restart a high pressure injection (073) precedes failure to control level with the lower pressure condensate system.

List of adjusted time delay overrides to address inappropriate sequencing of EPU HEPs due to conservative MAAP timing.

- A time delay override was used to sequence event HFA_0071LVL8_TRIP after HFA_0071MANLEVEL since failure to manually control level precedes failure to trip the pump prior to reaching L8.
- A time delay override was used to sequence event HFA_0073LVL8_TRIP after HFA_0073MANLEVEL since failure to manually control level precedes failure to trip the pump prior to reaching L8.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0HCIINIT30 since failure to initiate high pressure injection precedes failure to control level with the lower pressure condensate system.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0071L8RESTART since failure to restart a high pressure injection (071) precedes failure to control level with the lower pressure condensate system.
- A time delay override was used to sequence event HFA_0002RPV_LVL after HFA_0073L8RESTART since failure to restart a high pressure injection (073) precedes failure to control level with the lower pressure condensate system.

In addition, an assumed time delay override of 24 hours (and identical tcog, texec, and tsw) was used for all the human errors that used a screening value and appeared in combinations in the HRA dependency file. This method makes all screening values completely dependent on each other, and independent from other HEPs. This means that if a cutset has multiple screening values, a maximum of one is credited. The list of these events is shown below:

1. HFA_0074UNITXTIE
2. HFA_0032LEAK_ISO
3. HFA_0LPIINIT15
4. HFA_0085ALIGNFCV
5. HFA_0085REFILLCST

A time delay override was not used for the following screening events since there were no combinations in the HRA dependency file containing these events.

1. HFA_0_LCISTARTATWS
2. HFA_0_SPRAYIOOV
3. HFA_0_SPRAYLLOCA
4. HFA_0_VSSDEP
5. HFA_0099MGRESET
6. HFA_0HCIINIT10
7. HFA_0HCIINIT15
8. HFA_0LPIINIT06
9. HFA_PARALLEL_DG

Finally, a sensitivity was done for a Unit 3 modification that impacted the automatic transfer of power to the unit bus (DCN #51052) during specific electrical configurations. An operator action was added to represent failure to align the alternate supply when 480V SDBD is fed by transformer TS3E. However, the sensitivity yielded no cutsets even when the event was set to a probability of 1.0. The event is listed below.

- HFA_ALTP_UB

Appendix C – Description for Human Error Events

Table C-1 provides a description of the human error basic events used in the Internal Events PRA.

Table C-1 - Human Error Probability Events for Internal Events PRA	
Operator Error	**Description**
HFA_0_ADSINHIBIT	Failure To Inhibit ADS During An ATWS Event
HFA_0_ATWSLEVEL	Operator Fails To Run Back RFPs And Maintain Level At TAF
HFA_0_LCISTARTATWS	Operator Fails To Properly Control Start Of LPI
HFA_0_MSLOCADEP	Failure To Initiate Reactor-Vessel Depressurization (Medium Steam LOCA)
HFA_0_MWLOCADEP	Failure To Initiate Reactor-Vessel Depressurization (Medium Liquid LOCA)
HFA_0_RXMAJORFLOOD	Mitigation Of Major RCW Or EECW Rx Bldg Flood
HFA_0_SPRAYIOOV	Operator Fails To Spray Drywell During IOOV W Failed Suppression Pool
HFA_0_SPRAYLLOCA	Operator Fails To Spray Drywell During LLOCA W Failed Suppression Pool
HFA_0_SPRAYMLOCA	Operator Fails To Spray Drywell During MLOCA W Failed Suppression Pool
HFA_0_VSSDEP	Operator Fails To Depressurize RPV W Failed Suppression Pool
HFA_0001HPRVD1	Failure To Initiate Reactor-Vessel Depressurization (Transient Or ATWS)
HFA_0001HPRVD1_L	Failure To Initiate Reactor-Vessel Depressurization (Late Transient Or ATWS)
HFA_0001HPRVD2	Operator Fails To Initiate Depressurization (LERF)
HFA_0001HPRVD2_L	Operator Fails To Initiate Depressurization (LERF)
HFA_0001HPRVD5	Operator Fails To Initiate Depressurization (LERF)
HFA_0001MSIVATWS	Operator Fails To Bypass Low Level MSIV Closure Setpoint
HFA_0002CND_START	Operator Fails To Start Late Condensate
HFA_0002LVLCNTRL	Operator Fails To Control Hotwell Level
HFA_0002RPV_LVL	Operator Fails To Maintain RPV Level
HFA_0003P_START_A	Operator Fails To Start Standby/Tripped RFW Pumps - ATWS
HFA_0003PMP_START	Operator Fails To Restart RFW After Level 8 Trip
HFA_0003RXLVLATWS	Operator Fails To Maintain RPV Level (Non-ATWS)
HFA_0023ALIGNEECW	Operator Fails To Align Backup EECW Pump
HFA_0023ALIGNEECW_L	Operator Fails To Align Backup EECW Pump
HFA_0023SBCI	Failure To Initiate Standby Coolant Injection
HFA_0023STARTEECW	Failure To Start Standby EECW Pump
HFA_0024RCW_START	Operator Fails To Start Backup RCW Pumps
HFA_0024RCWINTAKE	Failure To Clear Debris At Intake Before Reactor Scram
HFA_0024RCW-M	Failure To Isolate RCW Flood Before Depth Fails Pumps In TB
HFA_0031STARTHVAC	Failure To Start Standby Control Building HVAC
HFA_0032CMP_START	Operator Fails To Manually Start Compressor
HFA_0032LEAK_ISO	Operator Fails To Isolate Leak

Table C-1 - Human Error Probability Events for Internal Events PRA	
Operator Error	**Description**
HFA_0032MSIV_N2	Operator Fails To Align Backup Nitrogen To MSIVs
HFA_0033HVACDOOR	Failure To Open Doors And Install Fans After HVAC Failure
HFA_0063SLCINJECT	Failure To SLC In Response To An ATWS Event
HFA_0064DWVENT	Operator Fails To Initiate Drywell Vent
HFA_0064HWWV	Failure To Use Hardened Wetwell Vent For Long-Term DHR
HFA_0064PCICLOSE	Operator Fails To Manually Close Primary Containment Isolation Valves
HFA_0064SCVBCLOSE	Operator Fails To Close Torus Vacuum Breaker Isolation Valves
HFA_0071CTLPOWER	Operator Fails To Transfer To Backup Power
HFA_0071L8RESTART	Operator Fails To Restart RCIC After Level 8 Trip
HFA_0071LVL8_TRIP	Failure To Trip HPCI Or RCIC Upon Reaching RPV Level 8
HFA_0071MANLEVEL	Operator Fails To Manually Control Level With RCIC
HFA_0073L8RESTART	Operator Fails To Restart HPCI After Level 8 Trip
HFA_0073LVL8_TRIP	Failure To Trip HPCI Upon Reaching RPV Level 8
HFA_0073MANLEVEL	Operator Fails To Manually Control Level With HPCI
HFA_0074ALIGN_DWS	Failure To Align Drywell Spray And Gain Spray Valve Control
HFA_0074HPSPC1	Failure To Align RHR For Suppression Pool Cooling (Non-ATWS/IORV)
HFA_0074HPSPC2	Failure To Align RHR For Suppression Pool Cooling (ATWS Or IORV)
HFA_0074RHR_CST	Operator Fails To Align RHR Pumps To CST
HFA_0074SDC_ALIGN	Operators Fails To Align SDC
HFA_0074SPCLATE	Failure To Align RHR For Suppression Pool Cooling In The Long Term
HFA_0074UNITXTIE	Operator Fails To Initiate Unit To Unit Cross Tie
HFA_0075CSCST	Operator Fails To Align Core Spray Pumps To CST
HFA_0084CADALIGN	Operator Fails To Align Cad Backup To DCA
HFA_0084PCAALIGN	Operator Fails To Align PCA Backup To DCA
HFA_0085ALIGNCST	Failure To Align Additional Inventory For CST - Crosstie & Levelize CSTs
HFA_0085ALIGNFCV	Operator Fails To Un-Isolate Flow Control Train
HFA_0085ALIGNFLT	Operator Fails To Un-Isolate Standby CRD Filter
HFA_0085CRDALIGN	Operator Fails To Align CRD System For Injection
HFA_0085MANUALFCV	Operator Fails To Take Manual Control Of CRD Flow Control Valves 11a Or 11b
HFA_0085MAXCRD	Failure To Maximize CRD Flow For RPV Injection
HFA_0085REFILLCST	TSC Operator Action To Refill The CST
HFA_0085TESTOPEN	Operator Fails To Detect Low CRD Flow And Open Test Line
HFA_0099MGRESET	Operator Action To Reset MG Sets
HFA_02114KVCRSTID	Failure To Crosstie De-Energized 4kv Shutdown Board To Energized SD BD (Battery Depletion)
HFA_02114KVCRSTIE	Failure To Crosstie De-Energized 4kv Shutdown Board To Energized

Table C-1 - Human Error Probability Events for Internal Events PRA	
Operator Error	**Description**
	Shutdown Board
HFA_0231480SDBTID	Failure To Transfer 480V Shutdown Board To Alternate Source (Battery Depletion)
HFA_0231480SDBTIE	Failure To Transfer 480V Shutdown Board To Alternate Source
HFA_0248ALNALTCHG	Failure To Align Alternate Battery Charger
HFA_0248ALNPWRSUP	Operator Fails To Align Alternate Power Supply
HFA_0248ALNSPRCHG	Operator Fails To Align To Spare Charger (0-Chga-248-000s)
HFA_0268480CRSTIE	Failure To Transfer Deenergized 480v Board To Alternate Supply
HFA_0280ALNALTBBD	Operator Fails To Align Alternate Feeder
HFA_0EDGTRIP	Operator Fails To Trip EDG If No Cooling Available
HFA_0FD2	Operator Fails To Manually Initiate Injection For Containment Flooding
HFA_0HCIINIT10	Operator Fails To Initiate HPI (10 Min)
HFA_0HCIINIT15	Operator Fails To Initiate HPI (15 Min)
HFA_0HCIINIT30	Operator Fails To Initiate HPI (30 Min)
HFA_0IR1_HPI	Operator Fails To Manually Initiate Injection For In-Vessel Recovery
HFA_0IR2_LPI	Operator Fails To Manually Initiate Injection For In-Vessel Recovery
HFA_0LKISL-CS	Failure To Isolate ISLOCA: Leak In CS Injection Path
HFA_0LKISL-RHR	Failure To Isolate ISLOCA: Leak In RHR (LPCI) Injection Path
HFA_0LPIINIT06	Operator Fails To Manually Initiate LPI (6 Min)
HFA_0LPIINIT10	Operator Fails To Manually Initiate Low Pressure Injection (10 Min)
HFA_0LPIINIT15	Operator Fails To Manually Initiate LPI (15 Min)
HFA_0LPIINIT30	Failure To Establish Low-Pressure Injection Given Loss Of High Pressure Injection
HFA_0SBDALTDC	Operator Fails To Isolate SD BD D And Align Alternate Dc
HFA_0SMSL_RHR	Failure To Isolate ISLOCA: Break In RHR (LPCI) Injection Path
HFA_0SMSL-CS	Failure To Isolate ISLOCA: Break In CS Injection Path
HFA_0SMSL-RHR	Failure To Isolate ISLOCA: Break In RHR (LPCI) Injection Path
HFA_0TD2_HPI	Operator Fails To Manually Initiate Injection Into Drywell After Core Damage
HFA_0TD2_LPI	Operator Fails To Manually Initiate Injection Into Drywell After Core Damage
HFA_ALTP_UB	Failure To Align Alt Supply When 480v SD BD Feed By TS3E
HFA_PARALLEL_DG	Failure To Parallel Diesel Generator