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July 15, 2013

Serial: BSEP 13-0070

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805 (NRC TAC Nos. ME9623 and ME9624)

References:

1. Letter from Michael J. Annacone (Carolina Power & Light Company) to U.S. Nuclear Regulatory Commission (Serial: BSEP 12-0106), *License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)*, dated September 25, 2012, ADAMS Accession Number ML12285A428

2. Letter from Michael J. Annacone (Carolina Power & Light Company) to U.S. Nuclear Regulatory Commission (Serial: BSEP 12-0140), *Additional Information Supporting License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)*, dated December 17, 2012, ADAMS Accession Number ML12362A284

3. Letter from Christopher Gratton (USNRC) to Michael J. Annacone (Carolina Power & Light Company), *Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805 (TAC Nos. ME9623 and ME9624)*, dated May 15, 2013, ADAMS Accession Number ML13123A231

Ladies and Gentlemen:

By letter dated September 25, 2012 (i.e., Reference 1), as supplemented by letter dated December 17, 2012 (i.e., Reference 2), Duke Energy Progress, Inc., formerly known as Carolina Power & Light Company (CP&L), submitted a license amendment request to adopt a new risk-informed performance-based (RI-PB) fire protection licensing basis for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

During the week of April 8 through April 12, 2013, the NRC conducted an audit at BSEP to support development of questions regarding the license amendment request. On May 15, 2013 (i.e., Reference 3), the NRC provided a set of requests for additional information (RAIs) regarding the license amendment request. That letter divided these RAIs into 60-day, 90-day,

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and 120-day responses. In subsequent telephone calls with the NRC Project Manager for BSEP, the following modifications were agreed to regarding the RAI response schedule shown in the May 15, 2013, letter:

- The 60-day RAI responses will be submitted by July 1, 2013 (i.e., 60 days following the May 2, 2013, clarification call that was conducted with the NRC). These responses were actually submitted by letter dated June 28, 2013. Probabilistic Risk Assessment (PRA) RAIs 1A, 1B, 1C, 1D, 1F, 1G, 1I, 1K, 1N, 1O, 1P, 1Q, 1R, 4, 5, 9, 10, 17, and 18, which were included in the set of 60-day RAIs, will be addressed in a separate submittal due by July 15, 2013 (i.e., 60 days following the date of the letter).
- The 90-day RAI responses will be submitted by July 31, 2013 (i.e., 90 days following the May 2, 2013, clarification call). Fire Protection Engineering RAI 1, which was included in the set of 60-day RAIs, will be addressed as part of the 90-day RAI responses.
- The 120-day RAI responses will be submitted by August 30, 2013 (i.e., 120 days following the May 2, 2013, clarification call). PRA RAI 1H will be addressed as part of the 120-day RAI responses, rather than with the 60-day RAI responses.

The response to RAI 1K is not being provided with the enclosed 60-day RAI responses. The response to RAI 1K will be provided as part of the 90-day RAI responses which will be submitted by July 31, 2013.

A tabulation of the individual RAIs and the planned response submittal dates is provided in Enclosure 1. Duke Energy's responses to the set of 60-day PRA RAIs (i.e., due by July 15, 2013) are provided in Enclosure 2.

This document contains no new regulatory commitments.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager – Regulatory Affairs, at (910) 457-2487.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on July 15, 2013, 2013.

Sincerely,

George T. Hamrick

Enclosures:
1. Revised Response Schedule to NFPA 805 Request for Additional Information
2. Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805

WRM/wrm

cc (with enclosures):

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Revised Response Schedule to NFPA 805 Request for Additional Information

Revised Response Schedule		
Section Title	**Question Number(s)**	**Submittal Date**
60-Day Response - Non-PRA		
Programmatic	1, 2, 3, 4, 5, 6, 7	July 1, 2013
Safe Shutdown Analysis	3, 4, 6, 7, 8, 10, 12	
Fire Modeling	1A, 1E, 1F, 1G, 1H, 2A, 2B, 5A, 5B	
60-Day Response – PRA		
Probabilistic Risk Assessment	1A, 1B, 1C, 1D, 1F, 1G, 1I, 1N, 1O, 1P, 1Q, 1R, 4, 5, 9, 10, 17, 18	July 15, 2013
90 Day Response		
Radiation Release	1, 2, 3	July 31, 2013
Fire Protection Engineering	1, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21	
Safe Shutdown Analysis	1, 2, 5, 9, 11, 13, 14	
Probabilistic Risk Assessment	1J, 1K, 1M, 2, 3, 6, 7, 11, 12, 13, 14, 15, 16	
Fire Modeling	1B, 2C, 5C	
120 Day Response		
Fire Protection Engineering	2	August 30, 2013
Safe Shutdown Analysis	15	
Probabilistic Risk Assessment	1E, 1H, 1L, 8	
Fire Modeling	1C, 1D, 1I, 2D, 3, 4, 6	

Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805

By letter dated September 25, 2012, as supplemented by letter dated December 17, 2012, Duke Energy Progress, Inc., formerly known as Carolina Power & Light Company, submitted a license amendment request (LAR) to adopt a new risk-informed performance-based (RI-PB) fire protection licensing basis for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

During the week of April 8 through April 12, 2013, the NRC conducted an audit at BSEP to support development of questions regarding the license amendment request. On May 15, 2013, the NRC provided a set of requests for additional information (RAIs) regarding the license amendment request. That letter divided the RAIs into 60-day, 90-day, and 120-day responses. In subsequent telephone calls with the NRC Project Manager for BSEP, the following modifications were agreed to regarding the RAI response schedule shown in the May 15, 2013, letter:

- The 60-day RAI responses will be submitted by July 1, 2013 (i.e., 60 days following the May 2, 2013, clarification call that was conducted with the NRC). These responses were actually submitted by letter dated June 28, 2013. Probabilistic Risk Assessment (PRA) RAIs 1A, 1B, 1C, 1D, 1F, 1G, 1I, 1K, 1N, 1O, 1P, 1Q, 1R, 4, 5, 9, 10, 17, and 18, which were included in the set of 60-day RAIs, will be addressed in a separate submittal due by July 15, 2013 (i.e., 60 days following the date of the letter).
- The 90-day RAI responses will be submitted by July 31, 2013 (i.e., 90 days following the May 2, 2013, clarification call). Fire Protection Engineering RAI 1, which was included in the set of 60-day RAIs, will be addressed as part of the 90-day RAI responses.
- The 120-day RAI responses will be submitted by August 30, 2013 (i.e., 120 days following the May 2, 2013, clarification call). PRA RAI 1H will be addressed as part of the 120-day RAI responses, rather than with the 60-day RAI responses.

Duke Energy's 60-day response to the PRA RAIs is provided below. The response to RAI 1K, which is not provided herein, will be provided as part of the 90-day RAI responses which will be submitted by July 31, 2013.

<u>Probabilistic Risk Assessment (PRA) Requests for Additional Information</u>

PRA RAI 1A

Clarify the following dispositions to fire F&Os and supporting requirement (SR) assessments identified in Attachment V of the LAR that have the potential to impact the FPRA results and do appear to be fully resolved:

a) F&O 1-8 against ES-A1 (Not Met), ES-A2 (Cat I/II/III), ES-A3 (Not Met), and FQ-A2 (Cat I/II/III):

   Attachment 8 of BNP-PSA-085 shows in a table whether, and in some cases, how internal event initiators were addressed in the FPRA. Describe how equipment, whose fire-induced failure could cause initiating events, was matched to the appropriate plant response models (i.e., internal events accident sequences). Given the cited sensitivity study results, justify treating the cited initiators as fire-induced failure of equipment

following a plant trip rather than using the internal events plant response models associated with internal event initiators.

Response

The fire probabilistic risk assessment (FPRA) was developed from the internal events PRA in which equipment, whose failure could cause an initiating event, was already matched to the appropriate plant response models. During component selection for the FPRA, these initiating events were reviewed for susceptibility to fire-induced failure, as documented in Attachment 8 of BNP-PSA-085, *BNP Fire PRA - Component Selection*, and the FPRA added the fire initiator to the initiating event logic. As described in Attachment V of the LAR for finding and observation (F&O) 1-8, significant tracing of the logic during the peer review confirmed that the initiating event logic was appropriately OR-gated with the system logic, in most cases. However, certain specific gates associated with inadvertent safety relief valve (SRV) opening were identified as exceptions and were subsequently corrected, as described in Attachment V of the LAR for the disposition of F&O 1-8. F&O 1-8 also cited a sensitivity study requested for two other initiators (i.e., loss of direct current (DC) power A1 and loss of offsite power (LOOP)). For the loss of offsite power, consequential LOOP was found to have been omitted in several locations resulting in additional cutsets and a slightly higher conditional core damage probability (CCDP) than assuming the subsequent failure of the equipment. Consequently, logic for fire-induced LOOP was added to the fault tree, where appropriate, as described in Attachment V of the LAR for the disposition of F&O 1-8. However, based on the results of the sensitivity study, no change was made for the loss of DC power, because both DC initiators and response model events are always adjacent in the fault tree, so there was no difference in cutsets.

PRA RAI 1B

b) F&O 1-9 against ES-A1 (Not Met), ES-A4 (Cat I/II), and FQ-A2 (Cat I/II/III):

The disposition to this F&O indicates that the independence of High-Pressure Coolant Injection (HPCI) and RCIC is a source of uncertainty. Explain how the dependency between HPCI and RCIC was accounted for in the FPRA, including a discussion on uncertainty as appropriate.

Response

Contrary to what may be conveyed by the stated disposition of F&O 1-9 in Attachment V of the LAR, the results of the multiple spurious operation (MSO) Expert Panel review, as documented in Attachment 3 of BNP-PSA- 085, *BNP Fire PRA - Component Selection*, did not identify a dependency between HPCI and reactor core isolation cooling (RCIC), and therefore no such dependency is represented in the FPRA. The source of uncertainty is more accurately identified as the reactor pressure vessel (RPV) water level at which the quality of the steam would cause an overspeed trip of either the HPCI turbine or the RCIC turbine, depending on which MSO is postulated, rather than to suggest concurrent failures of both HPCI and RCIC.

A clarification to the disposition of F&O 1-9 in the Attachment V will be included in the LAR update.

PRA RAI 1C

c) F&O 1-14 against PRM-B4 (Cat I/II/III):

This F&O indicates that cable tracing was not performed in some cases. In areas where cable tracing was not performed, identify the assumptions made about possible plant trips and fire induced failures. Was an "exclusionary approach" used that assumes cable failure in any areas where the presence of cable cannot be ruled out?

Response

The decision for which cables to trace was based on particular equipment rather than on particular areas. Consistent with the guidance in NUREG/CR-6850, cable selection and circuit analysis was performed for equipment identified during component selection and added to that already existing for safe shutdown equipment. As documented in Table 4 of BNP-PSA-085, *BNP Fire PRA - Component Selection*, the main feedwater system; condensate system; circulating water system; and turbine control system were not included for cable tracing, and the FPRA assumes these systems to be failed for every fire scenario. Practically any area might contain cables that were not traced, either because the system is considered failed in the FPRA or because the associated piece of equipment was never credited for the PRA or for safe shutdown.

As described in Attachment 10 of BNP-PSA-080, *BNP Fire PRA – Quantification*, the likelihood of a plant trip due to fire was evaluated on an area basis and considered both the equipment located in the area and the equipment that could be affected by traced cables that traverse the area. The evaluation considered the possible effects on the plant given a fire in the area but not the likelihood of a fire in that area or the assumed status of the system in the FPRA. One of three possible conditional trip probabilities (i.e., 1.0, for near certain; 0.1, for reduced likelihood; or 0.01, for not likely) was assigned. To accommodate operator discretion, no area was assigned a conditional trip probability of zero, even if the area contained no equipment important to plant operation.

PRA RAI 1D

d) F&O 1-19 against FSS-A1 (Not Met):

The disposition for this F&O explains that the ZOI associated with a 143 kilo-watt (kW) heat release rate (HRR) (75th percentile) transient fire was used in all fires areas, except the turbine building where a ZOI for a 317 kW HRR (98th percentile) fire was used. The disposition provides the basis for this lower HRR as existing and planned administrative controls, plant experience, and insights from a bounding sensitivity study. Provide further justification for the use of 143 kW transient fires, given that both 143 kW and 317 kW are taken from the same HRR distribution. Include further description of the administrative controls used in the different areas for managing transient combustibles, the results of reviewing plant experience and records of violations of transient combustible controls, other key factors for this reduced fire size, and the results of the bounding sensitivity study referred to in the disposition. Also, confirm that 143 kW and 317 kW HRRs were the only transient fire sizes used in the FPRA.

Response

Where the zone of influence (ZOI) for a 143 kW HRR was used to identify targets for the transient walkdowns, the FPRA used the same target sets for both the 75th percentile and the 98th percentile fires. Effectively, no two-point HRR distribution was used for these scenarios. By comparison, in the turbine building where the ZOI used for the walkdowns was based on a 317 kW HRR, the affected components for the two-point HRR distribution were different for less than 10% of the transient ignition sources. Therefore, the only transient fire sizes used in the FPRA are a single-point 143 kW HRR and a two-point 143 kW and 317 kW HRR.

As stated in Attachment V of the LAR for the disposition of F&O 1-19, additional transient walkdowns have been performed using a 317 kW HRR, but those results have not yet been incorporated into the FPRA. To support the use of a lower HRR in specific areas, an evaluation was performed, as documented in BNP-PSA-086, *BNP Fire PRA - Fire Scenario Data*, Attachment 25, to determine a reasonably realistic and bounding HRR for transient combustibles. These results will be incorporated into FIR-NGGC-0009, *NFPA 805 Transient Combustibles and Ignition Source Controls Program,* which, post-transition to NFPA 805, will be used to limit the placement of transient combustibles and ignition sources near equipment and cables. During interaction with the Peer Review Team, an unofficial bounding sensitivity study was performed to estimate the risk associated for a transient with a larger ZOI in a particular area by using the CCDP for a hot gas layer with 10% of the transient frequency. However, these results are not available because the sensitivity study was not maintained.

The results of a new sensitivity study will be submitted in the 120-day responses with the results of other sensitivity studies.

PRA RAI 1F

f) F&O 1-26 against HR-G1 (Cat I), FSS-B2 (Cat II), and HRA-C1 (Cat II):

Describe how the Human Reliability Analysis (HRA) was performed for alternate shutdown following MCR abandonment. Include in this description:

i. Identification of events or conditions that prompt the decision to transfer command-and-control from the MCR to the alternate shutdown station. Clarify how the loss-of-control due to fires in the MCR or Cable Spreading Room was modeled.

ii. Explanation of how timing was established (i.e., total time available, time until a cue is reached, manipulation time, and time for decision-making) and which fire or fires were used as the basis for the timing. Include in the explanation the basis for any assumptions made about timing.

iii. Discussion of how different core damage end-states defined by the Abandonment HRA Event Trees presented in Attachment 10 of BNP-PSA-084 were incorporated into the FPRA, given that some sequences resulted in early and others resulted in late core damage.

iv. Description of how the feasibility of the operator actions supporting alternate shutdown was assessed.

v. Justification for assuming that continuous communication and coordination will occur during implementation of 0ASSD-02 by the different operators at their different locations. Include consideration of actions that require taking off headsets or the unavailability of phone systems.

vi. Description of how the impact of complexity on coordination of actions and operator performance in 0ASSD-01 and 0ASSD-02 was addressed.

vii. Description of the treatment of potential dependencies between individual actions, including discussion of operator actions that can impact the actions of other operators.

Response

F&O 1-26 concerns the use of point estimates for CCDP and conditional large early release probability (CLERP) rather than a detailed task analysis for main control room abandonment (MCRA). As described in Attachment V of the LAR, the disposition of F&O 1-26 involved the use of a HRA to develop a detailed Human Event Probability (HEP) for the control room abandonment scenario. This HEP was used to obtain more realistic estimates of core damage frequency (CDF) and large early release frequency (LERF) for MCRA based on the results of a Consolidated Fire and Smoke Transport (CFAST) computer model of the time required to reach the main control room (MCR) habitability threshold values for temperature and visibility, as described in Attachment 16 of BNP-PSA-080, *BNP Fire PRA – Quantification*.

i. In crediting operator actions for alternate shutdown following control room abandonment, the decision to transfer command-and-control from the MCR to the alternate shutdown station is prompted by postulated MCR habitability limitations. In particular, these habitability limitations correspond to the threshold values for temperature and visibility in the hot gas layer, as identified by NUREG/CR-6850. No operator action is otherwise credited for recovery of loss-of-control due to fires in the MCR or the Cable Spreading Room.

ii. To establish the timing for MCRA based on habitability, the CFAST computer model considered electronic equipment fires and ordinary combustible fires within the Unit 1 and Unit 2 MCR and electronic equipment rooms with different configurations of control room heating, ventilation, and air conditioning (HVAC) and boundary doors. To establish a frequency for MCRA, the convolution of these times conservatively assumed the boundary doors are closed and 1% unavailability for the control room HVAC.

After the decision to abandon the MCR, the system window timing for individual human failure events (HFEs) was taken from the 0ASSD-02, *Alternative Safe Shutdown Procedure: Control Building*, procedure or Modular Accident Analysis Program (MAAP) calculations referenced in Attachment 10 of BNP-PSA-084, *BNP Fire PRA - Human Reliability Analysis*, (and in the individual HFEs). The only cues that are important in this analysis are the initial cue to evacuate the control room, the time it takes to reach a new step in the procedure, or waiting for verification from another operator before performing an action. Train B will be the safe shutdown train for fires in the control building, excluding battery rooms, and this procedure is written assuming the equipment will be available as only random failures will result in the unavailability of Train B equipment and the probability of that equipment failing is much smaller than the probability of the

operator's failing to perform the procedure. For this reason the procedure is primarily written as a set of steps that do not have a recovery action. The manipulation/travel times were obtained from information gathered during a walkthrough of the 0ASSD-02 procedures with the operators. Any assumptions made in terms of timing were based on gathered plant simulator/walkdown information.

iii. The CCDP of the operators failing to perform the alternative safe shutdown (ASSD) actions in the 0ASSD-02 was calculated by summing the probability of reaching each individual core damage end state, without consideration for whether the sequence resulted in early or late core damage. Given a fire that results in abandonment, 12 sequences have an end state that is expected to result in core damage. Separate from the rest of the FPRA, the total CDF for MCRA was determined as the product of this CCDP and the MCRA frequency determined from the CFAST habitability results. As described in Section 3.6.1 of BNP-PSA-080, *BNP Fire PRA – Quantification*, the risks associated with MCRA was added (i.e. similar to the treatment of risks from the multi-compartment analysis) to the results of the FPRA to obtain the Fire CDF and Fire LERF.

iv. Operators train on the ASSD procedures once every two years in order to remain familiar with the ASSD actions and equipment. Time requirements are stated in the procedures, and the ability to perform the actions within the stated times was part of the walk-through with operators to ensure the feasibility of the actions. Also based on the time window evaluations presented in the HRA calculator, which included travel path time considerations, simulator timing information, and assumptions based on known similar actions at BSEP, all actions were deemed feasible.

v. Because Section A of 0ASSD-01, *Alternative Safe Shutdown Procedure Index,* provides guidance for the use of alternate and diverse means (e.g., hand-held portable radios, plant paging system) to support continuous communication and coordination, removal of the headset or unavailability of the sound-powered phones was not considered to be a concern.

vi. The impact of complexity on coordination between operators and the evaluation of these interactions is addressed by the MCR Abandonment Timelines. These timelines show when one operator, working in one procedure, must perform an action prior to another operator beginning a step in another procedure. If an operator is required to wait for another operator to successfully perform an action in the procedure, then the time it takes the first operator to completely finish their action will be considered as the delay time (T_{delay}) for the second action. The coordination of actions in operator performance is also addressed in the fault trees that show how if one operator fails to perform an action then the goal of multiple actions, such as starting an emergency diesel generator (EDG), will be failed.

vii. The treatment of dependency between individual actions is addressed in the fault trees that show how if one operator fails to perform an action then the goal of multiple actions, such as starting an EDG, will be failed. This dependency is also treated in the timelines. If an action appears before another action in the timeline then the first action must be successful in order for the second one to occur. The structure of the 0ASSD-01 and 0ASSD-02 procedures is such that there is only one recoverable action per unit, in which exit from 0ASSD-01 and entry into 0ASSD-02 provides an opportunity for an intervening success between the actions. Consequently, a zero dependency level is applied, and

these two HFEs are placed under an AND gate. All other actions appear under OR gates, because the failure of any single action other than these will result in a core damage sequence.

PRA RAI 1G

g) F&O 1-30 against FSS-A1 (Not met):

Describe the approach and assumptions used to model fires in open and closed cabinets, and the sensitivity study on motor control centers (MCCs) presented in Section 4.8.3.1 of the LAR. Include in this description:

i. Confirmation that walkdowns were performed to determine open and closed cabinets.

ii. Given an MCC cubicle fire, identification of the cubicles in the MCC assumed to fail.

iii. Explanation of why the sensitivity study shows no impact on Unit 1 LERF and ΔLERF, and Unit 2 CDF, ΔCDF, LERF, and ΔLERF, while showing an increase in Unit 1 CDF and ΔCDF.

Response

Fire is not generally postulated to propagate from closed cabinets, as described in Attachment 3 of BNP-PSA-080, *BNP Fire PRA - Quantification*. Consequently, fires in closed cabinets located outside the MCR were assumed not to contribute significantly to fire risk, and that risk is not quantified. However, open cabinets and closed cabinets located in the MCR and the remote shutdown panels were quantified as "self" fires where the contents of the cabinet made up the target-set for the scenario, as described in Section 3.2.4.3 of BNP-PSA-080. For open cabinets, the risk associated with fire scenarios involving targets within the zone of influence is also quantified in a manner similar to other ignition sources.

As described in Section 3.6.5.4 of BNP-PSA-080, closed MCCs were assigned a probability of 0.1 to account for the fraction of time that the fire is postulated to propagate outside the closed MCC. This probability directly reduced the Scenario Event Frequency (SEF) for all scenarios associated with the closed MCCs. The target-sets for these scenarios were developed in the same manner as other ignition sources. The sensitivity analysis in Section 4.8.3.1 of the LAR presented results with the probability of 0.1 removed, effectively treating fires in the closed MCCs as always propagating outside the MCCs.

i. The majority of closed cabinets and closed MCCs were identified as being closed in the source walkdowns as documented in Attachments 6, 11, and 17 of BNP-PSA-086, *BNP Fire PRA - Fire Scenario Data*. As described in Attachment 6 of BNP-PSA-086, equipment qualification package (i.e., QDP 67) was also cited as a basis for determining about a dozen MCCs to be closed. Four additional MCCs were identified as being closed based on a previous evaluation, which included a walkdown, as documented in Attachment 25 of BNP-PSA-080.

ii. MCC fires were modeled as the entire MCC, and not down to the individual MCC cubicle level. No individual MCC cubicle was assumed to fail from the fire. The failures induced by a MCC fire were based on cables identified in the zone of influence for the entire MCC.

iii. As documented in BNP-PSA-095, *BNP Fire PSA - Sensitivity Analyses,* the sensitivity study shows that the results are largely insensitive to the assumption that closed MCCs are open 10% of the time. This assumption was applied to only 34 of 2401 sources evaluated in the FPRA. For Unit 1 LERF, Unit 1 ΔLERF, Unit 2 CDF, Unit 2 ΔCDF, Unit 2 LERF, and Unit 2 ΔLERF, these sources simply were not significant enough to cause a change in the base value when expressed to two significant figures. Had more significant figures been used the resulting percentage change may have, in some cases, shown a non-zero increase, but the conclusion that the results are largely insensitive to the assumption would not have changed.

PRA RAI 1I

i) F&O 1-38 against LE-G2 (Not Met), LE-F3 (Not Met), UNC-A1 (Not Met), FQ-E1 (Not Met), FQ-F1 (Not Met) combined with F&O 4-18 against QU-E3 (Cat I), QU-A3, UNC-A4 (not Met), and FQ-A4 (Cat I/II/III):

Explain how parametric data uncertainty was propagated and the state of knowledge correlation (SOKC) was evaluated for fire CDF and LERF. Identify fire-PRA-specific parameters (e.g., hot short probabilities, fire frequencies) that can appear in FPRA cutsets and how they were correlated. NUREG-1855 states that all basic events (regardless of system) must be correlated if their failure rates for a given failure mode are derived from the same data set. Therefore, if SOKC was applied only to basic events within the same system, provide a justification.

Response

The BSEP FPRA quantifies individual cutsets for each fire source scenario. These cutsets were combined into a single cutset file, as described in Attachment 38 of BNP-PSA-080, *BNP Fire PRA - Quantification*. Basic events (BEs) were renamed in a manner to allow random failures, fire induced failures, and fire induced hot shorts for the same component to have unique names within the same combined cutset file. Once combined, the cutset basic events were reviewed to determine if appropriate uncertainty parameters were used for each basic event, and additions were made where appropriate.

The UNCERT (i.e., part of the Electric Power Research Institute (EPRI) Computer Aided Fault Tree Analysis (CAFTA) Suite) program was run to evaluate the combined cutset file. Based on a Monte-Carlo sampling approach, the mean CDF/LERF was generated along with a probability distribution histogram, which can be found in BNP-PSA-080, Attachment 38. The mean CDF/LERF calculations, based on the data uncertainty, were consistent with the CDF/LERF, based on the point estimated solution.

The UNCERT solution for the parametric uncertainty can evaluate the impact of the SOKC for correlated data if a single uncertainty parameter is applied to multiple basic events, as is the case for reliability data using "type coded" values. No other basic event uses a single uncertainty parameter for multiple components. Because the mean point estimate CDF/LERF

results match those of the CDF/LERF values from the UNCERT solution, it is concluded that there are no correlated random failure events requiring SOKC additions to frequency.

For the fire PRA, various events in the combined solutions were reviewed for evaluation for SOKC. The following table, taken from Table 3.2.5-1 in BNP-PSA-080, Attachment 38, discusses the identification of fire data types reviewed for SOKC.

	Area of Uncertainty	Discussion
1.	Fire ignition frequency	The BSEP fire scenarios are based on single ignition sources. Therefore, there are no correlated ignition frequencies within an individual cutset, precluding SOKC occurrence concerns.
2.	Non-detection probabilities	A generic non-detection probability is used in quantifying the scenario frequencies. Multiple detectors are not credited, so that for individual scenarios, there is no correlated data.
3.	Non-suppression probabilities	There is no correlation between various types of suppression, in that they are uniquely different.
4.	Heat release rate severity factor/split fraction	See Item 1. In addition, the source target relationship is based on a single distance that is used to calculate the HRR severity factors. The split of the generic HRRs is quantified as two individual scenarios, precluding any correlated data in a single cutset.
5.	Circuit failure probabilities	With the exception of basic events where the hot short probability of 1.0 is used, cutsets including the same component type and failure mode with the same hot short probabilities are assumed completely correlated. The UNCERT code does not address this correlation, so an analysis showing the potential change in CDF/LERF is supplied below.

Based on the conclusions of the above table, only those combinations of events with hot short failure probabilities less than 1.0 are evaluated for SOKC.

The merged cutsets were reviewed to identify correlated hot short failure combinations above the truncation of 1.0E-9 for CDF and 1.0E-10 for LERF. The types of events, presented in the cutsets requiring SOKC consideration, are spurious operation of air-operated valves (AOVs) and motor-operated valves (MOVs). A SOKC multiplier, based on the number of correlated events in a cutset and calculated using the standard deviation for the hot short probabilities, was developed and multiplied by the CDF/LERF Fussell-Vesely (F-V) contribution of each SOKC combination to calculate the increase in CDF/LERF due to SOKC.

No cutset was found with correlated events from multiple systems. Only correlated failures within the same systems were present within the cutset solution, so that the SOKC multipliers affect one system only. This is expected, given cutsets are based on loss of functions, and functions typically have redundant components at the system level. Also, this may be due in part to the limited number of altered events that are used in the BSEP FPRA. A more extensive use

of hot short probabilities with altered events could potentially result in correlated events from different systems existing within the same cutset.

PRA RAI 1K

k) F&O 2-14 against FSS-D7 (Cat I):

Clarify whether information from the System Health Reporting and System Notebook processes, or other sources, shows data for more than 1 year to confirm that the Fire Detection and Suppression Systems have not experienced "outlier behavior." If only 1 year of data was used, justify why this is sufficient.

Response

The response to this question will be provided as part of the 90-day RAI responses which will be submitted by July 31, 2013.

PRA RAI 1N

n) F&O 4-14 against FSS-E3 (Cat I), FSS-H5 (Cat I), FSS-H9 (Cat I/II/III), UNC-A2 (Cat I/II/III):

Explain how uncertainty was treated with respect to CDF, LERF, ΔCDF, and ΔLERF. Clarify the extent to which statistical quantification of uncertainty was used to evaluate fire CDF, LERF, ΔCDF, and ΔLERF. Identify significant fire scenarios where uncertainty was characterized qualitatively. For these scenarios, explain (per Supporting Requirement QU-E4) how the FPRA is affected by these sources of uncertainty.

Response

As outlined in the response to PRA RAI 1I, parameter uncertainties are assigned to basic events in the FPRA model to account for the *aleatory* uncertainty, based on randomness of elements in the FPRA, for fire induced failures. As described in Section 3.2.4 of Attachment 38 to BNP-PSA-080, *BNP Fire PRA – Quantification*, uncertainty parameters were propagated to obtain an uncertainty distribution on the calculated CDF and LERF using Monte Carlo methods. There was no appreciable contribution to risk associated with parametric uncertainty or state of knowledge correlation. In that regard, no evaluation of aleatory uncertainty was performed for ΔCDF or ΔLERF, as it would not be expected to provide new insights.

The risk importance of *epistemic* uncertainties, based on level of knowledge of FPRA elements, is best assessed via sensitivity analyses by assuming an alternative outcome or method for each modeling issue or combinations of issues. Thus, the risk importance of a given epistemic uncertainty was assessed by calculating the change in CDF or LERF using an alternate modeling assumption. The procedure outlined in NUREG/CR-6850 for the Fire PRA uncertainty and sensitivity analysis (i.e., Task 15) was used to identify the important epistemic uncertainty issues associated with the BSEP Fire PRA.

BNP-PSA-095, *BNP Fire PSA - Sensitivity Analyses*, provides the following quantitative sensitivities for CDF, LERF, ΔCDF, and ΔLERF:

- Ignition Frequency from NUREG/CR-6850 versus Supplement 1 to NUREG/CR-6850,

- Removal of credit for control power transformer affect on alternating current (AC) circuit failure probabilities, and
- Treatment of closed MCCs ignition sources as opening for fire.

Engineering Change (EC) 89666, *FPRA NUREG/CR-6850 Appendix L Sensitivity Study for BNP Main Control Boards with In-Cabinet Incipient Detection*, provides quantitative sensitivities for CDF, LERF, ΔCDF, and ΔLERF for the use of incipient detection in the main control boards (MCBs) (i.e., Unit 1 only).

As listed in Section 3.6.6 of BNP-PSA-080, sources of model uncertainty and related assumptions were qualitatively characterized by category, with the associated impact on the FPRA but usually without specific linkage to a particular significant fire scenario listed in Section 3.6.3 of BNP-PSA-080.

Category	Item	Conservatism
Ignition Frequency	Ignition Frequencies are based on data in the Fire Events Database that includes "potentially challenging" fires and not actual observed fires.	Yes Clearer and more detailed collection of generic data may reduce ignition frequency (IGF).
Ignition Frequency	Method of apportioning a plant IGF between several pieces of equipment does not take into account equipment history, maintenance practices, standby vs. active, etc. By this method the same equipment in the same configuration at two different plants would have different ignition frequencies.	Yes/No Can vary depending on factors involved.
Source HRR	Bounding values from NUREG/CR-6850 were typically used for the 98th percentile file based on the HRR case. For a limited number of sources (e.g., cabinets) these values were adjusted based on fire modeling insights.	Yes It is not likely that the actual source configurations could support the default HRRs.
Source HRR	Closed cabinet treatment for MCCs. BSEP assumes certain MCCs are closed sources; however, guidance indicates that 480V MCCs can experience energetic faults which can create openings to support fire growth. To account for this, an additional factor of 0.1 is applied to the scenario specific ignition frequency (SSIF) to account for the fraction of the time the MCC stays closed	No The data for the guidance is interpreted conservatively. At best, only a small portion of the MCC fires would lead to an open cabinet situation.
Source HRR	Oil fires involving oil are assumed to instantly spill and instantly ignite to maximum HRR with no fire growth.	Yes This method is conservative, especially for treatment of large quantities of oil.

Category	Item	Conservatism
Source HRR profile	No credit is given for the incipient/smoldering stages of fire growth.	Yes Allowing for these phases will provide more time for manual suppression credit.
Source HRR profile	Most fires use a 12 minute ramp to peak HRR, 8 minutes at peak, and 19 minutes decay. No methodology for consideration of combustible loading and other factors is provided.	Yes Consideration of other factors will likely shorten duration or reduce peak HRR for most scenarios.
Target Selection	Use of multi-point fires is not based on target importance.	Yes More than two points or more targeted selection of the two points may reduce CCDP.
Target Selection	Fire modeling target selection only uses single point fires.	Yes Using multi-point fires for fire modeling may reduce CCDPs
Target Selection	The MCB probabilistic safety analysis (PSA) panel was evaluated using a single fire impacting the full panel. It is possible to separate fire scenarios in this panel into multiple scenarios	Yes Multi-scenario approach may result in a less conservative estimate of risk.
Target Selection	The identification of target sets is based on the flame and heat products of the fire. Damage due to smoke is not quantified in this analysis.	No However, vulnerable components are typically contained in cabinets/panels. The enclosure provides some protection against smoke damage for fires originating outside the enclosure. Vulnerable components in MCR cabinets or remote shutdown (RSD) panels are already failed in "self" fires and failure of components in Main Control Boards is mitigated by incipient detection.

Category	Item	Conservatism
Target Selection	"Sensitive" electronics (e.g., integrated circuits employing pin-grid arrays but not electro-mechanical devices, and not solid state components in power applications, and not discrete solid state components) are not failed due to heat damage.	No However, sensitive electronics are typically contained in cabinets/panels and the enclosure will provide some amount of protection from external fires. For fires inside MCR cabinets or RSD panels, failure of the components is already assumed in the "self" fires. For Main Control Boards, the failure of sensitive electronic is mitigated by incipient detection. For sensitive electronics not contained in enclosures, there is a strong possibility the cables to the components are already failed in the scenarios even though they are not the limiting failure for the component.
Target Selection	Change Package BNP-0224 (BNP-PSA-080, Attachment 37) identified three raceways (i.e. 2FVDI/CA, 2FVG2/CA and 2FCP/CB) required for breaker coordination that are not in the Brunswick Cable Management System (BCAMs). No location information is given for these raceways. No basic events are failed for hot gas layer (HGL) scenarios based on these raceways	No While the assumption (i.e. that failure of these raceways in HGL scenarios is not significant) is potentially non-conservative, it is likely that other failures in the HGL scenario will also fail the respective power supplies if the raceways are in locations where a HGL is plausible.
Target Selection	The coordination study, Change Package BNP-157 (i.e., BNP-PSA-080, Attachment 13) ignores cable drop length, but credits the entire length of the endpoint cable tray. It is assumed the length difference between the drop and additional lengths of partial tray runs is negligible.	Yes/No The actual cable length could be shorter or longer than actually used in the coordination study. This difference is expected to be within the uncertainty of the coordination study method and application in the PRA model.
Damage Time	Target damage is based on 625°F, although a small portion of cables are Thermoplastic. Note that this is consistent with the temperature used for generating the ZOIs.	No A lower damage threshold will provide less time for suppression.

Category	Item	Conservatism
Damage Time	Target damage does not credit conduit.	Yes More time should be available for suppression due to the use of conduit.
HGL possibility in Electrical Tunnel (ET) and Pipe Tunnel (PT)	This calculation assumes an HGL is not possible in the Electrical Tunnel and Pipe Tunnel.	No If a HGL is possible, the calculated CDF value could be non-conservative.
Time to HGL	See target selection and damage time items	Yes Extending the time to the first tray igniting will extend the time to HGL formation and provide more time for suppression.
Time to HGL	The cable tray growth model in NUREG/CR-6850 has limited applicability and appears to be conservative when applied outside the limits.	Yes
Time to HGL	Fire spread within a cable tray is offset by decay.	Yes/No Can vary depending on factors involved.
Non-Suppression	Suppression data and fire burn times do not differentiate between time required to control and time required to completely suppress a fire.	Yes Once a fire is controlled, the damage set is no longer increasing, which may occur long before the fire is suppressed.
Non-Suppression	When the presence of a wet-pipe or CO_2 system is indicated in an NFPA 805 Location, it is assumed that the system is effective in suppressing any fire in the NFPA 805 Location.	No. Change package BNP-0186 (i.e., BNP-PSA-080, Attachment 34) identifies locations where suppression systems may not be effective in preventing formation of a HGL. No credit was taken for these systems.
Circuit Analysis	Detailed circuit analysis to determine probabilities of spurious BEs due to cable faults is performed based on risk contribution.	Yes Further detailed circuit analysis may reduce probability of some spurious BEs.
HRA	HRA Screening - No credit is given to operator manual actions (OMAs).	Yes Detailed analysis of OMAs may reduce risk.
HRA	Many ex-control room actions are not credited through screening analysis. Detailed HRA Analysis is performed only for selected HFEs based on quantified importance.	Yes Further detailed analysis of HFEs may reduce HEPs for some events.

Category	Item	Conservatism
HRA	HFEs performed outside the control room are assumed to be unsuccessful if they require traversing or performing an action in a compartment with a fire.	Yes For large compartments, small fire scenarios, and actions with long performance time available, the environmental effects of a fire are limited and some credit could be given.
Quantification	Power dependencies for spurious operation.	Yes In some cases, the power may not be available in the time period needed to support some fire induced spurious events.
Quantification Tools	Utilizing Min-Cut Upper Bound approximation with large numbers of 1.0 Basic Events causes overestimation of results.	Yes

PRA RAI 1O

o) F&O 5-13 against QU-D2 (Not Met), QU-F3 (Cat I), FQ-E1 (Not Met), and FQ-F1 (Not Met):

The disposition to this F&O indicates that a review was performed on FPRA modeling to confirm that no inconsistencies were created between sequence and system modeling, or between the FPRA and how the plant is operated. This discussion of this review is not apparent in the cited documentation (BNP-PSA-085). Describe this review and its conclusions, and identify where it is documented.

Response

Documentation of this review is distributed through different sections of BNP-PSA-085, *BNP Fire PRA - Component Selection*, with conclusions consisting of statements of acceptability or descriptions of the required changes.

Section 3.3.1.4 of BNP-PSA-085 describes the review of the PRA Internal Events model Accident Sequence Notebook (i.e. BNP-PSA-029, *PRA Model Event Tree and Accident Sequence Delineation*) and Level 2 Accident Sequence Notebook (i.e. BNP-PSA-049, *PRA Model Sections 7-9 Level 2 Analysis*). These PRA notebooks address the plant response and the event trees developed for that response. The review determined that the FPRA will be maintained as part of the Internal Events PRA Model of Record and concluded that the use of the same PRA models as for the internal events sequence quantification ensures that interdependencies are modeled consistently and appropriately. Because the BSEP PRA uses functional tops for the event trees, and functions are modeled by initiating events and by system models, a detailed sequence by sequence review is not required and would provide no benefit. The fire events cannot change the modeled functions, and the conclusion is always that the accident sequence logic is adequate because the functions are always the same: reactivity control, RPV integrity, inventory control, pressure control, decay heat removal, and containment integrity.

Section 3.3.1.4 of BNP-PSA-085 also describes the review of applicable initiating events (i.e., BNP-PSA-032, *PRA Model Appendix C Initiating Events Assessment*) and the relevance to the fire model (i.e., BNP-PSA-085, Attachment 8). Consideration of possible additional initiating events that might be unique to the FPRA is documented in Table 3-2 of BNP-PSA-085. Attachments 3 and 4 of BNP-PSA-085 document the review and disposition of various fire-induced MSOs postulated by plant and industry personnel to have potential impact on mitigation functions and systems. This review resulted in certain FPRA model changes as documented in Attachments 9 and 12 of BNP-PSA-085 and included the creation of a simplified bypass event tree for a main steam isolation valve (MSIV) MSO, as described BNP-PSA-085, Section 3.3.1.4 and Attachment 13.

For the Level 2 review, a detailed review of the containment isolation is performed in BNP-PSA-085, Attachment 6. Subsequently the review of the fire impact on the Level 2 accident sequences and phenomenological events was performed as documented in Attachment 13 of BNP-PSA-085.

PRA RAI 1P

p) F&O 5-15 against QU-F2 (Cat I/II/III), QU-F3 (Cat I), QU-D6 (Cat I), QU-D7 (Not Met), FQ-E1 (Not Met), and FQ-F1 (Not Met) combined with F&O 5-16 against LE-F1 (Not Met), LE-F2 (Cat I), LE-G3 (Not Met), UNC-A1 (Not Met), FQ-E1 (Not Met), and FQ-F1 (Not Met):

   Describe the assessment performed to determine the significant risk contributors and risk importance events and failures for CDF and LERF. Clarify how the insights from importance analysis were used to review the correctness and reasonableness of the FPRA modeling.

Response

As described in response to PRA RAI 1I, the individual scenario cutsets were merged into a single cutset file and Section 3.4 of Attachment 38 to BNP-PSA-080, *BNP Fire PRA – Quantification,* lists, by cutset probability, the top contributors to both CDF and LERF. Prior to merging the cutsets, the correctness and reasonableness of the FPRA modeling were reviewed based on detailed cutset reviews for individual scenarios, such as that documented in Attachment 39 of BNP-PSA-080.

Based on the merged cutsets, the risk contributors and risk importance events and failures were assessed, as described in Section 3.3 of Attachment 38 to BNP-PSA-80, and included rankings in the following categories:

- Fire Compartments
- Fire Scenarios
- Fire Accident Sequences
- Containment Failure Types (i.e., LERF only)
- Operator Actions
- Fire Induced Equipment Failure Modes
- Random Component Failures
- Systems

- Component Type Failures

For each of these categories other than Containment Failure Types, the top contributors were ranked for both CDF and LERF according to the percent contribution to risk. For Containment Failure Types, the assessment only considered the contributions to LERF. The importance ranking results in each of these ranking categories are generally used in addressing which portions of the FPRA model need further refinement.

Insights from importance analysis were used to review the correctness and reasonableness of the FPRA modeling by comparing the results against what is normally understood about plant response. Attachment 38 of BNP-PSA-080 provides the following insights for reviewing the correctness and reasonableness of the FPRA model:

- The MCR and cable spreading rooms dominate risk contributions from Fire Compartments;
- Scenarios involving fixed ignition sources, rather than transient combustibles, are major contributors;
- All transient and station blackout (SBO) sequences for fire result in either loss of makeup events or loss of decay heat removal events that result in a loss of makeup;
- With emergency power blackout associated with the dominant cause of core damage, failures of fail-safe containment isolation valves may not contribute as much to LERF as presented;
- Control room abandonment for habitability is one of the more important operator actions;
- Lack of a specific method for evaluating fire-induced instrument faults is evident in the results;
- Random component failure rankings show test and maintenance unavailability is important to fire risk;
- The plant system that contributes most to fire risk is the AC power system, with
- AC breakers as contributing components.

PRA RAI 1Q

q) F&O 5-18 against LE-G2 (Not Met), LE-F3 (Not Met), LE-G4 (Not Met), UNC-A1 (Not Met), UNC-A2 (Cat I/II/III), FQ-E1 (Not Met), and FQ-F1 (Not Met):

These F&Os note that uncertainty and importance analysis was not performed for fire LERF. Describe the sources of uncertainty and results of importance analyses of fire LERF.

Response

The sources of model uncertainty and related assumptions which are listed in the response to PRA RAI 1N are considered applicable to LERF. Of these, the most significant area of epistemic uncertainty with regard to LERF is in the area of circuit analysis, specifically as related to spurious operation of containment isolation valves. At the time of the LAR submittal, two types of failures were not addressed by guidance in NUREG/CR-6850. These include the treatment of low voltage instrumentation loops and the likelihood of grounding or clearing hot shorts in DC circuits. Consequently, each of these is treated with a value of 1.0 in the BSEP FPRA. The

resulting cutsets are dominated by signal failures causing valve spurious operations or primary containment isolation valves (PCIVs) remaining spuriously open, even though their design is to fail safe in the closed/isolated position. A more realistic assessment of these affects would greatly reduce LERF. No additional area of uncertainty was found that is unique to the fire FPRA that is not already addressed for internal events.

The sources of aleatory uncertainty were evaluated for LERF, and a detailed results analysis was performed for LERF and documented as Attachment 38 to the BNP-PSA-080, *BNP Fire PRA – Quantification*. This analysis includes evaluation of parametric uncertainty for a combined LERF solution and various importance evaluations for the same solution. In particular, parametric uncertainty was evaluated for the following:

(a) Fire scenario event frequencies (i.e., initiators),

(b) Component failure probabilities (i.e., random faults and hot short probabilities),

(c) Component maintenance unavailability,

(d) Human error probabilities,

(e) Common cause failures, and

(f) Recovery Actions (i.e., main control room abandonment from environmental causes)

As described in response to PRA RAI 1I, the individual scenario cutsets were merged into a single cutset file. Merging the cutsets required differentiation between the naming convention of basic events that are due to fire failures, hot shorts, or random failures.

UNCERT (i.e., the EPRI CAFTA Suite) was run using the cutset file. Random failure uncertainty parameters were based on the internal events data analysis. Ignition source error factors were based on NUREG/CR-6850, Supplement 1, Chapter 10. The hot short probability error factors were based on NUREG/CR-6850 Tables 10-1 and 10-2. Other event error factors, such as those for HRAs, were based on their specific data analysis. Based on a Monte-Carlo sampling approach, the code determined the mean LERF and associated uncertainty distributions.

Each mean LERF from the uncertainty simulation compared favorably with its associated mean point estimate LERF, providing confidence in the published risk results. As taken from BNP-PSA-080, Attachment 38, the uncertainty histograms for the frequency density distribution and the statistics for the confidence internals are provided for each of the units risk metrics in the following figures and tables.

Figure 3.2.4-3. BSEP Unit 1 FPRA LERF Parametric Uncertainty Histogram
[BNP-PSA-080, Rev. 3, Attachment 38]

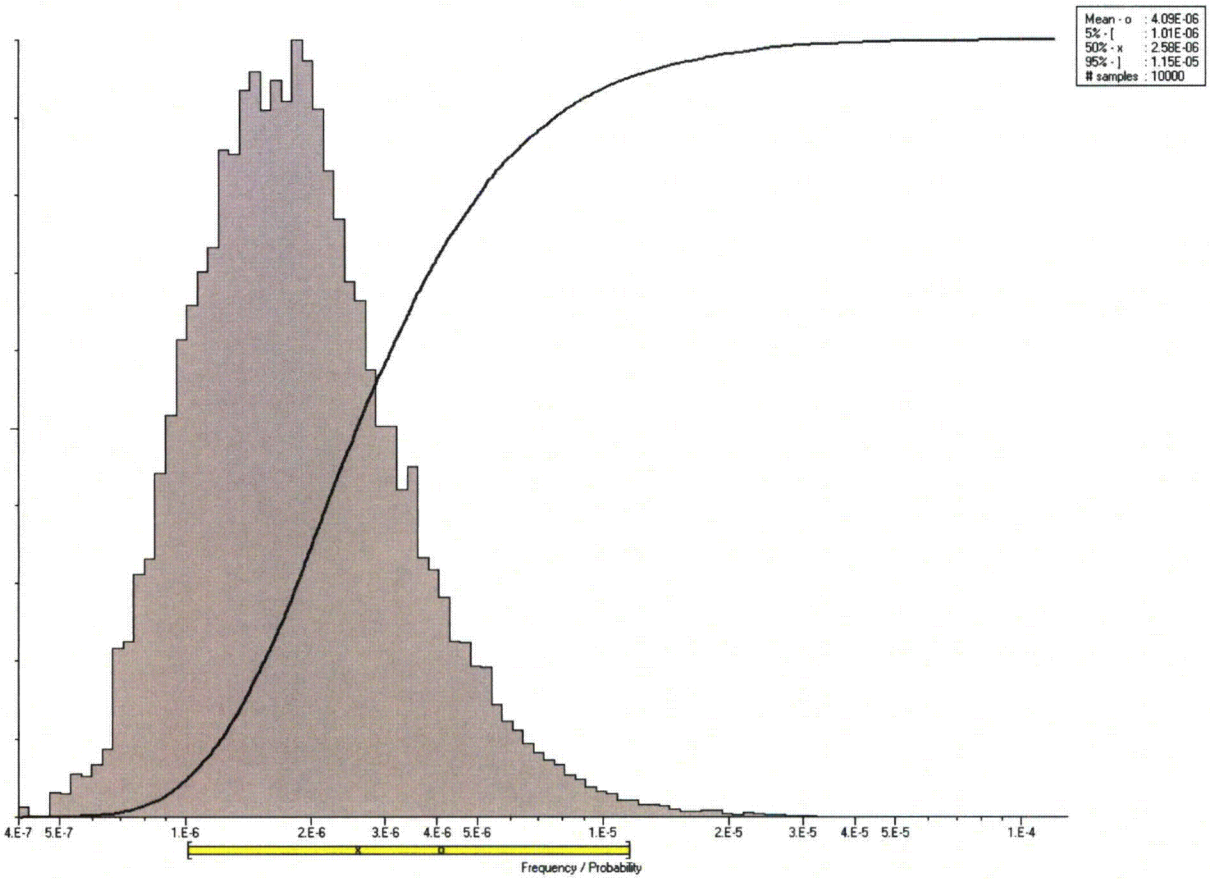

Table 3.2.4-3 - BSEP Unit 1 FPRA LERF - Parametric Uncertainty Statistics
[BNP-PSA-080, Rev. 3, Attachment 38]

	5% Confidence	Mean	95% Confidence
Point Estimate		4.08E-06	
Mean	3.98E-06	4.09E-06	4.20E-06
5%	9.95E-07	1.02E-06	1.04E-06
Median	2.54E-06	2.59E-06	2.63E-06
95%	1.10E-05	1.16E-05	1.21E-05
Standard Deviation		5.56E-06	
Skewness		7.68E+00	
Kurtosis		9.92E+01	

Figure 3.2.4-4. BSEP Unit 2 FPRA LERF Parametric Uncertainty Histogram
[BNP-PSA-080, Rev. 3, Attachment 38]

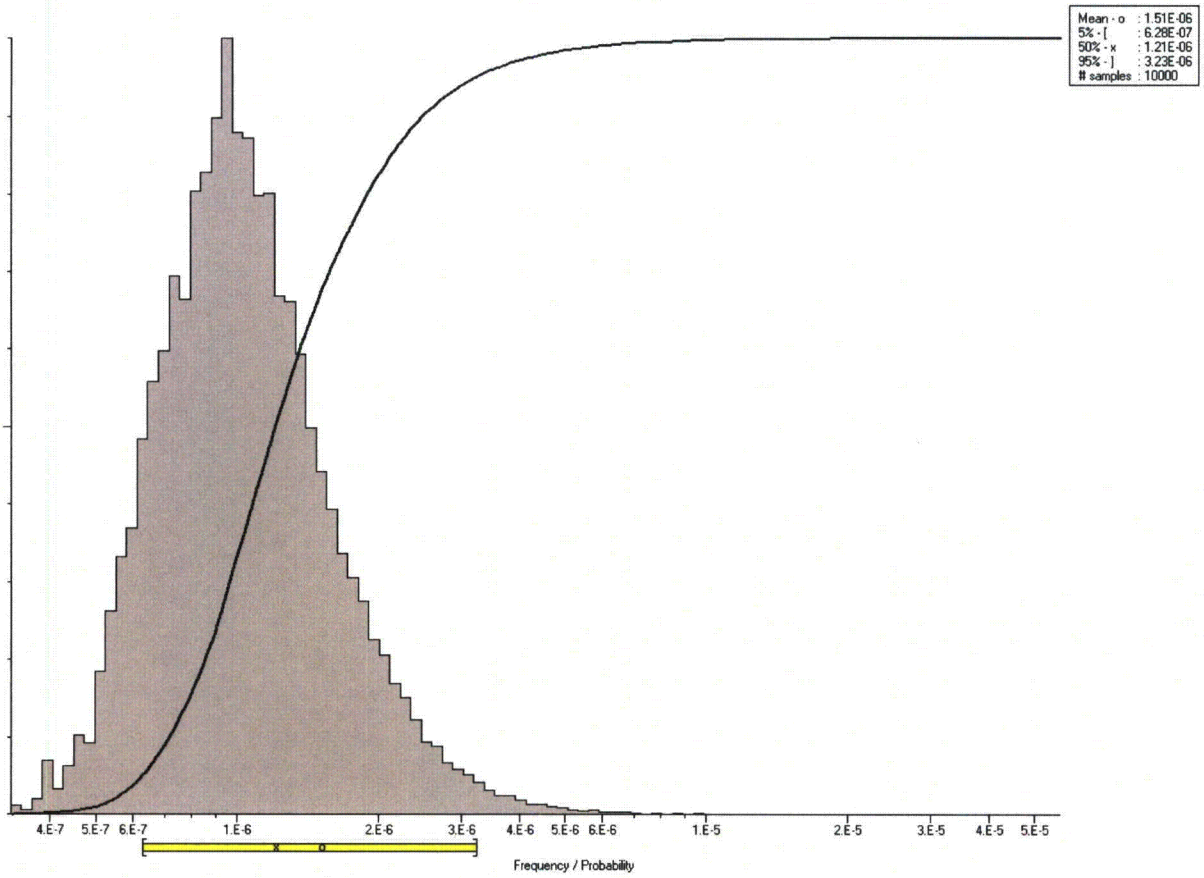

Table 3.2.4-4 - BSEP Unit 2 FPRA LERF - Parametric Uncertainty Statistics
[BNP-PSA-080, Rev. 3, Attachment 38]

	5% Confidence	Mean	95% Confidence
Point Estimate		1.51E-06	
Mean	1.49E-06	1.51E-06	1.54E-06
5%	6.20E-07	6.28E-07	6.40E-07
Median	1.19E-06	1.21E-06	1.22E-06
95%	3.12E-06	3.23E-06	3.33E-06
Standard Deviation		1.32E-06	
Skewness		1.24E+01	
Kurtosis		3.64E+02	

SOKC was evaluated for LERF cutsets as part of the uncertainty analysis. The UNCERT program results above demonstrated that there is no apparent increase in risk over the point estimate risk, so it was concluded that there is no significant data correlation from type coded data events.

However, the potential for non-type coded data events specific to the fire analysis was examined. The following areas of uncertainty were examined for data correlation as recommended by the December 2011 FPRA Peer Review:

Table 3.2.5-1: Identification of Fire Data Types Reviewed for State of Knowledge Correlation [BNP-PSA-080, Rev. 3, Attachment 38]

	Area of Uncertainty	Discussion
1.	Fire ignition frequency	The BSEP fire scenarios are based on single ignition sources. Therefore, there are no correlated ignition frequencies within an individual cutset, precluding SOKC occurrence concerns.
2.	Non-detection probabilities	A generic non-detection probability is used in quantifying the scenario frequencies. Multiple detectors are not credited, so that for individual scenarios, there are no correlated data.
3.	Non-suppression probabilities	There is no correlation between various types of suppression, in that they are uniquely different.
4.	Heat release rate severity factor/split fraction	See Item 1. In addition, the source target relationship is based on a single distance that is used to calculate the HRR severity factors. The split of the generic HRRs is quantified as two individual scenarios, precluding any correlated data in single cutsets.
5.	Circuit failure Probabilities	With the exception of basic events where the hot short probability of 1.0 is used, cutsets including the same component type and failure mode with the same hot short probabilities are assumed completely correlated. The UNCERT code does not address this correlation, so an analysis showing the potential change in LERF is supplied below.

The state of knowledge correlation impact on the CAFTA solved cutsets was assessed by use of a multiplier to the combined correlated failure probabilities. For BSEP, no more than two correlated hot short failure events showed up in a single cutset. The multiplier for two correlated failures can be expressed as:

$$\frac{\left((E(\lambda))^2 + Var(\lambda)\right)}{(E(\lambda))^2}$$

Where: $E(\lambda)$ = the expected value or mean of (λ)
λ = the failure rate being evaluated for the correlated failure data.

Variance was calculated from the standard deviation (Sigma) using the following formula.

$$\mathrm{Var}(\lambda) = \lambda^2 \times (e^{\sigma^2} - 1)$$

BSEP cutsets include basic events with only the hot short probabilities of 0.6 and 0.3. Using a sigma of 0.35 (i.e., Table 3.2.2-2 of BNP-PSA-0080, Attachment 38), the variances are 0.047 and 0.012, respectively. Using this information, the SOKC multiplier for two combined hot short probability events was calculated as follows using the equation above.

Table 3.2.5-2: Calculation of State of Knowledge Correlation Multiplier
[BNP-PSA-080, Rev. 3, Attachment 38]

Hot Short Prob. (HSP) (Lambda)	Standard Deviation (Sigma)	Variance	SOKC Multiplier	SOKC Adder
0.6	0.35	0.047	1.13	0.13*(LERF)
0.3	0.35	0.012	1.13	

The merged cutset were reviewed to identify correlated hot short failure combinations above the truncation of 1.0E-10 for LERF. The potential increase in LERF if the SOKC multiplier was applied is the product of the baseline LERF, the F-V importance of affected cutsets, and the SOKC adder. The following tables summarize the hot short event combinations and potential risk increase from SOKC.

Table 3.2.5-5: BSEP Unit 1 LERF - Impact of State of Knowledge Correlation
[EC 92418]

HSP	Event 1*	Event 2*	Combination F-V Importance	LERF Contribution (of 4.08E-6)	Addition to LERF
0.6	CAC1AOV-CO-V10_A1	CAC1AOV-CO-V9_A1	3.14E-03	1.28E-08	1.67E-09
0.6	MSS1AOV-CO-F003_A1	MSS1AOV-CO-F004_A1	1.69E-03	6.90E-09	8.96E-10
				Total LERF Addition	2.56E-09 (0.06%)

* Suffix of "_A#" is added during cutset merging to identify altered events.

Table 3.2.5-6: BSEP Unit 2 LERF - Impact of State of Knowledge Correlation
[EC 92418]

HSP	Event 1*	Event 2*	Combination F-V Importance	LERF Contribution (of 1.51E-6)	Addition to LERF
0.6	CAC2AOV-CO-V10_A1	CAC2AOV-CO-V9_A1	1.40E-01	2.11E-07	2.75E-08
0.6	MSS2AOV-CO-F022A_A1	MSS2AOV-CO-F028A_A1	1.38E-02	2.08E-08	2.71E-09
0.6	MSS2AOV-CO-F022B_A1	MSS2AOV-CO-F028B_A1	1.38E-02	2.08E-08	2.71E-09
0.6	MSS2AOV-CO-F022C_A1	MSS2AOV-CO-F028C_A1	1.38E-02	2.08E-08	2.71E-09
0.6	MSS2AOV-CO-F022D_A1	MSS2AOV-CO-F028D_A1	1.38E-02	2.08E-08	2.71E-09
				Total LERF Addition	3.83E-08 (2.54%)

* Suffix of "_A#" is added during cutset merging to identify altered events.

The SOKC analysis showed minimal impact on the LERF solutions and small impact on the Unit 2 LERF solution. All the contribution came from the control room complex, for which

adding credit for the first few steps of the abandonment procedure to secure power would correct these spurious events. This suggests that removing conservatisms from the analysis should be prioritized versus making SOKC additions.

Analysis of FPRA Important LERF Contributors:

The merged FPRA cutsets were analyzed to rank several categories of contributors to LERF. The following categories were reviewed and presented in BNP-PSA-080, Attachment 38:

- Fire compartments important to risk,
- Fire scenarios important to risk,
- Accident sequence types important to risk
- Containment failure types important to risk,
- Operator actions important to risk,
- Fire induced equipment failure modes important to risk,
- Component types important to risk, and
- Failure groupings important to risk

An abbreviated summary of the results is provided here for the top contributors.

Fire Compartment Importance

The following tables rank the fire compartments by contribution to risk for each for the plant units and risk metrics. The flag events identifying the fire compartments were used to assemble the ranking.

Table 3.3.1-3 - BSEP Unit 1 FPRA LERF – Fire Compartment Risk Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Cumulative Contribution	Percent Contribution	LERF	Probability	NFPA-805 Location Description
FL_FC210	63.40%	63.40%	2.59E-06	1.00E+00	CB-05 - U1 Cable Spreading Rm 23' Elev
FL_FC230	98.50%	35.10%	1.43E-06	1.00E+00	CB-23 - Control Rm 49' Elev

Table 3.3.1-4 - BSEP Unit 2 FPRA LERF – Fire Compartment Risk Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Cumulative Contribution	Percent Contribution	LERF	Probability	NFPA-805 Location Description
FL_FC230	81.50%	81.50%	1.23E-06	1.00E+00	CB-23 - Control Rm 49' Elev
FL_FC211	95.20%	13.70%	2.07E-07	1.00E+00	CB-06 - U2 Cable Spreading Rm 23' Elev
FL_FC402	96.50%	1.30%	1.96E-08	1.00E+00	TB2-01C/D/E/F/G/H - U2 Turbine Building (TB) Equipment Areas: 20ft

Fire Ignition Source Scenario Importance

The following tables rank the fire ignition source scenarios by contribution to risk for each unit and risk metric. The ignition source contribution to hot gas layer is not included in the risk

ranking but rather by an event for the total contribution of all sources in the compartment to hot gas layer. The percent contribution column is representative of the Fussell-Vesely importance measure.

Table 3.3.2-3 – BSEP Unit 1 FPRA LERF – Fire Scenario Risk Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Cumulative Contribution	Percent Contribution	LERF	Frequency	Fire Source Description
%FC210_BHGL	58.2%	58.2%	2.4E-06	2.4E-06	FC210 HGL
%FC230_4801_B75	73.9%	15.7%	6.4E-07	6.4E-07	1-11A - NODE HY3: CTRL BLDG 125V DC DISTRIBUTION PANEL 11A
%FC230_4801_B98	79.1%	5.2%	2.1E-07	2.1E-07	1-11A - NODE HY3: CTRL BLDG 125V DC DISTRIBUTION PANEL 11A
%FC210_4525_BFM	83.0%	3.9%	1.6E-07	4.7E-05	1-COM-C - 480V UNIT SUBSTATION COM-C
%FC230_MCRA	85.5%	2.5%	1.0E-07	1.0E-07	FC230 - LOSS OF MCR HABITABILITY WITH FAILED ABANDONMENT
%FC230_4747_B98	87.1%	1.6%	6.4E-08	6.4E-08	1-H12-P606 - NODE JG5, JH9: RADIATION MONITORING CABINET

Table 3.3.2-4 – BSEP Unit 2 FPRA LERF – Fire Scenario Risk Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Cumulative Contribution	Percent Contribution	LERF	Frequency	Fire Source Description
%FC230_4819_SELF	23.5%	23.5%	3.6E-07	1.9E-05	2-I2A-HZ3 - NODE HZ3: CTRL BLDG 125VDC DISTRIBUTION PANEL
%FC230_4807_BFM	32.0%	8.5%	1.3E-07	1.3E-07	2-H12-P609 - NODE JM6, JH3: RPS TRIP SYSTEM A, MAIN CONTROL ROOM PN
%FC230_4845_BFM	40.2%	8.3%	1.2E-07	3.2E-07	2-H12-P608 - NODE JC5, JC6, JC7, JP7, JP8, JP9: POWER RANGE NEUTRON
%FC230_MCRA	47.1%	6.8%	1.0E-07	1.0E-07	FC230 - LOSS OF MCR HABITABILITY WITH FAILED ABANDONMENT
%FC211_4572_BFM	52.5%	5.4%	8.2E-08	3.7E-05	2-2L - 480V UNIT SUBSTATION 2L BUS
%FC230_4873_BFM	57.8%	5.3%	8.1E-08	1.3E-07	2-H12-P617 - NODE JE6: RHR A RELAY VERTICAL BOARD
%FC211_4568_BFM	62.9%	5.1%	7.7E-08	3.7E-05	2-COM-D - UNIT SUBST COMMON 'D'
%FC230_4829_BFM	66.2%	3.3%	5.0E-08	1.3E-07	2-H12-P618 - NODE JH0: RHR RELAY VERTICAL BOARD
%FC230_4878_B98	69.4%	3.2%	4.8E-08	4.8E-08	2-H12-P630 - NODE JE1: REACTOR ANNUNCIATOR CABINET
%FC230_4814_B75	71.8%	2.4%	3.6E-08	6.4E-07	2-XU-30 - NODE H61: DIESEL GEN 4 ESS LOGIC CABINET
%FC230_4897_SELF	72.9%	1.1%	1.6E-08	2.0E-05	2-12B - NODE HZ5: CTRL BLDG 125VDC DISTRIBUTION PANEL

Table 3.3.2-4 – BSEP Unit 2 FPRA LERF – Fire Scenario Risk Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Cumulative Contribution	Percent Contribution	LERF	Frequency	Fire Source Description
%FC230_4806_B98	73.9%	1.1%	1.6E-08	1.6E-08	2-XU-47 - NODE JS8: TERMINAL CABINET FOR RIP
%FC230_4835_B98	75.0%	1.1%	1.6E-08	1.6E-08	2-XU-48 - NODE JS9: TERMINAL CABINET FOR RIP
%FC230_4876_B98	76.0%	1.1%	1.6E-08	1.6E-08	2-XU-46 - NODE JL9: BOP LOGIC ANNUNCIATOR CAB FOR UA-23 & 24
%FC230_4877_B98	77.1%	1.1%	1.6E-08	1.6E-08	2-XU-52 - NODE JW5: ANNUNCIATOR LOGIC CABINET FOR UA 25 & 27

Fire Accident Sequence Contributions

Accident Sequences were reviewed by major event types. All transient and SBO sequences result in either loss of makeup events or loss of decay heat removal events that result in a loss of makeup. The specific sequences are not identified because the sequence flag events are not used during FPRA quantification. However, the cutsets contain sufficient information to identify sequence contributions based on the failed events. The following tables provide a breakdown of the sequence contributions for both units for LERF.

Table 3.3.3-3 – BSEP Unit 1 FPRA LERF – Accident Sequence Type Risk Ranking (>0.25%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Accident Sequence Type	Percent Contrib.	LERF	Comment
Transients	87.7%	3.6E-06	Includes control room abandonment, loss of makeup, loss of decay heat removal, and LOOP or loss of emergency power.
MSIV Loss of Cooling Accident (LOCA)	11.8%	4.8E-07	Bypass sequence
Station Blackout	0.4%	1.4E-08	SBO does not include non-LOOP sequences with emergency bus failures.

Table 3.3.3-4 – BSEP Unit 2 FPRA LERF – Accident Sequence Type Risk Ranking (>0.25%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Accident Sequence Type	Percent Contrib.	LERF	Comment
MSIV LOCA	56.5%	8.5E-07	Bypass sequence
Transients	40.0%	6.0E-07	Includes control room abandonment, loss of makeup, loss of decay heat removal, and LOOP or loss of emergency power.
Station Blackout	2.5%	3.7E-08	SBO does not include non-LOOP sequences with emergency bus failures.
Small LOCA	1.0%	1.6E-08	Associated with RPV Head Vent MSOs

Plant damage states (PDS) are not specifically identified, as the flag events are set to true during quantification. The standard suggests, as an example, to identify plant damage state importance as a means of calculating LERF contribution. While this may be important for models that calculate LERF through PDS split fractions developed in models external to the fault tree solution, it is not relevant to the BSEP LERF model that has all the PDS contributors explicitly modeled. Therefore, the importance factors to LERF can be evaluated without identifying PDS flags.

Containment Failure Ranking

Unit 1 LERF is dominated by a single scenario associated with the Unit 1 cable spread room hot gas layer. For this scenario, the calculation method does not provide insights into the contributors to LERF. It is assumed that a containment isolation failure occurs from one or more of many possible penetrations. However, the dominant cause of core damage is associated with emergency power blackout. Therefore, it is highly likely, that the LERF would be less than the presented value due to isolation valves closing in the safe state after power is lost. This specific contribution to LERF is an area of uncertainty. Similarly, the analysis for control room abandonment does not identify specific contributors to LERF. For both cases, the scenario is presented as the cause of LERF and not the specific type of containment failure.

The relative importance of types of containment failures contributing to LERF are provided in the following tables.

Table 3.3.3-5 – BSEP Unit 1 FPRA LERF – Containment Failure Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Accident Sequence Type	Percent Contrib.	LERF	Comment
U1 Cable Spread Rm. HGL	58.2%	2.37E-06	The HGL scenario assumes a 100% contribution to LERF due to containment isolation faults. This is an area of uncertainty.
PCI Failures – Fire Induced	26.6%	1.09E-06	Containment isolation failures associated with CAC and DW drain systems.
MSIV LOCA	11.8%	4.80E-07	Bypass sequence from direct valve cable hot shorts or loss of isolation signal. Value includes random failures.
MCR Abandonment	2.5%	1.03E-07	The failure of control room abandonment assumes a 10% contribution to LERF.

Table 3.3.3-6 – BSEP Unit 2 FPRA LERF – Containment Failure Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Accident Sequence Type	Percent Contrib.	LERF	Comment
MSIV LOCA	56.5%	8.5E-07	Bypass sequence from direct valve cable hot shorts or loss of isolation signal. Value includes random failures.
PCI Failures – Fire Induced	33.2%	5.0E-07	Containment isolation failures associated with Containment Atmospheric Control (CAC) and drywell (DW) drain systems.
MCR Abandonment	6.8%	1.0E-07	The failure of control room abandonment assumes a 10% contribution to LERF.
Level 2 Phenomena	2.5%	3.8E-08	Includes over-pressure phenomena such as hydrogen deflagration, steam explosions, direct containment heating, and vapor suppression failures. Includes containment breach from induced ISLOCA, RPV pedestal attack and missiles.
PCI Failures – Non-fire	1.4%	2.2E-08	Associated with RPV Head Vent MSOs

Operator Actions Important to FPRA Risk

The FPRA cutsets were reviewed to determine and rank important operator actions. This includes post initiation type (i.e., Cp) and non-procedural type (i.e., Cr) actions that are part of the emergency operating procedure (EOP) network. The only alternate shutdown action that is modeled is control room abandonment. The following tables present the important operator actions for LERF (i.e., contributing > 1%) for each of the two units. All actions are performed in whole or in part away from the control room unless otherwise indicated by bold text.

Table 3.3.4-3 – BSEP Unit 1 FPRA LERF – Human Action Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name*	Percent Contrib.	LERF	Description
OPER-VALVECLOSE	21.40%	8.73E-07	FAILURE TO CLOSE VALVE GIVEN FAILURE OF SIGNAL
OPER-MCRA	2.52%	1.03E-07	FAILURE OF MAIN CONTROL ROOM ABANDONMENT
OPER-FPS1	2.46%	1.00E-07	FAILURE TO ALIGN FIREWATER FOR COOLANT INJECTION FLOW (ONE UNIT)
OPER-DEPRESS	1.46%	5.96E-08	FAILURE TO MANUALLY INITIATE AND ALIGN LOW-PRESSURE SYSTEMS

* Bolded text indicated control room action

Table 3.3.4-4 – BSEP Unit 2 FPRA LERF – Human Action Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name	Percent Contrib.	LERF	Description
OPER-FPS1	20.70%	3.13E-07	FAILURE TO ALIGN FIREWATER FOR COOLANT INJECTION FLOW (ONE UNIT)
OPER-MCRA	6.82%	1.03E-07	FAILURE OF MAIN CONTROL ROOM ABANDONMENT
OPER-4160X	2.45%	3.70E-08	FAILURE TO ALIGN POWER FROM OPPOSITE UNIT
OPER-SWRHR-O	2.24%	3.38E-08	FAILURE TO LOCALLY OPEN THE DISCHARGE VALVES FOR RHR INJECTION
OPER-RCICEXT	2.04%	3.08E-08	FAILURE TO EXTEND RCIC OPERATION BY MANAGING HCTL AND DEFEATING TRIPS
OPER-480X	1.73%	2.61E-08	FAILURE TO CONNECT UNIT 1 SUBSTATIONS E5 AND E6
OPER-SWRHR-C	1.71%	2.58E-08	FAILURE TO LOCALLY CLOSE SW VALVES FOR FW INJECTION

Fire Induced Component Failures Important to Risk

The FPRA cutsets were reviewed to determine and rank important fire induced component failures. The following tables present failure modes contributing greater than 2% to LERF. Failure events are sorted by Fussell-Vesely importance (i.e., % contribution). Bolded events are those failure modes that are spurious events. These events have values of 1.0, typically indicating that the sum of hot short probabilities for multiple control cables is approximately 1.0 or that detailed circuit analysis was not performed. Additional circuit analysis may provide improved risk results. A specific method for evaluating instrument faults has not been identified at this time.

Table 3.3.5-3 – BSEP Unit 1 FPRA LERF – Fire Induced Component Failure Mode Ranking (>2%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name*	Percent Contrib.	LERF	Prob.	Description
ACP0BKR-CO-AU9_T1	61.80%	2.52E-06	1.00E+00	CIRCUIT BREAKER AU9 FAILS TO REMAIN CLOSED
ACP0BKR-CO-AV4_T1	58.80%	2.40E-06	1.00E+00	CIRCUIT BREAKER AV4 FAILS TO REMAIN CLOSED
ICC1PTT-HI-N21A_T1	24.50%	1.00E-06	1.00E+00	PRESSURE TRANSMITTER B21-PT-N021A FAILS HIGH
ICC1PTT-HI-N21B_T1	24.50%	1.00E-06	1.00E+00	PRESSURE TRANSMITTER B21-PT-N021B FAILS HIGH
ADS-CHANAB_T1	23.90%	9.75E-07	1.00E+00	SPURIOUS OPERATION OF ADS CHANNEL A OR B
ICC1LTT-HI-N017A_T1	21.20%	8.65E-07	1.00E+00	REACTOR WATER LOW LEVEL #1 LEVEL TRANSMITTER B21-LT-N017A FAILS HIGH
ICC1LTT-HI-N017C_T1	21.20%	8.65E-07	1.00E+00	REACTOR WATER LOW LEVEL #1 LEVEL TRANSMITTER B21-LT-N017C FAILS HIGH
ICC1PTT-LO-N002A_T1	21.20%	8.65E-07	1.00E+00	DRYWELL HIGH PRESSURE TRANSMITTER C71-PT-N002A FAILS LOW
ICC1PTT-LO-N002C_T1	21.20%	8.65E-07	1.00E+00	DRYWELL HIGH PRESSURE TRANSMITTER C71-PT-N002C FAILS LOW
ICC1LTT-HI-N024B_T1	6.09%	2.48E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N024B FAILS HIGH
ICC1LTT-HI-N025B_T1	6.09%	2.48E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N025B FAILS HIGH
DCP1BDC-LP1AP_T1	4.34%	1.77E-07	1.00E+00	FAILURE OF DCP 125V DC SWITCHBOARD 1A BUS P

Table 3.3.5-3 – BSEP Unit 1 FPRA LERF – Fire Induced Component Failure Mode Ranking (>2%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name*	Percent Contrib.	LERF	Prob.	Description
CAC1SOV-CO-V49_T1	2.84%	1.16E-07	1.00E+00	SOLENOID VALVE CAC-V49 TRANSFERS OPEN
CAC1SOV-CO-V50_T1	2.84%	1.16E-07	1.00E+00	SOLENOID VALVE CAC-V50 TRANSFERS OPEN

* Bolded text indicated control room action

Table 3.3.5-4 – BSEP Unit 2 FPRA LERF – Fire Induced Component Failure Mode Ranking (>2%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name*	Percent Contrib.	LERF	Prob.	Description
DCP2BDC-LP4A_T1	34.10%	5.15E-07	1.00E+00	FAILURE OF DCP 125V DC DISTRIBUTION PANEL 4A
ACP0TFM-LP-SAT2_T1	25.30%	3.82E-07	1.00E+00	TRANSFORMER SAT #2 FAILURE NO POWER
ICC2LTT-HI-N024A_T1	25.00%	3.78E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N024A FAILS HIGH
ICC2LTT-HI-N025A_T1	25.00%	3.78E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N025A FAILS HIGH
RCI2TME-HI-N021B_T1	19.40%	2.93E-07	1.00E+00	TEMPERATURE ELEMENT E51-TE-N021B SPURIOUS OPERATION
RCI2TME-HI-N022B_T1	19.40%	2.93E-07	1.00E+00	TEMPERATURE ELEMENT E51-TE-N022B SPURIOUS OPERATION
CAC2AOV-CO-V10_A1	14.00%	2.12E-07	6.00E-01	CAC-V10 FAILS TO REMAIN CLOSED
CAC2AOV-CO-V9_A1	14.00%	2.12E-07	6.00E-01	CAC-V9 FAILS TO REMAIN CLOSED
ICC2LTT-HI-N024B_T1	14.00%	2.12E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N024B FAILS HIGH
ICC2LTT-HI-N025B_T1	14.00%	2.12E-07	1.00E+00	REACTOR LOW LEVEL TRANSMITTER B21-LT-N025B FAILS HIGH
SRV2CBL-DCP-NORM_T1	13.80%	2.08E-07	1.00E+00	NORMAL DCP SUPPLY CABLES BETWEEN DP 4B AND SRV RELAY PANEL DAMAGED BY POWER due
ICC2PTT-HI-N21D_T1	11.70%	1.77E-07	1.00E+00	PRESSURE TRANSMITTER B21-PT-N021D FAILS HIGH
CAC2SOV-CO-V49_T1	10.70%	1.62E-07	1.00E+00	SOLENOID VALVE CAC-V49 TRANSFERS OPEN
CAC2SOV-CO-V50_T1	10.70%	1.62E-07	1.00E+00	SOLENOID VALVE CAC-V50 TRANSFERS OPEN
ICC2LSO-NO-N037_T1	9.71%	1.47E-07	1.00E+00	Reactor Fuel Zone Level Recorder B21-LR-R615 (B21-LT-N037) Fails
SRV2SOV-CC-F013A_T1	9.27%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013A FAILS TO OPEN
SRV2SOV-CC-F013B_T1	9.27%	1.40E-07	1.00E+00	SOV FOR NON-ADS SRV B21-F013B FAILS TO OPEN
SRV2SOV-CC-F013C_T1	9.27%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013C FAILS TO OPEN
SRV2SOV-CC-F013D_T1	9.27%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013D FAILS TO OPEN
SRV2SOV-CC-F013E_T1	9.27%	1.40E-07	1.00E+00	SOV FOR NON-ADS SRV B21-F013E FAILS TO OPEN
SRV2SOV-CC-F013H_T1	9.27%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013H FAILS TO OPEN
SRV2SOV-CC-F013J_T1	9.27%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013J FAILS TO OPEN
SRV2SOV-CC-F013L_T1	9.25%	1.40E-07	1.00E+00	SOV FOR ADS SRV B21-F013L FAILS TO OPEN
SRV2SOV-CC-F013K_T1	9.09%	1.37E-07	1.00E+00	SOV FOR ADS SRV B21-F013K FAILS TO OPEN
DCP2BDC-LP12A_T1	8.23%	1.24E-07	1.00E+00	FAILURE OF DCP 125V DC DISTRIBUTION PANEL 12A
MSS2AOV-CO-F003_T1	7.19%	1.09E-07	1.00E+00	RPV HEAD VENT VALVE B21-F003 SPURIOUSLY OPENS
MSS2AOV-CO-F004_T1	7.19%	1.09E-07	1.00E+00	RPV HEAD VENT VALVE B21-F004 SPURIOUSLY OPENS
CAC2AOV-CO-V216_T1	6.17%	9.32E-08	1.00E+00	CAC-V216 FAILS TO REMAIN CLOSED
CAC2AOV-CO-V7_A1	6.07%	9.17E-08	6.00E-01	CAC-V7 FAILS TO REMAIN CLOSED
ACP0BKR-CO-AV4_T1	5.72%	8.64E-08	1.00E+00	CIRCUIT BREAKER AV4 FAILS TO REMAIN CLOSED
FIRE-LOCOND	5.69%	8.60E-08	1.00E+00	FIRE INDUCED LOSS OF CONDENSER or CWS or TCS

Table 3.3.5-4 – BSEP Unit 2 FPRA LERF – Fire Induced Component Failure Mode Ranking (>2%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Event Name*	Percent Contrib.	LERF	Prob.	Description
ACP0BKR-OO-AX1_T1	5.65%	8.54E-08	1.00E+00	MCC CIRCUIT BREAKER AX1 FAILS TO CLOSE
RCI2PPS-SA-N012A_T1	5.65%	8.54E-08	1.00E+00	PRESSURE SWITCH E51-N012A SPURIOUS OPERATION
RCI2PPS-SA-N012C_T1	5.65%	8.54E-08	1.00E+00	PRESSURE SWITCH E51-N012C SPURIOUS OPERATION
ACP0BKR-CO-AZ1_T1	5.49%	8.29E-08	1.00E+00	CIRCUIT BREAKER AZ1 FAILS TO REMAIN CLOSED
ACP0BKR-CO-AZ5_T1	5.38%	8.13E-08	1.00E+00	CIRCUIT BREAKER AZ5 FAILS TO REMAIN OPEN
ACP0BKR-OO-A10_T1	5.32%	8.04E-08	1.00E+00	CIRCUIT BREAKER A10 FAILS TO CLOSE
RCI2PPS-SA-N19B_T1	5.29%	7.99E-08	1.00E+00	PRESSURE SWITCH E51-N019B SPURIOUSLY ACTUATED
RCI2PPS-SA-N19D_T1	5.29%	7.99E-08	1.00E+00	PRESSURE SWITCH E51-N019D SPURIOUSLY ACTUATED
SWS2MOV-OC-V37_T1	5.26%	7.95E-08	1.00E+00	MOTOR-OPERATED VALVE SW V37 FAILS TO REMAIN OPEN
SWS2PPS-SAP3213L_T1	4.92%	7.43E-08	1.00E+00	PRESSURE SWITCH PS3213 SPURIOUS OPERATION FAILS LOW ISOLATES HEADER
ICC2PTT-HI-N21B_T1	4.88%	7.37E-08	1.00E+00	PRESSURE TRANSMITTER B21-PT-N021B FAILS HIGH
HPC2PPS-SA-N12A_T1	3.97%	6.00E-08	1.00E+00	PRESSURE SWITCH E41-N012A SPURIOUSLY ACTUATES
HPC2PPS-SA-N12C_T1	3.97%	6.00E-08	1.00E+00	PRESSURE SWITCH E41-N012C SPURIOUSLY ACTUATES
ACP0BKR-OO-2AC6_T1	2.46%	3.72E-08	1.00E+00	CIRCUIT BREAKER FROM SAT #2 TO 2C (2-AC6) FAILS TO CLOSE
ACP1BAC-LP-32A_T1	2.21%	3.34E-08	1.00E+00	120V AC DISTRIBUTION PANEL 32A FAILURE (NO POWER)
ICC2PTT-HI-N09A_T1	2.21%	3.34E-08	1.00E+00	PRESSURE DIFFERENTIAL TRANSMITTER B21-PDT-N009A FAILS HIGH
ACP2BAC-LP-2D_T1	2.03%	3.07E-08	1.00E+00	120V AC DISTRIBUTION PANEL 2D FAILURE (NO POWER)
IAN2SOV-OCSV5481_T1	2.01%	3.04E-08	1.00E+00	SOLENOID VALVE SV 5481 TRANSFERS CLOSED

* Bolded text indicated control room action

System Importance Ranking

The FPRA cutsets were reviewed to determine and rank important systems to fire risk. The ranking is based on the basic event three letter prefix for the system and includes both random faults and fire-induced faults. It should be understood that feedwater and main steam are linked fault trees to condensate, so that this contribution should be considered to include those systems. The following tables present the results of the analysis.

Table 3.3.7-3 – BSEP Unit 1 FPRA LERF – System Importance Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

System	Percent Contrib.	LERF	System Prefix
AC Power System	65.26%	2.66E-06	ACP*
Instrumentation and Control Pseudo-system	31.82%	1.30E-06	ICC*
DC Power System	5.62%	2.29E-07	DCP*
Main Steam System	5.59%	2.28E-07	MSS*
Containment Atmospheric Control System and Hardened Wetwell Vent	5.21%	2.12E-07	CAC*

Table 3.3.7-3 – BSEP Unit 1 FPRA LERF – System Importance Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

System	Percent Contrib.	LERF	System Prefix
Conventional and Nuclear Service Water System	3.50%	1.43E-07	SWS*
Reactor Core Isolation Cooling System	3.03%	1.24E-07	RCI*
High Pressure Coolant Injection System	2.81%	1.15E-07	HPC*
Safety Relief Valves and ADS	2.57%	1.05E-07	SRV*
Condensate System	1.95%	7.94E-08	CDS*
Reactor Building Closed Cooling Water System	1.10%	4.50E-08	RCC*

Table 3.3.7-4 – BSEP Unit 2 FPRA LERF – System Importance Ranking (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

System	Percent Contrib.	LERF	System Prefix
Instrumentation and Control Pseudo-system	50.14%	7.57E-07	ICC*
AC Power System	47.08%	7.11E-07	ACP*
DC Power System	38.08%	5.75E-07	DCP*
Containment Atmospheric Control System and Hardened Wetwell Vent	34.99%	5.29E-07	CAC*
Reactor Core Isolation Cooling System	33.61%	5.08E-07	RCI*
Safety Relief Valves and ADS	23.83%	3.60E-07	SRV*
Conventional and Nuclear Service Water System	20.28%	3.06E-07	SWS*
Main Steam System	18.11%	2.74E-07	MSS*
Condensate System	6.10%	9.22E-08	CDS*
High Pressure Coolant Injection System	5.51%	8.32E-08	HPC*
Instrument Air and Nitrogen	4.17%	6.31E-08	IAN*
Residual Heat Removal	3.23%	4.88E-08	RHR*
Control Rod Drive System (Makeup)	1.92%	2.90E-08	CRD*
Turbine Building Closed Cooling Water	1.17%	1.77E-08	TBC*

Analysis of Component Types and Failure Types Importance

The FPRA cutsets were reviewed to determine the risk ranking of important types of component (e.g., valves and pumps) and types of failure modes (i.e., common cause failures and human failures). The ranking is based on the basic event naming conventions, where the three-letter prefix for the system and includes both random faults and fire-induced faults. The following tables present the results of the analysis.

Table 3.3.8-3 – BSEP Unit 1 FPRA LERF – Component and Failure Type Rankings (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Component Type/ Failure Type	Percent Contrib.	LERF	BE String
Valves			
All Valves	12.42%	5.07E-07	??????V-*
Pneumatic Valves	8.59%	3.50E-07	????AOV-*
Solenoid Valves	4.55%	1.85E-07	????SOV*
Motor Operated Valves	1.54%	6.27E-08	????MOV-*
Main Drivers			
All Pumps	1.83%	7.46E-08	?????DP-*
Motor-Driven Pumps	1.81%	7.40E-08	????MDP-*
AC Power Components			
AC Breakers	64.26%	2.62E-06	ACP?BKR*
AC Transformers	1.55%	6.33E-08	????TFM*
DC Power Components			
DC Distribution Panels	5.06%	2.07E-07	????BDC*
Instrumentation and Relays			
Level Transmitters	27.66%	1.13E-06	????LTT*
Pressure Transmitters	26.71%	1.09E-06	????PTT*
Pressure Switches	5.28%	2.15E-07	????PPS*
Temperature Elements	1.11%	4.53E-08	????TME*
Failure Type Contributions			
HRAs - Type Cp/Cr	28.52%	1.16E-06	OPER-*
HRAs- Post Processed	25.95%	1.06E-06	XOP-*
Common Cause Failures	2.04%	8.31E-08	*-CF*
Type A - Pre-Init HRAs	1.16%	4.73E-08	????XHE-MN*

Table 3.3.8-4 – BSEP Unit 2 FPRA LERF – Component and Failure Type Rankings (>1%)
[BNP-PSA-080, Rev. 3, Attachment 38]

Component Type/ Failure Type	Percent Contrib.	LERF	BE String
Valves			
All Valves	62.02%	9.37E-07	??????V-*
Pneumatic Valves	41.13%	6.21E-07	????AOV-*
Solenoid Valves	23.97%	3.62E-07	????SOV*
Motor Operated Valves	17.85%	2.70E-07	????MOV-*
Manual Valves	3.58%	5.41E-08	????XVN-*
Main Drivers			
Compressors	1.11%	1.68E-08	????MDC*
All Pumps	1.00%	1.51E-08	?????DP-*
AC Power Components			
AC Transformers	26.45%	4.00E-07	????TFM*
AC Breakers	17.32%	2.62E-07	ACP?BKR*
AC Switchgears/Buses/MCCs	5.06%	7.64E-08	????BAC*
DC Power Components			
DC Distribution Panels	36.47%	5.51E-07	????BDC*
DC Batteries	1.10%	1.66E-08	????BAT*
Instrumentation and Relays			
Level Transmitters	39.52%	5.97E-07	????LTT*
Pressure Transmitters	20.46%	3.09E-07	????PTT*
Temperature Elements	20.40%	3.08E-07	????TME*
Pressure Switches	18.32%	2.77E-07	????PPS*
Group Contributions			
HRAs - Type Cp/Cr	33.98%	5.13E-07	OPER-*
HRAs- Post-processed	27.05%	4.09E-07	XOP-*
Common Cause Failures	5.11%	7.72E-08	*-CF*
Type A - Pre-Init HRAs	4.39%	6.63E-08	????XHE-MN*

PRA RAI 1R

r) F&O 6-1 against CS-B1 (Cat II) and CS-C4 (Not Met):

It is unclear from the documentation whether the breaker coordination studies for Brunswick Units 1 and 2 are complete. Section 3.3.1.7 of the LAR states that "short circuit and coordination calculations shall be updated as necessary" and it is noted that there are several breaker coordination change packages, and revised packages documented in BNP-PSA-080. Attachment 36 of BNP-PSA-080 states that three raceways could not be routed. In light of these observations:

i. Clarify how the breaker coordination study assessed the three raceways that could not be routed, given that breaker coordination is assessed based on length of cable.

ii. Clarify that all panels modeled in the FPRA have been evaluated and whether the breaker coordination study is complete.

Response

i. Attachment 36 (i.e., BNP-0218, Page C-1) of BNP-PSA-080, *BNP Fire PRA – Quantification*, identifies two raceways that could not be routed on drawings. However, Section 5.d of Attachment 36 describes how a reasonable approximation for the fire zones that the raceways traverse was possible based on the raceways adjacent to the raceway in question, the cable start and end equipment, and general plant layout knowledge.

Attachment 37 (i.e., BNP-0224, Page C-4) of BNP-PSA-080 identifies three other raceways that could not be routed. While not immediately apparent from the information in Attachment 37, the raceways that are missing information are located in the Unit 2 electrical equipment room (i.e., Control Building 49'). The raceway for 2-2A-120V involves a cable running between two panels in adjacent rows in the Unit 2 electrical equipment room (i.e., Control Building 49'). The two raceways for 2-2D-120V involve a cable running from a panel in the Unit 2 electrical equipment room (i.e., Control Building 49') to a panel in the Unit 2 cable spreading room (i.e., Control Building 23'), which is where the two other raceways are known to be located.

These three raceways are identified as a source of uncertainty in Section 3.6.6 of BNP-PSA-080, and the risk associated with their assumed failure is qualitatively addressed as a non-conservative assumption (i.e., Section 3.1.3.44) that is likely mitigated in the HGL scenario by other failures for the respective power supplies.

ii. All panels modeled in the FPRA were included within the scope of the breaker coordination study, as described in BNP-0217, which has been completed.

PRA RAI 4

Per NUREG/CR-6850 Section 11.5.1.6, transient fires should at a minimum be placed in locations within the plant physical analysis units (PAUs) where conditional core damage probabilities are highest for that PAU (i.e., at "pinch points"). Pinch points include locations of redundant trains or the vicinity of other potentially risk-relevant equipment, including the cabling associated with each. Transient fires should be placed at all appropriate locations in a PAU where they can threaten pinch points. Hot work should be assumed to occur in locations where hot work is a possibility, even if improbable, keeping in mind the same philosophy. Describe how transient and hot work fires are distributed within the PAUs at your plant. In particular, identify the criteria for your plant used to determine where an ignition source is placed within the PAUs. Also, if there are areas within a PAU where no transient or hot work fires are postulated because those areas are considered inaccessible, describe the criteria used to define "inaccessible." Note that an inaccessible area is not the same as a location where placement of a transient is simply unlikely. If there are "inaccessible" locations where hot work or transient fires are improbable and these locations are pinch points, provide a sensitivity study to determine the possible risk increase reflecting the possible size and frequency of fires in these locations.

Response

As described in Section 9.2.4 of Attachment 22 of BNP-PSA-083, *BNP Fire PRA - Plant Partitioning and Ignition Frequency*, the transient ignition sources are identified by walkdowns, using a specific methodology. This methodology assumes that transient ignition sources can be placed anywhere within a compartment; however, only those sources with potential targets within the specified ZOI actually result in fire scenarios that can be modeled in the FPRA. Consequently, potential targets identified with this methodology encompass “pinch points” as described in Section 11.5.1.6 of NUREG/CR-6850. However, if an area has no potential target within the ZOI of any postulated ignition source, that area has no fire scenario for a transient ignition source. As described in Section 9.1.1 of BNP-PSA-086, the ZOI used during the walkdown is based on thermoset cables, as would be appropriate for the plant, with a large trash bag on the floor under or near a raceway or cable being the mental model of the most likely transient ignition source. In practice, transient ignition sources are not postulated on top of plant equipment or wedged between cable trays because that is generally not a realistic representation of housekeeping at the plant.

Without establishing specific criteria, Section 9.2.4 of Attachment 22 of BNP-PSA-083 describes a Locked High Radiation area as "inaccessible." Because this represents an administrative control with a physical barrier and health threat, a sensitivity study is not considered to be required.

As described in Section 9.2.4 of BNP-PSA-086, *BNP Fire PRA - Fire Scenario Data*, cable fires due to cutting and welding are assigned no target sets because a continuous fire watch with an extinguisher is required by 0FPP-005, *Fire Watch Program*, to be present during hot work activities and is assumed to extinguish such a fire before it can spread beyond the original tray. Transient fires due to cutting and welding are assumed to involve the same target sets as the general transients.

PRA RAI 5

The sensitivity study presented in Section 4.8.3.6 of the LAR removes credit for incipient detection, also known as, the Very Early Warning Fire Detection System that will be installed in the MCR main control boards (MCBs). Explain why the sensitivity study results indicate no change (i.e., 0%) in ΔCDF but relatively significant change (i.e., +48%) in ΔLERF.

Response

The change in the ΔLERF value is predominantly caused by scenario FC230_8951 BCR (i.e., see EC 89666). This scenario is a fire in panel P601 which is not suppressed and damages external targets. The frequency of this scenario is increased in the sensitivity study due to the removal of credit for incipient detection which increases the probability of not suppressing the fire prior to external damage.

The scenario is a ΔLERF contributor due to cables in the ZOI related to the MSIVs. These cables are identified as variances from deterministic requirements (VFDRs) because the MSIVs are required to close for RPV inventory control. The PRA conservatively assumes an open inboard and outboard MSIV in the same line to be a LERF pathway because the status of the Turbine Control Valve and Turbine Stop Valve are unknown. Removing the MSIV cables from the target set effectively reduces the CLERP from 1.0 to nearly zero. The increase in the frequency for the scenario causes an increase in the ΔLERF.

This scenario does not contribute to ΔCDF because the target set for the scenario is sufficient to cause a CCDP of 1.0 regardless of whether it contains the VFDR cables for the MSIVs.

PRA RAI 9

Identify if any VFDRs in the LAR involved performance-based evaluations of wrapped or embedded cables. If applicable, describe how wrapped or embedded cables were modeled in the FPRA, including assumptions and insights on how these cables contribute to the VFDR delta-risk evaluations.

Response

BSEP does not have any VFDR that involves performance-based evaluations of wrapped or embedded cables.

Protected cables are modeled in the PRA by removing the associated failures from the target sets of the respective sources. While not applicable to VFDRs, the FPRA credited cable protection when it was unrelated to deterministic compliance but still appropriate for risk reduction. Cases where cable protection was credited are as follows:

- Conduit 16IL1/BA was assumed to be protected from 1-1L-480V Unit Substation 1L, as described in Section 3.1.3.43 of BNP-PSA-080, *BNP Fire PRA – Quantification*. This protection will be achieved through modification, Table S-1, item 5 and will be further discussed in the response for RAI-SSA-02.

- Four cables in the MCR were assumed to be protected from specific MCR sources, as described in Sections 3.1.3.46, 3.1.3.47, and 3.1.3.48 of BNP-PSA-080. This protection will be achieved through modification, Table S-1, item 7 and will be further discussed in the response for RAI-SSA-02.
- Raceways in DG-05 (i.e., Diesel Generator Cell 1) which are encased in Pyrocrete were assumed to be protected from sources in the area, as described in Section 3.1.3.34 of BNP-PSA-080. This will be further discussed in the response for RAI-FPE-15

PRA RAI 10

Attachment W of the LAR presents the total CDF and LERF for Units 1 and 2 and specifies the CDF from each of the following contributors: "Internal Events (including internal flooding)," "External Flood," "High Wind," "Seismic," and "Fire." The seismic CDF (6.2 E-8/yr for Unit 1 and 6.5E-8/yr for Unit 2) used in this estimate is low compared to the seismic CDF estimate (1.5E-5/yr) presented in a memorandum from NRC staff dated September 2010 providing updated results for Generic Issue 199 (memo titled: Safety/Risk Assessment Results for Generic Issue 199, Implication for Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United states on Existing Plants"). Also, the CDF provided for internal events (7.9E-6/yr) is much lower than the internal events CDF (4.2E-5/yr) reported in NUREG-1437, Supplement 25, dated 2006, for the BSEP license renewal environmental report. Identify the bases for the internal and seismic event CDFs and LERFs presented in the LAR, and justify the adequacy of these risk estimates for this application.

Response

The CDF and LERF for the internal events PRA, as presented in Table W-1 of the LAR, were estimated from model of record (MOR) 2011, as described in Section 3.2.5 of BNP-PSA-030, *PRA Model Sequence Quantification*. By contrast, Section G.2.1 of NUREG-1437 identifies MOR03 as the source that reported CDF. As documented in Attachment 8 of BNP-PSA-068, *BNP – PSA BWROG F&O Resolutions*, numerous changes have been made to the internal events PRA model in the intervening years.

BSEP Model of Record Update Summary

MOR	REV	Unit	CDF (freq/yr)	LERF (freq/yr)
MOR04	4	Unit 1	4.11E-05	6.31E-07
		Unit 2	4.04E-05	6.29E-07
MOR05	5	Unit 1	3.59E-05	6.31E-07
		Unit 2	3.32E-05	6.29E-07
MOR06	6	Unit 1	3.34E-05	2.13E-06 (Based on MOR03)
		Unit 2	3.07E-05	2.13E-06 (Based on MOR03)
MOR07	7	Unit 1	1.22E-05	2.13E-06 (Based on MOR03)
		Unit 2	1.17E-05	2.13E-06 (Based on MOR03)
2007 Self Assessment				
MOR08	8	Unit 1	1.09E-05	2.13E-06 (Based on MOR03)
		Unit 2	1.10E-05	2.13E-06 (Based on MOR03)
MOR2010	9	Unit 1	8.91E-06	1.08E-07
		Unit 2	7.79E-06	1.11E-07
2010 Peer Review				
MOR2011	10	Unit 1	7.87E-06	5.67E-07
		Unit 2	7.86E-06	5.65E-07

MOR04 addresses findings and observations from 2001 Peer Review along with failure and unavailability data and success criteria update.

MOR05 updated HRAs and answered findings and observations from 2001 Peer Review to support MSPI.

MOR06 used the on-line EOOS-06 model as its starting point. The major enhancement associated with the EOOS-06 model was the incorporation of the ability to cross-tie service air between units. Changed MOV and AOV data from time based failure to demand based failure.

MOR07 changes included several items: (1) implemented a new diesel room heat analysis, additionally added two independent generators to support DC power; (2) removed safety bus initiators; (3) updated HRA events for post initiators and included consideration of improved battery life analysis; (4)added new initiating event for loss of intake structure; (5) update EDG data to 2007; (6) update LOOP to 2007; (7) air operated valve data update; (8) incorporated a conversion from calendar years to reactor years per the PRA standard; (9) added additional water source for circulation water pump seals; and (10) provided logic to credit swing turbine building closed cooling water system for both units.

MOR08 involved improved logic for the instrument air system modification and improved anticipated transient without scram (ATWS) logic, and resolved most of the issues from the 2007 Self Assessment.

MOR2010 incorporates updates to the common cause failure analysis for batteries and other equipment with major revisions to meet Regulatory Guide 1.200 improvements, including revisions to accident sequences, data and initiating event updates, and plant configuration

through 2009. The remaining items identified in 2007 Regulatory Guide 1.200 Self Assessment were also resolved.

MOR2011 incorporated completely revised internal flooding and new high wind analysis to meet Regulatory Guide 1.200, and incorporated model changes for multiple spurious logic for NFPA 805 and fire PRA.

Table D-1 of the September 2010 NRC memorandum (i.e., ML12335A421, *Safety/Risk Assessment Results For Generic Issue 199, "Implications Of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants"*) presents the safety/risk assessment results for Generic Issue 199 based on the 2008 United States Geological Survey (USGS) Seismic Hazard Curves. In particular, a point estimate for seismic CDF was developed by integrating the mean seismic hazard curve and the mean plant level fragility curve for each nuclear power plant. However, lacking a realistic plant-level fragility curve for BSEP, that seismic CDF was conservatively based on the high confidence, low probability of failure (HCLPF). By contrast, Table W-1 of the LAR presented seismic CDF and LERF developed from a seismic margins analysis using a 0.16 g earthquake, rather than a seismic PRA, since the Individual Plant Examination of External Events (IPEEE) used a seismic margins analysis. Although these values do not reflect the USGS information considered for Generic Issue 199, the seismic CDF and LERF, as presented in Table W-1 of the LAR, are considered the best available information for BSEP.

PRA RAI 17

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009 describes when changes to a PRA should be characterized as a "PRA upgrade." Identify any such changes made to the internal events or FPRA subsequent to your most recent full-scope peer review. Also, address the following:

a) If any changes are characterized as a PRA upgrade, indicate if a focused-scope peer review was performed for these changes consistent with the guidance in ASME/ANS-RA-Sa-2009, and describe any findings and their resolution.

b) If a focused-scope peer review has not been performed for changes characterized as a PRA upgrade, describe what actions will be implemented to comply with the ASME/ANS standard.

Response

No change made to either the internal events PRA or the FPRA since the last respective full scope peer review is considered to be characteristic of an upgrade as described in ASME/ANS-RA-Sa-2009, Section 1-A.

PRA RAI 18A

Please clarify the following dispositions to internal events PRA F&Os identified in Attachment U of the LAR that have the potential to impact the FPRA results and do not appear to be fully resolved:

a) F&O 6-8 against SC-C2 (Cat I/II/III):

   Identify software codes other than MAAP that were used to establish success criteria (e.g., GOTHIC), and describe any limitations of these codes to support success criteria used in the PRA.

Response

As described in BNP-PSA-070, *PSA Related GOTHIC Thermal/Hydraulic Codes for BNP*, GOTHIC was used in room heat-up analyses for the PRA model. Section 6.3.3 of Attachment 3 of BNP-PSA-070 evaluated GOTHIC action items, or potential errors, reported to the user community since the release of GOTHIC 7.2b (QA) and concluded that none of the outstanding reported errors impacts or affects the results of the analyses.

PRA RAI 18B

b) F&O 4-5 against SY-A13 (Cat I/II/III):

   This F&O states that failure of feedwater check valve F032A or F032B can lead to flow diversion that defeats HPCI or RCIC. The disposition to this F&O argues that these valves each have a failure probability two orders of magnitude lower than other HPCI or RCIC failures and therefore do not need to be modeled. Given that check valves fail at approximately 2E-4/demand, it is not clear why these failures can be dismissed per guidance in SR SY-A15. Provide further justification for dismissing these failures.

Response

The postulated HPCI/RCIC flow diversion path is through either B21-F032A or B21-F032B then through both FW-V10 and FW-FV-177 to the condenser. F032A and F032B are normally-open stop check valves, while FW-V10 is a normally-closed, motor-operated valve and FW-FV-177 is a normally-closed air-operated valve. As listed in Table 5 of BNP-PSA-004, *PSA Model Appendix B Component Failure Database*, the nominal fail-to-close probability of a check valve is approximately 1E-4/demand, while the nominal fail-to-remain-closed rates for an MOV and AOV are approximately 5E-8/hour and 2E-7/hr, respectively. Over the 24-hour mission time, the probability of some combination of these valves resulting in the postulated flow diversion is about 1E-15.

PRA RAI 18C

c) F&O 3-3 against HR-E3 (Cat I): F&O 3-4 against HR-E4 (Cat I):

Annex E4 of BNP-PSA-034 (Human Reliability Analysis) presents an "Operator Interview Worksheet" form and an "engineering review," but no operator interview results. Describe how and where interviews with plant operators and training staff for the purpose of confirming procedure interpretation in support of the PRA modeling are documented. Likewise, describe where and how talk-throughs with plant operators or simulator observations for the purpose of confirming the response models for the scenarios modeled in the PRA are documented. If these interview or talk-throughs do not exist as part of the FPRA documentation, provide the interview and talk-through results.

Response

Procedure talk-throughs were included as part of the operator interviews for which Annex E4 to Attachment 1 of BNP-PSA-034, *PRA Model Appendix E Human Reliability Analysis*, was used as both an outline of major topic points and a convenient form for capturing specific information. After the interviews, the collected information was then added to the respective fields of the appropriate sections (e.g., Scenario Description, Key Assumptions, Procedure and Training, Operator Interview Insights) of the HRA Calculator to document the analysis of the HFE. The electronic file for the HRA Calculator is documented as an output of BNP-PSA-034. The conduct of simulator observations and the subsequent assessment of the data are documented in Section E1.2 of Annex E1 to Attachment 1 of BNP-PSA-034.

PRA RAI 18D

d) F&O 2-3 against HR-12 (Cat I/II/III):

Describe the Human Failure Events (HFEs) screening process. Explain how HFEs that were screened out of the internal events PRA but could impact FPRA results were evaluated.

Response

HFEs were not screened out in the internal events PRA. F&O 2-3 concerns a documentation issue related to the assignment of a relatively high screening HEP for use in identifying those HFEs to be subjected to more detailed analysis based on their resultant risk significance. However as described in Attachment 3 of BNP-PSA-034, *PRA Model Appendix E Human Reliability Analysis*, even those HFEs which were not identified for more detailed analysis were retained in the model in case they should become more important for alternate plant configurations or for specific initiating events such as fire.

PRA RAI 18E

e) F&O 2-2 against DA-C8 (Cat I):

The F&O states that plant specific data concerning standby time is not collected and used in the PRA. Explain how the requirement to determine component standby time

(i.e., DA-C8) using operational records is met. Alternatively, justify why meeting this requirement at Capability Category II is not needed.

Response

As described in Section B.2.3 of BNP-PSA-004, *PSA Model Appendix B Component Failure Database*, in response to F&O 2-2, a review of plant specific operating data over a period of several years showed the standby times to be sufficiently balanced among multiple trains of a system to be appropriately represented by the approximate split of 50% for a two-train system or 33% for a three-train system.

PRA RAI 18F

f) F&O 6-12 against LE-G5 (Not Met):

It is not clear what was done to resolve this F&O. Characterization of LERF uncertainty is presented in BNP-PSA-075, but limitations in the LERF analysis do not appear to be provided in this document or elsewhere. Clarify what the specific limitations in the LERF analysis are for this application.

Response

Since the FPRA is based on the internal events PRA, the LERF analyses have some common limitations related to timing and containment damage. As noted in Section 6.7 of BNP-PSA-049, *PRA Model Sections 7-9 Level 2 Analysis*, minimal or no credit is provided for the recovery of failed equipment and loss of adequate injection at the time of containment failure due to the limited information on human response and equipment capability under such adverse conditions. Additional limitations which result from conservative modeling assumptions are specific to the FPRA and are common between the CDF analysis and the LERF analysis. Feedwater, Condensate, Circulating Water (i.e., Condenser), and Turbine Control are assumed failed for all fire scenarios because the related cables were not routed, as described in Section 3.1.3.42 of BNP-PSA-080, *BNP Fire PRA – Quantification*. As also described in Section 3.2.11 of BNP-PSA-080, the truncation limit of 1E-10/yr for LERF was on the edge of what is practical for the FPRA using a ONES solution, because of excessive quantification time and additional runs that the quantification engine could not solve.

PRA RAI 18G

g) F&O 1-22 against IFSO-A4 thru IFQU-B2 (Many SRs are Not Met):

For nearly all internal flooding findings presented in Attachment U of the LAR, the dispositions state that internal flooding can have no impact on the FPRA. A number of scenarios listed in Tables W-2-1 and W-2-2 of the LAR supplement result in LOCAs. In general, spurious actuations have the potential to cause internal flooding. Clarify whether any fire event can result in internal flooding. If flooding can occur as a result of a fire event, then further justify why these F&Os and other internal flooding F&Os can have no impact on fire CDF, LERF, ΔCDF, and ΔLERF.

Response

As described in Attachment 3 to BNP-PSA-085, *BNP Fire PRA - Component Selection*, the MSO review resulted in the addition to the FPRA of some relatively small Interfacing Systems Loss of Coolant Accident (ISLOCA) scenarios to radwaste or the condenser. The potential damage due to flooding was considered, but no additional risk due to internal flooding was identified to result from any fire event. Consistent with Section F.0 of Attachment 2 to BNP-PSA-035, *PSA Model Appendix F Internal Flooding Analysis,* had there been any risk resulting from fire-induced internal flooding, it would be quantified by the FPRA, rather than in the internal flooding model, and attributed to fire because fire was the initiating event.

PRA RAI 18H

h) F&O 6-16 against IFSN-A6 (not Met) and F&O 1-33 against IFQU-A9 (Not Met)

   Since spurious actuations have the potential to cause spray effects, clarify whether any fire event can result in spray effects impacting components modeled in the PRA. If so, justify why these F&Os can have no impact on fire CDF, LERF, ΔCDF, and ΔLERF.

Response

Spray effects due to the inadvertent actuation of the fire protection system were evaluated in BNP-PSA-035, *PSA Model Appendix F Internal Flooding Analysis,* but the fire-induced actuation of the fire protection system would not be considered "spurious." Otherwise, no additional risk due to spray effects was identified to result from any fire event. Consistent with Section F.0 of Attachment 2 to BNP-PSA-035, had there been any risk resulting from fire-induced spray effects, it would be quantified by the FPRA, rather than in the internal flooding model, and attributed to fire because fire was the initiating event.