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U.S. Nuclear Regulatory Commission
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Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding License Amendment Request for Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program

References:

1. Letter from William R. Gideon (Duke Energy) to U.S. Nuclear Regulatory Commission, *Application For Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program*, dated December 21, 2015, ADAMS Accession Number ML16004A249

2. NRC E-mail Capture, *Brunswick Unit 1 and Unit 2 Request for Additional Information related to LAR to Relocation of Specific Surveillance Frequency Requirements to Licensee Controlled Program (CAC NOS. MF7206 and MF7207)*, dated June 15, 2016, ADAMS Accession Number ML16167A174

Ladies and Gentlemen:

By letter dated December 21, 2015 (i.e., Reference 1), Duke Energy Progress, Inc., submitted a license amendment request (LAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendment would modify the Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." Additionally, the change would add a new program, the Surveillance Frequency Control Program, to TS Section 5.5, "Programs and Manuals." The changes are consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," Revision 3.

On June 15, 2016, by electronic mail (i.e., Reference 2), the NRC provided a request for additional information (RAI) regarding the LAR. The proposed questions were discussed by telephone with the NRC on June 13, 2016. It was agreed that a response would be provided within 30 days of receipt of the RAI, except for Question 4.c to which a response will be

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provided within 60 days. Duke Energy's 30-day responses are provided in Enclosure 1 of this letter.

A new regulatory commitment, as described in Enclosure 2, is contained in this letter.

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 13, 2016.

Sincerely,

Karl Moser for

William R. Gideon

MAT/mat

Enclosures:

1. Response to Request for Additional Information
2. List of Regulatory Commitments

cc (with enclosures):

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Response to Request for Additional information

By letter dated December 21, 2015 (i.e., Reference 1), Duke Energy Progress, Inc., submitted a license amendment request (LAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendment would modify the Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." Additionally, the change would add a new program, the Surveillance Frequency Control Program, to TS Section 5.5, "Programs and Manuals." The changes are consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," Revision 3.

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NRC RAI 1:

The following requests for additional information apply to the internal events Facts and Observations (F&Os) reported in Tables 1 and 2 of Enclosure 2 to the License Amendment Request (LAR):

a. Open F&O 3-12 related to supporting requirement (SR) LE-C3 states that the process for identification of recovery and repair actions that can terminate or mitigate the progression of a severe accident was incorporated into the original analysis, rather than performing a review of significant accident progression sequences and then incorporating repair, as required by the SR. Please clarify whether any credit for repair is taken. If any credit for repair is taken, please justify how this credit for repair is consistent with SR LE-C3 at Capability Category II.

b. Open F&O 3-12 related to SR LE-C10 and LE-C12 was entered because the peer review team could not find any evidence that significant accident sequences were reviewed to determine if engineering analyses could support continued equipment operation or operator actions that could reduce Large Early Release Frequency (LERF). Please clarify whether credit for equipment survivability or human actions in adverse environments is taken. If any credit is taken, please justify how it satisfies SR LE-C10 and LE-C12 at Capability Category II.

c. Open F&O 3-13 related to SR LE-C13 indicates that "scrubbing by the reactor building is treated in a conservative method." The resolution to this F&O indicates that Capability Category I of the SR is met, which implies that no credit for scrubbing is taken in a containment bypass event. Please confirm whether credit for scrubbing is taken. If any credit for scrubbing is taken, please justify how it satisfies SR LE-13 at Capability Category II.

d. Open F&O 6-8 related to SR SC-C2 was entered because there was no documentation of computer codes limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) as required by the SR. Please identify computer codes that were used to establish PRA success criteria and describe any limitations of these codes to support PRA success criteria.

e. F&O 1-3 related to SR IE-C1 was entered because the peer review team found that initiating event frequencies for pipe breaks outside containment were a factor of 100 to 1000 too low. In the resolution to this F&O the licensee stated that, as a result of an analysis performed using updated pipe break frequencies, Main Steam Line Break (MSLB) should be included as an initiator. Please confirm that MSLB has been included in the internal events initiators or otherwise justify its exclusion.

f. F&O 6-12 related to SR LE-G5 was entered because the peer review team could not find sufficient documentation on the limitations of the LERF analysis that could impact different applications, as required by the SR. Please identify the specific limitations in the LERF analysis for this application.

Response:

a. The model does include two operator recovery actions for cases of instrumentation or control problems specifically for LERF consequences:

   - **Operators fail to restore reactor pressure vessel (RPV) injection.** If the RPV injection systems were unavailable to prevent core damage, the model allows for an operating crew to restore a system to operation within the short window available between core damage and vessel breach. The failure probability for system recovery is assessed to be high (i.e., a value of 0.91 is assigned) because less than one hour is credited. If failed systems have not been repaired/restored within an hour, it is judged unlikely that the system(s) will be restored before RPV melt-through. System restoration is conservatively assessed because it is considered to be more involved than simply bypassing a system trip from the control room.

   - **Operators fail to recover low pressure systems for accidents IA and IIIA.** Following failure to recover injection systems to prevent vessel breach, a small time window is available (i.e., approximately 2.5 to 3 hours) for an operating crew to restore a system to operation before releases to the environment could result in LERF. A value of 0.84 is assigned for the conditional probability that recovery is not successful in the short time window. Recovery is not considered possible for accidents in which the primary containment has already failed.

   The probabilities for these actions are based on an exponential model that uses the overall time to failure and mean time to repair as inputs:

$$P_f = e^{-\lambda t}$$

where:

P_f = Final basic event probability,
λ = 1/12 hours characteristic of an instrument or control problem that can be remedied, and
t = Overall time to failure.

In this case lambda is equal to 1/MTTR (i.e., mean time to repair), where MTTR for instrumentation is 6 hours. In order to bound the uncertainty in the data, a MTTR of 12 hours was used. The time to failure is scenario-specific and is developed from thermal-hydraulic analyses. Plant conditions were reviewed to ensure it is feasible to perform the action based on the scenario. No additional credit is taken for repair of equipment that can terminate or mitigate the progression of a severe accident.

b. No credit for equipment survivability or human actions in adverse environments is taken that would satisfy SR LE-C10 or LE-C12.

c. No credit is taken for scrubbing in the reactor building.

d. Success criteria for the BSEP PRA were determined by using the Modular Accident Analysis Program, Version 4 (MAAP4). The PRA Groundrules and Assumptions Notebook documents the software limitations and states that success criteria are defined with sufficient margin to allow for: the limitations of the computer code, the sophistication of the models, and the uncertainties in the results. Specific code limitations include:

- MAAP4 uses a simple node breakage model. The sensitivity of the results to break size and location should be assessed for all loss-of-coolant accident (LOCA) sequences.
- Because the run of pipe between the primary system and the discharge location outside of containment will be substantial, the area of the break and the discharge coefficient should be determined for interfacing system LOCAs to represent the equivalent length of the pipe.
- MAAP4 inadequately models the early portion of a large break LOCA (LBLOCA) with reflood. MAAP4 should not be used to derive minimum injection requirements for LBLOCAs during the period before the initial recovery of the core (i.e., reflooding) has been completed.
- MAAP4 uses a simple containment model. MAAP4 should not be used to determine the detailed containment response (e.g., the time to reach a containment pressure set point) during the period before recovery of the core because the code will tend to under-predict the mass and energy release to the containment from the primary system.
- MAAP4 cannot be used to determine the number of Safety-Relief Valves (SRVs) that are needed to respond to the transient following the closure of the Main Steam Isolation Valves (MSIVs). An assessment of the number of valves that is needed ties into the probability that not all of the valves will open.

- The extent of steam voiding due to injection and the resulting water displacement into the downcomer may not be fully represented during the early portion of a sequence (i.e., the first four minutes). The predicted water level response may be delayed due to the fact that the model for water circulation between the downcomer, core and core bypass are simplified, with one node for the combined downcomer, jet pump and recirculation pump water and one node for the combined core and core bypass water.
- MAAP4 cannot model reactivity and cannot be used for restart.
- MAAP4 cannot be used for finite element core failure. MAAP4 models only macro core damage.

In order to properly treat the limitations of MAAP4 when developing success criteria the following methodologies were used:

- Core damage will be assumed if one of the following occurs: The hottest core node temperature exceeds 1800 degrees F as defined by a best-estimate thermal-hydraulic calculation using MAAP, or the collapsed liquid water level in the downcomer falls below the lower third of the active core. These assumptions provide sufficient margin to ensure that limitations in the code are accounted for to provide a realistic estimation of core damage.
- In areas of large break LOCA and Anticipated Transient Without Scram (ATWS), the design basis analyses are used.

For specific cases where additional analysis is needed for room heat-up calculations, GOTHIC analyses were performed. These calculations were used to determine whether or not it was necessary to model room cooling in various areas of the plant following severe accidents. The limitations of the GOTHIC code and models are documented within the individual GOTHIC calculations.

e. The MSLB initiator is not included in the internal events initiators for the BSEP model of record. The MSLB is assessed as a LOCA outside of containment with an updated pipe break frequency calculated from the EPRI *Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments* methodology. The updated frequency analysis replaced the old, segment-based methodology for estimating pipe break frequencies. In addition, a conservative assessment of a MSLB outside containment that causes a high energy line break (HELB)-induced loss of offsite power (LOOP) demonstrated that the risk increase due to inclusion of the MSLB initiator is on the order of 1E-10/year. Based on this value, it was justified to track the potential model change for update at a later date based on insignificant contribution to risk relative to the base Core Damage Frequency (CDF) and LERF values.

f. A summary of model uncertainties and limitations of the LERF analysis is presented in Table 1. The uncertainty risk impact is a qualitative assessment of the LERF contribution for the identified system and the relative risk associated with the uncertainty/limitation. No limitations were identified that would impact the BSEP Surveillance Frequency Control Program (SFCP) as changes in surveillance test intervals do not affect the LERF phenomena identified by these limitations.

Table 1: LERF Limitations and Uncertainties

Item No.	Description	Basis	Uncertainty Risk Impact to the SFCP
1	Containment isolation Signal failure probability not modeled in detail.	The 1 E-5/demand estimate is based on judgment and considers the multiple signal failures that must occur.	Low
2	A Single Control Rod Drive (CRD) pump is assumed inadequate to cool the debris in the bottom head. CRD is not considered a viable system for averting vessel melt through.	Even with enhanced flow (starting a second CRD pump), successful debris coolability is marginal.	Low
3	Drywell failure occurring despite the presence of water is omitted as a failure mechanism for BSEP.	Generic estimate considered too conservative due to the BSEP containment design differences and 1,000 gpm of water is sufficient to quench core debris.	Low
4	Potential scrubbing of the releases due to flooding containment utilizing drywell sprays is not credited.	Conservative modeling decision.	Low
5	Contingency methods for containment heat removal (e.g., spent fuel pool heat exchangers) are not credited.	There is limited heat removal capability and lack of procedural guidance.	Low
6	Loss of adequate in-vessel injection coupled with inadequate containment heat removal results in moderately high temperatures and also conditional probability of leakage versus rupture.	The temperature failure regime is "intermediate."	Low
7	The size of containment failures may vary over the spectrum of accident sequences.	The characterization of failure sizes of negligible, small, and large are adequate to encompass the major effects on pressure, temperature, and radionuclide release.	Low

Table 1: LERF Limitations and Uncertainties

Item No.	Description	Basis	Uncertainty Risk Impact to the SFCP
8	All ATWS sequences are assumed to result in large containment failures with a probability of 1.0.	The rate of pressure increase in these cases has been shown in industry studies to be beyond containment heat removal and venting capabilities [C-1]. The energy deposition to the containment and the resultant pressure rise is greater than that which can be relieved by a "small" hole (~ > 16" dia.); therefore, the break in these cases is always classified as large.	Moderate
9	For unmitigated ATWS sequences, the primary cause of containment failure is judged to be torus dynamic loads.	Containment failure is found to be in the torus a large percentage of time.	Moderate

NRC RAI 2

Most of the F&Os related to internal flooding reported in Table 1 of Enclosure 2 to the LAR arose from the 2010 peer review when the internal flooding model or documentation was incomplete. As a result, in most instances, the review team was unable to review the supporting requirements. This is summarized in F&O 1-31 (related to many documentation SRs such as IFSN-B1), which states that "documentation of internal flooding was not conducive to supporting PRA applications, maintenance and upgrade, or peer review." The peer review team identified specific examples of areas where the models or documentation were incomplete, such as:

- The flood scenario development documentation did not provide all of the information needed to fully describe the scenario development process.
- Various items noted in the F&Os related to the IFSN-A SRs, such as:
  - no documentation of flood alarms, blowout panels or HVAC dampers (IFSN A2)
  - no identification of automatic flood isolation features or operator indications (IFSN-A3)
  - no documentation of qualitative assessment of pipe whip, jet impingement, humidity, condensation and temperature concerns in the flooding analysis (IFSN-A6)
  - no discussion of drain paths as possible propagation paths and propagation through wall penetrations, cable trays and HVAC ducts were not considered (IFSN-A8)
  - flood sources screened without discussion of indication or isolation (IFSN-A14)

- The following items listed in SR IFSN-B2 were not included in the documentation:
  - assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability
  - screening criteria used in the analysis
  - flooding scenarios considered, screened, and retained
  - description of how the internal event analysis models were modified to model these remaining internal flood scenarios
- A listing of the specific components assumed to be failed in each flood area was not provided.
- Documentation of risk-important components within each flood area appeared incomplete in the walk-down notes.
- Documentation of characteristics of flood pathways between zones was weak and/or incomplete.
- Documentation and discussion of flood pathways other than doors was limited.
- Assumptions on door behavior, such as pressures doors would withstand or flood propagation rates through door gaps, were not documented.
- Operator actions to be modeled in the Human Reliability Analyses (HRA) for flooding were not clearly identified and described, including documentation of alarms that would be considered for specific flooding scenarios.
- Review of plant operating experience pertinent to these HRA analyses were not documented.
- It was not possible to verify proper development of the HRA Calculator Files, as discussed under IFQU-A5.
- The flooding analysis did not provide information concerning modeling of system/component failures due to pipe failures in that system, as opposed to failure due to flooding and flood propagation, as discussed in IFQU-A1.

The licensee's dispositions to the internal flooding F&Os indicate that they have been addressed, as documented in various plant-specific reports. However, there is little discussion of how these resolutions were done other than reference to specific reports, unavailable for review. The NRC staff needs the result of either one of the two analysis to complete our review of this application:

a. Demonstrate quantitatively that the contribution to Core Damage Frequency (CDF) and LERF from internal flooding scenarios in which basic events potentially affected by the requested changes in surveillance frequency is negligible, or

b. Conduct a focused-scope peer review for the internal flooding technical elements based on the enhancements made since the 2010 peer review and adequately disposition any remaining or new F&Os.

Response:

Duke Energy will conduct a focused-scope peer review to close the resolved F&Os in accordance with Regulatory Guide 1.200, Revision 2, for the internal flooding technical elements based on the enhancements made since the 2010 peer review.

NRC RAI 3:

In the license amendment for transition to the National Fire Protection Association (NFPA) Standard 805 dated January 28, 2015 (ADAMS Accession Number ML14310A808), license condition # 2 requires the implementation of all plant modifications "by the startup of the second refueling outage for each unit after issuance of the [NFPA-805] safety evaluation". Since the NFPA-805 implementation period has not ended, please address how the fire PRA that will be used for the surveillance frequency control program (SFCP) reflects the as-built, as-operated plant.

Response:

Both risk-related plant modifications, as listed in Table S-1 of the NFPA 805 LAR, have been completed and are being addressed through the normal PRA maintenance process as described in the response to Fire PRA RAI 24.01 (i.e., ADAMS Accession Number ML14191A672). As described in Section 2 of Enclosure 2 to the LAR, Duke Energy procedures provide the guidance, requirements, and processes for maintenance, update, and upgrade of the PRA. For the SFCP, no other plant modification required by license condition #2 has any risk impact.

NRC RAI 4:

The following RAIs apply to the internal fire F&Os reported in Table 3 of Enclosure 2 to the LAR:

a. F&Os 1-19 and 1-20 related to SR FSS-A1 identified that transient fires in the battery room were not developed where the transient damages or ignites the batteries (e.g., was the fire ignition frequency increased by the transient fire frequency in addition to the fire frequency of the batteries themselves?). Please justify the exclusion of batteries as transient fire targets or perform a sensitivity evaluation incorporating the effect of treating batteries as transient fire targets. Please assess the impact on the SFCP.

b. Resolution to F&O 1-24 related to SR FSS-B1, HRA-A2, HRA-C1 and HRA-D1 states that possible conservatism associated with not modeling other ASSD actions are not considered to be significant. Please justify why not modeling other ASSD actions are not considered to be significant for the SFCP.

c. Resolution to F&O 1-36 related to SR QU-B2, QU-F2, QU-B3, FQ-B1 and FQ-F1 states that effective truncation values of 1E-09/yr for CDF and 1E-10/yr for LERF are used for scenario quantification. It further states that the process for establishing truncation limits does not demonstrate that the overall model results converge and SR QU-B3 is assessed as 'Not Met'. Please provide the results from a sensitivity analysis that expands the truncation levels to those comparable for internal events, typically at least 1E-12 for CDF, as the necessary justification why the chosen truncation levels have no impact on the SFCP.

d. F&O 2-10 related to SR QU-E4, UNC-A1 and PRM-B10 is concerned with PRA items that were assumed failed for the component selection. The resolution to this F&O states that the assumption of items considered always failed in the fire PRA represents a conservatism in the calculated fire CDF. Assuming a component failed may be non-conservative for the SFCP, as it could underestimate the potential increase in CDF and, as a result, underestimate the surveillance frequency for that component. Please confirm that this "always failed" assumption cannot lead to underestimating the surveillance frequency for any component.

e. F&O 2-19 related to SR FQ-A1 identified that some non-conservatism could exist for some Hot Gas Layer (HGL) scenarios (whole room burnout) where the mapping tables do not cover all the components that are affected by the fire-induced failures. Please confirm that after the resolution of this F&O, which modified the mapping tables, all the components affected by the fire-induced failures are failed for both the HGL scenarios and the individual scenarios.

f. F&O 2-20 related to SR FQ-A1 identified an assumption that cable trays with solid bottoms, which are present above some transformers in the Diesel Generator Basement (sources 5010 through 5017), will prevent damage to cables for ignition sources with heat release rates (HRRs) of 69 kW or less based on the discussion provided in section Q.2.2 of NUREG/CR-6850. As per Section Q.2.2 of NUREG/CR-6850, while solid bottom barriers for cable trays will delay damage to qualified cables, they do not preclude it. Therefore, to justify the judgment that "the spray shields installed above these transformers will prevent thermal damage to the target cable trays, thus damage resulting from a transformer fire need not be postulated." Please clarify the following:

   i. The basis for limiting the 98th percentile HRR for sources 5010 through 5017 to 69 kW, recognizing that fires from transient combustibles which may be present have a 98th percentile HRR of 317 kW (Note the potential relevance of Item (2), "Clarification for Transient Fires," in the June 21, 2012, letter from Joseph Giitter (NRC) to Biff Bradley (NEI), ADAMS Accession No. ML12171A583) and

   ii. The reason why the "test fire model run with FDS" cited by the Peer Review team as acceptably demonstrating that "cable trays with solid bottoms will prevent damage to cables for ignition sources with HRR 69 kW or less" is not cited as the basis for the judgment in the F&O resolution (alternatively, cite this "test fire model" as the basis in conjunction with the discussion for part [i] above).

g. F&O 3-6 related to SR SF-A3 noted that common cause failure of gaseous suppression systems is not addressed. The F&O resolution justifies that there is no potential for common cause failures between the CO2 system for Unit 1 and the CO2 system for Unit 2 because their supplies are separated by a large, open distance. Please confirm that the CO2 systems for the two units are widely separated with regards to both supply tanks and the connecting piping.

h. F&O 4-1 related to SR FSS-A1 identifies that the fire PRA uses a severity factor of 0.1 for cabinet breaching factor for well-sealed Motor Control Centers (MCCs). Resolution to this F&O and F&O 1-30 references Frequently Asked Question (FAQ) 14-0009, "Treatment of Well-Sealed Electrical Panels Greater than 440V," which assigns a breaching factor of 0.23

for well-sealed MCCs above 440 V (if not well-sealed, the breaching factor is always 1.0). Please confirm that the accepted value of 0.23 is used in the fire PRA, as applicable, or alternatively, confirm that the sensitivity studies on breaching factors, similar to those performed for the NFPA-805 application (summarized in response to RAI 1.g, ADAMS Accession No. ML 13205A016, where a factor of 1.0 was used) will be performed for the surveillance frequency calculations.

i. F&O 5-4 related to SR ES-A5, ES-A6, ES-B2, ES-D1 and PRM-B9 identified six fire-induced spurious events that were screened, but could cause a plant trip (or manual shutdown) and impact equipment that is credited for accident mitigation in the fire PRA. Resolution to this F&O justified screening of the cited six potential trip inducers by the fact that they do not introduce any scenarios not already modeled, but did not address impact on initiating event frequencies. Please provide either the results of a sensitivity analysis including these events or justify quantitatively why these potential trip inducers do not impact the initiating event frequencies.

j. F&O 5-8 related to SR IGN-A4 and IGN-B4 identified that the bases for excluding certain historical plant specific fire events should be strengthened to support the use of generic ignition frequency data. The F&O identifies a list of historical fires related to the heater drain pumps (items #1, 2, 5 and 7 in the peer review comments) that need more justification for their exclusion. The resolution to this F&O states that the plant documentation "was revised to include additional discussion of the plant history and corrective actions related to the heater drain pumps." Please clarify whether after resolution of this F&O the heater drain pump fire events are still excluded from potential plant-specific effects on ignition frequencies and provide the basis, or alternatively perform a sensitivity analysis to assess impact.

k. The resolution to F&O 5-16 related to SR LE-F1, LE-F2, LE-G3, UNC-A1, FQ-E1 and FQ-F1 implies that the noted unit asymmetry whereby 98.1% of the Unit 2 fire LERF is due to fires in the Unit 2 Main Control Room (MCR), vs. the approximate 60% contribution from the Unit 1 MCR to the Unit 1 fire LERF, is an accurate representation. Given that the ignition frequency for a Main Control Board fire is now six times higher (NUREG-2169) than that from Supplement 1 to NUREG/CR-6850 (and twice as high as that from the original NUREG/CR-6850), Please clarify, preferably quantitatively, the effect on the MCR fire contribution to the fire LERF at each unit and the status of the asymmetry and whether it remains an accurate representation.

l. Resolution to F&O 6-1 related to SR CS-B1 states that three raceways in the Unit 2 electrical equipment room that could not be routed are identified as a source of uncertainty. It further states that the risk associated with the assumed failure of these raceways is "qualitatively addressed as a non-conservative assumption [...] that is likely mitigated in the HGL scenarios by other failures for the respective power supplies." Please assess the impact of this assumption on the SFCP to confirm it is conservative.

m. F&O 6-7 related to SR CF-A1 identifies that cable failure probabilities are based on the Chapter 10 tables in NUREG/CR-6850. Updated failure probabilities were published in NUREG/CR-7150. Please confirm that the NUREG/CR-7150 cable failure probabilities will be used in the surveillance frequency calculations, and that these probabilities will be applied to both the base and the adjusted PRA when calculating change in CDF/LERF for the SFCP.

Response:

a. As described in the response to Fire PRA RAI 4 (i.e., ADAMS Accession Number ML13205A016), transient fires were postulated consistent with the guidance in NUREG/CR-6850. Although NUREG/CR-6850 identifies batteries as ignition sources, there is no industry guidance identifying batteries as targets and establishing a damage criteria or ignition criteria from which to determine the applicable Zones of Influence (ZOIs) for a transient fire. As provided in Section H.2 of NUREG/CR-6850, fire vulnerability for major components is assumed to be limited by the vulnerability of power, control, and/or instrument cables supporting the component. Section H.2 also provides guidance for the treatment of sensitive electronics, but batteries are not included as sensitive electronics in the final version of FAQ 13-0004 (i.e., ADAMS Accession Number ML13322A085).

   Since Operating Experience does not support a trend of transient fires damaging or igniting batteries, this treatment promotes realistic risk assessments in support of the SFCP.

b. The fire-specific procedures were systematically reviewed to identify both beneficial and detrimental actions which could be potentially relevant to the Fire PRA. Where the beneficial actions were not modeled, the potential credit was considered small. The detrimental actions (e.g., pre-emptive removal of power to equipment) were not modeled but were eliminated from the procedures or conditioned on indications of abnormal equipment performance in response to the fire. Since the Fire PRA already fails equipment whose performance is impacted by fire, modeling those detrimental actions is unnecessary. With regard to the SFCP, Alternate Safe Shutdown (ASSD) actions that are not modeled would be likewise not modeled in both the base and the adjusted PRA.

c. It was agreed that a response would be provided within 30 days of receipt of the RAI, except for Question 4.c to which a response will be provided within 60 days.

d. For clarity, please note that the "always failed" assumption was for the Fire PRA and does not apply to other hazards. In Section 1.0 of Enclosure 2 of the SFCP LAR (i.e., ADAMS Accession Number ML16004A249), BSEP indicated that the SFCP would follow the methodology provided in NEI 04-10. Step 8 of the SFCP change process described in Section 4.0 of NEI 04-10 requires an assessment of whether the surveillance test interval (STI) change can be adequately characterized by the PRA. Where the STI change is determined to be adequately characterized, the "always failed" assumption would not lead to underestimating the surveillance frequency because, to the extent that the SFCP change process requires quantification of the Fire PRA, the "always failed" assumption would be applied to the same components in both the base and the adjusted PRA. Where the STI change is determined <u>not</u> to be adequately characterized, Step 10 of the SFCP change process provides guidance for performing qualitative or bounding risk analysis, with a detailed account of how to approach fire events.

e. F&O 2-19 describes a problem with a mapping table where an individual ignition source in a room would appear to have more affected components than the number of components affected by a burnout of the whole room. The resolution of F&O 2-19 limited the components affected by an individual ignition source to those that could actually be affected by a fire in the room. Consequently, the HGL scenarios and the individual scenarios have fire-induced failures appropriate for each particular scenario.

f. i. Because ignition sources 5010 through 5017 are dry transformers (i.e., Bin 23b) rather than transients (i.e., Bin 25), the 98th percentile heat release rate (HRR) of 69kW was appropriately assigned consistent with the guidance in NUREG/CR-6850. Table 11-1 of NUREG/CR-6850 recommends assuming that a dry transformer HRR is equivalent to that of electric motors, which Table E-1 identifies at the 98th percentile HRR as 69kW.

Transient ignition sources are located as described in the response to Fire PRA RAI 4 (i.e., ADAMS Accession Number ML13205A016) and assigned HRRs as described in the response to Fire PRA RAI 1.d (i.e., ADAMS Accession Number ML13205A016), as supplemented by the reviews described in the response to Fire PRA RAI 1.d.01 (i.e., ADAMS Accession Number ML14079A233) and clarified in the response to the Fire PRA RAI 1.d.02 (i.e., ADAMS Accession Number ML14191A672).

ii. The test fire model run with Fire Dynamic Simulator (FDS) was an ad-hoc demonstration that was shared with the Peer Review team to support the treatment of the spray shields but was not retained in plant records. It is not cited in the response to F&O 2-20 because the Peer Review team wanted better documentation. A formal, rigorous FDS analysis of the spray shields resulting in the same conclusions was performed and documented in the PRA notebook.

g. The piping and supply cylinders for the Unit 1 CO_2 system are widely separated from the piping and supply cylinders for the Unit 2 CO_2 system. For Unit 1, the CO_2 cylinders are in the bottle storage area behind the Unit 1 Reactor Building and the piping runs to the Unit 1 High Pressure Coolant Injection (HPCI) room inside the Unit 1 Reactor Building. For Unit 2, the CO_2 cylinders are in the bottle storage area behind the Unit 2 Reactor Building and the piping runs to the Unit 2 HPCI room inside the Unit 2 Reactor Building. The bottle storage area for Unit 1 is separated from the bottle storage area for Unit 2 by more than the length of the Radwaste Building.

h. The Fire PRA does not currently use a breaching factor of 0.23 for well-sealed Motor Control Centers (MCCs). However, the update to the final version of FAQ 14-0009 is being addressed through the normal PRA maintenance process as described in the response to Fire PRA RAI 24.01 (i.e., ADAMS Accession Number ML14191A672). As described in Section 2 of Enclosure 2 to the LAR, Duke Energy procedures provide the guidance, requirements, and processes for maintenance, update, and upgrade of the PRA.

In Section 1.0 of Enclosure 2 of the SFCP LAR (i.e., ADAMS Accession Number ML16004A249), BSEP indicated that the SFCP would follow the methodology provided in NEI 04-10. Step 5 of the SFCP change process described in Section 4.0 of NEI 04-10 includes "Gaps" to Capability Category II requirements as inputs to identifying appropriate sensitivity cases in Step 14. Consequently, until the accepted breaching factor is updated in the Fire PRA, the appropriate breaching factor to use in the sensitivity study would be 0.23.

i. In the Fire PRA, the initiating event is the fire, and the initiating event frequencies are the fire ignition frequencies. The scenario for progression through the accident sequences are defined by the fire-affected equipment and the plant responses. The potential trip inducers were screened from further evaluation because the applicable scenarios were already modeled. As described in the response to Fire PRA RAI 01.c (i.e., ADAMS Accession Number ML13205A016), a plant trip is postulated for every fire. The fire-affected equipment does not impact the frequency of the fire.

j. After resolution of F&O 5-8, the heater drain pump fire events were still excluded from potential plant-specific effects on ignition frequency, consistent with the guidance in Section 6.5.2 of NUREG/CR-6850 for effective correction of a common cause. The resolution found:

   In late 2000, the root cause investigation for NCR 24699 identified the previous instances and determined the cause to be the stator windings shorted turn-to-turn due to thermal aging of the insulation system. This normal aging was accelerated by the extremely dirty condition of the windings. The dirt clogged the air passages around the coils and through the stator core vent ducts, preventing airflow. The corrective actions included replacement of the motors, modifications to provide additional cooling/air flow, and better maintenance practices. Since there has been no subsequent failure in 11+ years [as of February 2012], the cause of these previous fire events is considered corrected.

k. The current Fire PRA does not use NUREG-2169 ignition frequencies. However, the update to those NUREG-2169 ignition frequencies is being addressed through the normal PRA maintenance process as described in the response to Fire PRA RAI 24.01 (i.e., ADAMS Accession Number ML14191A672). Also included are offsetting changes like the lower Heat Release Rates from NUREG-2178. Without speculating on the net effect on the MCR contribution to LERF, the updated Fire PRA is expected to remain an accurate representation within acceptable limits of uncertainty. As described in Section 2 of Enclosure 2 to the LAR, Duke Energy procedures provide the guidance, requirements, and processes for maintenance, update, and upgrade of the PRA.

   Currently, the Main Control Room (MCR) contributions to LERF are approximately 60% and approximately 72% for Unit 2 and Unit 1, respectively. The Main Control Board (MCB) contributions to LERF are approximately 14% for Unit 2 and approximately 13% for Unit 1 and are included in the respective stated MCR contributions. The response to Fire PRA RAI 06.02 (i.e., ADAMS Accession Number ML14191A672) explained how the Main Control Board (MCB) contributions to LERF reflect scenarios that are postulated to grow large enough to damage not only the entire panel but also external cables in trays/conduits above the panel. The response to Fire PRA RAI 13 (i.e., ADAMS Accession Number ML13220B041) described how asymmetries result from different cables being routed above different panels.

l. To the extent that the SFCP change process described in Section 4.0 of NEI 04-10 requires quantification of the Fire PRA, the Fire PRA would provide acceptable estimates of the risk impact of the proposed STI adjustment because the same fire-induced failures of the cables in raceways in the Unit 2 electrical equipment room would be applied to both the base and the adjusted PRA.

m. The current Fire PRA does not use NUREG-7150 cable failure probabilities. However, the update to the NUREG/CR-7150 cable failure probabilities is being addressed through the normal PRA maintenance process as described in the response to Fire PRA RAI 24.01 (i.e., ADAMS Accession Number ML14191A672). As described in Section 2 of Enclosure 2 to the LAR, Duke Energy procedures provide the guidance, requirements, and processes for maintenance, update, and upgrade of the PRA.

   To the extent that the Fire PRA will be used to calculate changes in CDF/LERF in support of the surveillance frequency calculations, these cable failure probabilities would be applied to both the base and the adjusted PRA.

NRC RAI 5:

The following RAIs apply to the high winds and external flooding F&Os reported in Table 4 of Enclosure 2 to the LAR:

a. F&O WPR-A5-02 related to SR WPR-A5 and WPR-A8 identifies two potential errors in utilizing the multiplier criteria for increasing the likelihood of human errors due to high winds. First, ex-control room actions that do not traverse through areas impacted by winds and performed in areas not impacted by winds do not need to be considered at a higher failure probability. Second, some ex-control room action may have a guaranteed failure. Please provide an explanation on how this F&O was addressed and, if not completely addressed, assess the impact on the SFCP.

b. F&O WPR-A11-01 related to SR WPR-A11 states that if system recoveries are credited in the model, then their potential to be impacted by the high winds conditions needs to be evaluated. Please confirm that all recoveries credited in the model were assessed for impact from the high wind conditions and summarize the assessment.

c. F&O WPR-B2-01 related to SR-WPR-B2 states that the uncertainties in each of the inputs and for all important dependencies and correlations have not been assessed, as required by the SR. Please confirm that the uncertainties due to all important dependencies and correlations have been assessed and documented and provide a summary of the assessment.

d. F&O WPR-C3-01 related to SR WPR-C3 was entered because the sources of uncertainty and assumptions were not identified as required by the SR. Please describe the sources of uncertainty and assumptions and assess their impact on the model and on the results for the SFCP.

Response:

a. For those operator actions to be performed inside category I buildings (e.g., control room, reactor and turbine buildings) where there would be no impact by high wind events, the operator failure probabilities are not changed as compared with the internal events Human Error Probabilities (HEPs).

   For actions that take place in unprotected areas, the HEP takes into account equipment failure or inaccessibility due to high wind events in conjunction with the time needed to perform the actions. As a result, for those actions taking place in unprotected areas during the first hour of the high wind event, the operator actions are assumed failed.

b. All recovery actions credited in the model were assessed for impact from the high wind conditions. The operator recovery actions that need to be performed outside Category I buildings during the first hour of high wind events are always failed, so the probabilities of those recovery actions are set to 1.0. For those recovery actions to be performed outside Category I buildings after the first hour of the event, the HEP is re-evaluated in conjunction with the adjusted equipment failure due to high wind impact. The operator recovery actions inside the Category I buildings are not impacted by high wind event and thus not changed. The availability of equipment to be recovered is driven by its fragility.

c. The uncertainties due to all important dependencies and correlations have been assessed and documented. Uncertainties in the exceedance probabilities for the high wind hazard curves and for equipment fragilities were assessed at the mean value and at the 5^{th} and 95^{th} percentiles. The uncertainty parameters for missile hit probabilities were also assessed at the mean value, 5^{th}, and 95^{th} percentiles. An error factor of 10 has been assigned to other basic events.

   Model uncertainties, and uncertainties due to system dependencies and correlations were assessed and documented as part of the peer reviewed internal events analysis. Since high wind events have no impact on the internal events model, the assessed uncertainties are unchanged. Similarly, the uncertainty due to dependencies and correlations among operator actions inside Category I buildings are the same as those of the peer reviewed internal events model. Recovery actions outside the protected buildings within the first hour are set to 1.0, so no dependencies are applicable. Recovery actions performed outside Category I buildings after the first hour of the event are re-evaluated in conjunction with the adjusted equipment failure due to high wind impact. The results of the uncertainty analyses are shown in the following tables for Units 1 and 2, CDF and LERF.

Table 1. BSEP U1 High Wind CDF Distribution Parameters			
	5% Conf.	mean	95% Conf.
Point Est		1.60E-05	
Mean	1.60E-05	1.63E-05	1.67E-05
5%	4.15E-06	4.29E-06	4.38E-06
Median	1.15E-05	1.17E-05	1.19E-05
95%	4.07E-05	4.23E-05	4.41E-05
StdDev		1.68E-05	
Skewness		7.47E+00	
Kurtosis		1.36E+02	

Table2 . BSEP U1 High Wind LERF Distribution Parameters			
	5% Conf.	mean	95% Conf.
Point Est		2.44E-07	
Mean	2.30E-07	2.35E-07	2.41E-07
5%	5.46E-08	5.61E-08	5.77E-08
Median	1.60E-07	1.62E-07	1.65E-07
95%	5.97E-07	6.22E-07	6.54E-07
StdDev		2.97E-07	
Skewness		1.10E+01	
Kurtosis		2.69E+02	

Table 3. BSEP U2 High Wind CDF Distribution Parameters			
	5% Conf.	mean	95% Conf.
Point Est		1.56E-05	
Mean	1.52E-05	1.55E-05	1.58E-05
5%	4.07E-06	4.18E-06	4.29E-06
Median	1.10E-05	1.13E-05	1.15E-05
95%	3.84E-05	4.01E-05	4.19E-05
StdDev		1.46E-05	
Skewness		4.44E+00	
Kurtosis		3.62E+01	

Table 4. BSEP U2 High Wind LERF Distribution Parameters			
	5% Conf.	mean	95% Conf.
Point Est		2.08E-07	
Mean	1.97E-07	2.02E-07	2.06E-07
5%	4.74E-08	4.87E-08	4.97E-08
Median	1.40E-07	1.43E-07	1.46E-07
95%	5.18E-07	5.39E-07	5.55E-07
StdDev		2.26E-07	
Skewness		1.01E+01	
Kurtosis		2.77E+02	

d. BSEP is a well bunkered facility with few external targets that contribute to CDF/LERF for the high winds PRA. For those systems, structures, and components (SSCs) outside category I buildings, the impacts of the uncertainties on the model and results have been assessed. The sources of uncertainty include:

- Duration of the high wind event (e.g., storm, hurricane) affecting the site. The duration of the largest peak winds during a hurricane is 6 - 8 hours.

- Ability of the diesel generators to withstand a high wind event. The diesel exhausts will break away, as designed, during high wind and missile scenarios.
- Ability of the Severe Accident Mitigation Alternative (SAMA) diesels to survive high wind events. These diesels are being replaced with larger, bunkered diesels.
- Capacity of the SAMA diesel fuel oil to last for the duration of the storm without refueling. The diesels have a fuel capacity to run for approximately 24 hours which is longer than the peak winds of the storm.
- Failure of the metal reactor building siding which requires high, straight winds.

None of these sources of uncertainty will impact the results of the SFCP as a high wind event affects both the base and modified cases. A change in the surveillance test intervals will not impact the results of the uncertainty assessments of CDF/LERF as the assessed impact is negligible on any STI due to the types of uncertainty.

NRC RAI 6:

Please confirm that, when citing conservatism in the base PRA model (as for example in internal events F&Os 3-12 and 3-13, internal fire F&Os 1-24, 2-10, and 4-13), calculation of the differential risk for the application is also conservative (i.e., the risk estimated for the before versus after condition uses the same assumptions, etc., except for the change to any basic event values affected by the application, ensuring that the before value is not overestimated such that subtracting it from the after value could underestimate the risk increase).

Response:

The same conservatisms and assumptions used in the base PRA model will be used in the calculation of differential risk for the application, except for the change to any basic event values affected by the application.

NRC RAI 7:

The LAR states that hazard screening performed for the Individual Plant Examination of External Events (IPEEE) studies will be used to assess hazards from transportation and nearby facility accidents. Please confirm that since the IPEEE there were no changes to the plant site that could invalidate the conclusions of the IPEEE study with regards to hazards from transportation and nearby facility accidents.

Response:

Transportation and nearby facility accidents were included in the IPEEE to account for human errors outside the normal operation of BSEP. The types of hazards identified for analysis included:

- Aircraft Impact
- Industrial Accidents
- Military Accidents
- Pipeline Accidents
- Hydrogen Storage Failures
- Transportation Accidents

For the IPEEE submitted in 1995, BSEP concluded that potential accidents associated with nearby air traffic, runways, roads, railways, waterways, pipelines, and fixed military and industrial facilities are not considered a significant hazard. Based on a review of the Environmental Report, prepared in support of BSEP License Renewal in 2004, and the most recent Evacuation Time Estimate Analysis prepared in 2012, in accordance with 10 CFR 50, Appendix E, Section IV, it was confirmed that there have been no significant changes to the plant site and surrounding areas that could invalidate this conclusion. There are no new airports, pipelines, or hydrogen storage facilities. Likewise there are no new industrial facilities that would result in industrial or transportation accidents potentially affecting BSEP.

List of Regulatory Commitments

The following table identifies the actions in this document to which the Brunswick Steam Electric Plant (BSEP) has committed. Statements in this submittal, with the exception of those in the table below, are provided for information purposes and are not considered commitments.

Please direct questions regarding these commitments to Mr. Lee Grzeck, Manager - Regulatory Affairs, at (910) 457-2487.

Commitment	Type (Check One)		Scheduled Completion Date/Event
	One-Time Action	Continuing Compliance	
1 Duke Energy will conduct a focused-scope peer review in accordance with Regulatory Guide 1.200, Revision 2, for the internal flooding technical elements based on the enhancements made since the 2010 peer review.	X		7/13/2017