UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 12, 2014

Mr. George T. Hamrick, Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, Inc.
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 –REQUEST FOR ADDITIONAL INFORMATION REGARDING VOLUNTARY RISK INITIATIVE NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805 - PROBABILISTIC RISK ASSESSMENT (TAC NOS. ME9623 AND ME9624)

Dear Mr. Hamrick:

By letter dated September 25, 2012, as supplemented by letter dated December 17, 2012, Duke Energy Progress, Inc. (the licensee) proposed to amend the operating license for the Brunswick Steam Electric Plant, Units 1 and 2, by adopting a new risk-informed performance-based fire protection licensing basis in accordance with National Fire Protection Association Standard 805. By letter dated May 15, 2013, the Nuclear Regulatory Commission (NRC) requested additional information needed to complete its review. The licensee responded by letters dated June 28, July 15, July 31, August 29, and September 30, 2013.

The NRC staff has reviewed the licensee's application and responses to the staff's request for additional information (RAI) and determined that further information is needed to complete its evaluation of the proposed change.

On December 23, 2013, the NRC staff forwarded via electronic mail a draft of the second set of Probabilistic Risk Assessment (PRA) RAIs to the licensee. On January 14, 2014, the NRC staff and representatives of the licensee held a conference call to provide the licensee with an opportunity to seek clarification regarding any portion of the information request and to discuss the response schedule.

The NRC staff's finalized second set of PRA RAIs is enclosed. This request was discussed with Mr. Bill Murray of your staff on January 15, 2014, and it was agreed that the licensee would respond consistent with the following schedule:

<u>March 14, 2014:</u>
PRA RAI 1.d.01, 1.f.ii.01, 1.f.iii.01, 14.01, 15.01, 16.01, 18.g.01, 22, and 24

<u>April 11, 2014:</u>
PRA RAI 1.l.01, 6.01, 8.01, and 23

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. Please note that review efforts on this task are continuing and additional RAIs may be forthcoming.

If you have any questions regarding this letter, please feel free to contact me at (301) 415-1447.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION

VOLUNTARY FIRE PROTECTION RISK INITIATIVE

DUKE ENERGY PROGRESS, INC.

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

By letter dated September 25, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12285A428), as supplemented by letter dated December 17, 2012 (ADAMS Accession No. ML12362A284), Duke Energy Progress, Inc. (the licensee) proposed to amend the operating license for the Brunswick Steam Electric Plant, Units 1 and 2, by adopting a new risk-informed performance-based fire protection licensing basis in accordance with National Fire Protection Association Standard 805. By letter dated May 15, 2013 (ADAMS Accession No. ML13123A231), the Nuclear Regulatory Commission (NRC) requested additional information needed to complete its review. The licensee responded by letters dated June 28, July 15, July 31, August 29, and September 30, 2013 (ADAMS Accession Nos. ML13191B271, ML13205A016, ML13220B041, ML13246A276, and ML13277A040, respectively).

The NRC staff has reviewed the licensee's application and responses to the staff's request for additional information (RAI) and determined that further information is needed to complete its evaluation of the proposed change.

Probabilistic Risk Assessment (PRA) Request for Additional Information (RAI) 1.d.01

In letter dated July 15, 2013, the licensee responded to PRA RAI 01, but did not describe the results of its reviews regarding plant experience and records of violations of transient combustible controls per the request to augment justification for reducing the transient heat release rate (HRR) from 317 kW to 143 kW. Also, it was explained that the transient combustibles and ignition source controls program (i.e., FIR-NGGC-009) will be modified to support use of a lower HRR for specific areas of the plant, that change is not listed in license amendment request (LAR) Attachment S, Table S-2.

By letter dated September 30, 2013, the licensee presented the results of a sensitivity study showing factors of 3.4 and 3.1 increase in the fire core damage frequency (CDF) and factors of 1.8 and 1.9 increases in the fire large early release frequency (LERF) for Units 1 and 2, respectively. However, the changes-in-risk were not included; furthermore, it appears that the cited sensitivity results for CDF and LERF are based on a simplified calculation. The Fire PRA (FPRA) Sensitivities Report (BNP-PSA-095) states that assessment of sensitivity was performed by increasing the hot gas layer (HGL) conditional core damage probability/conditional large early release frequency by 10 percent for areas where the reduced HRR is credited; and explains that the basis for this simplification is a determination that the turbine building target set would increase by 10 percent if an HRR rate of 317 kilowatts (kW) were used (although BNP-PSA-095 states this is "[A]necdotally" derived). The staff notes that administrative controls should not be the sole basis to reduce the transient fire HRR, and reducing the transient fire

HRR should not substitute for reducing the frequency of occurrence with appropriate transient fire weighting factors.

Provide the results of reviewing plant experience and records of violations of transient combustible controls and add an implementation item to LAR Attachment S, Table S-2, to change the transient combustibles and ignition source controls program to support use of the lower transient fire HRRs used in the FPRA. Alternatively, demonstrate that the increase in total CDF and LERF and the increase in change-in-risk are acceptable based on using real plant and cable specific configurations and an HRR of 317 kW for transient fires.

PRA RAI 1.f.ii.01

By letter dated July 15, 2013, the licensee responded to PRA RAI 1.f.ii, but did not describe how the total available time (i.e., T_{SW}) or how the time to the initial cue to evacuate the Main Control Room (i.e., the initial Tdelay) was determined. The NRC staff notes that the bases and assumptions made about these times affect quantification of human error probabilities used to determine the failure probability of abandonment actions.

Describe the bases for calculating the total time available and how the time to the initial cue to evacuate the control room was determined, and how these timing assumptions include margin for potential delays.

PRA RAI 1.f.iii.01

By letter dated July 15, 2013, the licensee responded to PRA RAI 1.f.iii and clarified that the failure probability to perform safe shutdown was determined by summing associated probabilities of individual core damage end states without regard to whether damage was early or late. In light of this assumption, explain how LERF was determined and provide justification that the result is a realistic or conservative representation of LERF.

PRA RAI 1 i.01

By letter dated July 15, 2013, the licensee responded to PRA RAI 1.i and explained that the state of knowledge correlations (SOKC) were omitted for fire ignition frequencies and nonsuppression probabilities, and implied that SOKC was only applied within cutsets. This response also explains that an SOKC multiplier was calculated using the standard deviation for hot short probabilities. Provide an estimate of the impact of considering SOKC for fire ignition frequency, nonsuppression probability, and circuit failure probabilities on the risk estimates (i.e., CDF, LERF, delta (Δ)CDF, ΔLERF) across cutsets.

PRA RAI 6.01

By letter dated July 31, 2013, the licensee responded to PRA RAI 6 and explained that for main control boards (MCBs) with incipient detection that the risk from "self" scenarios was assumed to be negligible and not modeled in the FPRA. By letter dated September 30, 2013, the licensee presented the results of a sensitivity study that credited the currently installed in-panel ion smoke detection system rather than incipient detection, and applied the NUREG/CR 6850, "EPRI [Electric Power Research Institute]/NRC-RES [Nuclear Regulatory Research] Fire PRA Methodology for Nuclear Power Facilities," approach to determine the frequency of "self-fires" in the MCB. Use of the NUREG/CR-6850 Appendix L approach appears not to have been previously used in the FPRA (i.e., the baseline approach excludes in-cabinet MCB fires based on installation of incipient detection), and use of this approach produces significant differences in accident sequence risk estimates (e.g., the CDF increases from 3.5E-5/yr for Units 1 and 2 to 6.3E-5/yr and 6.6E-5/yr, respectively for each unit).

The licensee's application of the Appendix L method appears to deviate from the NUREG/CR-6850 guidance, based on the description presented in the FPRA Sensitivities report (i.e., BNP-PSA-095, Revision 1) that was issued after the audit. The initiating event fire frequencies for 27 MCB "self" scenarios presented in Table 20 of the FPRA Sensitivities report were determined by dividing the EPRI MCB Bin 4 fire frequency by 27, leading to a frequency of 6.104E-5/yr per MCB. This conflicts with the counting guidance presented in Section 6.5.6 of NUREG/CR-6850, which presumes that counting will yield only one or two MCBs. Furthermore, when applying the Appendix L method, the frequency of a scenario involving specific target damage in the MCB should be determined by multiplying the probability of target damage, as defined by Figure L-1 of NUREG/CR-6850, by the entire MCB fire frequency. When using this approach partitions or segmentation cannot be used to justify subdividing the MCB fire frequency, although they might be used to preclude certain scenarios involving targets separated by partitions.

Use of the NUREG/CR-6850 Appendix L modeling approach meets the definition of a model upgrade as defined by American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." In light of the fact that use of incipient detection to preclude internal MCB cabinet fires is not consistent with Frequently Asked Question (FAQ) 08-0046 and implementation of Appendix L appears to deviate from NUREG/CR-6850 guidance, the Appendix L analysis of the MCB needs to be re-performed to meet guidance in NUREG/CR-6850 by applying the whole MCB fire frequency to each MCB scenario. In addition, a focused scope peer review on the use of the NUREG/CR-6850 Appendix L modeling approach is needed prior to using the FPRA for self-approval.

Provide the reanalysis of the MCBs as part of PRA RAI 23, and provide a proposed implementation item, to address that a focused-scope peer review will be performed, and any findings will be resolved, before self-approval of post transition changes.

PRA RAI 8.01

By letter dated August 29, 2013, the licensee responded to PRA RAI 8 and explained that the conditions under which a fire has the potential to fail additional equipment due to increased room temperature is "very limited," but acknowledged that such a condition might exist in an enclosed room with an ignition source. The response further argues that cables for equipment in such an enclosed space may already be in the zone of influence of the ignition source. The heat-up analysis report (Attachment 21 to BNP-PSA-067), in support of the success criteria for the internal events PRA, identifies the following maximum ambient operating temperatures for temperature limited equipment:

- 131°F for the limiting component in the Battery Room
- 235°F for the limiting component in the Core Spray Room
- 165°F for the limiting component in the High Pressure Coolant Injection Room
- 180°F for the limiting component in the Residual Heat Removal Room.

Discuss whether these components can be affected by fire producing ambient operating temperatures above the maximum temperature for the component. If so, provide the impact on the risk estimates (i.e., CDF, LERF, ΔCDF, and ΔLERF).

PRA RAI 14.01

By letter dated July 31, 2013, the licensee responded to PRA RAI 14 and stated that the determination of frequency of MCR abandonment scenarios is based on a "conservatively-selected time distribution, assuming that the control room boundary doors remain closed during the fire and assuming 1% unavailability for the control room HVAC system." It is not clear how 1 percent unavailability of the control room heating, ventilation, air conditioning (HVAC) system was incorporated into the calculation of abandonment times; whether modeling of the MCR boundary configuration is conservative; or if the 1 percent unavailability is based on plant experience. Clarify how control room HVAC system unavailability is addressed in abandonment scenarios; whether MCR boundary configuration is conservatively modeled; and the basis for the control room HVAC unavailability of 1 percent.

PRA RAI 15.01

By letter dated July 31, 2013, the licensee responded to PRA RAI 15 and explained that "[T]ables W-4-1 and W-4-2 report "N/A" for the additional risk of recovery actions (RAs), because possible risk reductions for other RAs were not quantified." The meaning of this explanation is not clear. The additional risk of RAs should be calculated for fire areas where RAs are credited, unless they are only credited for defense-in-depth. By letter dated September 30, 2013, the licensee showed an entry in LAR Attachment W, Table W-4-2 for the Turbine Bay fire area in the "RAs" column that was changed from "Yes" in the original LAR Supplement dated December 17, 2012 (ADAMS Accession No. ML12362A284) to "No" in the response. LAR Attachment G indicates that RAs are credited in the Unit 2 Turbine Bay for defense-in-depth. Clarify which fire areas have RAs credited for risk reduction and provide or confirm that additional risk of RA values exist for these fire areas.

PRA RAI 16.01

By letter dated July 31, 2013, the licensee responded to PRA RAI 16 and explained that in accordance with the post-transition change process, an implementation item (i.e., Item 9) was added to provide for identification and evaluation of features that result in the as-built (i.e., after modifications are complete) change in risk exceeding LAR Attachment W, Table W-4-1 or W-4- 2 values. Add an implementation item to evaluate the change in risk estimates due to completed implementation items (e.g., completion of fire procedures), and a plan of action if the as-operated change in risk exceeds risk acceptance guidelines.

PRA RAI 18.g.01

In the disposition to Internal Events PRA Peer Review Findings on Flooding Supporting Requirements IFSN-A6 and IFQU-A9 (see LAR Attachment U, Table U-1), neither of which was originally met, the licensee cites "no impact on the Fire PRA [from Internal Flooding]." Both findings cite the need to address potential detrimental effects from spray rather than submergence or other flooding mechanisms. Since fire-induced flooding events, such as interfacing system loss of coolant accidents (ISLOCAs) that create spray effects, might be more likely than the same scenarios initiated randomly, the disposition needs to also address this possibility in order to justify the conclusion of "no impact on the Fire PRA." Provide a discussion regarding the potential increased likelihood, if any, for flooding-induced spray effects that could be detrimental to equipment as a result of fire-induced failures to justify the original disposition, including, but not limited to, ISLOCAs.

PRA RAI 22

By letter dated September 30, 2013, the licensee provided a set of CDF, LERF, ΔCDF, and ΔLERF values in LAR Attachment W, Tables W-4-1 and W-4-2 of Enclosure 4. As illustrated by the table below, these values are different from the values presented in Tables W-4-1 and W-4-2 of the LAR Supplement dated December 17, 2012 (ADAMS Accession No. ML12362A285). This is also true for the fire area values as well as the total values. The source of these differences does not appear to be explained in any of the licensee's letters dated July 15, July 31, August 29, and September 30, 2013.

Source Document	Fire CDF (yr^{-1})	Fire LERF (yr^{-1})	Total CDF (yr^{-1})	Total LERF (yr^{-1})	ΔCDF (yr^{-1})	ΔLERF (yr^{-1})
Unit 1 – Dec 17, 2012	1.6E-5	4.0E-6	3.0E-5	4.6E-6	2.8E-6	4.7E-7
Unit 1 – Sep 30, 2013	2.1E-5	4.4E-6	3.5E-5	5.0E-6	2.1E-6	1.0E-7
Unit 2 – Dec 17, 2012	1.4E-5	1.5E-6	2.8E-5	2.1E-6	3.3E-6	5.2E-7
Unit 2 – Sep 30, 2013	2.1E-5	4.1E-6	3.5E-5	4.8E-6	3.6E-6	9.6E-8

In addition to the different risk values reported in Fire Area Risk Summary tables, LAR Attachment W, Tables W-2-1, W-2-2, W-3-1, and W-3-2 report, a different listing of dominant scenarios and, in some cases, different fire scenario construction. The change in the listing might be attributed to an unexplained modeling refinement that changed risk values across fire areas, but it is not clear why the HGL contribution was added to certain scenarios where HGL contribution to CDF or LERF was not originally considered (i.e., FC211_4568_BFM2 and

FC230_4807_BFM2 in Table W-2-2, and FC211_4572_BFM2, FC211_4568_BFM2, and FC230_4807_BFM2 in Table W-3-2). In light of these observations:

a. Explain the reason for the differences in the fire area and plant risk values reported in LAR Attachment W included in the September 30, 2013, letter versus the original LAR Supplement dated December 17, 2012. Include an explanation for the asymmetries in the changes (i.e., CDF values went up in all cases and ΔCDF values went down for Unit 1, but went up for Unit 2. LERF values went up in all cases but ΔLERF went down for Units1 and 2).

b. Explain why HGL contribution is determined differently for certain dominant scenarios between the original LAR submittal dated September 25, 2012 (ADAMS Accession No. ML12285A428) and in the letter dated September 30, 2013, and explain whether there are any other differences in quantification of scenarios reported in the September 30, 2013, letter versus the LAR supplement dated December 17, 2012. If there are differences, provide the rationale for those differences. (For example, in Table W-2-2 of the September 30, 2013, letter, scenario "FC11_4568_BFM2" includes HGL contribution, whereas this scenario, as described in the same table in the December 17, 2012, LAR submittal does not. Same for scenarios "FC11_4568_BFM2" and "FC230_4807_BFM2" in Table W-3-2.)

c. Identify the specific changes made to baseline FPRA since the time of the audit , the reason for those changes, and explanation of whether any of the changes represent application of a new method or a change in scope in application of a previously used method that constitute a model upgrade. Also, explain if further refinements to the FPRA will continue to be made and whether those changes may represent PRA model upgrades. Indicate which, if any, of these changes are considered a PRA upgrade as defined in the PRA standard. For those considered to be a PRA upgrade, provide a proposed implementation item, that a focused-scope peer review will be performed, and any findings will be resolved, before self-approval of post transition changes.

PRA RAI 23

Section 2.4.3.3 of National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, (NFPA 805) incorporated by reference into Title 10, *Code of Federal Regulations* (10 CFR) Section 50.48(c) states that the probabilistic safety assessment (PSA) (PSA is also referred to as PRA) approach, methods, and data shall be acceptable to the authority having jurisdiction, which is the NRC. Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," identifies NUREG/CR-6850, Nuclear Energy Institute 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), Revision 2," and the ongoing FAQ process as documenting acceptable methods to the staff for adopting a fire protection program consistent with NFPA-805.

The RAI and sensitivity study that are listed below address PRA methods that have not been accepted by the NRC staff. Although the licensee demonstrated that the individual effect of removing a specific method did not result in exceeding the guidelines in the RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The licensee neither showed the impact of the combined effect nor provided adequate justification to demonstrate acceptability of these methods. Therefore, unless a method is eventually found to be acceptable by the NRC, that method needs to be replaced by a previously acceptable method, or the FPRA of record for the transition needs to be based on the sensitivity studies listed below, which would limit the ability to exercise "self-approval."

- PRA RAI 1.d-01 regarding use of transient fire HRRs less than 317 kW (unless the alternative response described in followup RAI 1.d-01 is provided);
- Sensitivity analysis reported in Section 4.8.3.2 of the LAR removing credit for Control Power Transformers in circuit failure analysis;
- PRA RAI 6-01 regarding the sensitivity analysis reported in Section 4.8.3.6 of the LAR removing credit for incipient detection in MCBs;
- PRA RAI 8-01 regarding evaluation of temperature limited components (unless the response to PRA RAI 8-01 shows the risk to be negligible);
- PRA RAI 1.i-01 regarding SOKC for FPRA specific parameters.

a. Provide the results of an aggregate analysis that provides the integrated impact on the fire risk (i.e., the total transition CDF, LERF, ΔCDF, ΔLERF) of replacing the above methods with methods that are acceptable to the NRC. In this aggregate analysis, for those cases where the individual issues have a synergistic impact on the results, a simultaneous analysis must be performed. For those cases where no synergy exists, a one-at-a-time analysis may be done. For those cases that have a negligible impact, a qualitative evaluation may be done.

PRA RAI 24

By letter dated September 30, 2013, the licensee presented the results of the following sensitivity analyses for methods that have not been accepted by the NRC staff, but in doing so demonstrate that the effect of removing credit these methods have negligible impact on the risk estimates (i.e., ΔCDF, ΔLERF, CDF and LERF):

- Sensitivity analysis reported in Section 4.8.3.1 of the LAR addressing use of panel factors for motor control centers;
- Sensitivity analysis reported in Section 4.8.3.7 of the LAR addressing use of a fire HRR of less than 211 kW for motor pumps;
- Sensitivity analysis reported in Section 4.8.3.8 of the LAR addressing incorporation of a quantitative evaluation of potential sensitive electronics fire damage;
- Sensitivity analysis reported in Section 4.8.3.10 of the LAR addressing the assumption of multiple versus single bundle fires in the MCBs;
- Sensitivity analysis reported in Section 4.8.3.11 of the LAR addressing incorporation a maintenance influencing factor of 50.

While use of the methods addressed in these sensitivity analyses has negligible impact on the change in risk for the post-transition plant, they may have greater impact in future plant-change evaluations. Include a new implementation item in LAR Attachment 3, Table S-3 to make PRA model improvements necessary to incorporate acceptable modeling (e.g., the model adjustments from these sensitivity analyses), to inform the NRC if the change results in the risk metrics (i.e., CDF or LERF) exceeding the RG 1.205 criteria for changes requiring NRC review and approval, and that a focused scope peer review will be performed on changes that are PRA upgrades as defined in the PRA standard, and any findings will be resolved, before self-approval of post-transition changes.

If you have any questions regarding this letter, please feel free to contact me at (301) 415-1447.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

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