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

June 18, 2015

Mr. Steven D. Capps
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2: REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO IMPLEMENT A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM (TAC NOS. MF2934 AND MF2935)

Dear Mr. Capps:

By letter dated September 26, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13276A126), Duke Energy Carolinas, LLC (Duke) submitted a license amendment request to change its fire protection program to one based on the National Fire Protection Association (NFPA) Standard-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

The U.S. Nuclear Regulatory Commission staff is continuing its review and has determined that additional information is needed in the probabilistic risk assessment area as discussed in the Enclosure.

Sincerely,

Bob Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: As stated

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION (RAI)

ADOPTION OF NATIONAL FIRE PROTECTION ASSOCIATION

STANDARD 805 FOR FIRE PROTECTION

DUKE ENERGY CAROLINAS, LLC (DUKE)

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369, 50-370

By letter dated September 26, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13276A126), Duke submitted a license amendment request to change its fire protection program to one based on the National Fire Protection Association Standard-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition (NFPA 805). The U.S. Nuclear Regulatory Commission (NRC) staff is continuing its review and has determined that additional information is needed in the probabilistic risk assessment (PRA) area as follows.

Follow-ups to November 12, 2014, December 12, 2014, and January 26, 2015, February 27, 2015, and March 13, 2015 PRA McGuire RAI Responses

PRA RAI 03.a.01

The response to PRA RAI 03.a provides the results of an uncertainty analysis on the total fire core damage frequency (CDF) and large early release frequency (LERF), that includes accounting for state of knowledge correlations, but no corresponding results are provided for delta-risk. Provide the results of the uncertainty analysis impact on the delta risk results and discuss if the RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" risk guidelines are met using the mean results.

PRA RAI 03.b.01

The response to PRA RAI 03.b identifies that the following RAI responses resulted in changes to the PRA:

- PRA RAI 01.a
- PRA RAI 01.b
- PRA RAI 01.c.i
- PRA RAI 02.f
- PRA RAI 02.i
- PRA RAI 10
- PRA RAI 16
- PRA RAI 17

- PRA RAI 18
- PRA RAI 19
- PRA RAI 22
- FM RAI 01.a through 01.g
- FM RAI 04

The response mentions the following RAIs as being discussed in parts A or D:

- PRA RAI 01.d
- PRA RAI 06
- PRA RAI 09
- PRA RAI 12
- PRA RAI 13
- PRA RAI 20
- PRA RAI 21
- PRA RAI 23
- FM RAI 01.j

The response lists some RAIs which were "addressed" in the aggregate PRA, other RAIs for which the PRA was "adjusted" in the aggregate PRA, and finally some RAIs that are "discussed." The difference between addressed, adjusted, and discussed is not explained and oftentimes the responses to the original RAIs do not describe what specific changes were or could be made to the PRA. Part b of RAI 3 requested a summary of the disposition of each issue in the aggregate PRA and the post-transition PRA. This was not provided. Please provide a table with the requested information.

PRA RAI 12.01

In PRA 12, the NRC staff noted the discussion in license amendment request (LAR) Section V.2.7 that describes two main control room (MCR) abandonment on loss-of-habitability scenarios. The NRC staff requested "[a]n explanation of how the [conditional core damage probabilities] CCDPs account for the range if probabilities for properly shutting down the plant, and discussion of how they were applied in the scenario analysis." Three different levels of fire severity were provided as examples illustrating the source of the range of shutdown probabilities. The response stated, in part, that:

"Each MCR abandonment scenario encompasses the range of results from few functional failures to multiple functional failures, each condition (b.i, b.ii, & b.iii) leading to the most severe end state where the SSF is the sole remaining success path after abandonment. In the MNS Fire PRA, for the abandonment scenarios, the number of fire induced failures and spurious operations is based on the panel of origin that produces the highest conditional core damage probability (CCDP). Therefore, the abandonment scenarios account for the worst case impacts on the SSF regardless of a potentially more favorable outcome."

The response further clarifies that:
"... main control board frequency was applied in the quantification of the abandonment scenario for the main control board (MCB) fire. The remaining fire area wide ignition frequency (including

electrical cabinet and transient frequency), was applied to the abandonment scenario for the non-MCB fires in the control room."

Although the response to states that two scenarios are modelled (one following MCB fires and another following non-MCB fires) it is unclear whether a single CCDP/CLERP is used for the two abandonment scenarios. No discussion or justification was provided as to why not accounting for the range of probabilities in the fire PRA will result in a well characterized or conservative risk change-in-risk estimate. The NRC staff requests the following information to determine whether accounting for the range of probabilities for properly shutting down the plant following loss of MCR habitability would change the acceptable change-in-risk estimates to unacceptable estimates.

a) Identify the fire frequency, CCDP, and conditional large early release probability (CLERP) assigned to each abandonment scenarios for both the compliant and the variant plant.

b) Explain any differences between the compliant and the variant plant PRA models for these abandonment scenarios.

c) A claim of "worst case impact" is insufficient when the meaning of worst case can vary as it does with change-in-risk calculations. Summarize how the change in risk calculation is performed and justify that the change-in-risk estimates from these loss of habitability abandonment scenarios is well characterized or conservative.

PRA RAI 13.01

The response to PRA RAI 13 discussed how the change-in-risk was calculated for fire areas (other than the MCR and cable room) that are designated as safe shutdown facility (SSF) areas in accordance with 10 CFR 50, Appendix R Section III.G.3. This response was augmented with information provided in the slides for the public meeting on April 14, 2015 (ADAMS Accession No. ML15099A587), which included further explanation about how the compliant and post-transition plants for these areas were modelled in the Fire Probabilistic Risk Assessment (FPRA). Based on the methods used by McGuire in the FPRA as described in the meeting, please provide the following:

a) Confirm that the equipment damaged by each fire in each SSF area has been identified using the McGuire fire damage methodology and is assumed to fail in the FPRA.

b) Confirm that all applicable equipment undamaged by each fire is nominally available to mitigate the fire, i.e., is credited in the PRA model. Summarize any differences between the nominally available equipment in the complaint versus the variant plant.

c) Confirm that each of the SSF fire areas has been reviewed by the NRC and has been determined to meet the alternative shutdown option in Section III.G.3 of Appendix R and all the criteria laid out in Section 2.4 b) of RG 1.205, "Risk-Informed, Performance Based Fire Protection For Existing Light-Water Nuclear Power Plants." Provide any limitations and conditions associated with any of the areas and, if any, clarify why such issues are addressed by or not relevant to the FPRA analysis.

d) Summarize the procedures and process used by the operating crews to mitigate fires in the SSF areas using the surviving equipment in the area.

e) Provide the procedural step(s) including the decision guidelines directing the operating crews to make the decision to activate the SSF and subsequently establish control of the plant at the SSF.

f) Summarize the proceduralized steps directing the operating crews to transfer command and control from the MCR to the SSF.

g) Summarize the operating training in the activation and the use of the SSF.

h) LAR Section V.2.6 and the response to PRA RAI 13 state that the MCR is only abandoned (i.e., command and control is transferred to the SSF) on loss of control room habitability. Clarify this statement in light of the April 14, 2015, presentation and the statement that the SSF is a PCS as defined in RG 1.205 that indicates that command and control must be transferred to the SSF on loss of control.

i) Explain how the variance from deterministic requirements (VFDRs) within the SSF areas were identified. LAR Section W.2.1 and the response to PRA RAI 13 explains that, generally, the compliant plant for both abandonment and non-abandonment scenarios was evaluated by "toggling" off or excluding basic events to remove the fire-induced failures associated with the VFDRs. Clarify, relative to the variant plant model, how excluded basic events are identified and removed from the compliant plant quantification.

PRA RAI 13.02

The response to PRA RAI 13 states that the compliance assessment for fire areas 01 (Ul and U2), 02, 03, 04 (U1 and U2), 13 (U1 and U2), 14 (U1 and U2), 19, 20, 21 (U1 and U2), 24 (U1 and U2), and 25 (Ul and U2) relies upon transfer of primary command and control to the Standby Shutdown Facility (SSF) as the Nuclear Safety Performance (NSP) success strategy (i.e., the NSP success path).

a) Clarify how the 10 CFR 50.48(c) rule (including the NFAP-805 Standard as incorporated by reference) and associated guidance, allows the assignment of the remotely located SSF facility as the single NSP success for all fires in some fire areas outside of the MCR.

b) The response to RAI 13.b states, the, "[p]er MNS plant procedures, only control room habitability due to the fire will cause a complete abandonment of the MCR (as opposed to implementing SSF functions while maintaining command and control in the MCR as discussed in the response to PRA RAI 13.a)." According to the guidance in RG 1.205, all actions to recover NSP successes strategies and taken outside of the MCR while maintaining command and control in the MCR are recovery actions. Why are there no recovery actions identified for these areas?

c) How and when (i.e., in what PRA accident sequences) is the NSP SSF success strategy for the aforementioned fire areas modelled in the compliant plant PRA model, and in the post-transition plant PRA model.

PRA RAI 19.a.01

The response to PRA RAI 19.a states that the response to PRA RAI 03 would use the methods in draft FAQ 14-0009 to evaluate propagation of fires outside of well-sealed and robustly-secured motor control centers (MCCs) greater than 440 volts. The analyses described in draft FAQ 14-0009 has changed considerably since the first draft dated July 1, 2014. To resolve this issue use the method described in the final version of FPRA FAQ 14-0009 (ADAMS Accession No. ML15119A176) in the FPRA and provide updated results as part of the aggregate change-in-risk analysis requested in PRA RAI 03.

PRA RAI 19.b.01

The response to PRA RAI 19.b states that internal spurious operations (hot shorts) within well-sealed MCCs were removed from the FPRA when the unrelated severity factor of 0.2 was removed. The RAI response asserts that the results are overly conservative after removing the severity factor to justify removing hot shorts. A general claim of overly conservative is insufficient to justify changing modelling assumptions. Guidance on including fire-induced spurious hot short in the panel wiring's conductor bundles within a cabinet is available in NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 1, Section 6.6.3, and Volume 2, Section 7.4. To resolve this issue apply the guidance in NUREG/CR-7150 in the FPRA and provide updated results as part of the aggregate change-in-risk analysis requested in PRA RAI 03.

PRA RAI 24

During final review of the implementation items, it was discovered that McGuire's Implementation Item 12 program does not provide confidence that the final change in risk from transition meets the acceptance guidelines. The licensee proposes that "If the revised Fire PRA shows a risk increase of greater than 1E-07 for CDF or 1E-08 for LERF then enter the results into the corrective action program to determine the cause of the risk increase and determine corrective actions." Entering an increase greater than the self-approval guidelines into the corrective action program does not provide confidence that the final result of any corrective action will be a transition change in risk that is consistent with the acceptance guidelines. Furthermore, unanticipated risk increases greater than the self-approval guideline generally need to be reduced by fixing the cause of exceedance (i.e., the change itself), otherwise the results and the proposed change (if any) should be submitted to the NRC staff according to the license condition. Provide an Implementation Item to verify that the cumulative change-in-risk does not exceed RG 1.174 guidelines once all the modifications and procedural implementation items are completed.

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NAME	BMartin	SFigueroa	SRosenberg* (LFields for)	RPascarelli	BMartin
DATE	06/18/15	05/28/15	06/18/15	06/18/15	06/18/15

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