UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
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May 22, 2014

Mr. K. Henderson
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

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

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST RE: METHODOLOGY REPORT DPC-NE-3001-P, REVISION 1, MULTIDIMENSIONAL REACTOR TRANSIENTS AND SAFETY ANALYSIS PHYSICS PARAMETERS METHODOLOGY (TAC NOS. MF3119, MF3120, MF3121, AND MF3122)

Dear Mr. Henderson and Mr. Capps:

By letter dated November 14, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13325B142), Duke Energy Carolinas, LLC (Duke, the licensee), submitted a license amendment request for Catawba Nuclear Station, Units 1 and 2, and McGuire Nuclear Station, Units 1 and 2. The proposed amendment requested review and approval to the Methodology Report DPC-NE-3001-P, Revision 1, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology."

The Nuclear Regulatory Commission (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The RAI questions were provided in draft form to Duke on April 22, 2014 (ADAMS Accession No. ML14129A284). The draft questions were sent to facilitate a teleconference to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed.

Enclosure 1 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 2, this document is decontrolled.

On May 6, 2014, during the teleconference, it was identified that information sufficient to resolve the NRC staff needs was already docketed. The question has been maintained to preserve numbering, but edited to reflect that no further response is needed. Additionally, during the call, the licensee indicated that they could respond by June 20, 2014.

The NRC staff has determined that Enclosure 1 contains proprietary information pursuant to 10 CFR 2.390. Accordingly, the NRC staff has prepared a redacted, publicaly available non-proprietary version (i.e., Enclosure 2).

If you have any questions regarding this matter, I may be reached at (301) 415-2481 or by e-mail at ed.miller@nrc.gov.

Sincerely,

G. Edward Miller, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413, and 50-414

RAIs for DPC-NE-3001P, Rev. 1, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology"

SNPB-RAI-1) The main change made to Section 4.2.1.2, described in Change 4-1, proposes the use of Version 2 of SIMULATE-3K. The new models listed on page A-11, especially including burnup-dependent gap conductance, fuel thermal conductivity, and radial power profile. Were benchmarks of Version 2 of SIMULATE-3K performed relative to Version 1. If so, please provide a summary of the benchmarks. If not, please justify why benchmarking is not needed.

SNPB-RAI-2) **PROPIN:** The changes to Section 4.2.2.2, described in Change 4-3, allows for the use of [[ ]] Since the [[ ]] model is described in detail in DPC-NE-3000-PA, please provide additional details about the [[ ]] model.

SNPB-RAI-3) The technical justification for Change 4-13 states "The power level uncertainty has been reduced from 2% to approximately 0.3% as part of the measurement uncertainty recapture (MUR) uprate at McGuire and Catawba." While McGuire has received approval for an MUR uprate, Catawba has not. Please clarify this statement as appropriate.

SNPB-RAI-4) One of the revisions to Section 4.3.2, described in Change 4-14, allows for the use of both the high flux and high flux positive rate trips for the rod ejection analysis. Please justify why it is now reasonable to credit the high flux positive rate trip when it was not previously credited.

SNPB-RAI-5) There are several changes made to Section 5.2.1.2, contained in Appendix B to the licensee's submittal, which do not have a technical justification in Appendix A. The staff therefore has the following questions:

5.a) The existing subsection discussing steam generator (SG) renodalization is amended such that it only applies to the hot zero power (HZP) steam line break (SLB) case. Is SG renodalization required for the hot full power (HFP) SLB case? Why or why not?

5.b) **PROPIN:** In the subsection regarding [[ ]] for the HFP SLB analysis, it is stated that [[ ]] What impact does this modeling change have on the long-term cooling capabilities of the faulted steam generator?

Enclosure 1

5.c) **PROPIN:** Section 5.2.1.2 notes that one of the major differences between Catawba and McGuire is in their steam generators. Given that this is the case, are the model modifications proposed in the [[ ]] applicable to and/or necessary for both sites? When the changes are made, are they then consistent between the two sites?

5.d) **PROPIN:** One of the subsections discusses the [[ ]] According to EPRI NP-1850-CCM, Volume 5, EPRI's RETRAN-02 modeling guidelines, the primary purpose of [[ ]] Given that [[ ]] is extremely important for a steam line break analysis, please discuss the impact, if any, that this change will have on the accuracy of the HFP SLB analysis.

SNPB-RAI-6) A subsection describing the HFP core physics parameters is added to Section 5.2.2.1.

6.a) It is stated that "The limiting core physics parameters are dependent on the break size." How is this dependence characterized? Is it that the parameter which is limiting changes depending on the break size? Please clarify.

6.b) It is also stated that "The moderator temperature coefficient (MTC) is determined by sensitivity analyses within the range of the least negative beginning-of-cycle value and the most-negative end-of-cycle value." What kind of sensitivity analysis is performed on the MTC, and how does that sensitivity analysis inform the SLB analysis?

SNPB-RAI-7) In Section 5.2.2.2, both the HZP and HFP SLB cases are revised such that the three-dimensional power distribution will be used as an input to the centerline fuel melt limit evaluation. Please describe in more detail how this will be accomplished.

SNPB-RAI-8) Section 5.2.3.2, described in Change 5-6, proposes the use of the WRB-2M and BWU-N critical heat flux (CHF) correlations for HFP SLB analyses and the W-3S and WLOP CHF correlations for HZP SLB analyses.

8.a) What statistical methodology is used for the BWU-N departure from nucleate boiling ratio (DNBR) limit? The WRB-2M correlation is explicitly used with the statistical core design (SCD) methodology, but the limit

listed for BWU-N is simply a correlation limit. Considering that both correlations are being used to analyze the same event, why do they not use the same statistical methodology?

8.b) The WLOP correlation is being added as an alternate for the W-3S correlation. How does Duke propose to determine which correlation to use? Will one be a primary correlation, with the other used in applications outside the parameter range of the primary?

8.c) The DNBR correlation limit for the WRB-2M correlation is listed in Technical Specifications (TS) Section 2.1.1, "Reactor Core Safety Limits," for both Catawba and McGuire. Please explain why the limits for BWU-N, W-3S, and WLOP are not also required to be listed in the TS.

SNPB-RAI-9) Section ~~5.2.3.1~~ 5.3.3.1[1] is revised to add a methodology to compute excore flux detector error due to overcooling for the HFP SLB case; Section 6.3.2.3 is revised in the same way for the Dropped Rod analysis. In both sections it is stated that "A conservative attenuation factor is assumed as a function of the change in reactor vessel downcomer density resulting from the change in temperature." How is this attenuation factor chosen or calculated?

NRC Staff Note: During a conference call to discuss the draft RAIs, Duke noted that they had previously docketed information responsive to this question in a letter dated May 21, 2003 (ADAMS Accession No. ML031490471). This letter is also proprietary to Duke and, as such, is not publically available. The NRC staff reviewed the information contained in the May 21, 2003 letter and has no further informational needs with respect to this RAI.

SNPB-RAI-10) **PROPIN:** There are a number of changes to the temperature feedback model for the HZP power case in Section 5.3.2.6, described in Change 5-8. One key change is [[

]]

10.a) Provide clarification on [[

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10.b) In what way is [[

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10.c) The technical justification for Change 5-8 states: [[

[1] Error on reference to section number noted by licensee.

]]

10.d) In determining what is conservative, are the [[

]]

10.e) In the sample SLB analysis performed using the revised methodology, Table 5-1 states that criticality occurs at 25 seconds, but Figure 5-7 seems to indicate criticality occurring at approximately 20 seconds. Please clarify the apparent discrepancy.

SNPB-RAI-11) Change 6-1 revises Section 6.1.3 to remove language specifying a particular fuel performance code, replacing it with language saying that an "appropriate" fuel performance code will be used. How will the appropriateness of the fuel performance code be ensured? What will be done for transition cores where more than one fuel performance code may be applicable?

SNPB-RAI-12) Change 6-2 revises Section 6.3.1 to change the initial condition for the dropped rod analysis to use minimum rather than maximum average fuel temperatures. Please justify why it is conservative to use minimum values.

SNPB-RAI-13) Change 6-8 revises Section 6.3.2.1 so that weighting factors are now used to account for the relative importance of the fuel assemblies in generating the excore detector response. How are these weighting factors determined?

SNPB-RAI-14) Change 6-9 revises Section 6.3.2.2 to include the high flux, over-temperature ΔT, and over-power ΔT trips for the dropped rod analysis. What has changed in the methodology that now allows these trips to be credited?

SNPB-RAI-15) In Figure 6-10, which is a flowchart of the dropped rod accident analysis, the box labeled "Fuel Melt Limits" is not connected to any other boxes. How do the fuel melt limits factor in to the analysis?

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If you have any questions regarding this matter, I may be reached at (301) 415-2481 or by e-mail at ed.miller@nrc.gov.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413, and 50-414

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