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

June 21, 2013

Mr. Scott Batson
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT APPLICATION RELATED TO REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMIT CURVES (TAC NOS. MF0763, MF0764, AND MF0765)

Dear Mr. Batson:

By letter dated February 22, 2013, Duke Energy Carolinas, LLC (Duke), submitted an application for a proposed amendment for the Oconee Nuclear Station, Units 1, 2, and 3, which would revise the Technical Specifications for the Reactor Coolant System pressure-temperature limit curves.

The Nuclear Regulatory Commission (NRC) staff is reviewing the 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). On June 10, 2013, the Duke staff indicated that a response to the RAI would be provided within 60 days of the date of this letter.

If you have any questions, please call me at 301-415-2901.

Sincerely,

John P. Boska

John P. Boska, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

<u>REQUEST FOR ADDITIONAL INFORMATION (RAI)</u>

<u>REGARDING TECHNICAL SPECIFICATION CHANGE REQUEST</u>

<u>FOR PRESSURE-TEMPERATURE LIMIT CURVES</u>

<u>DUKE ENERGY CAROLINAS, LLC</u>

<u>OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3</u>

<u>DOCKET NOS. 50-269, 50-270 AND 50-287</u>

By letter dated February 22, 2013, Agencywide Documents Access and Management System Accession No. ML130580060, Duke Energy Carolinas, LLC (Duke), submitted an application for a proposed amendment for the Oconee Nuclear Station, Units 1, 2, and 3, which would revise the Technical Specifications for the Reactor Coolant System pressure-temperature limit curves. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

RAI-1 (generic)

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, Paragraph IV.A states that, "*The pressure-retaining components of the reactor coolant pressure boundary [RCPB] that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth* ... [in paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]." Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel (RV) beltline (neutron fluence $\geq 1 \times 10^{17}$ n/cm^2, E > 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence $< 1 \times 10^{17}$ n/cm^2, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the American Society of Mechanical Engineers Code (ASME Code), Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all reactor coolant pressure boundary (RCPB) components specified in Section III, NB-2332(b).

The P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials due to the following factors:

1. RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.

2. Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P-T limits than those for the RV beltline shell materials.

Enclosure

Consequently, please describe how the current P-T limit curves at 54 effective full power years (EFPY) for the Oconee units and the methodology used to develop these curves considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G in the proposed update. Your description shall include the following:

- Using a proposed composite heatup curve and a proposed cooldown curve as examples, point out the segments that were limited by closure head, outer nozzle, and beltline.
- Confirm availability of material data (initial RT_{NDT} and copper and nickel contents) for all non-beltline materials for all three unit RPVs and demonstrate that none of them will become limiting under the 54 EFPY fluence.
- Confirm that the lowest service temperatures (LSTs) for all Ferritic RCPB components that are not part of the RV have been established for all three Oconee units, and the lowest temperature of 60 °F in the proposed P-T limits are higher than these LSTs.

RAI-2

The fluence values at the one-quarter thickness (1/4T) and 3/4T of the RV in Table 3-1 of ANP-3127, Revision 1 (the ANP, Enclosure 2 of the February 22, 2013, submittal) appear to not be based on the same RV thickness for the same unit. Please confirm that you used Equation (3) in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to calculate the fluence at 1/4T and 3/4T of the RV. Please also provide the values that you used for "x" in Equation (3) in the ART calculations at 1/4T and 3/4T for the limiting materials shown in Table 3-1 and the RV radius that you used in the P-T limit calculations. Provide the same information for Units 2 and 3.

RAI-3

Table 3-2 of the ANP indicated that the copper value of the lower nozzle belt to the upper shell circumferential weld of Heat 406L44 for Unit 2 is 0.27% and for the same type of weld of Heat 821T44 for Unit 3 is 0.24%, both are 0.01% lower than the corresponding values in the approved license renewal application. Since the accumulative effect of several small changes over the years can be significant, please provide justification for the revision.

RAI-4

Section 4.2 of the ANP states that, "A ¼ t_{NB} (t_{NB} - the thickness at the nozzle belt) deep corner flaw is postulated on the inside surface of the reactor vessel inlet and outlet nozzles and core flood nozzle corner." This suggests that your nozzle methodology is more complete than that of BAW-10046, Revision 2, which performed analysis on outlet nozzles only, based on the belief that the outlet nozzle is the most limiting nozzle. Please confirm that actual calculations have been performed on inlet, outlet, and core flood nozzles to determine the most limiting P-T limits for nozzles in this license amendment request (LAR).

RAI-5

Section 4.4 of the ANP states, "The Pressure-Temperature limits derived for the reactor vessel head-to-flange conservatively bounds the minimum required temperature requirements as given in Table 1 of the Appendix G to 10 CFR Part 50." Please use a figure from Figure 7-1 to Figure 7-9 to support and explain this statement.

RAI-6

Section 4.6 of the ANP indicated that both ramped and stepped transient definitions are modeled for normal operation heatup and cooldown. Please confirm whether the thermal stresses are based on (1) the stepped transient, which is likely to be limiting, or (2) the stepped and the ramped transients depending on time. If Case (2) applies, please provide a discussion of why the stepped transient may not be limiting.

RAI-7

Section 6 of the ANP discussed pressure corrections with the ΔP listed in Table 6-1. Please confirm whether this ΔP should be added to or subtracted from the calculated pressure values based on the ASME Code, Section XI, Appendix G methodology. Likewise, how do you adjust the calculated 1/4T metal temperature to the "indicated reactor coolant system (RCS) inlet temperature?"

RAI-8

Figures 7-1 and 7-2 of the ANP illustrate the P-T limits for heatup and cooldown, along with pressure temperature pairs of typical points along the P-T limit curves (on the left side of Figures 7-1 and 7-2). To assist the NRC staff verify the proposed heatup and cooldown curves, please use these pressure temperature pairs as examples and provide the corresponding thermal stress intensity factors (K_{It}s).

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Sincerely,

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John P. Boska, Senior Project Manager
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ADAMS Accession No.: ML13165A147 ***See memo dated 5/30/13**

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NAME	JBoska	SFigueroa	SRosenberg	RPascarelli	JBoska
DATE	6/17/13	6/19/13	5/30/13	6/21/13	6/21/13

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