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

May 6, 2020

Mr. J. Ed Burchfield, Jr.
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 RE: REVISION OF LICENSING BASIS FOR HIGH ENERGY LINE BREAKS OUTSIDE OF THE CONTAINMENT BUILDING (EPID L-2019-LLA-0184)

Dear Mr. Burchfield:

By letter dated August 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19240A925), Duke Energy Carolinas, LLC. (Duke Energy, or the licensee) submitted a license amendment request (LAR) for Oconee Nuclear Station, Units 1, 2, and 3 (Oconee). The LAR proposes to revise the Oconee current licensing basis for high energy line breaks (HELBs) outside of the containment building.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed as discussed in the Enclosure. During a clarification call on May 1, 2020, Mr. Tim Brown of your staff agreed that Duke Energy would respond within 45 days of the date of this letter. Please note that the NRC staff's review is continuing and further requests for information may be developed.

The request for additional information (RAI) contains proprietary information as originally submitted in the August 28, 2019, license amendment request. Proprietary information is identified by text enclosed within double brackets as shown here **[[ ]]**. A non-proprietary version of the RAI is also enclosed.

Enclosure 1 transmitted herewith contains SUNSI. When separated from Enclosure 1, this transmittal document and Enclosure 2 is decontrolled.

If you have any questions, please contact me at 301-415-3867 or via e-mail at Michael.Mahoney@nrc.gov.

Sincerely,

/RA/

Michael Mahoney, 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

Enclosures:
1. Request for Additional Information (Proprietary)
2. Request for Additional Information (Non-Proprietary)

cc: w/o enclosure 1: Listserv

ENCLOSURE 2

NON-PROPRIETARY VERSION

REQUEST FOR ADDITIONAL INFORMATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DUKE ENERGY CAROLINAS, LLC.

LICENSE AMENDMENT REQUEST TO REVISE THE LICENSING BASES FOR HIGH ENERGY LINE BREAKS OUTSIDE OF THE CONTAINMENT BUILDING

DOCKET NOS. 50-267, 50-270, AND 50-287

PROPRIETARY VERSION

REQUEST FOR ADDITIONAL INFORMATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DUKE ENERGY CAROLINAS, LLC.

LICENSE AMENDMENT REQUEST TO REVISE THE LICENSING BASIS FOR HIGH

ENERGY LINE BREAKS OUTSIDE OF THE CONTAINMENT BUILDING

DOCKET NOS. 50-269, 50-270, AND 50-287

By letter dated August 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19240A925), Duke Energy Carolinas, LLC. (Duke Energy, or the licensee) submitted a license amendment request (LAR) for Oconee Nuclear Station, Units 1, 2, and 3 (Oconee). The LAR proposes to revise the Oconee current licensing basis for high energy line breaks (HELBs) outside of the containment building.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed as discussed in the Enclosure.

Regulatory Basis:

Oconee's UFSAR, Section 3.1.6, "Criterion 6 - Reactor Core Design (Category A)," states, "The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power."

UFSAR, Section 3.1.9, "Criterion 9 - Reactor Coolant Pressure Boundary (Category A)", states, "The reactor coolant pressure boundary shall be designed and constructed to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime."

Title 10 of the *Code of Federal Regulations* (CFR), Section 50.46(b)(4), "Coolable geometry," states, "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The request for additional information (RAI) contains proprietary information as originally submitted in the August 28, 2019, license amendment request. Attachment 4 of the August 28, 2019, letter, is the proprietary (non-public) version of "Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," and Attachment 5 is the non-proprietary (public) version. Proprietary information is identified by text enclosed within double brackets as shown here **[[ ]]**. A non-proprietary version of the RAI is also enclosed.

RAI-1 (Nuclear Systems Performance Branch (SNSB))

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," Section 1.1, third paragraph states:

> RELAP5/MOD2 is selected for these analyses based on the potential for sustained two-phase conditions in the RCS [reactor coolant system] piping. The MS [main steam] HELB analysis can result in sufficient overcooling that leads to two-phase conditions in the RCS piping which can potentially interrupt natural circulation. To accurately predict this phenomenon the RELAP5/MOD2 code was selected to perform this analysis. The FDW [Feedwater] HELB analyses can also result in sustained two phase conditions, indicating that RELAP5 based methods are more appropriate to perform the analysis. RETRAN based methods are selected for analyses where sustained two phase conditions are not expected.

As stated above, two phase conditions in the RCS are expected in both MS and FDW HELB overcooling and overheating analysis respectively. RELAP5/MOD2 is suitable for analysis in which the two-phase conditions are expected for accurately predicting the results, and RETRAN should be selected where two-phase conditions are not expected. For the thermal-hydraulic (T-H) analysis described in Section 3.0 and 4.0 of Attachment 6, specify the code used for each analysis. In case RETRAN was used for any of the analyses, justify that sustained two-phase condition in the RCS did not exist as this code is not suitable for predicting accurate results for this condition.

RAI-2 (SNSB)

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," Section 3.1 "Ambient Heat Losses", first paragraph, last sentence states:

[[

]]

Describe the **[[**

]]

RAI-3 (SNSB)

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," Section 3.2 describes reactor vessel (RV) head axial conduction modification in the RELAP5/MOD2-B&W ONS base model described in DPC-NE-3003-PA, Revision 1. This modification is not described in any of the Subsections under Section 2.1 of Attachment 5. Provide responses to the following:

(a) Describe the **[[** **]]**, and what each one represents.

(b) Since this modification is not included in those listed in any of the Subsections under Section 2.1 of Attachment 5, confirm this modification was performed in the RELAP5/MOD2-B&W ONS base model for the overheating analysis.

(c) Since this feature was not included in the RELAP5/MOD2-B&W ONS base model, justify the **[[** **]]** for overheating analysis.

(d) The last sentence in Attachment 5, Section 3.2 states:

> The RV upper head includes numerous axial structures, a portion of which are modeled to allow heat transfer across node boundaries.

Explain why there is a need for additional heat structure even though the existing ones allow heat transfer across node boundaries.

RAI-4 (SNSB)

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," Sections 3.7 and 5.0, first paragraph in both sections' state:

> The modeling approach for several of these features considers the impact of asymmetric loop conditions on the performance of the individual boundary condition.

Explain what is meant by "impact of asymmetric loop conditions on the performance of individual boundary conditions". Provide examples of asymmetric loop conditions and the boundary conditions that are impacted.

RAI-5 (SNSB)

Attachment 6, for scenarios analyzed in Sections 3.1.1 and 3.1.2 in which the minimum nucleate boiling ratio (DNBR) is required to meet the acceptable fuel design limits, provide the following:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the parameter showing that the minimum departure from DNBR meets the specified acceptable fuel design limits.

RAI-6 (SNSB)

Attachment 6, for scenarios analyzed in Sections 3.1.1, 3.1.2, 3.1.3, 3.2, and 3.3, provide the following for the RCS pressure analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the RCS pressure developed comparing it with 110% of design pressure.

(d) Please provide the following figures in the response:

1. **[[**
2.
3. **]]**

(e) Confirm that the **[[**
]] are conservative. These tables are not necessary to be added in the response.

RAI-7 (SNSB)

Attachment 6, for scenarios analyzed in Sections, 3.1.1, 3.1.2, 3.1.3, 3.2, and 3.3, provide the following for the core coolable geometry analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs of the parameters involved and their numerical limiting values showing the core remains intact and the coolable geometry is maintained.

(d) During a clarification call with the licensee it was stated: The collapsed liquid level above the core and RVLIS [reactor vessel level indicating system] indications are tracked as a function of time to validate the core remains covered for overcooling events. Please provide the following:

1. From the above statement it appears that the reactor level is tracked during the event. If so, how, where, and according to which procedure?
2. Explain how the analysis validates that the core remains covered during the event. Provide analysis results of the reactor level showing that the core remains covered during overcooling events.

RAI-8 (SNSB)

Attachment 6, for scenarios analyzed in Sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2 in which the minimum DNBR is required to meet the acceptable fuel design limits, provide the following:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the parameter showing that the minimum DNBR meets the specified acceptable fuel design limits.

(d) Please provide following figures:

1. **[[**
2.
3.
4. **]]**

RAI-9 (SNSB)

Attachment 6, for scenarios analyzed in Sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2, provide the following for the RCS pressure analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values showing the RCS remains within acceptable pressure and temperature limits.

(d) Please provide the following figures:

1. **[[**
2.
3.
4.
5. **]]**

RAI-10 (SNSB)

For all the scenarios analyzed in Reference 1, Attachment 6, Sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2, provide the following for the steam generator (SG) tube stress analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Maximum induced stress in the SG tubes and the allowable stress for the SG tube material.

RAI-11 (SNSB)

Attachment 6, Section 3.2 states:

> Pressurizer level increases with increasing RCS temperatures and goes off-scale high. The pressurizer does not become water-solid prior to SSF [standby shutdown facility]

> ASW [auxiliary service water] being aligned to the SGs at 14 minutes, and liquid relief is not predicted through the PSVs [pressurizer safety valves] or PORV [power operated relief valve]. Maintaining a steam space in the pressurizer is dependent on the timing of providing a heat sink.

Attachment 6, Section 3.3 states:

> Pressurizer level increases with increasing RCS temperatures and goes off-scale high. The pressurizer does not become water solid prior to PSW [protected service water] being aligned to the SGs, and liquid relief is not predicted through the PSVs or PORV when PSW is started within 14 minutes. Maintaining a steam space in the pressurizer is dependent on the timing of providing a heat sink.

In above statements, it is predicted that the pressurizer would not become water-solid prior to the alignment of SSF ASW or PSW system for the mitigation of FDW HELB in the turbine building (TB) during the overheating event so that liquid relief would not take place through the PSVs or PORVs. Explain how it is determined that the pressurizer would not become water-solid. Describe the analysis and results on which this prediction is based.

RAI-12 (SNSB)

Attachment 6, Section 3.2 and Section 3.3 states the analysis is based on "(ANS [American National Standard]-79 with uncertainty)" decay heat. The decay heat and its uncertainty are not specified in any of the remaining analyses.

(a) Specify what uncertainty value was used in the analysis.

(b) State the decay heat standard with its uncertainty value used for the remaining thermal-hydraulic analysis described in Attachment 6.

(c) Please provide following figures:

1. **[[ ]]**
2. OSC-11638 Figure 9.3.7-4

RAI-13 (SNSB)

Attachment 6, Section 4.1.2, fourth paragraph, page 7 states:

> Since the RCPs continue to operate in this case, the DNBR is bounded by the double MS HELB without offsite power case where RCPs [reactor coolant pumps] are lost immediately.

Describe the relationship of the DNBR with RCP in operation. Explain why the DNBR for a double MS HELB with offsite power available is bounded with offsite power not available during the same scenario.

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