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

August 31, 2012

Mr. Preston Gillespie
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - REQUEST FOR ADDITIONAL INFORMATION AND SUSPENSION OF REVIEW OF LICENSE AMENDMENT REQUEST FOR POWER UPRATE (TAC NOS. ME7164, ME7165, AND ME7166)

Dear Mr. Gillespie:

By letter dated September 20, 2011, Duke Energy Carolinas, LLC (Duke or the licensee) submitted a license amendment request (LAR) to the Nuclear Regulatory Commission (NRC) for a measurement uncertainty recapture (MUR) power uprate for Oconee Nuclear Station, Units 1, 2, and 3 (ONS 1/2/3). By letter dated July 31, 2012, Duke stated that the implementation of the new protected service water (PSW) system had developed some issues, and requested that the schedule for implementation of the PSW system be delayed by two years. The PSW system is credited in the MUR LAR (e.g., for mitigation of certain high-energy line breaks). As the PSW system had been scheduled to be operational before the end of 2012, the NRC staff had proceeded with the MUR LAR with that understanding. Now that the PSW system has been delayed, the NRC staff finds that it is not possible to issue an MUR power uprate amendment without credit for the PSW system. Therefore, the NRC staff has suspended the review of the MUR amendment.

Attached is a request for additional information (RAI) on the MUR LAR that had been developed prior to the suspension of the review. When the PSW system is operable per the ONS 1/2/3 Technical Specifications, please submit the response to this RAI and the NRC staff will resume the review of the MUR LAR.

If you have any questions, please contact me at 301-415-2901 or via e-mail at John.Boska@nrc.gov.

Sincerely,

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, 50-287

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REGARDING MUR POWER UPRATE

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

By letter dated September 20, 2011, as supplemented by letters dated November 21, 2011, March 16, 2012, and April 4, 2012, Duke Energy Carolinas, LLC (Duke or the licensee) submitted a license amendment request (LAR) to the Nuclear Regulatory Commission (NRC) for a measurement uncertainty recapture (MUR) power uprate for Oconee Nuclear Station, Units 1, 2, and 3 (ONS 1/2/3). The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

RAI 35 SRXB

The licensee identified two postulated events, which are not covered by the reload analyses, and which would require re-evaluation. In the LAR, these events were identified as the high energy line break and the double main steam line break (MSLB). In response to Request for Additional Information (RAI)-33, the licensee provided additional clarification. The high energy line break consists of several events, including: (1) MSLB; (2) Feed water line break (FWLB); (3) Auxiliary steam for startup.

a) The licensee previously dispositioned the main steam line break by stating that the Final Safety Analysis Report (FSAR) analyses are performed at 102-percent power. In fact, Updated FSAR (UFSAR) Table 15-34 indicates that this is the case. However, the LAR enclosure and UFSAR Table 15-33 both refer to the accident analysis methodology at DPC-3005PA. The methodology states that the MSLB is performed at a nominal power level. Please clarify the apparent discrepancy in this information. In light of the potentially conflicting information, please also provide excerpts from the applicable calculation notes to demonstrate which analytic treatment is being applied to this event.

b) The LAR and RAI response refer to OS-73.2 as the licensing basis for the FWLB, noting that the power level for the analysis is not clearly defined. On page E2-40 of the submittal document, it is stated that emergency feedwater (EFW) injection is required within 15 minutes and high pressure injection (HPI) is required within 60 minutes. It is also stated that there is no assurance that HPI would be available within the required time.

In response to RAI 33, the licensee stated that a RELAP-5 main feedwater line break analysis assuming EFW restoration in 30 minutes and HPI restoration in 8 hours successfully demonstrates that general design criterion 6 is satisfied. The RAI response also notes that this analysis is not the licensing basis analysis, because the NRC staff has not reviewed and approved it.

The submittal and the RAI response consider the RIS 2002-03 requested information regarding the main feedwater line break, considering the event in terms of both a postulated event that is within the bounds of the current reload methodology that could be re-analyzed, and a postulated event not within the scope of the reload methodology. It appears that the event requires analysis, but that the reload analytic methodology does not apply to the event.

The licensee has not provided the following information concerning an analysis that establishes an acceptable licensing basis for measurement uncertainty recapture (MUR) conditions with respect to the main feedwater line break:

i) Important analytic inputs and assumptions.
ii) Confirmation that the analysis was performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology.
iii) Description of the sequence of events and explicit identification of the events that would change as a result of the power uprate.
iv) Description and justification of the chosen single failure assumption.
v) Plots of important parameters.
vi) Results and acceptance criteria for the analyses, including any changes from the previous analyses.

Given that RELAP-5 is approved to calculate mass and energy releases for postulated loss of coolant accidents, it is not clear that the RELAP-5 method is NRC-approved for use to calculate the effects of a postulated feed water line break at Oconee. Because the RELAP-5 calculation extends for a period of 8 hours, it is also not clear whether the code is being used to calculate the effects of a postulated accident, or whether the code is being used to evaluate acceptable operator actions to mitigate an accident.

Please provide an acceptable analysis of the feedwater line break that applies to the MUR conditions. The analysis should include a minimum of operator actions, and should provide the immediate results of the accident. The analysis should also confirm that the plant is left in a safe end state, and that the radiological consequences of any fuel or reactor coolant pressure boundary failures are acceptable.

If presently proposed hardware modifications are required to produce an acceptable analysis, please consider deferring the MUR request until the NRC staff has completed its review of the proposed hardware modifications.

c) Please provide similar information concerning the double MSLB, the results for which are discussed in response to RAI 33. The staff requires the information listed in Item III.3 of RIS 2002-03 in order to review the analysis to determine whether it is acceptable with respect to plant operation at the proposed, uprated power level.

RAI 36 SRXB

Reactor vessel neutron fluence is discussed in Section IV.1.C.ii of the LAR. Please provide the following clarifying information:

a) The evaluation refers to "fluence rates." Please define this term and explain how the fluence rate values were derived or approximated. Explain whether one could reasonably interchange this term with flux, and clarify any differences.

b) The discussion states that fluence values were calculated using core follow information. Please describe the process that was used to obtain the core follow information, and how the core follow information was input to the fluence calculation.

c) The application states, "The pre-MUR time period used was the cycle(s) after the last full fluence transport analysis was performed for the cycle the MUR is assumed to start in." Please explain whether fluence transport is analogous to neutron transport, and further clarify the meaning of this sentence.

d) The application excerpts the following from a BAW-2241 safety evaluation:
BAW-2241 was approved by NRC letter from Frank Akstulewicz to J.J. Kelly (B&W Owner's Group), "Acceptance for Referencing of Licensing Topical Report BAW-2241-P, "Fluence and Uncertainty Methodologies" (TAC No. M98962)," undated.

The application also states that the fluence calculations adhere to Regulatory Guide (RG) 1.190; however, the TAC No. above likely predates RG 1.190's issuance. Please provide the date that the neutron transport calculations were performed, and the revision of BAW-2241(P) that best describes the neutron transport analysis.

RAI 37 SRXB

If the fluence values supporting the MUR are less than those that support current licensing basis material evaluations such as pressure and temperature limits, low temperature overpressure protection system setpoints, or those provided to the NRC to support license renewal, please explain the differences.

RAI 38 EEEB

In its response to RAI-1.b, the licensee stated that "in order to prevent the Unit 1 and 2 Isolated Phase Bus (IPB) duct from reaching their temperature alarm set points in the summer months, a temporary modification is performed that provides chilled water to a heat exchanger."

a) Clarify why the modification is temporary and provide a commitment to develop and implement a permanent long-term solution that addresses MUR conditions.

b) Is the forced cooling system of the Unit 3 IPB permanent? Explain whether the additional measures will prevent the Unit 3 IPB from reaching their temperature alarm setpoints in the summer months.

RAI 39 EEEB

In its response to RAI-8, on page 11 and 13, the licensee stated that non-safety systems are not subject to the EQ requirements. This statement is related to the temperature and pressure conditions for the Integrated Control System on page 11, and the radiation condition in the

Radwaste Building on page 13, which the licensee identified as being non-safety systems. The licensee, in Enclosure 2, Section V.1.C, of the license amendment request, stated that the ONS environmental qualification (EQ) Program is guided by the regulations detailed in 10 CFR 50.49. 10 CFR 50.49 (b) defines electrical equipment important to safety covered by 10 CFR 50.49, and includes non-safety related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions of the safety-related equipment.

Provide a discussion to clarify that the non-safety related items mentioned above meet the 10 CFR 50.49 (b)(2) requirement for the EQ of non-safety related equipment.

RAI 40 EVIB

A number of entries in Tables IV.1.C-1 and IV.1.C-2 of the submittal dated September 20, 2011, on reactor vessel materials, neutron fluence, and related parameters, do not appear to align with previously submitted values. Indicate and clarify the origin of all information presented in these tables. Specify which values have previously been reported to the NRC, when, and where. Also specify if any values are being submitted to the NRC for the first time in this application.

RAI 41 EICB

In the submittal dated September 20, 2011, Enclosure 2, Item I.1.D, Criterion 1 from ER-1 57P requires the licensee to justify continued operation at the pre-failure power level for a predetermined time and the decrease in power that must occur following that time. The response provided in the LAR states that an engineering evaluation was performed to justify an allowed outage time (AOT) upon loss of the signal from the leading edge flow meter (LEFM). Also, the response provided for this criterion states that the analysis performed established a bounding uncertainty of 0.037% RTP, rounded to 0.04% RTP, over a 7-day period for Oconee Unit 3 at operating levels above 90% RTP. This result would allow Oconee to maintain the new power level for up to 7 days following a failure of the LEFM.

In its RAI response dated March 16, 2012, Duke explained how they performed the bounding analysis to calculate the 0.04% bias (Cameron report ER-932), to allow Oconee to maintain the new power level when the LEFM is degraded.

For previous MUR applications, the NRC staff position has been to allow licensees to maintain the new power level for up to 72 hours when the LEFM failed, which is consistent with Cameron's analysis and recommendation to operate with a failed LEFM. Further, this AOT is consistent with the already established and well understood timeframe within industry to allow repair or replacement of an inoperable LEFM. Please explain specifically why Duke needs additional time, seven days at the higher power levels, when the equipment that justifies operation at this higher level is inoperable.

RAI 42 EICB

a. In document ER-972, revision 2, Appendix A, item 7 does not match with the data in the approved topical report, ER-157P. Please explain this discrepancy.

b. In document ER-972, revision 2, Appendix B, explain why some values are inconsistent with the values in the approved topical report. For example, items 11, 12, and 13 are listed as negative values, while the topical report lists positive values. Please explain this discrepancy.

c. In document ER-972, revision 2, Appendix A.5, item 5, a clock accuracy of 0.02% is used. However, the assumption states that the clock accuracy is 0.01%. Please explain this discrepancy.

RAI 43 SRXB

Please provide the following information:

a. Distances from the LEFM to the next downstream non-straight pipe element in the Alden Laboratory test setups.

b. Plant feedwater drawings similar to the drawings provided to illustrate the Alden Laboratory test setup drawings. These drawings should cover from the feedwater pumps to about 10 pipe diameters downstream of the CheckPlus instruments. If such drawings are not readily available, then provide plant isometric drawing showing the CheckPlus locations for the same distance.

If you have any questions, please contact me at 301-415-2901 or via e-mail at John.Boska@nrc.gov.

Sincerely,

/ra/

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, 50-287

Enclosure:
Request for Additional Information

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