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

April 28, 2016

MEMORANDUM TO: Michael T. Markley, Chief
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Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Jeffrey A. Whited, Project Manager
Plant Licensing Branch II-1
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Office of Nuclear Reactor Regulation

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST TO REVISE TS 5.5.2 "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MF7265 AND MF7266)

The attached request for additional information (RAI) was transmitted on April 22, 2016, to Duke Energy Carolinas, LLC (the licensee). This RAI was transmitted in order to clarify the licensee's license amendment request (LAR) submitted on January 18, 2016, for the Catawba Nuclear Station Units 1 and 2 (Catawba). In its LAR the licensee proposes to revise Catawba Technical Specification Section 5.5.2, "Containment Leakage Rate Testing Program", for Permanent Extension of Type A and Type C Integrated Leakage Rate Test Frequencies.

The attached RAIs were sent to the licensee to ensure that the questions are understandable, the regulatory basis for the questions is clear, and to determine if the information was previously docketed. The licensee intends to respond to these RAIs within 45 days of the date of this memorandum. This memorandum and the attachment do not convey or represent an NRC staff position regarding the licensee's request.

Docket Nos. 50-413 and 50-414

Attachment:
Draft RAI

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST

CONTAINMENT LEAKAGE RATE TESTING PROGRAM

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

By letter dated January 18, 2016,[1] Duke Energy Carolinas, Inc., (Duke, the licensee) submitted a license amendment requested (LAR) for the Catawba Nuclear Station, Units 1 and 2 (Catawba). The LAR proposes to revise Catawba Technical Specification Section 5.5.2, "Containment Leakage Rate Testing Program", for Permanent Extension of Type A and Type C Integrated Leakage Rate Test (ILRT) Frequencies. Specifically, the LAR proposes to increase the existing Type A ILRT program test interval from 10 years to 15 years and to increase the existing Type C containment isolation valve leakage testing frequency from 60 months to 75 months. Responses to the request for additional information (RAI) questions listed below are needed to support the U.S. Nuclear Regulatory Commission (NRC) staff's continued technical review of the proposed LAR.

RAI-01: The following question refers to Section 3.7, "Evaluation of Risk Impact," of the enclosure to the licensee's LAR.

Limitation/Condition 2 in Table 3.7.1-1 on page 61 of the enclosure to the licensee's LAR, states that the increase in population dose for Catawba is 0.026 person-rem/y;[2] and that in Conditional Containment Failure Probability (CCFP) is 0.502%. Table 6-1 and Section 7.0 in Attachment 5, "Evaluation of Risk Significance of Permanent ILRT Extension," 54003-CALC-02, of the licensee's LAR, give corresponding values of (1) for the extension to 1 in 15 years vs. the base case of 3 in 10 years, 0.109 person-rem/yr (1.47%) and 0.888%, respectively; and (2) for the extension to 1 in 15 years vs. the extension to 1 in 10 years, 0.0453 person-rem/y (0.61%) and 0.370%, respectively. On pages 65-66 of the enclosure to licensee's LAR, the reported results are as follows: (1) large early release frequency (LERF) increase = 1.13E-7/y (Electrical Power Research Institute (EPRI) guidance) or 1.14E-7/y (including effect of steel liners) for a change in test interval from 3 in 10 years to 1 in 15 years; (2) baseline LERF = 1.12E-6/y; (3) increase in total integrated plant risk = 0.026 person-rem/y;[3] and (4) increase in CCFP = 0.502% for a change in test interval from 3 in 10 years to 1 in 15 years. All these differ from the results reported in Table 6-1 and Section 7.0 in Attachment 5, 54003-CALC-02, each of which is higher there, as follows: (1) LERF increase = 4.68E-7/y from 3 in 10 years to 1 in 15 years; (2) baseline LERF = 1.67E-6/y; (3) increase in total integrated plant risk = 0.109 person-rem/yr (1.47%) from 3 in 10 years to 1 in 15 years; and (4) increase in CCFP = 0.888% from 3 in 10 years to 1 in 15 years.

[1] Agencywide Documents Access and Management System (ADAMS) Accession Number ML16026A048.
[2] This value is also reported on Page 69 in Section 4.3.
[3] This value is also reported on Page 69 in Section 4.3.

a. Which values are correct and are the conclusions regarding the acceptability of the ILRT extension affected?

RAI-02: The following questions refer to Attachment 5 to the licensee's LAR. The pages noted at the beginning of each question refer to a page in Attachment 5 to the licensee's LAR.

a. Page 9. What is the basis for assuming no impact on the reliability of containment isolation valves to close when demanded by an isolation signal given the test interval is increased? If there is an impact, but the assumption is that it is negligible, provide justification, preferably quantitative.

b. Page 16. Accident sequences involving large and small isolation failures, including "failure-to-seal" events for the latter, are cited as not being affected by the ILRT frequency change. Do any of these failures potentially result from components whose failure probability is test-interval dependent? If not, confirm. If so, justify the statement that there is no effect. Page 19. This assumption is repeated for Class 6 Sequences, citing dominance due to misalignment of containment isolation valves following test/maintenance.

c. Pages 28-29. A seismic core damage frequency (CDF) of 1.65E-5/y from the Catawba Individual Plant Examination of External Events (IPEEE) is cited. More recent updates per letter dated September 2, 2010, "Safety/Risk Assessment Results for Generic Issue 199,"[4] estimate a seismic CDF using the 2008 United States Geological Survey (USGS) Seismic Hazard Curves for Catawba of 3.7E-5/y (Table D-1, weakest link model). The Catawba analysis used a seismic CDF of 1.15E-5/y to estimate the Class 3b frequency. If the Generic Issue (GI)-199 results were used instead, the results would be as shown below.

Class	Multiplier	Failure Rate	Seismic CDF	LERF/CDF Ratio	Frequency
3b	1.00E+00	2.29E-03	3.70E-05	3.17E-02	8.22E-08
3b-10	3.33E+00	2.29E-03	3.70E-05	3.17E-02	2.74E-07
3b-15	5.00E+00	2.29E-03	3.70E-05	3.17E-02	4.11E-07

These exceed the frequencies calculated using the IPEEE by 5.67E-8/y, 1.89E-7/y and 2.83E-7/y for the three intervals, respectively. These result in the following changes to Tables 5-17 and 5-18.

Hazard	3 per 10	1 per 10	1 per 15	LERF Increase
External	1.71E-07	5.70E-07	8.54E-07	6.83E-07
Internal	1.17E-07	3.90E-07	5.85E-07	4.68E-07
Combined	2.88E-07	9.60E-07	1.44E-06	1.15E-06

[4] ADAMS Accession NO. ML100270582.

Hazard	3 per 10	1 per 10	1 per 15	LERF Increase
External	1.73E-07	5.75E-07	8.61E-07	6.88E-07
Internal	1.17E-07	3.90E-07	5.85E-07	4.68E-07
Combined	2.90E-07	9.65E-07	1.45E-06	1.16E-06

The combined results now slightly exceed the allowed total change in LERF of 1E-6/y for "small" changes. The total LERFs for the two units using the GI-199 results are now as follows:

U1 = 1.67E-6/y + (3.7E-5/y x 0.0317) + 3.41E-6/y +6.48E-7/y + 1.15E-6/y = 8.05E-6/y
U2 = 1.67E-6/y + (3.7E-5/y x 0.0317) + 3.48E-6/y +6.48E-7/y + 1.16E-6/y = 8.13E-6/y

These remain below 1E-5/y. Page 29 of Attachment 5 to the licensee's LAR states, in part, that:

> Although the total change in LERF is somewhat close to the Regulatory Guide 1.174 limit [when calculated using the Seismic CDF from the IPEEE] when external event risk is included, several conservative assumptions were made in this ILRT analysis, as discussed in Sections 4.0, 5.1.3, 5.2.1, and 5.2.4; therefore the total change in LERF is considered conservative for this application.

Given the delta-LERF now slightly exceeds the RG-1.174 threshold for "small" changes, address the role of the cited conservatisms in justifying the acceptability of the LERF increases. Alternatively, provide a reassessment of the seismic risk based on the more recent USGS Seismic Hazard Curves in lieu of that used in the GI-199 reference, such that the RG-1.174 threshold is not exceeded.

RAI-03: The following questions refer to Attachment 1 in Attachment 5 to the licensee's LAR. The pages noted at the beginning of each question refer to a page in Attachment 5 to the licensee's LAR.

a. Page 42. A fact and observation (F&O) for IE-A8 remains Open, but is determined not to impact the ILRT extension because it is "unlikely" that plant personnel interviews would uncover any new initiating events. In performing the "extensive search" for initiating events referenced in the disposition, was simulator experience or other types of experience that might be obtained only through personnel interviews considered?

b. Page 47. Although not cited as an F&O at the time of the 2002 Peer Review, an action required for IE-C14 remains Open due to the need to incorporate updated industry guidance on removing credit for motor operated valves (MOVs) that could impact the LERF. With External Events included, the total LERF is ~ 7E-06/y for each unit (~ 8E-06/y when the GI-199 seismic CDFs are incorporated, as per RAI 02.c). Provide a basis, preferably quantitative, to justify that the expected increase in CDF/LERF with credit for the MOVs removed is small enough that the risk metrics would remain in Region II.

c. Pages 52, 57-60. F&Os for AS-A8, SC-A1 and SC-A2 are Dispositioned, but it appears that confirmation that the results from using a 2000F criterion for core damage vs. 2500°F via the Modular Accident Analysis Program (MAAP) have not been evaluated, at least not via MAAP itself. Other justification for not revising the criterion for Catawba (i.e., similarity to McGuire) is cited, but it is not clear that this MAAP confirmation has been performed for the McGuire Nuclear Station, Units 1 and 2, either. Explain if the MAAP confirmation has been done; if not, justify why basing Catawba success criteria on MAAP runs using 2500°F vs. 2000°F for core damage remain valid.

d. Pages 54-55, 136-137. F&Os for AS-B1 and QU-B6 remain Open and cited the need to correct the probabilistic risk assessment (PRA) to correctly account for the dependency of the turbine-driven pump (TDP) on the steam generator tube rupture (SGTR) initiator, scheduled for incorporation into the Rev. 3 PRA. Confirm that this correction has been incorporated. If not, provide a sensitivity analysis of the effect on LERF and metrics for the ILRT extension that incorporates this correction.

e. Pages 56 and 62-63. F&Os for AS-B5 and SC-A4 are Dispositioned. One item of concern was failure to model the degraded condition of the supply to the TDP given an SGTR. The disposition cites an update to reflect the correct success criteria due to SGTR loss of the auxiliary feedwater (AFW) pump. Confirm that this update corrected the deficiency related to the degraded supply to the TDP given an SGTR.

f. Pages 89-90. An F&O for SY-B14 remains Open, although concerns related to high-energy line breaks are considered resolved. However, Revision 2 of Regulatory Guide (RG)-1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,"[5] adds the following example to the supporting requirement (SR): "(h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage." Confirm that consideration of this additional example does not affect the conclusion that the ILRT extension is not impacted.

g. Pages 101-102, 104-105. F&Os for HR-F2 and HR-G4 remain Open, citing the need to incorporate updated information related to operator actions into the PRA model. While citing no significant changes to the success criteria as the basis for negligible impact on the ILRT extension, it is not clear whether there are potentially other aspects of the PRA besides success criteria that might be affected by the needed incorporation of the operator actions. Explain further the conclusion of negligible impact despite the need to still incorporate these operator actions into the PRA model.

h. Pages 109-113. F&Os for DA-A1 and DA-A4 are Dispositioned regarding the use of outdated generic data based on "minor changes to random failure rate[s] of the components [are] not significant in the risk evaluations." However, it is unclear whether more recent generic data sources were reviewed such that the assertion that any changes would be "minor" is justified. Pages 114-116. An F&O for DA-C1 remains Open but appears to address the concern in the previous two F&Os in that it cites use of NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial

[5] ADAMS Accession No. ML090410014.

Nuclear Power Plants,"[6] (through 2010) as the primary data source for generic parameter estimates. If this explanation is applicable to the previous two Dispositioned F&Os, confirm. If not, explain what reviews were performed, even if the generic data were not updated, to confirm that any changes would be "minor."

i. Pages 116-118. An F&O for DA-C2 remains Open, citing collection of plant-specific failure data from Maintenance Rule documents through 2005. This suggests that the current failure data used in the PRA are at least 10 years old. The basis for "negligible impact" on the ILRT extension is that "minor changes" to random failure rates are not significant. Provide a basis for concluding that the plant-specific data used in the PRA, current only through 2005, remain representative of the past 10 years of operation at Catawba such that the conclusion that any changes to failure data remain "minor" is justified. Note: Related to this F&O are three for DA-C11, C12 and C13 on Pages 122-123. While remaining Open due to the need for documentation, please justify all references to the 2005 limit date for collection of plant-specific failure data such that the similar conclusion of "negligible impact."

j. Page 128. Although not cited as an F&O at the time of the 2002 Peer Review, an action, cited as "documentation," required for DA-D5 remains Open. However, it is unclear that this is only a documentation issue, as it discusses the use of a "modified" multiple Greek letter (MGL) method for common-cause failure (CCF) analysis. It is unclear how far from the standard MGL method this "modification" diverges or whether it is adequately representative. Non-mandatory Appendix 1-A of ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008: Standard for Level 1/LERF PRA for Nuclear Power Plant Applications," ASME 2009, discusses PRA Maintenance and Upgrade and cites "new treatment of common cause failure" as a potential type of PRA Upgrade. Explain whether or not this "modified" MGL method constitutes a PRA Upgrade and why. If it constitutes an Upgrade, provide a sensitivity evaluation of its effect until a Focused-Scope Peer Review can be completed.

k. Pages 128-129. Although not cited as an F&O at the time of the 2002 Peer Review, an action cited for DA-D6 remains Open. Although it is assumed that any effect on CCF rates would be minor, thereby negligibly impacting the ILRT extension, this needs to be confirmed by comparing the component boundaries used in the CCF generic estimates with those assumed for the PRA. Confirm that the component boundaries assumed for the PRA assure that the generic CCF estimates are adequate.

l. Pages 138-139. An F&O for QU-D4 remains Open citing the need to compare the Catawba PRA results with those from similar plants. Provide assurance that, at least for those results which are relevant to the ILRT extension, the Catawba results are consistent with similar plants or, where not, the difference can be adequately explained.

m. Pages 149-157. Although considered Dispositioned, numerous F&Os for IFPP-A2 through IFEV-B1 justify an insignificant impact on the ILRT extension due to "internal flooding represent[ing] such a small portion of the internal events risk." However, Tables 5-1 and 5-2 in the enclosure to the licensee's LAR indicate that internal flooding contributes (3.92E-5)/(5.27E-5) = 0.74 to total internal CDF and (5.58E-7)/(1.67E-6) = 0.33 to total

[6] ADAMS Accession No. ML070650650.

internal LERF, i.e., it is the dominant contributor in each case. If, as cited, internal flooding resolutions do not significantly impact the ILRT extension, provide appropriate justifications for all these F&Os being Dispositioned.

n. Pages 160-161. An F&O for FSS-A2 is Dispositioned, but it is unclear whether the clarification from Revision 2 of RG-1.200 was addressed. This clarification adds "including spurious operation" to the requirement to specify failure modes for equipment and cables in the target sets. Confirm that this additional failure mode was addressed.

o. Pages 162-163. An F&O for HRA-D2 (referencing HR-H2) is Dispositioned based on the operator action in question not being a "recovery action" in the context of National Fire Protection Association (NFPA) Standard 805. The NFPA-805 definition of a "recovery action" is not relevant when dispositioning "recovery actions" in the context of PRA/human reliability analysis (HRA). If this action constitutes a "recovery" in the context of PRA/HRA, typically post-processed after cut-set generation, then the requirements of HR-H2 apply. Given this action has been credited, indicate (1) if it is proceduralized and trained on, as required for crediting under HR-H2; (2) if not, provide the basis for crediting it; or (3) why, despite its meeting neither (1) nor (2), it does not pose more than a negligible impact on the ILRT extension.

RAI-04: The following questions refer to Section 3.4 of the Enclosure to the licensee's LAR.

a. Section 3.4.1 of the licensee's LAR discusses the observations made in IN 2010–12, "Containment Liner Corrosion,"[7] and Technical Letter Report on "Containment Liner Corrosion Operating Experience Summary,"[8] Revision 1. Specifically, the licensee discusses actions performed at Catawba in responses to these documents, including an examination of coating failures on the exterior faces of the steel containment vessel (SCV) of both units in 1990. All identified failed coatings were cleaned and reapplied. Specifically, the licensee stated that concrete interface was repaired and removed and moisture barrier caulking was replaced.

   Please confirm, based on recent inspection findings of the SCV, that the previously identified problems have not recurred, or if they have, they are under active monitoring and the degradation/corrosion is under control by measures adopted.

b. Section 3.4.2 of the licensee's LAR discusses the inspection of containment leak chase channel systems at Catawba. In its LAR, the licensee proposes to inspect these leak chase channels by performing a VT-3 visual examination in accordance with procedure "Visual Examination (VT-1 and VT-3) of Metal and Concrete Containment," stating, in part, that these examinations may be scheduled and performed as follows:

   > 100% of the containment interior concrete floors shall be examined during each inspection interval. Approximately 1/3 of the floor surface areas shall be examined during each inspection period to determine the condition of all leak

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[7] ADAMS Accession No. ML100640449.
[8] ADAMS Accession No. ML112070867.

chase channel bronze caps and test channel drain plugs installed in the floor within the examination area.

Please provide additional information to justify why inspection of 100% of the leak chase test channels each interval is acceptable to comply with the containment inservice inspection requirements of 10 CFR 50.55a(g)(4) as opposed to the 100% visual inspection each period as discussed in Information Notice (IN) 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner."[9] Include a summary of the results of any inspections completed to date.

c. Section 3.4.4 of the licensee's LAR discusses Problem Identification Reports from the results of recent containment inspections at Catawba. Based on the description provided, there are several instances where the same or similar problems were identified in two consecutive inspections, once in 2010 for 2EOC18 and another in 2013 for 2EOC20. For example,

    i. CNS evaluation in Indication Number 2–SCVI–0006.2010.1 and Indication Number 2–SCVI–0006.2013.1.
    ii. CNS evaluation in Indication Number 2–SCVI–0007.2010.1 and Indication Number 2–SCVI–0007.2013.1.
    iii. CNS evaluation in Indication Number 2–SCVI–0008.2010.1 and Indication Number 2–SCVI–0008.2013.1.
    iv. CNS evaluation in Indication Number 2–SCVI–0009.2010.1 and Indication Number 2–SCVI–0009.2013.1.

It is not clear from the description whether the problem(s) identified during the first inspection in 2010 were or were not completely rectified at the time of closing Work Order #00986330–01 or if the problem(s) recurred later. Please provide a discussion of whether or not the problem(s) were the original ones or recurred subsequent to the closure of the work order. The information should include the bases for concluding the work was complete under Work Order #00986330–01 in 2010 for 2EOC18. Additionally, please indicate when Work Order #02036636 was opened as the LAR states that this work order was still open during 2013 inspection for 2EOC20.

[9] ADAMS Accession No. ML14070A114.

April 28, 2016

MEMORANDUM TO: Michael T. Markley, Chief
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Office of Nuclear Reactor Regulation

FROM: Jeffrey A. Whited, Project Manager ***/RA/***
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