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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
Response to Request for Additional Information
License Amendment Request
Permanent Extension of Type A and Type C Leak Rate Test Frequencies

By letter dated December 19, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16363A349), Duke Energy submitted the subject license amendment request to the U. S. Nuclear Regulatory Commission (NRC) for approval.

By letter dated May 25, 1917 (ADAMS No. ML17156A563), Duke Energy provided the response to a request for additional information.

By electronic mail dated November 13, 2017 (ADAMS No. ML17317B093), the NRC requested additional information. The enclosed document provides the requested information.

This submittal contains no regulatory commitments.

If you have any questions or require additional information, please contact P.T. Vu of Regulatory Affairs at (980) 875-4302.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 7, 2017.

Thomas D Ray

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Enclosure

A001
NRR

xc:

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License Amendment Request

Response to Request for Additional Information

ENCLOSURE

MNS ILRT RAI-02

Question:

Section 3.1.6.3 on page 16 of Enclosure 1 of the LAR states, "The EPRI methodology used to estimate the increase in LERF is conservative." Section 7.0 on page 35 of Attachment 5 of the LAR also states, "The EPRI methodology used to estimate the increase in LERF is conservative. Therefore, even though the increase in LERF is near the Regulatory Guide 1.174 threshold, the conservative methodology adds margin."

Section 3.1.6.3 on page 16 of Enclosure 1 and page 36 of Attachment 5 of the LAR both state, "Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the MNS risk profile."

Based on the discussion above, address the following:

a. With the total LERF for both Unit 1 (7.83E-6/yr) and Unit 2 (8.59E-6/yr) being close to the 1E-5/yr threshold, confirm that the "conservative methodology" made in this ILRT analysis maintain ΔLERF within Region II of RG 1.174. When citing conservatism in the base PRA model, confirm that calculation of the differential risk for the application is also conservative (i.e., the risk estimated for the before versus after condition uses the same assumptions, etc., except for the change to any basic event values affected by the application, ensuring that the before value is not overestimated such that subtracting it from the after value could underestimate the risk increase).

b. Even if the conservative methodology maintains ΔLERF within Region II of RG 1.174, the change to the MNS risk profile is not considered to be very small, and therefore not insignificant. Justify the reasoning for considering this LAR to be insignificant to the risk profile and to represent a very small change if the LAR is within Region II of RG 1.174.

Response:

Part a:

The conservatism cited applies to the methodology provided in EPRI 1018243 regarding the estimation of the class 3b failure probability and categorizing a class 3b leak as a large, early release. These conservatisms are utilized in both the "before" and "after" conditions (where the "before" and "after" conditions are the base 3 in 10 ILRT interval and the extended 1 in 15 ILRT interval, respectively).

The class 3b failure probability is estimated using the Jeffreys Non-Informative Prior distribution and is considered a conservative estimate of the probability of a large containment leak. The effect of the conservatism is to increase the estimated frequency of a class 3b leak expected from a given ILRT interval. Because the same probability is used in both conditions, changing the method of estimating the class 3b failure probability will result in a similar change to the change in class 3b frequency. For example, if a given plant had a CDF of 1E-06, the "before" class 3b frequency would be 1E-06 times 2.3E-3 (probability of class 3b) = 2.3E-09. The "after" class 3b frequency would be 1E-06 times 2.3E-3 times 5 (for the increase in test interval) = 1.15E-08. The change in class 3b frequency would be 9.2E-9. If the class 3b probability were decreased by a factor of 2, the change in class 3b would be 5.75E-9 – 1.15E-9 = 4.6E-9. The change in class 3b is reduced by a factor of 2. If the original probability of class 3b increased by a factor of 2, the change in class 3b frequency would be 1.84E-8. Thus, conservatism in the methodology increases the estimate of the class 3b frequency because the frequency of a class 3b accident is multiplied by a factor of 5 for the "after" case. The other metrics that depend on the change in class 3b frequency are similarly affected.

By assigning the entire class 3b frequency to LERF, the change in LERF is conservatively estimated if some class 3b accidents would not be large in release magnitude. Changing this assumption would mean a lower change in LERF could be estimated.

The combined effect of the cited conservatisms results in conservative differential risk metrics for the application.

Part b:

The appropriate term for the change in LERF is "small" per the guidance in RG 1.174. The remaining risk metrics (CCFP, change in dose) that make up the plant risk profile meet the definition of "small" or "very small" per the discussion in section 2.0 of Attachment 5 to the LAR. Therefore, in Section 3.1.6.3 on page 16 of Enclosure 1 and page 36 of Attachment 5 of the LAR, the discussion should state, "Therefore, increasing the ILRT interval to 15 years is considered to be a small change to the MNS risk profile."

MNS ILRT RAI-03

Question:

According to Regulatory Issue Summary 2007-06 the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200 by the end of its implementation period (i.e., one year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (ASME/ANS RA-Sa-2009).

On page 13 of Enclosure 1 of the LAR, the licensee states, "The technical adequacy of the MNS PRA is consistent with the guidance of Regulatory Guide 1.200 Revision 2." Given the guidance listed above:

a. Confirm that the peer reviews of the Internal Flood PRA in September 2011, the LERF PRA in December 2012, and the Internal Events PRA in June 2015, cited on page 37 of Attachment 5 of the LAR were reviewed against the 2009 ASME/ANS PRA Standard, as clarified by RG 1.200, Revision 2. If not, please identify any gaps between the peer review and the requirements in RG 1.200, Revision 2.

b. Identify when the peer review of the Fire PRA and the High Winds PRA was performed, and confirm that the peer review of the High Winds PRA, cited on page 37 of Attachment 5 of the LAR was reviewed against the 2009 ASME/ANS PRA Standard, as clarified by RG 1.200, Revision 2. If not, please identify any gaps between the peer review and the requirements in RG 1.200, Revision 2.

Response:

Part a

Internal Flooding: Confirmed

The peer review report states that the Internal Flooding Probabilistic Risk Assessment (PRA) was performed against the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, *"Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,"* American Society of Mechanical Engineers, New York, NY, February 2009 and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200.

LERF: Confirmed

The peer review report states that the LERF PRA was performed against the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, *"Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,"* American Society of Mechanical Engineers, New York, NY, February 2009 and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200.

Internal Events: Review Below

The peer review report states that the Internal Events PRA was performed against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard *ASME/ANS RA-Sb-2013, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for NuclearPower Plant Applications,"* American Society of Mechanical Engineers, New York, NY, September 2013 and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200.

The peer review team utilized a database that contained the wording for supporting requirements (SRs) from ASME/ANS RA-Sa-2009 and as such was aware of the differences between the two standards. Duke Energy made an assessment of the differences between ASME/ANS RA-Sa-2009 and ASME/ANS RA-Sb-2013 and the impacts to RG 1.200 Revision 2 with respect to Clarifications and Qualifications. Of the SRs in Part 2, (excluding LERF), 16 represented changes to SRs potentially significant enough to require further investigation.

Detailed review of these 16 SRs indicated no gaps were identified between the internal events peer review and the requirements in RG 1.200 Revision 2.

Part b

Fire PRA

The McGuire Fire PRA model received a peer review against the requirements of the ASME/ANS RA-Sa-2009 PRA Standard and Regulatory Guide 1.200, using NEI 07-12 in September, 2009.

High Wind: Review Below

The peer review of the High Winds PRA model was performed from October 6 to October 10, 2014.

The peer review report states that the High Wind PRA was performed against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard *ASME/ANS RA-Sb-2013, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,"* American Society of Mechanical Engineers, New York, NY, September 2013 and any Clarifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200. The peer report further states that while NRC has currently only endorsed Addendum A, a review of Addendum A versus Addendum B showed no substantive differences, therefore Addendum B was judged to be valid for this review.

Duke Energy made an assessment of the differences between ASME/ANS RA-Sa-2009 and ASME/ANS RA-Sb-2013 and the impacts to RG 1.200 Revision 2 with respect to Clarifications and Qualifications. Of the SRs in Part 7 only two represented changes to SRs potentially significant enough to require further investigation. Detailed review of these two SRs indicated no gaps were identified between the high wind peer review and the requirements in RG 1.200 Revision 2.

MNS ILRT RAI-04

Question:

For the disposition of F&O WPR A3-1 in Attachment 5 on page 58 of the LAR, the licensee states, "Failure of the Main Steam and Feedwater lines due to wind pressure or missile is modeled as failing both diesel generators." As a result, wind pressure fragilities are a major contributor to the MNS plant CDF. Next, the licensee states, "The re-evaluation provided a significantly higher wind loading. Incorporating this re-evaluation, along with adding separate failure modes for wind pressure and missiles in to the PRA, reduced the calculated CDF / LERF risk."

Based on the information provided in the disposition, it is unclear how a significantly higher wind loading for the Main Steam and Feedwater lines would reduce CDF and LERF, unless the licensee meant the higher wind loading is actually a higher threshold for withstanding wind loading. It is also unclear how adding separate failure modes would reduce CDF and LERF. Clarify how assuming a higher wind load and how adding separate failure modes for the Main Steam and Feedwater lines reduces the plant CDF and LERF.

Response:

The F&O disposition contains discussion of the resolution of the F&O and historical information relating to the significance of the results and the evolution of modeling changes made to resolve the F&O.

The peer review identified the need to model the failure mode described in the F&O. The failure mode is failure of the main steam or main feedwater lines, via wind pressure or wind missile effects, that may result in steam intake into the Emergency Diesel Generator(s) (EDGs) which may disable the generators. Subsequently, the failure mode was added to the High Wind PRA by modeling the existing wind pressure and wind missile fragilities as failing the EDGs. Modeling the failure of the EDGs when the main feedwater and main steam lines fail due to wind pressure or wind missile effects resolves the F&O by adding the missing risk contributor identified by the peer team, and the High Wind CDF and LERF used in the ILRT extension includes the risk contributor, so the F&O has no impact to the ILRT extension analysis risk metrics in this application.

The specific discussion of high wind loading and the reduction in CDF and LERF are part of the narrative of the work accomplished to resolve the F&O, but do not impact the ILRT extension analysis risk metrics. The following discussion clarifies the F&O disposition.

The discussion of the significance of the failure mode to the High Wind CDF and LERF and subsequent re-evaluation of high wind "loading" is historical in nature and describes the evolution of modeling the main feedwater and main steam line fragilities to fail the EDGs. In this case, high wind "loading" refers to the capacity to resist wind loading, as noted in the RAI. The main feedwater and main steam lines were originally combined into a single high wind pressure screening fragility. When the EDGs were initially added as targets to the originally combined high wind pressure screening fragility, the results showed a relatively elevated plant CDF and LERF. The discussion of "adding separate failure modes" pertains to the refined wind pressure fragility evaluation, which provided a separate main feedwater pressure fragility and a separate main steam line pressure fragility, rather than the original combined screening fragility. Modeling of the separate pressure fragilities, which were based on the higher capacity to resist wind loading, was performed in response to the initial significance of the screening fragility to the High Wind CDF and LERF. Thus, when separation and refinement of those main feedwater lines and steamline wind pressure fragilities was done, the results showed a reduction to the plant CDF and LERF when compared to modeling them as single high wind pressure screening fragility after including EDGs as targets. Main feedwater and main steam line missile fragilities were not refined because they were already modeled separately for the main feedwater and main steam lines and the risk contribution from the missile fragilities did not warrant additional fragility refinement.

In summary, high wind "loading" refers to the capacity to resist wind loading. Since the capacity to resist wind loading increased, less likely events are required before the lines will fail, which results in a decrease in CDF and LERF. The addition of separate failure modes, using the higher wind loading capacity, reduced CDF and LERF relative to initial modeling of the combined screening fragility.

The High Wind model CDF and LERF used in the ILRT extension evaluation includes the risk of the wind pressure and wind missile fragilities modeled to result in steam line and feedwater line failure and Emergency Diesel Generator steam ingestion; thus, this F&O has no impact on the risk metrics for the application.

MNS ILRT RAI-05

Question:

For the disposition of F&O SF A5-01, the licensee states, "No impact on quantification of Fire PRA or Change Evaluations: seismic-fire interaction is purely qualitative per NUREG/CR-6850."

The NRC agrees there is no quantitative impact on the Fire PRA due to fire-seismic interactions, however this finding is qualitative in nature, and the licensee does not address the qualitative risk from lack of fire brigade training requirements on fire-seismic interactions. Identify how the training requirements in the fire brigade training program addresses fire-seismic interactions. If there is no requirement in the fire brigade training program, identify how the licensee's fire brigade is trained to handle fire-seismic interactions. If no such requirement or training exists, evaluate the qualitative impact of fire-seismic interactions on the Fire PRA.

Response:

Fire brigade members are trained and equipped to combat fire events in all areas of the plant during all modes of plant operation. As described in AD-TQ-ALL-0086, Fire Brigade Training, Duke Energy provides training to each brigade member and leader on firefighting fundamentals and specific elements regarding the tools and equipment provided such that they can gain access and egress to plant areas either as a matter of routine or using accepted fire service forcible entry methods and techniques. All of this fundamental knowledge and techniques could be applied to events and conditions up to and including large scale damage either natural (Seismic, Flex) or man-made (B.5.b). Procedures such as AD-EG-ALL-1532, NFPA 805 Pre-Fire Plans, detail the content of the Pre-Fire Plans including the requirement to contain all access and egress routes (i.e., stairs and doors). This is to allow for the potential of one route being blocked, as could be the case in a seismic event. Actual fire area details are described in the Pre-Fire Plans such as CSD-MNS-PFP-AB-0767-001, Auxiliary Building Elevation 767 Pre-Fire Plan, which provides both written and visual descriptions of the fire areas and their access and egress routes, along with the various details of the area, its hazards, and firefighting equipment that may be relied upon for an event.

The fire brigade makes up only a portion of the defense-in-depth strategy elements listed below:

1. Preventing fires from starting.

2. Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage.

3. Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

Active and passive features are employed to satisfy elements 2 and 3 along with the fire brigade response. In the event systems or features are not available, the brigade, comprised largely of operations personnel, are trained to assess and attempt to activate fire protection systems and features when appropriate. Fire drill scenarios assume conditions where systems and features are both operating and not operating.

Again, the brigade would use their available tools, procedures and training to affect firefighting operations for all plant areas and operating modes and conditions. The training and practice sessions conducted throughout the year are intended to allow the brigade to practice in a variety of conditions and situations such that they will be ready and effectively mitigate and control the hazards anticipated.