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RA-18-0097

10 CFR 50.90

August 17, 2018

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station (CNS), Units 1 and 2
Facility Operating License Numbers NPF-35 and NPF-52
Docket Numbers 50-413 and 50-414
Response to NRC Requests for Additional Information (RAIs)
License Amendment Request to Revise Technical Specification Section 3.7.8, "Nuclear Service Water System"

References:
1. Letter from Duke Energy to the NRC dated September 14, 2017, ADAMS Accession No. ML17261B255
2. Letter from the NRC to Duke Energy dated July 17, 2018, ADAMS Accession No. ML18198A195

The Reference 1 letter was submitted for the Catawba Nuclear Station (CNS), Units 1 and 2, Facility Operating License Numbers NPF-35 and NPF-52, Docket Numbers 50-413 and 50-414, License Amendment Request (LAR) to Revise Technical Specification Section 3.7.8, "Nuclear Service Water System." The Reference 2 letter transmitted Requests for Additional Information (RAIs) from the NRC associated with the subject matter LAR.

The purpose of this letter is to formally respond to the RAI questions contained in the Reference 2 letter. The enclosure to this letter constitutes Duke Energy's response to the RAIs. The format of the enclosure is to re-state each RAI question, followed by its associated response.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the significant hazards consideration analysis provided in the original submittal is not altered by the additional information provided.

There are no regulatory commitments contained in this letter or the enclosure or attachments.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this request by transmitting a copy of this letter and enclosure to the designated State Official.

Please direct questions on this matter to Carrie L. Wilson, Sr. Engineer, at (803) 701-3014.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 17, 2018.

Sincerely,

Tom Simril

Tom Simril
Vice President, Catawba Nuclear Station

Enclosure: Response to NRC Requests for Additional Information (RAIs)

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Enclosure

Response to NRC Requests for Additional Information (RAIs)

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION (RAIS)

LICENSE AMENDMENT REQUEST TO ADD NEW CONDITION TO TECHNICAL

SPECIFICATION 3.7.8, "NUCLEAR SERVICE WATER SYSTEM"

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION (CNS), UNITS 1 AND 2

DOCKET NUMBERS 50-413 AND 50-414

Request for Additional Information RAI-05

Facts and Observations Closure Process

In NRC letter dated May 3, 2017 (ADAMS Accession No. ML17079A427), NRC staff accepted, with conditions, the Nuclear Energy Institute (NEI) Appendix X guidance to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations" (ADAMS Accession No. ML17086A431), for use by licensees to close F&Os that were generated during a peer review process.

The NRC staff observed (ADAMS Accession No. ML18117A187) that independent reviews were performed to close F&Os for the Catawba internal events, large early release frequency (LERF), and internal flooding PRAs (e.g., reviews conducted in October 2015 (Phase I), July 2017 (Phase II), and August 2017). Section 3.2.2, "PRA Quality/Technical Adequacy," of the LAR states, "an F&O closure effort was completed in July 2017, for internal flood and LERF, to validate the F&O closure process met the Appendix X requirements (Ref. 3.2.8.10)." In addition, LAR Section 3.2.2.1, "Internal Events, CDF and LERF," states, "[t]here were 8 internal events PRA Findings which were considered to be open after the December 2015 Peer Review (Table 1 of Ref. 3.2.8.8) and the 2017 independent F&O closure technical review." The LAR Reference 3.2.8.10 (APC 17-13, "NRC Acceptance of Industry Guidance on Closure of PRA Peer Review Findings," dated May 8, 2017) is not available to NRC staff, and it is not clear whether the F&O closure effort was performed consistent with the NRC-accepted process discussed in the letter dated May 3, 2017. To address the observations above, the staff requests the following additional information.

a) Provide the following information to confirm that the independent reviews performed to close F&Os for the Catawba internal events, LERF, and internal flooding PRAs (e.g., reviews conducted in October 2015 (Phase I), July 2017 (Phase II), and August 2017) were performed consistent with the NRC-accepted process discussed in the letter dated May 3, 2017.

    i. Provide a summary with a timeline of the independent reviews performed to close F&Os for the internal events, LERF, and internal flooding PRAs. Explain how each of these independent reviews (or combinations thereof) are consistent with the NRC-accepted process for closing F&Os as discussed in the May 3, 2017 letter.

ii. Clarify whether a focused-scope peer review was performed concurrently with these independent reviews. If so, provide the following:

(1) Summary of the scope of the peer review, and

(2) Detailed descriptions of any new F&Os generated from the peer review and the associated dispositions for the application.

iii. Confirm that the independent review teams were provided with a written assessment and justification of whether the resolution of each F&O, within the scope of the independent assessments, constitutes a PRA upgrade or maintenance update, as defined in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. If the written assessment and justification for the determination of each F&O was not performed and reviewed by the independent review teams, discuss how this aspect of the F&O closure process was met consistent with the staff's acceptance as discussed in the May 3, 2017 letter.

iv. Appendix X, Section X.1.3, includes five criteria for selecting members of the F&O closure review team. Explain how the selection of members for the independent reviews summarized under Part (i) met the five criteria.

v. Section 4.2 of RG 1.200, Revision 2, states, "[i]f a requirement of the standard has not been met, the licensee is to provide a justification of why it is acceptable that the requirement has not been met." Explain how closure of the F&Os, as summarized under Part (i), was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet Capability Category (CC) II for supporting requirements (SRs) from ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

vi. It is unclear whether the scope of the independent reviews included all finding-level F&Os, including those finding-level F&Os associated with SRs that were met at CC II. Owing to their potential impact to the risk results of the proposed CT extension, provide all remaining finding-level F&Os and associated dispositions that were not closed from the independent reviews and not included in LAR Attachment 4, "PRA Peer Review Findings and Resolutions."

b) Alternatively to Part a, provide all finding-level F&Os and associated dispositions that were in scope of the independent reviews performed to close F&Os for the internal events, LERF, and internal flooding PRAs, including any finding-level F&Os associated with SRs that were met at CC II.

Duke Energy Response 5.a)

i. <u>Internal Flooding F&O Resolutions</u>

The Catawba Internal Flooding Peer Review was conducted in 2012. Subsequent to the 2012 Peer Review, the Internal Flooding PRA model and documentation was updated to resolve the Peer Review F&Os. As part of the December 2015 Internal Events Peer Review, an independent review was performed on the CNS Internal Flood F&O resolutions to determine if they were resolved and that the corresponding SRs are MET at CC-II or greater. This independent review was documented within the Internal Events Peer Review report. The independent review was performed prior to NRC acceptance of NEI Appendix X.

Upon NRC acceptance of NEI Appendix X, a gap assessment was performed internally by Duke Energy to assess the acceptability of the 2015 independent review for Internal Flooding F&O resolutions against the current industry guidance for independent reviews (NEI Appendix X) as accepted by the NRC. This gap assessment identified certain shortcomings of the 2015 report, specifically pertaining to the lack of justification that F&O resolutions are considered to be maintenance/updates as opposed to PRA upgrades.

In order to close the identified gap to meeting the NEI Appendix X requirements, the same individuals who performed the 2015 Independent Review were contracted again in 2017 to perform a second independent review, including an assessment of whether or not each F&O resolution constitutes an upgrade to the PRA. This 2017 independent review also included an assessment of how each requirement of NEI Appendix X and NRC Expectations was met by the 2015 and 2017 reviews. Duke Energy provided the 2017 independent review team with documentation of F&O resolutions and a self-assessment of whether or not each resolution could constitute a PRA upgrade.

The 2017 Independent Review report shows that each requirement of NEI Appendix X and NRC expectations for independent F&O closure assessments is met with no exceptions. Therefore, it is shown that all Internal Flooding F&Os determined to be closed by the 2015 and 2017 independent reviews are indeed closed in accordance with NEI Appendix X and NRC expectations.

LERF F&O Resolutions

The Catawba LERF Peer Review was conducted in 2012. Subsequent to the 2012 Peer Review, the LERF PRA model and documentation were updated to resolve the Peer Review F&Os. As part of the December 2015 Internal Events Peer Review, an independent review was performed on the CNS LERF F&O resolutions to determine if they were resolved and that the corresponding SRs were MET at CC-II or greater. This independent review was documented within the Internal Events Peer Review report. The independent review was performed prior to NRC acceptance of NEI Appendix X.

Upon NRC acceptance of NEI Appendix X, a gap assessment was performed by Duke Energy to assess the acceptability of the 2015 independent review for LERF F&O resolutions against the current industry guidance for independent reviews (NEI Appendix X) as accepted by the NRC. This gap assessment identified certain shortcomings of the 2015 report, specifically pertaining to the lack of justification that F&O resolutions are considered to be maintenance/updates as opposed to PRA upgrades.

In order to close the identified gap to meeting the NEI Appendix X requirements, the same individuals who performed the 2015 Independent Review were contracted again in 2017 to perform a second independent review, including an assessment of whether or not each F&O resolution constitutes an upgrade to the PRA. This 2017 independent review also included an assessment of how each requirement of NEI Appendix X and NRC Expectations was met by the 2015 and 2017 reviews. Duke Energy provided the 2017 independent review team with documentation of F&O resolutions and a self-assessment of whether or not each resolution could constitute a PRA upgrade.

The 2017 Independent Review report shows that each requirement of NEI Appendix X and NRC expectations for independent F&O closure assessments is met with no exceptions. Therefore, it is shown that all LERF F&Os determined to be closed by the 2015 and 2017 independent reviews are indeed closed in accordance with NEI Appendix X and NRC expectations.

<u>Internal Events F&O Resolutions</u>

The CNS Internal Events PRA Peer Review was performed in December of 2015. Subsequent to the 2015 Peer Review, the Internal Events PRA model and documentation were updated to resolve the Peer Review F&Os. An independent review of the F&O resolutions for Internal Events was conducted in August 2017. This review, being performed after the NRC acceptance of NEI Appendix X in the letter dated May 3, 2017, was performed and documented in accordance with, and meeting the guidance of NEI Appendix X. Out of this assessment, one upgrade to the PRA was discovered (related to ASME SRs HR-A1 through HR-C3), and a focused-scope peer review was performed in September 2017. The focused scope peer review shows that ASME standard supporting requirements HR-A1 through HR-C3 are fully met by the Catawba Internal Events (non-LERF) PRA at CC II or greater.

ii. For Internal Flooding and LERF, no focused-scope peer review was performed concurrent with the independent F&O closure assessments, thus there were no new F&Os generated as a result.

For Internal Events, as explained in the response to part i above, one upgrade was identified in the F&O closure independent assessment, which warranted a focused scope peer review for SRs HR-A1 through HR-C3. The focused scope peer review was conducted in September 2017. The focused scope peer review shows that ASME standard supporting requirements HR-A1 through HR-C3 are fully met by the Catawba Internal Events (non-LERF) PRA at CC II or greater, with no new F&Os.

iii. For internal flooding and LERF, the July 2017 independent review team was provided with a written justification that each F&O resolution was an update and not an upgrade to the PRA model as defined in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

The Internal Events August 2017 independent review team was provided with written justification that each F&O resolution was an update and not an upgrade to the PRA model as defined in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

iv. Each F&O closure assessment that was performed includes a statement of independence for the reviewers, and how they meet the appropriate Peer Review qualifications as stated in the ASME/ANS PRA Standard. Each F&O was reviewed by more than one qualified reviewer, and the total number of finding F&Os was considered when selecting the total number of reviewers. All reviewers were knowledgeable about the F&O independent assessment process. Reviewer resumes are also provided for each F&O closure assessment. Thus, all independent assessment member selection requirements are met by each F&O closure assessment for Internal Events, LERF, and Internal Flooding.

v. The independent F&O closure assessments reviewed the finding resolution and the applicable SR against which the finding was written. The applicable SR was then

assessed as to whether or not resolution of the finding in the PRA model causes the applicable SR to be MET at CC II or greater. This was done for the independent assessments for Internal Events, Internal Flooding and LERF.

vi. The scope of the independent reviews included all finding-level F&Os, including those finding-level F&Os associated with SRs that were met at CC II. Any remaining open finding-level F&Os are provided in the LAR submittal.

References to Duke Energy Documentation:

Internal Flooding

- Peer Review: *CNC-1535.00-00-0200, Rev. 6, Appendix C*
- F&O Resolutions: *CNC-1535.00-00-0197, Rev. 1*
- Pre-Appendix X F&O Closure Assessment: *CNC-1535.00-00-0200, Rev. 6, Appendix A*
- Duke Energy Gap Assessment and Update Justifications: *CNC-1535.00-00-0200, Rev. 6, Appendix E*
- Post-Appendix X Independent Assessment: *CNC-1535.00-00-0200, Rev. 6, Appendix G*

LERF

- Peer Review: *CNC-1535.00-00-0200, Rev. 6, Appendix D*
- F&O Resolutions, *CNC-1535.00-00-0178, Rev. 2*
- Pre-Appendix X F&O Closure Assessment: *CNC-1535.00-00-0200, Rev. 6, Appendix A*
- Duke Energy Gap Assessment and Update Justifications: *CNC-1535.00-00-0200, Rev. 6, Appendix F*
- Post-Appendix X Independent Assessment: *CNC-1535.00-00-0200, Rev. 6, Appendix G*

Internal Events

- Peer Review: *CNC-1535.00-00-0200, Rev. 6, Appendix A*
- F&O Resolutions: *CNC-1535.00-00-0220, Rev. 1*
- August 2017 Independent Review: *CNC-1535.00-00-0200, Rev. 6, Appendix H*
- Focused-scope Peer Review of HR-A1 - HR-C3: *CNC-1535.00-00-0200, Rev. 6, Appendix I*

Request for Additional Information RAI-06

Disposition of PRA F&Os

Attachment 4 of the LAR provides PRA peer review F&Os and dispositions for the internal events, internal flooding, high winds, and fire PRAs. Address the following questions related to the dispositions of the internal events and high winds F&Os that do not seem fully resolved for this LAR.

a) High winds F&O WPR-C3-01 is concerned with eight modelling assumptions. The disposition stated that four assumptions were removed from the analysis and the other four were revised and enhanced. Given that modeling assumptions can have a significant impact on core damage frequency (CDF) and LERF results, the NRC staff requests the following additional information:

   i. Describe and justify the revised and enhanced assumptions, or

   ii. Alternatively, explain why resolution of this F&O has a negligible impact on the risk associated with extending the NSWS CT.

b) Internal events F&O 22-7 states that the human reliability analysis (HRA) dependency analysis recovery rules were inappropriately implemented by applying them first to human error probability (HEP) combinations with low probabilities, rather than to higher order HEP combinations. The disposition to F&O 22-7 indicates that a new HRA dependency analysis was performed and new recovery rules have been implemented in the PRA model used for this application using the Electric Power Research Institute (EPRI) HRA calculator. The NRC staff's review of the disposition to F&O 22-7 has identified additional information shown below to understand full characterization of the risk estimates.

   i. Describe the change made to the HRA dependency analysis to resolve F&O 22-7. This description should be of sufficient detail for the NRC staff to determine whether the dependency analysis update is considered a PRA maintenance or PRA upgrade, as defined in ASME/ANS RA-Sa-2009 PRA standard, Section 1-5.4, as qualified by RG 1.200, Revision 2. Also, include in this discussion: (1) a summary of the original dependency analysis method and the new dependency analysis method; (2) changes in PRA scope that impact the significant accident sequences or the significant accident progression sequences; (3) changes in PRA capability that impact the significant accident sequences or the significant accident progression.

   ii. Indicate whether the dependency analysis update is a PRA maintenance or PRA upgrade, along with a justification for this determination.

   iii. If the dependency analysis update is determined to be a PRA upgrade, then discuss any focused-scope (or full-scope) peer reviews that have been performed for this upgrade and provide peer review findings-level F&Os and their associated dispositions as it pertains to this LAR. If a peer review(s) was not performed for this upgrade, then provide a quantitative evaluation (e.g., sensitivity or bounding analysis) of its effect on the results of this LAR until a focused-scope peer review can be completed. [Note, this sensitivity or bounding analysis should be based on the combined updates of PRA methods or treatments considered in the response to Probabilistic Risk Assessment Licensing Branch (APLA) RAI 11.]

iv. Explain how the fire and high winds PRAs incorporated the dependency analysis update performed for the internal events PRA in response to F&O 22-7. If the fire and high winds PRAs did not incorporate this update, then justify that the fire and high winds PRAs meet PRA quality expectations prescribed in RG 1.200, Revision 2, for risk-informed applications.

Alternatively, incorporate the dependency analysis update performed for the internal events PRA into the fire and high winds PRA models used for this LAR, as appropriate, that aggregate the PRA updates requested in APLA RAI 11.

Duke Energy Response:

a) High Winds F&O WPR-C3-01 asks for additional clarification for multiple assumptions pertaining to the HWPRA. Standard Requirement (SR) WPR-C3 requires the PRA to "DOCUMENT the sources of model uncertainty and related assumptions associated with the high wind plant response model development."

The peer review team determined that the assumptions identified in the F&O required clarification. Three of the assumptions listed in the F&O were not applicable to the peer-reviewed model but were incorrectly included in the peer-reviewed report; these assumptions were Assumptions 1, 6, and 7 of Revision 0 (the peer-reviewed version) of the report and were subsequently removed in Revision 1 of the report. Assumption 5 was removed in Revision 1 of the report due to a model change associated with a different F&O. These assumptions are described in detail below.

<u>Specifically:</u>

Assumption 1: Deleted in Revision 1 -

Assumption 1 stated that a functional failure of the main transformers required two missile hits. This was an assumption used early on in the fragility analysis that was later abandoned due to possible non-conservatism and was not included in the fragility analysis that supported the peer-reviewed model. The assumption was included in earlier drafts of the report and should have been removed for Revision 0.

Assumption 5: Deleted in Revision 1 -

Assumption 5 stated that a reactor trip was only assumed following a high wind-induced failure of an SSC. The peer-reviewed model was revised per the resolution of F&O WPR-A1-02 to assume a plant trip for all high wind events; and therefore, Assumption 5 was removed in Revision 1 of the report. The provided F&O resolution for WPR-C3-01 discusses the model change, but only because it is associated with the removal of Assumption 5, and thus, Assumption 5 was no longer applicable to the model.

Assumption 6: Deleted in Revision 1 -

Assumption 6 stated that conservatism was introduced by ORing the high wind-induced LOOP events with the internal events LOOP event, %T3. This assumption is not needed, as the high wind-induced LOOP events should be mapped to the internal events LOOP initiating event, in order to impact the correct model logic. The assumption was included

in earlier drafts of the report and should have been removed for Revision 0 as it is not an assumption but represents the model logic as developed.

Assumption 7: Deleted in Revision 1 -

Assumption 7 stated that some components were modeled in the high winds analysis but had no representation in the fault tree, specifically the MSSVs and MSIVs. The purpose of this assumption was to indicate that fragilities were calculated for all MSSVs and MSIVs, but the internal events model, which the high wind model was built on, only included a SGTR on SG "B", so only the fragilities for the "B" MSSVs and MSIVs were included in the high winds fault tree model. This assumption was removed because this is not an assumption made for the high winds analysis, it was an internal events simplification to only model the "B" train. Since high wind fragilities were generated for each MSSV and MSIV, they were included in the high wind analysis. The assumption was included in earlier drafts of the report and should have been removed for Revision 0. Furthermore, the internal events and high winds models have been revised and include model logic for all steam line trains. The train-specific MSSV and MSIV high wind fragilities are included in the current high winds model, which is documented in Revision 2 of the HWPRA report.

The remaining four assumptions were Assumptions 4, 8, 11, and Assumption 1 of Appendix A Section B.1, which were enhanced for Revision 1 of the report to clarify the basis for the assumption and discuss the impact that the assumption has on the HWPRA results. These are described in detail below:

Assumption 4:

Assumption 4 stated that a LOOP is assumed for all F2 and greater high wind events, but did not provide the basis for this assumption. The assumption was enhanced in Revision 1 to explain that failure of LOOP-related equipment, such as the switchyard structures, are likely to fail at F2 and greater wind speeds; and therefore, it is realistic to assume a LOOP for all F2 and greater wind events.

Assumption 8:

Assumption 8 stated that the Drinking Water System (System YD) was assumed failed for all high wind events, but did not provide basis. In Revision 1, this assumption was enhanced to explain that the system is only a backup cooling system to the charging pumps, and the normal cooling supply from the component cooling water system is unlikely to fail. Also, the drinking water system requires offsite power, so it would not be available for F2 or greater high wind events or anytime there is a LOOP at F1 speeds. Therefore, failing of the system would have little impact on the risk. This conservative assumption was removed in Revision 2, since the operator action to align System YD for cooling was included in the analysis; however, it offers only a minor benefit.

Assumption 11:

Assumption 11 stated that the HWPRA analysis was for Unit 1 with shared Unit 2 SSCs, and that the analysis is assumed applicable to Unit 2 with shared Unit 1 SSCs. In Revision 1, this assumption was updated to state that this is realistic because there is a

high level of symmetry between the Units. This assumption was removed for Revision 2, since unit-specific HWPRA models were developed.

Assumption 1 in appendix A Sect B.1 (Revision 0):

Assumption 1 of Appendix A Section B.1 stated that straight line or tornado wind conditions will not prevent access to the SSF after one hour, but no basis was provided. In Revision 1, the assumption was moved to Section G.2.1 and was enhanced to explain that the duration of the high wind events is expected to be less than one hour that multiple travel pathways are available for the operators to take to the SSF, and debris from F1 wind events are not expected to block access to the SSF. For Revision 2, the assumption is in Section G.1.2.

In summary, as this is a documentation issue only, resolution of F&O WPR-C3-01 does not have any impact on the results of the HWPRA. Therefore, the resolution of this F&O does not impact the results of the High Winds analysis performed in the LAR submittal to revise the CNS TSs associated with NSWS CT.

b) This F&O was written against the internal events PRA model for Catawba. The external hazards HRA combination recovery rules were ordered by way of number of events, beginning with combinations containing the most events and ending with those containing the least number of events, so no such F&O was written for those models. Therefore, in addressing F&O 22-7 to order the HRA combinations in the recovery rules by number of events, the external hazards recovery rules are not impacted. This issue thus only affects Internal Events CDF and LERF results.

   i. No changes were made to the Internal Events HRA dependency analysis to resolve F&O 22-7. Thus, the F&O is still unresolved. This response will serve as the disposition of F&O 22-7 for the 30-Day CT LAR submittal.

      The F&O was written against the fact that the CNS dependency analysis applies HRA combination recovery rules by order of lowest probability to highest probability. The F&O is concerned that this may result in an underestimation of risk. For this F&O to be resolved, the HRA combination recovery rules must be ranked first by decreasing number of events (combinations with more events first and fewest events last). A sensitivity study is performed to determine the effect of the updated dependency analysis on the results from the original LAR submittal, which are shown in Table 1 below.

Table 1: Results from Original LAR Submittal

		Base	30 Day CT	**Change in Risk**
U1	CDF	7.23E-06	7.50E-06	**2.70E-07**
	LERF	3.42E-07	3.39E-07	**-3.00E-09**
U2	CDF	7.33E-06	7.39E-06	**6.00E-08**
	LERF	4.01E-07	3.68E-07	**-3.30E-08**

To perform a sensitivity study, the HRA recovery rules for the CNS Internal Events CDF and LERF models were re-ordered by way of number of events per combo (higher events first and fewer events last), per the finding. The base models and the models for the CT 30-day LAR for units 1 and 2 were re-solved using the revised recovery rules. The results are shown in Table 2 below:

Table 2: Results Using Revised Recovery Rules

		Base	30 Day CT	**Change in Risk**
U1	CDF	7.33E-06	7.60E-06	**2.70E-07**
	LERF	3.76E-07	3.73E-07	**-3.00E-09**
U2	CDF	7.39E-06	7.46E-06	**7.00E-08**
	LERF	4.13E-07	3.80E-07	**-3.30E-08**

The results from Table 2 show that, for Units 1 and 2 CDF, there is roughly a 1% increase in CDF when the revised recovery rules are applied, which is very small. For Unit 1 LERF, a 10% increase is shown and for Unit 2 LERF, a 3% increase is shown. However, in both of these cases, a risk *improvement* is still shown when applying the 30-Day CT LAR for LERF for both units. Furthermore, when looking at the change in CDF and LERF from the Base PRA results to the 30-Day CT LAR, there is no change observed in the delta between using the original submittal results and the results using the revised recovery rules for both units. The exception to this is for the Unit 2 CDF results, which show a change in the delta risk from 6.00E-08 using the original recoveries to 7.00E-08 when using the corrected recoveries, a change in the delta risk of 1.00E-08. This represents 0.1% of the Base CDF, which is negligible. The results from Table 2 show that all CDF and LERF risk insights from the 30-Day CT LAR that are presented in the original submittal are still valid. From this it can be shown that F&O 22-7 does not have a significant impact on the 30-Day CT LAR.

ii. As described in item (i) above, the issue that the finding has with the way in which the internal events dependency analysis is applied to the final cut sets in the PRA model. To obtain the results presented in (i), the recovery rules were simply re-sorted in a more appropriate manner. A sensitivity was performed in (i) to show that the impact of F&O 22-7 on the internal events model is such that the results of the 30-Day CT LAR are still valid. No revision was performed to the base model presented

in the original submittal, and as such there is no PRA upgrade (as defined by the ASME/ANS PRA Standard) to the PRA model.

iii. As described in item (ii) above, no PRA upgrade was performed to the CNS PRA model (as defined by the ASME/ANS PRA Standard). As such, no focused-scope peer review was performed in response to this issue, and no new F&Os exist.

iv. The external hazards HRA combination recovery rules were ordered by way of number of events, beginning with combinations containing the most events and ending with those containing the least number of events, so no such F&O exists for those models. Therefore, in addressing F&O 22-7 to order the HRA combinations in the recovery rules by number of events, the external hazards recovery rules are not impacted.

Request for Additional Information RAI-07

Updated Internal Events Logic Transferred to Other Hazard Models

Section 3.2.2 of the LAR states that a peer review was performed for the internal events PRA in 2015. It is not clear to what extent the internal events PRA was updated prior to this peer review and in response to F&Os from this peer review. It is generally understood that the mitigation logic (particularly system modeling) from the internal events PRA model is used as the basis for other PRA hazard models. The LAR indicates that the peer review for the high winds PRA was performed in August 2013 and a peer review for the fire PRA was performed in July 2010. Since these reviews occurred prior to the 2015 internal events PRA peer review, it is not clear how the fire and high winds PRAs incorporate internal events PRA updates needed to align with PRA quality expectations prescribed in RG 1.200, Revision 2. To address the above observations, provide the following information.

RAI-07.a
For the internal events, fire, and high winds PRAs used to support this LAR, explain how the fire and high winds PRAs appropriately incorporate the internal events PRA updates performed since the last peer review of the fire and high winds PRAs. Also, summarize these internal events PRA updates.

RAI-07.b

If the fire and high winds PRAs did not appropriately incorporate the internal events PRA updates, then justify how the fire and high winds PRAs meet PRA quality expectations prescribed in RG 1.200, Revision 2, for risk-informed applications. Alternatively, incorporate the updates performed for the internal events PRA, as applicable, into the fire and high winds PRAs used for this LAR that aggregate the PRA updates requested in APLA RAI 11.

Duke Energy Response:

RAI-07.a:

After the 2015 internal event peer review, the internal events model was updated and issued as Catawba Rev. 4. Significant internal events model changes between the previous internal events PRA model of record and revision 4 include the following:

- Updated model data
- Re-performed HRA and dependency analysis
- Developed unit-specific models.
- Developed a Condensate System model
- Developed a Condenser Circulating Water System model
- Added and deleted initiators
- Switched from single to multiple alignment system models
- Included support system initiator fault trees in the PRA model (upgrade)
- Switched from the Multiple Greek Letter approach to the alpha-factor method for quantifying common cause failure events (model upgrade).

The Catawba high winds PRA model has been revised to incorporate updates resulting from the 2015 internal event peer review. On the other hand, the Catawba fire PRA model has not yet been updated to incorporate these updates. Rather, it is based on the Catawba Rev. 3a internal events PRA model.

RAI-07.b:

The acceptability of the fire PRA model for the NSWS LAR is justified by the peer review on the model (see section 3.2.2.4 of the LAR) and the resolutions of Finding F&Os generated during those reviews, as discussed in Attachment 4, section 4.5 (Fire). Section 3.2.2.4 also notes that "multiple RAIs were generated by the NRC during the review of the NFPA 805 LAR. A number of these RAIs referenced the F&Os from the FPRA Peer Review. These RAI responses have been incorporated into the FPRA and are documented through various RAI responses, as well as the RAI 03 response which involved quantification of the FPRA after all RAI responses were incorporated. No new methods were introduced during the FPRA changes; therefore, no additional Peer Review is needed."

The fire PRA model is based on Rev. 3a internal events model, as noted in the response to RAI-07.a above. The 2015 peer review was performed on an early version of the Rev. 4 internal events model, which is significantly different from the Rev. 3 model. Thus, F&Os generated from the 2015 peer review are not necessarily applicable to the fire PRA model.

As mentioned in the response to RAI 7a. above, the Catawba high winds PRA model has been revised to incorporate updates resulting from the 2015 internal event peer review.

References

1. CNC-1535.00-00-0200, Rev. 6, Catawba Nuclear Station PRA Peer Review F&O Resolutions, April 2018.
2. CNC-1535.00-00-0219, Rev. 1, Risk Determination for Proposed Catawba RN LCOs to Implement Single Pond Return Header Operation, October 2017.
3. CNS-17-014, Catawba, Units 1 and 2, License Amendment Request to Revise Technical Specification Section 3.7.8, "Nuclear Service Water System" to add a new condition to allow Single Pond Return Header Operation of the NSWS with a 30-Day Completion Time, https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML17261B255.
4. CNC-1535.00-00-0180, Rev. 1, Catawba Nuclear Station Probabilistic Risk Assessment Section 8.0: Model Integration and Quantification, February 2017.
5. CNC-1535.00-00-0154, Rev. 2, Catawba Nuclear Station Units 1 and 2 High Wind/Missile PRA Analysis, February 2017.

Request for Additional Information RAI-08

Use of ASME/ANS RA-Sb-2013

Section 3.2.2 of the LAR explains that certain peer reviews were conducted with consideration of the changes between the ASME/ANS RA-Sa-2009 and ASME/ANS RA-Sb-2013 PRA standards (e.g., peer reviews of the internal events, internal flooding, and high winds PRAs). The technical adequacy of PRAs used for risk-informed activities is evaluated using RG 1.200, Revision 2, which endorses, with clarifications and qualifications, ASME/ANS RA-Sa-2009. The NRC did not endorse ASME/ANS RA-Sb-2013. Explain how the peer reviews that utilized the 2013 PRA standard meets the technical adequacy guidance in RG 1.200, Revision 2 (e.g., perform a comparison between ASME/ANS RA-Sb-2013 and ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2).

Duke Energy Response:

To clarify, the Catawba Internal Flooding, LERF, and High Winds PRA Peer Reviews utilized ASME/ANS RA-Sa-2009, which is endorsed by the NRC through RG 1.200, Revision 2.

The Catawba Internal Events PRA 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 Nuclear Power 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.

Request for Additional Information RAI-09

External Events Analysis

The Catawba LAR states that the proposed change to the NSWS TS CT has been evaluated using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3.2 of RG 1.177 states, "[t]he scope of the analysis should include all hazard groups (i.e., internal events, internal flood, internal fires, seismic events, high winds, transportation events, and other external hazards) unless it can be shown that the contribution from specific hazard groups does not affect the decision."

The LAR does not explain how it is concluded that the risk associated with the NSWS 30-day CT is not impacted by other external hazards (i.e., hazards other than external flooding, seismic, and those modelled in the PRAs).

a) Provide the results of a systemic assessment of other external hazards (such as those listed in Appendix 6-A of Part 6 of the PRA Standard ASME/ANS RA-Sa-2009) demonstrating that the conclusions of the LAR are not impacted by other external hazards. The systematic assessment should reflect the most current information for each hazard and reflect the as-built, as-operated plant.

b) If the conclusions of the LAR are impacted by other external hazards, incorporate the other external hazards, as applicable, into the risk evaluation used for this LAR that aggregate the PRA updates requested in APLA RAI 11

Duke Energy Response:

a) The Catawba site was extensively assessed against external hazards during the IPEEE evaluation[1]. Table 1 presents the initial external event listing given in the 1994 IPEEE submittal report. Table 2 provides the screening justification for the majority of these events.

   The remaining events were addressed in detail in the IPEEE submittal. Besides seismic, fire, high winds and flooding, Catawba was also analyzed for aircraft crashes, transportation events, impact of nearby military and industrial facilities, on-site storage of toxic materials, on-site storage of explosive materials and gas pipeline ruptures. Since the screening criteria found in SPR EXT-B1 of Section 6 in the ASME / ANS Standard[2] is essentially the same as that used in the IPEEE submittal, none of these hazards are deemed to be significant contributors to plant risk.

   Since the IPEEE response was submitted, updated fire and high winds analyses have been developed and peer-reviewed against the ASME / ANS Standard. Furthermore, as part of the Fukushima NTTF 2.1 response, external flooding concerns for Catawba were addressed via

[1] Duke Power Company, Catawba Nuclear Station, IPEEE Submittal Report, June 1994
[2] ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RS-S-2008 Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications

updated analyses and mitigating strategies[3] [4]. The sites were evaluated for flooding from the following sources:

- Local Intense Precipitation
- Flooding in Reservoirs
- Dam Failures
- Storm Surge and Seiche
- Tsunami
- Ice-Induced Flooding
- Channel Diversion
- Combined Effects

The results of these analyses demonstrate that Catawba external flooding events meets its licensing design basis for local intense precipitation and thus screens out per Section 6, SPR EXT-B1 of the ASME / ANS Standard.

The response to RAI 16 discusses consideration of the seismic hazard for the Catawba NSWS.

b) N/A - None of the conclusions of the LAR were impacted by other external hazards.

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[3] Duke Energy Letter, "Flood Hazard Reevaluation Report, Response to NRC 10 CFR 50.54(f) Request for Additional Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident", March 12, 2014 (ADAMS Accession Number ML 14077A054)

[4] Duke Energy Letter, Catawba Nuclear Station Flood Hazard Mitigating Strategies Assessment (MSA) Report Submittal, June 20, 2017

Table 1
Catawba Preliminary External Initiating Events List

1. Aircraft
2. Avalanche
3. Coastal Erosion
4. Drought
5. External Flooding
6. Extreme Winds and Tornadoes
7. Fire
8. Fog
9. Forest Fire
10. Frost
11. Hail
12. High Tide, High Lake Level, or High River Stage
13. High Summer Temperature
14. Hurricane
15. Ice Cover
16. Industrial or Military Facility Accident
17. Internal flooding
18. Landslide
19. Lightning
20. Low Lake or River Water Level
21. Low Winter Temperature
22. Meteorite
23. Pipeline Accident (gas, etc.)
24. Intense Precipitation
25. Release of Chemicals in On-site Storage
26. River Diversion
27. Sandstorm
28. Seiche
29. Seismic Activity
30. Snow
31. Soil Shrink-Well Consolidation
32. Storm Surge
33. Transportation Accidents
34. Tsunami
35. Toxic Gases
36. Turbine-Generated Missile
37. Volcanic Activity
38. Waves

Table 2
Catawba Screening Justifications for Other External Initiating Events

	Event	Remarks
1	Avalanche	There are no mountains in the vicinity of Catawba from which a significant avalanche could be generated.
2	Coastal Erosion	Catawba is located more than 150 miles from the nearest coastal area. However, to protect the lake edge from erosion, the yard areas subjected to waves are protected by riprap underlain by a thick subgrade of filter material. Therefore, lake edge erosion will not be a significant problem.
3	Drought, High Summer Temps., Low Lake or River Water Level	The effect of a drought, high summer temperatures, low lake level, or low river water level at Catawba is insignificant because there are upstream dams that provide water level control on Lake Wylie.
4	Fog	Accident data involving surface vehicles or aircraft would include the effects of fog.
5	Forest Fire	Bush and local forest fires are handled by the local fire department. Such fires are not considered to have any impact on the station because the site is cleared and the fire cannot propagate to station buildings or equipment
6	Frost, Hail, Snow, Ice Cover	Both the Reactor Building and the Auxiliary Building are designed for a combination of snow, ice, and rain. Low winter temperatures causing failure of instruments is included in the plant trip frequency data.
7	Hurricane	[Hurricanes are handled under the high winds analysis.] The effect of water from a hurricane is considered similar to the effect of intense precipitation.
8	Landslide	Landslides are considered an insignificant hazard at Catawba. The Standby Nuclear Service Water Pond (SNSWP) dam is the only natural or man-made slope which, upon failure, would prevent safe shutdown of the plant. Therefore, the SNSWP was statically designed for stability under all loading conditions.
9	Lightning	The most probable effect of lightning is the loss of off-site power due to a strike in the switchyard. These occurrences are accounted for in the loss of off-site power initiating event frequency.

Table 2
Catawba Screening Justifications for Other External Initiating Events

	Event	Remarks
10	Meteorite	This event has significantly lower frequency than other events with similar uncertainties. The occurrence of a meteorite event could not result in worse consequences than other external events of a higher frequency. Therefore, this event is excluded because it will not significantly influence the total risk.
11	Intense Precipitation	Per response to NTTF 2.1, Catawba meets its licensing basis for local intense precipitation and thus screens out per Section 6, SPR EXT-B1 of the ASME / ANS Standard.
12	River Diversion	No present means exist to divert or reroute the river flow through the dams other than insignificant amounts of water used for municipal supply.
13	Sandstorm	Catawba is located more than 150 miles from the nearest area with a large sand deposit. The likelihood of occurrence is insignificant
14	Seiche	Since the flood examined in the [U]FSAR[5] uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.
15	Soil Shrink-Well Consolidation	Per the Catawba [U]FSAR, hazards associated with soil shrink-well consolidation will be insignificant
16	Storm Surge	Since the flood examined in the [U]FSAR uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.
17	Tsunami	Catawba is located more than 150 miles from the nearest coastal area at an elevation of 760 ft. mean sea level. Therefore, tsunami effects are insignificant.
18	Turbine-Generated Missile	The majority of the structures at Catawba are located either along or within close proximity to the longitudinal centerlines of the respective turbines. Calculations on turbine missiles prepared for the Catawba [U]FSAR indicate that the contribution to plant risk from the turbines would be insignificant
19	Volcanic Activity	No active volcanoes exist within the vicinity of Catawba.
20	Waves	Since the flood examined in the [U]FSAR uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.

[5] Duke Power Company, Catawba Nuclear Station Final Safety Analysis Report, 1992 Revision

Request for Additional Information RAI-10

Risk Calculations for the NSWS CT Extension

The Catawba LAR states that the proposed change to the NSWS TS CT has been evaluated using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3 of RG 1.177 provides guidance on PRA modeling detail needed for TS changes. Section 2.3.3.1 of RG 1.177 states that the PRA "model should also be able to treat the alignments of components during periods when testing and maintenance are being carried out." It also states that "[s]ystem fault trees should be sufficiently detailed to specifically include all the components for which surveillance tests and maintenance are performed and are to be evaluated."

It is not clear how certain aspects of the risk evaluation in support of the LAR meet the guidelines in RG 1.177, Revision 1. Therefore, the NRC staff requests the following additional information:

a) Section 3.2.3.1, "Model Change Overview," of the LAR states:

> "For the High Winds, Fire and Internal Flooding models, the PRA model is a single unit model that generally assumes A-Train equipment is running with B Train equipment in standby."

However, the high winds PRA risk results reported in LAR Tables 3.2.4.4-1 and 3.2.4.4-2 are different across units for each case (i.e., base case, CT case, non-CT case). Similarly, the fire PRA risk results reported in LAR Tables 3.2.4.3-1 and 3.2.4.3-2 are different across units for each case. [Note, the non-CT case is similar to the base case except that it assesses the risk for normal at-power operation with the nominal component unavailability values applied over the time during the year when the NSWS is not in the CT configuration.]

i. For the high winds, fire, and internal flooding "single unit" PRA models (i.e., the single unit PRA is assumed to represent both units), explain how the PRA models are representative or bounding (e.g., the most limiting) for Units 1 and 2. [The NRC staff notes that the internal events results show a difference between units.] Include a discussion of how systems, structures, and components (SSCs) that are shared between both units were implicitly or explicitly modeled in the single unit PRA models, and how differences between the single unit PRA models and Units 1 and 2 for all risk significant systems do not change the conclusions of the LAR.

ii. If the single unit PRAs cannot be justified because the PRAs do not reflect the differences between units, then update the PRAs used for this LAR that aggregate the PRA updates requested in APLA RAI 11 to reflect the difference between units.

iii. While the LAR states that the PRA models for high winds and fire are single unit models, explain why the reported risk results for these hazards (i.e., LAR Tables 3.2.4.3-1, 3.2.4.3-2, 3.2.4.4-1, and 3.2.4.4-2) are different between units for each case. If the reported values are incorrect, provide the correct risk estimates determined for this application after new PRA results are generated in response to APLA RAI 11.

iv. Explain why the internal flooding risk results reported in LAR Tables 3.2.4.2-1 and 3.2.4.2-2 are the same between the base case and non-CT case for CDF. If the reported values

are incorrect, provide the correct risk estimates determined for this application after new PRA results are generated in response to APLA RAI 11.

b) Section 2.3.4 of RG 1.177, Revision 1, states:

"When calculating the risk impacts (i.e., a change in CDF or LERF caused by CT changes), the change in average CDF should be estimated using the mean outage times (or an appropriate surrogate) [i.e., use the average test and maintenance model] for the current and proposed CTs. If a licensee chooses to use the zero-maintenance state as the base case (i.e., the case in which no equipment is unavailable because of maintenance), an explanation stating so should be part of the submittal."

While LAR Section 3.2.3.2 does substantiate certain maintenance events being set to zero for the CT case, the LAR does not specify whether the average test and maintenance (TM) model or zero TM model was used for the risk evaluations associated with the base case, CT case, and non-CT case.

i. Specify whether the average TM model or zero TM model was used for the risk evaluations associated with the base case, CT case, and non-CT case in support of this LAR.

ii. If the zero TM model was used for any of these cases, then justify its use. If a justification cannot be provided, then demonstrate (such as via a sensitivity study using the combined PRA updates considered in the response to APLA RAI 11) that use of the average TM model would not change the conclusions of this LAR.

   Alternatively, incorporate the average TM model in the risk evaluation supporting the LAR that aggregate the PRA updates requested in APLA RAI 11.

iii. Clarify which SSCs from LAR Table 3.2.6-1, "Tier 2 SSCs," were assumed to be available (i.e., not unavailable due to TM, or the TM basic event was set to zero) for the CT case. For these SSCs that are assumed to be available in the CT case, explain how the licensing basis will ensure that these SSCs will be prevented from being taken out-of-service during the NSWS CT.

c) Clarify whether the Emergency Supplemental Power Source (ESPS) is credited in the risk evaluations associated with the base case, CT case, and non-CT case. If credited, provide a brief description of how it is credited.

Duke Energy Response:

a) i) The SSCs that are shared among Units 1 and 2 include:

- Nuclear Service Water (RN) System - NSWS
- Instrument Air (VI) System
- Standby Shutdown (SS) System electrical power

High Winds

Previous CNS high wind analysis models used Unit 1 SSCs and shared unit SSCs. The applicability of the results to Unit 2 was considered to be realistic due to the

high level of symmetry between units[6,7]. However, the current high wind analysis (i.e., the version used for the risk analysis to support the NSWS T.S. LAR)[8] is comprised of site-specific models for Unit 1 and Unit 2. Hence, RAI 10.a.i is not applicable for the high winds PRA since the two units are modeled explicitly.

Fire

The FPRA model was initially developed as a Unit 1 only model. Subsequently, a FPRA model was developed for Unit 2 using the Unit 1 fault tree but with Unit 2 equipment and cable routing mapped to their corresponding basic events in the Unit 1 fault tree. The Unit 2 basic event mapping (to components) process was performed consistent with the process followed for Unit 1. The component footprint (i.e., cable location information) necessary for performing a compartment analysis from a Unit 2 perspective was developed similar to the Unit 1 component footprint based on DATATRAK data. DATATRAK contains cable location information for both Units 1 and 2 as well as common areas. Therefore, the data presented for Unit 2 is consistent with a Unit 2 fire.

Internal Flood

The PRA model used to determine consequences from an internal flooding scenario only includes Unit 1 SSCs, with several instances of credit taken for Unit 2 SSCs that support Unit 1. This model was found to be applicable to both units with only minor differences noted and accounted for. The only significant difference identified is that the feedwater tempering line is not secured at power for Unit 1, which allows for potentially diverting flow from the auxiliary feedwater pumps should a break occur in the main feedwater piping inside the doghouse of Unit 1 that results in the loss of both main and auxiliary feedwater. This is the only difference that impacts accident sequences or success criteria and it does not significantly impact the quantification results. However, because of this difference between units, the Unit 1 internal flooding PRA results bound the results for Unit 2. This unit difference has been captured in the flood scenario characterization[9].

Conclusion

The Unit 1 results are deemed applicable to Unit 2 due to nearly identical SSC design and operation, and similar spatial configuration. Appropriate modeling changes were made in consideration of the differences between the units as described above. There are therefore no changes to the conclusions of the LAR risk analysis.

ii) N/A

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[6] CNC-1535.00-00-0154, Catawba Nuclear Station Units 1 and 2 High Wind / Missile PRA Analysis, Rev. 0

[7] CNC-1535.00-00-0154, Catawba Nuclear Station Units 1 and 2 High Wind / Missile PRA Analysis, Rev. 1

[8] CNC-1535.00-00-0154, Catawba Nuclear Station Units 1 and 2 High Wind / Missile PRA Analysis, Rev. 2

[9] CNC-1535.00-00-0151, Flood PRA Modeling and Quantification for Catawba Nuclear Station Units 1 & 2, Rev. 1

iii) **High Winds**

As stated in Part a) i) above, previous CNS high wind analysis models used Unit 1 SSCs and shared unit SSCs. The applicability of the results to Unit 2 was considered to be realistic due to the high level of symmetry between units. However, the current high wind analysis (i.e., the version used for the risk analysis to support the NSWS T.S. LAR) is comprised of separate models for Unit 1 and Unit 2. Hence, the values for CDF and LERF are not equal to each other.

Fire

While the Unit 1 and Unit 2 FPRA results are based on the same fault tree, the components and associated cable routings mapped to the basic events in the fault tree are unit-specific. This accounts for the difference in results between the two units.

iv) The Catawba internal flooding analysis was based upon the Rev. 3b internal events model, which is based on Unit 1 only. This model was found to be applicable to both units with only minor differences noted and accounted for. This explains why the internal flooding risk results reported in LAR Tables 3.2.4.2-1 and 3.2.4.2-2 are the same between the base case and non-CT case for CDF.

b) i) The average TM model was used for the risk evaluations associated with the base case, CT case, and non-CT case in support of this LAR.

ii) N/A - The average TM model was used.

iii) The SSCs from LAR Table 3.2.6-1 that are assumed to be available for the CT case were the NSWS, 4160V ac power (Essential Buses ETA and ETB) and the EDGs. The switchyard is not modelled in the PRA. The remainder of the SSCs assumed nominal maintenance during the CT.

Regarding the prevention of these SSCs from being taken out-of-service during the NSWS CT, as outlined in the Notes section of Section 3.2.1.14, entry in the NSWS Single Pond Return Header Operation is restricted to having both trains of EDGs, NSWS Pumps, and NSWS supplied equipment on both units operable. Performing scheduled or planned maintenance that renders the EDGs, NSWS pumps, or NSWS supplied equipment both inoperable and unavailable on either train of NSWS of either unit is prohibited while NSWS is aligned for Single Pond Return Header Operation with one exception. For the EDGs, a monthly periodic test is performed to confirm operability. Prior to starting the EDG, a "bar and roll" of the EDG is performed. This renders the EDG inoperable but available, and is allowed while the NSWS is aligned for Single Pond Return Header Operation.

Maintenance and testing of the 4160V ac Essential Buses is not performed during at-power conditions and will thus be maintained available during the CT. Likewise, protection measures will be in place so as to limit access to the switchyard during the CT.

Catawba site Regulatory Affairs is responsible creating and tracking all planned action items generated for this LAR submittal. As such, several Nuclear Task Management (NTMs) tracking items have been created to ensure they are implemented including, as necessary, incorporation into the licensing basis.

c) The Emergency Supplemental Power Source (ESPS) was NOT credited in the risk evaluations performed in support of this LAR.

Request for Additional Information RAI-11

Reasonableness of Human Error Probabilities for Operator Actions

The Catawba LAR states that the proposed change to the TS CT has been developed using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3.1 of RG 1.177 states that the technical adequacy of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. Section 2.3.2 of RG 1.174 states that the risk assessment supporting a risk-informed LAR should properly account for the effects of the changes on operator actions.
Section 3.2.3.2 of the LAR states:

> "For HRA considerations, recovery actions are planned to address events such as swapping the NSWS suction and discharge back to Lake Wylie, restoring power to key MOVs [motor-operated valves] for isolation and local operation of manual valves for isolation. The internal events, internal flooding and high winds analyses used a bounding analysis in which such potential accident sequence recoveries were added to the model but were not credited (i.e., HRA events set to 1.0). ... However, in assessing the fire CT risk results, it became necessary to quantify these values to address the spurious operation of MOVs."

Based on the LAR, additional recovery actions were credited in the fire CT risk evaluation to address spurious operation of MOVs. This demonstrates the importance of calculating realistic HEPs for these recovery actions. It is not clear whether the licensee has the applicable procedures in place for these recovery actions, which could query the validity of the analysis of these actions.

To understand the full characterization of the risk estimates, address Part (a) or Part (b) below:

a) Describe the new recovery actions credited in the fire PRA in support of the LAR. Also, describe any previously modeled recovery actions whose HEPs were modified in the fire PRA in support of the LAR. For the recovery actions identified above, provide the following additional information:

i. Explain how the recovery actions' HEPs were developed or modified. Provide sufficient details and numerical values to understand the basis for these HEPs, including:

- A discussion of the specific actions and instructions for these recovery actions, including the cues or indications operators will use to initiate these actions. Provide a timeline for these operator actions, and how the time available and time required to complete operator actions were estimated.

- Explain whether the modeling and feasibility study of these recovery actions were performed consistent with guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," July 2012 (ADAMS Accession No. ML12216A104). Otherwise, justify the basis for the HRA of these operator actions.

ii. If any recovery actions/HEPs discussed in Part (i) cannot be justified, then modify the HRA using a defensible basis and incorporate the results into the fire PRA used for this LAR that aggregate the PRA updates requested in APLA RAI 15. Explain how the HRA was modified and provide sufficient details to justify the basis for the modification(s).

iii. For each recovery action credited in the fire CT case but not credited in the fire base case, confirm the recovery action is new to the plant in support of the LAR (e.g., new changes made to the plant design/plant procedures as a result of this LAR and are not in the current licensing basis/plant procedures).

For each recovery action where confirmation cannot be provided, include credit for the recovery action in the fire base case used for this LAR that aggregate the PRA updates requested in APLA RAI 15.

iv. Propose a mechanism (e.g., a license condition) to complete the following prior to implementing the TS 30-day CT for Single Pond Return Header Operation of the NSWS: (1) provide/revise plant procedures and required training, as necessary, for the credited recovery actions; (2) update the HEPs, fire PRA, and risk estimates associated with this LAR, as needed, to be consistent with these procedures and training; and (3) confirm that the updated risk estimates associated with this LAR meet the RG 1.174 and RG 1.177 risk acceptance guidelines. Also, include in this mechanism a plan of action should the updated risk estimates associated with this LAR exceed the RG 1.174 and RG 1.177 risk acceptance guidelines. Alternatively, justify why this mechanism is not needed.

b) Alternatively, remove credit for these recovery actions in the fire PRA used for this LAR that aggregate the PRA updates requested in APLA RAI 15.

Duke Energy Response:

a. Two new operator actions were developed and credited in Fire PRA supporting the CNS Nuclear Service Water System Single Pond Return Header LAR:
- WRNFLDVDHE – Operator Fails to Address Flow Diversion
- WRNLKWYDHE – Operator Fails to Align Flow From Lake Wylie

No previously modeled recovery actions credited in the Fire PRA were modified in support of this LAR.

i. These operator actions were initially developed using a conservative screening value of 1.00E-02. Detailed analysis has since been performed for these HFEs utilizing guidance found in NUREG-1921 and the EPRI HRA Calculator tool. The development of each action is discussed below:

a) WRNFLDVDHE –

During the SNSWP return header alignment, if a fire event causes valve 1RN57A (normally-closed isolation valve) to transfer position, flow will be diverted away from the operating SNSWP return line and discharge

through open manways in the isolated (non-operating) return line. The spurious opening of 1RN57A coupled with normally-open isolation valve 1RN843B during this alignment can lead to draining of the SNSWP.

Discussions with CNS engineers and Operations staff determined that operators would receive multiple alarms / indications on the Operator Aid Computer (OAC) that would inform them of SNSWP level decreasing. Based on discussions with Operations, the SNSWP Level will typically be around 573 - 574 ft. The SNSWP low level alarm set point is 572 ft. and the low-low level alarm set point 571.5 ft. (Minimum Tech. Spec. level is 571 ft.)

Assuming the SNSWP level is at 573.5 ft at the start of the event, using estimated pond volumes from CNS Engineering, the volume of water between 573.5 ft and 571 ft is 75 acre-ft (~24 million gal.)[10]. At a manway discharge rate of 18591[1] gpm, it will take approximately 21.9 hours to reach 571 ft (2.5 ft decrease in pond level). Ops would receive a SNSWP low level alarm at 572 ft in 13.14 hrs. (1.5 ft decrease in pond level).

Thus, for HEP calculation purposes, the total time window available before the SNSWP reaches the minimum Tech. Spec. level is 21.9 hours and the operator cognitive time (when the low level indication is received) is conservatively applied to be 13.14 hrs. Further, it is assumed it will take 20 minutes to reach the Auxiliary Building from the Control Room and 5 minutes to manipulate the valve, based on discussions with Ops.

Using CBDTM & THERP methodologies in the EPRI HRA Calculator tool, the HEP value is calculated to be <u>3.09E-04</u>. (Note that using a full pond elevation of 574 ft. will not change this value.) As mentioned above, the modeling and feasibility of these actions were performed consistent with the guidance in NUREG-1921.

b) WRNLKWYDHE –

During the SNSWP return header alignment, if a fire event causes damage to the normally closed isolation valves on the return path to Lake Wylie, such that they spuriously open, this could result in flow being diverted to Lake Wylie and subsequently draining the SNWS Pond. Operators will have to manually close these valves.

Discussions with CNS engineers and Operations staff determined that operators would receive multiple alarms / indications on the Operator Aid Computer (OAC) that would inform them of SNSWP level decreasing. Based on discussions with Operations, the SNSWP Level will typically be around 573 - 574 ft. The SNSWP low level alarm set point is 572 ft. and the low-low, level alarm set point 571.5 ft. (Minimum Tech. Spec. level is 571 ft.)

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[10] Based on volume information provided in Duke calc. CNC-1223.24-00-0072.

Assuming the SNSWP level is at 573.5 ft at the start of the event, using estimated pond volumes from CNS Engineering, the volume of water between 573.5 ft and 571 ft is 75 acre-ft (~24 million gal.)[11]. At a header discharge rate of 20300[2] gpm, it will take approximately 20 hours to reach 571 ft (2.5 ft decrease in pond level). Ops would receive a SNSWP low level alarm at 572 ft in 12 hrs. (1.5 ft decrease in pond level).

Thus, for HEP calculation purposes, the total time window available before the SNSWP reaches the minimum Tech. Spec. level is 20 hours and the operator cognitive time (when the low level indication is received) is conservatively applied to be 12 hrs. Further, it is assumed it will take 20 minutes to reach the Auxiliary Building from the Control Room and 5 minutes to manipulate the valve, based on discussions with Ops.

Using CBDTM & THERP methodologies in the EPRI HRA Calculator tool, the HEP value is calculated to be <u>5.49E-03</u>. (Note that using a full pond elevation of 574 ft. will not change this value.) As mentioned above, the modeling and feasibility of these actions were performed consistent with the guidance in NUREG-1921.

ii. N/A.

iii. The operator actions developed for this response are new to the plant in support of the LAR.

iv. There will be a license condition prior to implementing the 30-day CT for the NSWSP Single Return Header to address the following:

a) The plant engineering process will be utilized to develop new plant procedures and required training to support this alignment and new operator actions credited in the PRA.

b) HEPs for the two new operator actions developed in support of this LAR will be updated as needed to be consistent with the updated procedural guidance and training. Risk estimates will also be updated to include the updated HEPs.

c) After the HEPs are updated, it will be confirmed that the risk estimates associated with this LAR are within the acceptance guidelines of RG 1.177 and RG 1.174. If the risk estimates are not within the acceptance guidelines of RG 1.177 and RG 1.174, additional risk reduction measures will be taken as needed to ensure that the acceptance guidance are met.

The values for these HFEs have been updated in the aggregate cutsets associated with the RN Single Pond Header Alignment. See table below for new FPRA numbers.

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[11] Based on volume information provided in Duke calc. CNC-1223.24-00-0072.

Request for Additional Information RAI-12

Tier 2, Avoidance of Risk-Significant Plant Configurations

The Catawba LAR states that the proposed change to the TS CT has been developed using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3 of RG 1.177, Revision 1, cites the need to avoid risk-significant plant configurations and discusses Tier 2 of the RG 1.177 three-tiered approach for evaluating risk associated with proposed TS CT changes. According to RG 1.177, Tier 2, the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service consistent with the proposed TS change. Once the specific plant equipment is identified, an assessment can be made as to whether certain enhancements to the TS or procedures are needed to avoid risk-significant plant configurations. In addition, RG 1.177, Section 2.4 states, as part of the acceptance guidelines specific to permanent CT changes, the licensee should demonstrate that there are appropriate restrictions on dominant risk-significant configurations associated with the change.

Section 3.2.6, "Tier 2 Component Evaluation," of the LAR provides a discussion of Tier 2 (i.e., avoidance of risk-significant plant configurations) and identifies in Table 3.2.6-1 those SSCs that are important to the 30-day NSWS CT. Furthermore, LAR Section 3.2.6 states that unavailability of these SSCs should be avoided during the CT. However, with the exception of NSWS and emergency diesel generator (EDG) SSCs, the LAR does not describe a mechanism or a set of controls that will be used by the plant to avoid the unavailability of these SSCs.

To address the observations above, explain how the unavailability of SSCs identified in LAR Table 3.2.6-1 (which represent high risk configurations) will be avoided during the 30-day NSWS CT. Include a detailed discussion of the mechanism that ensures these high-risk configurations will be avoided.

Duke Energy Response:

Duke Energy relies on several methods to limit work on high risk configurations. These methods consist of Technical Specifications (Tech Specs) and Selected Licensee Commitments (SLC), Cycle Schedule, Protected Equipment schemes, and the Electronic Risk Assessment Tool (ERAT).

Tech Specs and SLC specify requirements for SSCs to be operable or functional. Tech Specs and SLC specify a completion time (CT) for SSCs. Generally, when multiple trains are out of service, the CT is very short or a shutdown is required. In the case of NSWS Single Pond Return Header Operation, work will be performed per proposed new Condition D of Catawba NSWS TS 3.7.8 and entry into this Condition shall only be allowed for pre-planned activities as outlined in the accompanying Technical Bases.

During the CT, planned or discretionary maintenance that renders one or more NSWS pumps and / or the associated EDGs inoperable and unavailable on either train of NSWS is prohibited while in the Single Pond Return Header alignment <u>with one exception</u>. For the EDGs, a monthly periodic test is performed to confirm operability. Prior to starting the EDG, a "bar and roll" of the EDG is performed. This renders the EDG inoperable but available, and is allowed while the NSWS is aligned for Single Pond Return Header Operation. Protected

equipment plans will be developed for important SSCs. These plans are maintained by the Operations group. Duke procedure AD-OP-ALL-0201[12] (Protected Equipment) provides guidance for the management of protected equipment.

Duke Energy's online work management practices are described in AD-WC-ALL-0200[13] (On-Line Work Management.) A key provision of this practice is the use of a Cycle Schedule. "Plant systems are grouped in a rotating cycle of Work Weeks. System groupings are based on Technical Specification requirements, Probabilistic Risk Assessment (PRA) and resource loading."

Work on those SSCs which is not prohibited by Tech Specs or SLC, the Cycle Schedule, or the Protected Equipment Plan will be managed using the Electronic Risk Assessment Tool (ERAT). As outlined in Duke procedure AD-NF-ALL-0501[14] (Electronic Risk Assessment tool (ERAT)), Duke manages this process using a blended (i.e., quantitative and qualitative) configuration risk assessment approach. The ERAT calculates the CDF and LERF for equipment out of service. The tool displays the risk as one of four colors - Green (lowest), Yellow, Orange, or Red (highest.) Colors above Green represent a configuration where the ICCDP (ICLERP) could exceed 1.0E-06 (1.0E-07) within 7 days. Colors above Green receive extra review, consideration of risk management plans, and consideration of rescheduling to remove or reduce the color. Catawba uses the PARAGON software program to analyze plant risk via a "look-ahead" of plant configurations over a specified period of time. Prior to entering the extended CT, PARAGON operators will review the plant schedule to identify and correct any significant potential risk impacts occurring during the CT. During the CT, risk will be monitored and any emergent risk configurations will be addressed appropriately. Duke Energy's configuration risk management program requires the implementation of risk management actions to help alleviate risk when risk significant configurations are entered. Thus, plant risk will be effectively managed prior to and during the extended CT.

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[12] AD-OP-ALL-201, "Protected Equipment", Rev. 4
[13] Duke Procedure, AD-WC-ALL-0200, "On-Line Work Management", Rev. 13
[14] Duke Procedure, AD-NF-ALL-0501, "Electronic Risk Assessment tool (ERAT)", Rev. 1

Request for Additional Information RAI -13

Tier 3, Risk-Informed Configuration Risk Management

The Catawba LAR states that the proposed change to the TS CT has been developed using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3 of RG 1.177, Revision 1, cites the need to establish an overall configuration risk management program to ensure that other potentially lower probability, but nonetheless risk-significant configurations resulting from maintenance and other operational activities are identified and compensated for (Tier 3).

The LAR does not address Tier 3 during the NSWS 30-day CT. Explain how Tier 3 in RG 1.177 will be met during the NSWS 30-day CT. Discuss the mechanism (e.g., programs and procedures) that ensures: (1) risk-significant plant configurations resulting from maintenance or other operational activities are identified in a timely manner, (2) appropriate compensatory measures are taken to avoid risk-significant configurations, and (3) the associated risk impact is appropriately assessed and managed.

Duke Energy Response:

Tier 3 of RG 1.177 requires the licensee to provide assurance of compliance with 10 CFR 50.65(a)(4) to ensure the risk impact of taking equipment out of service is appropriately assessed and managed. As outlined in Duke procedure AD-NF-ALL-0501[15], Duke manages this process using a blended (i.e., quantitative and qualitative) configuration risk assessment approach with its Electronic Risk Assessment Tool (ERAT). Catawba uses the PARAGON software program to analyze plant risk via a "look-ahead" of plant configurations over a specified period of time. Prior to entering the extended CT, PARAGON operators will review the plant schedule to identify and correct any significant potential risk impacts occurring during the CT. During the CT, risk will be monitored and any emergent risk configurations will be addressed appropriately. Duke Energy's configuration risk management program requires the implementation of risk management actions to help alleviate risk when risk significant configurations are entered. Thus, plant risk will be effectively managed prior to and during the extended CT.

[15] Duke Procedure, AD-NF-ALL-0501, "Electronic Risk Assessment tool (ERAT)", Rev. 1

Request for Additional Information RAI-14

Sensitivity and Uncertainty Analyses Relating to Assumptions in Technical Specification Change Evaluations

The Catawba LAR states that the proposed change to the TS CT has been developed using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Section 2.3.5 of RG 1.177 states:

> "As in any risk-informed study, risk-informed analyses of TS changes can be affected by numerous uncertainties regarding the assumptions made during the PRA model's development and application. Sensitivity analyses may be necessary to address the important assumptions in the submittal made with respect to TS change analyses."

RG 1.177 relies on RG 1.200, Revision 2, for addressing the technical adequacy of the PRA. Regulatory Guide 1.200, Revision 2, Section 3.3.2, states, "for each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application."

The LAR does not address sensitivity and uncertainty analyses relating to assumptions made in the PRAs and risk evaluations in support of this LAR. The NRC staff's review of the information in the LAR has identified additional information needed to understand the full characterization of the risk estimates.

a) Describe the key assumptions and key sources of uncertainty identified in the PRAs used to support the LAR. [RG 1.200, Revision 2, Section 3.3.2, defines the terms "key assumption" and "key source of uncertainty."] Discuss how each key assumption and key source of uncertainty identified above was dispositioned for this application.

b) Describe the approach used to identify and characterize the "key" assumptions and "key" sources of uncertainty in the PRAs for this application.

Duke Energy Response:

RAI 14.a

Regulatory Guide 1.200, Revision 2, Section 3.2.2, defines the following terms:

A *key assumption* is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. For the base PRA, the term "different results" refers to a change in the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the

technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

A *key source of uncertainty* is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF estimates significant enough to affect insights gained from the PRA.

The generic sources of modeling uncertainties from EPRI Report 1016737 have been evaluated for the base PRA models supporting this application, as well as for the additional changes made for this application. Generic sources of uncertainty identified in WCAP-16304-P and plant-specific sources of uncertainty have also been evaluated for this application. Each source of modeling uncertainty, as well as each assumption, has been assessed for its potential impact on this application, and none of the items has been identified to be a key assumption or key source of uncertainty for the RN TS CT application.

RAI 14.b

Modeling assumptions and sources of uncertainty were reviewed even though Section 2.3.5 of Regulatory Guide 1.177 states that previous sensitivity studies have shown that TS CT changes are relatively insensitive to uncertainties due to the base case and change case being similarly changed. The process for identifying "key" assumptions and "key" sources of uncertainty followed NUREG-1855, as summarized below:

1. *Identification of Sources of Model Uncertainties and Related Assumptions of the Base PRA*–Both generic and plant-specific sources of model uncertainty and related assumptions for the base PRA are identified and characterized. These sources of uncertainty and related assumptions are those that result from developing the PRA model.

2. *Identification of Sources of Model Uncertainties and Related Assumptions Relevant to the Application* – The sources of model uncertainty and related assumptions in the base PRA that are relevant to the application are identified. This identification may be performed with a qualitative analysis. This analysis is based on an understanding of how the PRA is used to support the application and the associated acceptance criteria or guidelines. In addition, new sources of model uncertainty and related assumptions that may be introduced by the application are identified.

3. *Identification of Key Sources of Model Uncertainties and Related Assumptions for the Application* – The sources of model uncertainty and related assumptions that are key to the application are identified. Quantitative analyses of the importance of the sources of model uncertainty and related assumptions identified in the previous steps are performed in the context of the acceptance guidelines for the application. The analyses are used to identify any credible alternative modeling hypotheses that could impact the decision. These hypotheses are used to identify which of the sources of model uncertainty and related assumptions are key to the application.

Upon completion of the above described process, none of the generic or plant-specific assumptions or sources of uncertainty was identified as a key assumption or key source of uncertainty for the RN TS CT application.

References:

1. 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.
2. CNC -1535.00-00-0174, Catawba Nuclear Station Success Criteria Notebook, Revision 1.
3. CNC-1535.00-00-0175, CNS Accident Sequence Development Notebook, Revision 0.
4. CNC-1535.00-00-0201, CNS Loss of Offsite Power (LOOP) Restoration Assessment, Revision 2.
5. CNC-1535.00-00-0206, Catawba Internal Events Unit 1 PRA Working Model and Quantification-Task 1 ESPS Revision, Revision 0.
6. CNC-1535.00-00-0207, CNS Assessment of Need to Split Dynamic Human Error Events Used in Multiple Accident Sequence Models-Task 1 ESPS Revision, Revision 0.
7. WCAP-16304-P, Strategy for Identifying and Treating Modeling Uncertainties in PRA Models: Issues Concerning LOCA and LOOP, Revision 0.
8. ML13331A673, U.S. Nuclear Regulatory Commission Staff Review of the Documentation Provided by Duke Energy Carolinas, LLC for the Catawba Nuclear Station Units 1 and 2 Concerning Resolution of Generic Letter 2004-02 Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors, December 31, 2013.
9. CNC-1535.00-00-0176.16, Catawba Nuclear Station Probabilistic Risk Assessment, Appendix 16: Heating, Ventilation, and Air Conditioning System Notebook (VC, YC), Revision 1.
10. CNC-1535.00-00-0176.09, Catawba Nuclear Station Probabilistic Risk Assessment, Appendix 9: Standby Shutdown Facility Notebook (SSF), Revision 1.
11. CNC-1535.00-00-0215, Catawba Internal Events Unit 2 PRA Working Model and Quantification-Task 1 ESPS Revision, Revision 0.
12. CNC-1535.00-00-0217, Catawba Nuclear Station Post-Initiator Human Reliability Analysis, Revision 0.
13. CNC-1223.24-00-0072; RN Single Pond Return Header Design Basis; Rev. 0
14. EPRI Report 1016737, Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments, December 2008.
15. USNRC Regulatory Guide 1.177, An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications, Revision 1.

16. USNRC Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2.
17. USNRC NUREG 1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Vol. 1, March 2009.
18. CNC -1535.00-00-0180, Catawba Nuclear Station Probabilistic Risk Assessment Section 8.0: Model Integration and Quantification, Revision 1.
19. CNC-1535.00-00-0176.17, Catawba Nuclear Station Probabilistic Risk Assessment, Appendix 17: DC Power System Notebook (DCP), Revision 1.

Request for Additional Information RAI-15

Aggregate Update Analysis

Regulatory Guide 1.174, Revision 2, provides quantitative guidelines on CDF and LERF and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. Regulatory Guide 1.177, Revision 1, provides risk acceptance guidelines on incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) and identifies acceptable changes to these probabilities that result from proposed permanent TS changes. The NRC staff's review of the information in the LAR has identified additional information that is needed to understand full characterization of the risk estimates.

The PRA methods and treatments discussed in the following APLA RAIs may need to be revised to be acceptable by the NRC:

- APLA RAI 06.b.iv, regarding incorporating the dependency analysis update performed for the internal events PRA into the fire and high winds PRAs.
- APLA RAI 07.c, regarding incorporating internal events PRA updates into the fire and high winds PRAs.
- APLA RAI 09, regarding incorporating other external hazards.
- APLA RAI 10.a, regarding updating the single unit PRAs to reflect the differences between Unit 1 and Unit 2.
- APLA RAI 10.b, regarding incorporating the average TM model into the PRAs.
- APLA RAI 11.a.ii, regarding incorporating more appropriate recovery action HEPs into the fire PRA.
- APLA RAI 11.a.iii, regarding incorporating credit for recovery actions in the fire base case.
- APLA RAI 11.b, regarding removing credit for recovery actions in the fire PRA.

To address the RAIs cited above, provide the following:

a) For the PRA updates identified in response to the RAIs cited above, provide the results of an aggregate analysis that reflect the combined impact of these PRA updates on the LAR risk results (i.e., change in CDF, change in LERF, ICCDP and ICLERP).

b) For each APLA RAI listed above that resulted in changes to the PRA or risk assessment, summarize briefly how the issue(s) cited in the RAI was resolved in the PRA and/or LAR. If the resolution involved an update to the PRA models, then briefly summarize the PRA update.

c) Confirm that the updated results still meet the risk acceptance guidelines in RG 1.177, Revision 1, and RG 1.174, Revision 2.

d) If the risk acceptance guidelines are exceeded, then identify which risk acceptance guidelines are exceeded and provide qualitative or quantitative justification that support the conclusions of the LAR. If applicable, include discussion of conservatisms in the analysis and the risk significance of these conservatisms.

Duke Energy Response:

The RAI responses listed above were dispositioned as follows:

- APLA RAI 06.b.iv - In addressing F&O 22-7 to order the HRA combinations in the recovery rules by number of events, the external hazards recovery rules are not impacted.

- APLA RAI 07.b - As discussed in the associated RAI response, the acceptability of the fire PRA model for the NSWS LAR is justified by the peer review on the model and the resolutions of F&Os generated during those reviews. The high winds PRA is based on the Rev. 4 internal events model, which incorporated the 2015 internal events peer review.

- APLA RAI 09 - No additional external hazards were incorporated.

- APLA RAI 10.a - No updates were made to reflect the differences between Unit 1 and Unit 2.

- APLA RAI 10.b - The average TM model was used for the risk evaluations associated with the base case, CT case, and non-CT case in support of this LAR. No changes were made to the database used for the risk assessment regarding the maintenance values used.

- APLA RAI 11.a.ii - Not applicable since no additional recovery actions were incorporated into the fire analysis.

- APLA RAI 11.a.iii - Risk values reflecting updated HEPs are given in the response to the associated RAI and are included below.

- APLA RAI 11.b - No credits for recovery actions were removed from the fire PRA analysis.

a) The aggregate risk results presented in Section 3.2.4.8 of the LAR submittal were updated to reflect the impact of the fire PRA HEP update presented in RAI 11.a.iii as shown below:

3.2.4.8 ICCDP / ICLERP for 30-Day CT

The ICCDP and ICLERP for one entry into the T.S. are now computed. First, the delta CDF and LERF computed in the Sections A.4.1 thru A.4.4 above are tabulated below:

Table 3.2.4.8-1 – Base Case Risk, All Hazards

Risk Metric	Internal Events (/ yr.)	Internal Flood (/ yr.)	Fire (/ yr.)	High Winds (/ yr.)	Total (/ yr.)
U1 CDF	7.23E-06	4.08E-05	4.28E-05	8.13E-06	9.90E-05
U1 LERF	3.42E-07	7.64E-07	4.51E-06	2.16E-06	7.78E-06
U2 CDF	7.33E-06	4.08E-05	4.38E-05	7.43E-06	9.94E-05
U2 LERF	4.01E-07	7.64E-07	4.37E-06	2.19E-06	7.73E-06

Table 3.2.4.8-2 – 30-Day CT Risk, All Hazards

Risk Metric	Internal Events (/ yr.)	Internal Flood (/ yr.)	Fire (/ yr.)	High Winds (/ yr.)	Total (/ yr.)
U1 CDF	7.50E-06	4.05E-05	3.79E-05	6.29E-06	9.22E-05
U1 LERF	3.39E-07	7.32E-07	3.96E-06	1.53E-06	6.56E-06
U2 CDF	7.39E-06	4.05E-05	3.91E-05	5.94E-06	9.29E-05
U2 LERF	3.68E-07	7.32E-07	3.95E-06	1.54E-06	6.59E-06

Table 3.2.4.8-3 – Delta Risk, All Hazards

Risk Metric	Internal Events (/ yr.)	Internal Flood (/ yr.)	Fire (/ yr.)	High Winds (/ yr.)	Total (/ yr.)
U1 CDF	2.70E-07	-3.00E-07	-4.90E-06	-1.84E-06	-6.77E-06
U1 LERF	-3.00E-09	-3.20E-08	-5.50E-07	-6.30E-07	-1.22E-06
U2 CDF	6.00E-08	-3.00E-07	-4.70E-06	-1.49E-06	-6.43E-06
U2 LERF	-3.30E-08	-3.20E-08	-4.20E-07	-6.50E-07	-1.14E-06

Thus, for a 30-day CT, using the definitions for ICCDP and ICLERP presented earlier,

U1 ICCDP = $[(CDF_{CT\ Config.} - CDF_{Baseline})$ x (30 days) / 365 days/yr]
= (-6.77E-06) x 30 / 365
= <u>-5.56E-07</u>

U1 ICLERP = $[(LERF_{CT\ Config.} - LERF_{Baseline})$ x (30 days) / 365 days/yr]
= (-1.22E-06) x 30 / 365
= <u>-1.00E-07</u>

U2 ICCDP = $[(CDF_{CT\ Config.} - CDF_{Baseline})$ x (30 days) / 365 days/yr]
= (-6.43E-06) x 30 / 365
= <u>-5.28E-07</u>

U2 ICLERP = $[(LERF_{CT\ Config.} - LERF_{Baseline})$ x (30 days) / 365 days/yr]
= (-1.14E-06) x 30 / 365
= <u>-9.37E-08</u>

Therefore, for both units, the ICCDP is less than 1E-06 and the ICLERP is less than 1E-07; therefore, these risk metrics meet the acceptance guidelines of RG. 1.177.

b) The overall impact of this plant change results in a reduction in CDF, LERF, ICCDP, and ICLERP. Incorporating the changes from the RAI responses above results in a further reduction in CDF, LERF, ICCDP, and ICLERP. No update to the fire PRA model occurred.

c) As shown in the response to Part a) above, the updated fire HEP values resulted in a further reduction in risk and these values are still well within the risk acceptance guidelines in RG 1.177, Revision 1, and RG 1.174, Revision 2.

d) N/A - The risk acceptance guidelines were not exceeded.

Request for Additional Information RAI-16

Seismic Evaluation

Consistent with Regulatory Position 2.3.2 of RG 1.177, Revision 1, the scope of the analysis should include all hazard groups (e.g., seismic in this case) unless it can be shown that the contribution from specific hazard groups does not affect the decision.

The licensee stated in LAR Section 3.2.4.6 that structures such as NSWS pump structure as well as the Standby Nuclear Service Water Pond (SNSWP) intake and discharge structure were screened from the seismic PRA analysis performed for the Catawba Individual Plant Examination of External Events submittal. The licensee concluded that the consideration of risk from seismic events "while the plant is in the SNSWP single return header configuration is not a significant factor for this assessment." It is not clear whether the licensee's qualitative assessment has considered impact of seismic failures for all SSCs that could affect the change in risk for this application.

Describe the seismic ruggedness of SSCs of which seismic failures may affect this application and demonstrate that the impact of seismic events, considering the re-evaluated hazard, on these SSCs would not affect the decision on the proposed single pond return header configuration.

Duke Energy Response:

Regarding the seismic ruggedness determination of the SSCs, as stated in Section 3.1.4 of the Catawba IPEEE submittal[16], structures with a median fragility greater than 2.5g and components with a median fragility greater than 2.0g were screened out of the model. This was the criteria used as the basis of determining whether an SSC was considered seismically rugged. Specifically for the NSWS, the equipment considered for this application includes the NSWS Intake Structure, the NSWS Pumps and all associated piping and valves. Per the fragility report used as a source for the IPEEE submittal[17], the median fragilities for these SSCs are listed as follows:

- NSWS Intake Structure - 2.9g
- NSWS Pumps - 2.05g
- Piping - > 2g
- Valves - > 2g

Thus, the SSCs related to the NSWS were all determined to be seismically rugged and were screened from the model.

Consideration must also be given to the sources required to power the NSWS pumps and associated valves as well as the equipment the NSWS serves. Considering a generic High Confidence of Low Probability of Failure (HCLPF) of 0.1g for a seismic-induced LOOP event, the EDGs become the predominant SSCs of interest for seismic scenarios. Again, per the

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[16] Duke Power Company, Catawba Nuclear Station, IPEEE Submittal Report, June 1994
[17] Nuclear Technical Services, Seismic Fragilities of Structures and Components at the Catawba Nuclear Station, Report No. NTS 19202.01, March 1986

fragility report referenced previously, the EDGs are very rugged (fragility > 2g); however, some of the support components such as the control panels have lower fragilities and were therefore included in the IPEEE analysis. By observation of the IPEEE cutsets, many of the scenarios are composed of LOOPs with subsequent random and seismic-induced failures of the EDGs. This would be true regardless of whether the IPEEE plant hazard or the updated plant hazard[18] were used to perform a seismic risk evaluation. Hence, the impact to NSWS pumps and associated valves as well as the equipment it serves would be the same.

Therefore, whether the NSWS is in its normal at-power configuration or in the single pond return header configuration, the delta seismic risk is the same.

[18] Lettis Consultants International, Inc.; "Catawba Seismic Hazard and Screening Report; Calculation of Seismic Hazards for CEUS Sites"; Project No. 1041; October 2013