Tom Simril
Vice President
Catawba Nuclear Station
Duke Energy
CN01VP | 4800 Concord Road
York, SC 29745
o: 803.701.3340
f: 803.701.3221

CNS-18-026

10 CFR 50.59
10 CFR 50.4

May 10, 2018

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

Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Nos.50-413 and 50-414
Renewed License Nos.NPF-35 and NPF-52

Subject: 2017/2018 10 CFR 50.59 Evaluation Summary Report

Attached please find the 2017/2018 10 CFR 50.59 Evaluation Summary Report. The report contains a brief description of changes, tests, and experiments, including a summary of the safety evaluations for Catawba Nuclear Station, Units 1 and 2. This report is submitted pursuant to the provisions of 10 CFR 50.59(d)(2) and 10 CFR 50.4.

If there are any questions regarding this submittal, please contact Tolani E. Owusu at (803) 701-5385.

Sincerely,

Tom Simril

Tom Simril
Vice President, Catawba Nuclear Station

Attachment: Catawba Nuclear Station, Units 1 and 2 2017/2018 10 CFR 50.59 Evaluation Summary Report

cc: (with Attachments)
C. Haney, USNRC Region II - Regional Administrator
J. Austin, USNRC Senior Resident Inspector - CNS
M. Mahoney, NRR Project Manager
S.E. Jenkins, Manager, Radioactive and Infectious Waste Management (SC)

Attachment
Catawba Nuclear Station, Units 1 and 2
2017/2018 10 CFR 50.59 Evaluation Summary Report

Action Request: 02047559

The proposed activity being evaluated is a revision to the fuel rod analysis methodology (Section 4.2.5) in the Westinghouse Fuel Transition methodology report (DPC-NE-2009-P-A). The proposed activity replaces the existing ZIRLO cladding corrosion model in DPC-NE-2009 Revision 3b with a cladding corrosion model applicable to both ZIRLO and Optimized ZIRLO. This revision is necessary because Optimized ZIRLO cladding will be used with RFA fuel beginning with Catawba 1 Cycle 24 (C1C24).

Specifically, methodology report DPC-NE-2009 Revision 3c implements WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A (Optimized ZIRLO) and Addendum 2-A (Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO), both of which have been previously approved by the NRC. DPC-NE-2009 methods are considered part of the Updated Final Safety Analysis Report (UFSAR) via reference, and as such, are considered "as described in the UFSAR."

This 10 CFR 50.59 evaluation is performed in accordance with guidance in Revision 1 of NEI 96-07 endorsed by the NRC via Regulatory Guide 1.187. It was concluded that the proposed activity is not a departure from a method of evaluation described in the UFSAR, and thus does not require NRC review and approval.

Action Request: 02085006

Duke Energy is upgrading the version of the VIPRE-01 thermal-hydraulics computer code from mod 2 as modified by Duke to mod 2.5 as modified by Duke (mod2.5adke). This code is used by the Duke Energy safety analysis group to simulate thermal-hydraulic conditions in the core to determine whether a rod or group of rods is undergoing departure from nucleate boiling (DNB). The code remains substantially the same. However, the default water properties functions, which approximate the steam tables for VIPRE, have been modified slightly by the code vendor to better match with the steam tables. It is possible this change could result in more or less margin (i.e. the results might show more or less boiling in the core). Therefore, this code change screened in to the evaluation. The computer code was examined to determine whether changes in aggregate made a substantive difference to the results during the Software Quality Assurance process. This process determined that results were within 0.5% of previous results. Per NEI 96-07 Section 4.3.8.1, "Licensees may change one or more element of a method of evaluation such that results move in the nonconservative direction without prior NRC approval, provided the revised result is 'essentially the same' as the previous result. Results are 'essentially the same' if they are within the margin of error for the type of analysis being performed." As provided by the vendor code manual, the uncertainty of the VIPRE computer code is between 1 and 5%. The change to the results is within code uncertainty. Therefore, this 50.59 evaluation has concluded that the results are essentially the same, does not represent a departure from a method of evaluation, and no prior NRC approval is required.

Action Request: 02089032

Three software applications (computer codes) used in reload reactor core design and safety analysis for Catawba Nuclear Station were updated to more recent versions supplied by software vendor Studsvik Scandpower. The current versions and the updated versions are shown in the table below.

Application	Current Vendor Version	Updated Vendor Version
CASMO-4	2.05.07	2.05.17
CMS-LINK	1.16.13_DUKE	1.27.04_DUKE
SIMULATE-3	6.07.09_MOX_4	6.09.33_DUK_1

This evaluation only covered the use of these three software applications for reload design and safety analyses in the Nuclear General Office Nuclear Fuel Engineering division. Note that the SIMULATE-3 software is used in the GARDEL core monitoring software application being considered for installation at Catawba Nuclear Station. However, the use of SIMULATE-3 in GARDEL will be addressed in a separate 10 CFR 50.59 screen/evaluation.

No revision to the Catawba UFSAR, Tech Specs or Selected Licensee Commitments was required.

The proposed activity was determined to consist solely of changes to methods of evaluation. Therefore, the proposed activity did not require evaluation against the criteria in 10 CFR 50.59(c)(2)(i-vii).

None of the three updated software applications were new or different methods of evaluation. All of them were modifications to existing, approved methods of evaluation.

The justification to Question 8 showed that the results from the updated versions of the three software applications were essentially the same as the results from the earlier versions approved by the NRC in methodology reports DPC-NE-1005-PA and DPC-NE-3001-PA. A calculation file was written specifically to document the comparison of the results of the different software versions and to facilitate the justification. Therefore, no License Amendment was required.

Action Request: 02104045

A set of updates to the Catawba Nuclear Station (CNS) UFSAR Tables 2-105, 15-14, and 15-17 were completed following changes made to the Alternative Source Terms (AST) analysis of the main steam line break (MSLB). Some of the changes were found to be outside the scope of 10 CFR 50.59; two of them were evaluated fully under 10 CFR 50.59. These two changes, together denoted as the activity, were as follows:

1) The flow rate for the Steam Generator (SG) Power Operated Relief Valves (PORVs) used in the AST analysis of the MSLB was recalculated to provide lower and upper bound values.

The lower bound values were used to recalculate the profile for post-accident cool-down (times to average unit temperatures) while the upper bound SG PORV flow rates were used in setting the fission product release rates with long-term SG boil-off.

2) Margin was applied to each limiting post-MSLB radiation dose; each calculated dose was multiplied by 2.2 to obtain the value reported in the UFSAR.

Neither any plant modification nor procedure revision was completed in association with this method. The increases in post-MSLB radiation doses do not equate to more than a minimal increase in the consequences of a MSLB. The UFSAR may be updated as described on account of the activity without prior NRC approval pursuant to 10 CFR 50.59.

Action Request: 02110981

The Alternative Source Terms (AST) analysis of the Catawba Nuclear Station (CNS) locked rotor accident (LRA) was revised, prompting an update to CNS UFSAR Table 15-14 to report revised radiation doses for the LRA. The revised AST analysis accounts for partitioning of alkali metals and bromine in the water in the steam generators (SGs) and releases with boil-off. It also accounts for a recalculation of the flow rate for the SG Power Operated Relief Valves (PORVs). The recalculation yielded upper and lower bound flow rates in place of a "nominal" flow rate, used a correlation from the SG PORV manufacturer this calculation, and accounted for the piping immediately connected to each SG PORV. This prompted a revision to the profile for long-term cool-down following the LRA. The lower bound SG PORV flow rates were used in this effort, while the upper bound values were used to represent the SG boil-off rates for this phase of the accident. In addition, the profile was modified to add a state point and remove some of the margin embedded in the approach to the local boiling point (211 ^{o}F). The AST calculations with this set of changes produced small increases in the radiation doses at the boundary of the Low Population Zone (denoted as the LPZ) and in the control room, making it adverse.

The calculation indicated that for the LRA scenario limiting for control room dose (offsite power available and a Minimum Safeguards failure), the 2 hour radiation dose at the Exclusion Area Boundary (EAB) was seen to reach its maximum over the interval 0.1-2.1 hour. Reassigning the 0-2 hour values of the control room atmospheric dispersion factors (X/Qs) to this interval pursuant to Regulatory Guide 1.194 produced a drop in the control room dose. For this reason, the 0-2 hour values for the control room X/Qs were retained in the 0-2 hour interval in the calculation of the limiting control room dose for the CNS LRA.

These changes left the post-LRA EAB radiation dose unchanged while yielding small increases in the LPZ and control room doses for this accident. These recalculated doses equated to not more than a minimal increase in consequences. As seen above, the retention of the 0-2 hour values of the control room X/Qs in the first two hours was a conservative change to the method for calculation of control room doses and as such not a departure. These changes may be made and UFSAR Table 15-14 updated without prior NRC approval.

Action Request: 02114867

This evaluation addresses the addition of an Open Phase Protection (OPP) System for Catawba Unit 2 implemented under Engineering Change (EC) 402827. Open phase detection is to be installed on the primary winding of each of the two (2) Main Step Up (MSU) transformers 2A and 2B. The system will be armed to initiate a trip only when the unit is offline. To avoid potential damage to loads fed from the 4160-volt safety-related loads, actuation of the OPP system will separate the affected offsite power source from the auxiliary electrical distribution system. This change will not require a revision or addition to the Technical Specification. This change has been evaluated against the eight (8) questions required by 10CFR50.59, and because the change involves digital components, the supplementary questions contained in NEI 01-01 Appendix A. From this evaluation it is concluded that the change can be implemented under 10CFR50.59 without prior approval from the NRC.

Action Request: 02170349

This evaluation addresses the addition of an Open Phase Protection (OPP) System for Catawba Unit 1 implemented under Engineering Change (EC) 401499. Open phase detection is to be installed on the primary winding of each of the two (2) Main Step Up (MSU) transformers 1A and 1B. The system will be armed to initiate a trip only when the unit is offline. To avoid potential damage to loads fed from the 4160-volt safety-related loads, actuation of the OPP system will separate the affected offsite power source from the auxiliary electrical distribution system. This change will not require a revision or addition to the Technical Specification. This change has been evaluated against the eight (8) questions required by 10CFR50.59, and because the change involves digital components, the supplementary questions contained in NEI 01-01 Appendix A. From this evaluation it is concluded that the change can be implemented under 10CFR50.59 without prior approval from the NRC.