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10 CFR 50.59 (d)

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Renewed Facility License Numbers NPF-9 and NPF-17
Summary Report of Evaluations Performed Pursuant to 10 CFR 50.59
Changes, Tests, and Experiments

Pursuant to 10 CFR 50.59(d)(2), attached is a summary report of evaluations performed at McGuire Nuclear Station (MNS) for changes occurring during the period from January 1, 2016, to March 31, 2017. These evaluations demonstrate that the associated changes do not meet the criteria for license amendments as defined by 10 CFR 50.59(c)(2).

This submittal document contains no regulatory commitments.

If there are any questions or if additional information is needed, please contact Brian Richards of Regulatory Affairs at (980) 875-5171.

Sincerely,

Jimmy Hixon
for

Steven D. Capps

Attachment

IE47
NRR

xc:

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Attachment 1

McGuire Nuclear Station (MNS)
Changes Evaluated Under 10 CFR50.59

UFSAR Change in Support of Lifting Reactor Vessel Upper Internals Lift Rig
Action Request 01983884

This evaluation addresses the proposed use of the Containment Building Polar Crane to lift the Reactor Vessel Upper Internals Lift Rig during Modes 1 through 4 to support inspection of the lift rig. The rig is to be lifted a total vertical distance of approximately 35 feet 7 inches above the stand. A load drop analysis of the lift rig has demonstrated that the consequences of a load drop meet the guidelines of NUREG-0612, Section 5.1. Handling of the lift rig in Modes 1 through 4 has been found to be consistent with the heavy lifts program guidelines and controls described in UFSAR Section 9.1.5. Consistent with the Nuclear Regulatory Commission (NRC) endorsed guidelines of NEI 08-05, a revision to the UFSAR was generated to describe the inclusion of this lift into the Heavy Lifts Program. This proposed activity does not require a change to the Technical Specifications. Responses to each of the eight questions from 10 CFR 50.59.c(2) have been addressed in the evaluation with the conclusion that this activity can be performed without prior approval from the NRC.

Revision to Methodology Report DPC-NE-2009-P-A
Action Request 01991491

The proposed activity being evaluated is a revision to the fuel rod analysis methodology in the Westinghouse Fuel Transition methodology report (DPC-NE-2009-P-A). The proposed activity adds a mechanistically based Departure from Nucleate Boiling (DNB) propagation methodology to DPC-NE-2009-P-A. Specifically, methodology report DPC-NE-2009-P-A Revision 3a is being revised to implement Revision 1-A to WCAP-8963-P-A Addendum 1-A, Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology), which has been previously approved by the NRC. DPC-NE-2009 methods are considered part of the UFSAR via reference, and as such, are considered "as described in the UFSAR."

This 10 CFR 50.59 evaluation was 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.

Allow Replacement of Digital Rod Position Indication (DRPI) Controller
Engineering Change (EC) 113220
Action Request 02014647

EC 113220 involves replacement of obsolete Universal Input/Output (I/O) Controllers (UIOCs) in the DRPI system with new DRPI I/O Controllers (DIOCs). The function of the affected controller is to manage retrieval of the raw rod data to the main control room for further processing by the DRPI system.

The existing DRPI UIOCs no longer have spare parts available from the original manufacturer. Additionally, ongoing failures of the existing controllers have significantly decreased the reliability of the DRPI system.

Both the existing UIOC and the proposed replacement DIOC are digital components, and therefore, digital considerations apply to this 50.59 evaluation.

The activity was evaluated per 10 CFR 50.59, following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activity may be implemented under 10 CFR 50.59 without requiring prior NRC review or approval.

Revision to Methodology Report DPC-NE-2009-P-A
Action Request 02047562

The proposed activity being evaluated is a revision to the fuel rod analysis methodology 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 robust fuel assembly fuel beginning with McGuire 2 Cycle 25 (M2C25).

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 UFSAR via reference, and as such, are considered "as described in the UFSAR."

This 10 CFR 50.59 evaluation was 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.

Tune Distributed Control System (DCS) Controls Due to Hot Leg Streaming
EC 404368
Action Request 02050448

This activity is a temporary engineering change for cycle M1C25 on Unit 1 DCS for a robust tuning of reactor and feedwater controls. The tuning is desired to improve the Upper Plenum Anomaly (UPA) impact on two non-safety related control features. The UPA was observed during initial cycle run on M1C25, whereby hot leg streaming perturbations have resulted in rod movement and excessive thermal power oscillations. The UPA impact is distracting to the operator and prevents full power operation.

The results of the review demonstrate that the activity does not adversely affect design functions described in the UFSAR nor any other aspect of the current licensing basis. All eight questions were answered negatively with appropriate justifications evaluating the activity for accidents and malfunctions that could be initiated, impact on radiological consequences and fission product barriers, and use of approved methodologies. The potential impact upon the analysis of an accident was shown to be satisfactory with no increase in consequence. Therefore, this activity does not require a license amendment request to obtain prior NRC approval.

Allow Use of Upgraded VIPRE Code for Analysis
Action Request 02073877

Duke Energy is upgrading the version of the VIPRE-01 thermal-hydraulics computer code from mod 2 as modified by Duke Energy to mod 2.5 as modified by Duke Energy (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 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 concluded that the results are essentially the same, that they do not represent a departure from a method of evaluation, and that no prior NRC approval is required.

UFSAR Change in Support of Correcting Errors in Analysis for Steam Generator Tube Rupture (SGTR)
Action Request 02083777

A set of updates to the MNS UFSAR Tables 15-12 and 15-24 has been prepared. These updates are based on a revised Alternative Source Terms (AST) analysis of the MNS SGTR in which two errors were corrected. The corrections results in a lowering of the partition factors for ruptured steam generator (SG) boil-off and intact SG tube leakage. Although the corrections had adverse impact, the revised AST analysis of the MNS yielded radiation doses none of which equated to more than a minimal increase in the consequences of an accident or equipment malfunction. With the corrections, the AST analysis conforms to the NRC guidelines for AST analyses of the SGTR in a manner consist with the plant current license basis. The revised analysis may be put into place and the UFSAR updated without prior NRC review pursuant to 10 CFR 50.59.