September 6, 2001

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: REACTOR COOLANT LOOP ANALYSIS METHODOLOGY FOR STEAM GENERATOR REPLACEMENT (TAC NOS. MA9886, MA9887, AND MA9888)

Dear Mr. McCollum:

By letter dated August 28, 2000, Duke Energy Corporation proposed methodology for the analysis of the reactor coolant loop in support of steam generators replacement at the Oconee Nuclear Station, Units 1, 2, and 3. By letter dated July 26, 2001, Duke responded to the staff's request for additional information. The staff has completed its review of the subject methodology and, based on the information provided, the staff has concluded that the approach and methodology used for the reactor coolant loop re-analysis in support of the replacement of the steam generators is reasonable and, therefore, acceptable. One basis for our conclusion is that, prior to finalizing the piping design, you demonstrate that the leak before break application is valid for the extended period of operation.

Additionally, we request that you provide a summary of this evaluation and conclusion(s) to the staff when it is completed. Our Safety Assessment is enclosed.

Sincerely,

/RA/

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: As stated

cc w/encl: See next page

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SAFETY ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

METHODOLOGY FOR ANALYSIS OF THE REACTOR COOLANT LOOPS

FOR STEAM GENERATOR REPLACEMENT

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated August 28, 2000 (Reference 1), Duke Energy Corporation (Duke or the licensee) submitted for staff review and approval a proposed methodology for the analysis of the reactor coolant loop (RCL) for the replacement of the once through steam generators (OTSGs) at the Oconee Nuclear Station (ONS), Units 1, 2, and 3. Since the OTSGs that will be replaced incorporate a number of material changes that will reduce the operating weight of each SG, and the original skirt support of the OTSG will be replaced with a pedestal support, the changes have necessitated the re-analysis of the ONS reactor coolant loop using current methodology.

In 1985, the NRC approved the elimination of the dynamic effects from large break loss of coolant accidents (LOCAs). Therefore, the licensee indicated that the large break LOCAs need not be considered in the re-analysis. Design details for the replacement OTSGs will be provided by Babcock & Wilcox Canada (BWC). The complete re-analysis of the RCL using current analytical method has been performed by Framatome ANP (FRA-ANP).

The purpose of the re-analysis is to demonstrate that the RCL stresses are not exceeded by the introduction of the replacement OTSGs, and that the replacement OTSGs will not adversely affect the functions of the remainder of the reactor coolant system. In Reference 1 the licensee described the approach and methodology (only) that were employed in the re-analysis. The results of the re-analysis were not included in the submittal. Therefore, the staff's review is limited to the review of the summary of the approach and methodology for the re-analysis of the RCL system.

On October 23, 2000, the NRC transmitted a request for additional information. On January 25, 2001, a conference call was held to provide clarification to the NRC staff of the re-analysis methodology. From this discussion, two additional questions were raised. By letter dated July 26, 2001 (Reference 2), the licensee provided its response to these questions.

2.0 STRUCTURAL ANALYSIS

The licensee stated in Reference 2 that the purpose of the structural analysis is to demonstrate that the design basis requirements for the piping, components, and supports are still met with the replacement OTSGs in the system. This is demonstrated in one of two ways:

1. By showing that the loads acting on the piping, components, and supports do not increase beyond the design basis loads when the replacement OTSG is introduced into the Reactor Coolant System (RCS), or

2. By showing that the stresses, which are present after the replacement OTSG is introduced into the Reactor Coolant System, continue to meet the allowable stresses dictated by the applicable design codes.

The licensee's approach and methodology in performing the analysis included the development of full structural models of the ONS Reactor Coolant Systems using the FRA-ANP structural code BWSPAN. A separate model of Unit 1 is necessary due to the fact that Unit 1 has Westinghouse reactor coolant pumps (RCPs) and Units 2 and 3 have Bingham RCPs. These models include the RCS components, RCS piping, component supports, Control Rod Drive Mechanisms, Service Support Structure, the replacement OTSG internals, and the Interior Concrete Structure.

The licensee used the current modeling technique to develop and analyze the RCS, replacement OTSG and reactor building hydraulics models. The licensee stated that these modeling techniques and calculations include the use of current discharge correlations (Modified Zaloudek-Moody, for example) and the subcompartment modeling techniques discussed in NUREG 0609 (Asymmetric Blowdown Loads on PWR Primary Systems, January 1981) and Standard Review Plan Section 6.2.1.2 of NUREG 0800 (Subcompartment Analysis, Revision 2, July 1981).

The RCS, replacement OTSG secondary side and reactor building initial conditions are those at 100 percent power for the High Energy Line Break Accident analyses. Temperatures and pressures are taken from the replacement OTSG Certified Design Specification (CDS, Reference 3) or other documentation provided by BWC. The licensee stated that the structural loading analysis was performed by FRA-ANP considering the following load cases in the loading analysis of the Reactor Coolant System:

- Pressure: Design and operating (as appropriate).

- Deadweight: 100 percent power operating weight.

- Thermal Expansion: 0, 8, 15, and 100 percent power, reactor trip.

- Seismic: Operating Basis Earthquake and Safe Shutdown Earthquake.

- High Energy Line Break Accident.

All loading analyses are performed using the BWSPAN model of the Reactor Coolant System and use properties at 100 percent operating conditions unless otherwise noted.

Before the displacements and loads at key locations throughout the Reactor Coolant System are finalized, FRA-ANP performed a comparative analysis of the replacement OTSG and the original OTSG dynamic characteristics in lieu of generating new attachment point response spectra. A combined deadweight and seismic loading analysis was performed to show that the replacement OTSG will not adversely affect the connected piping and components.

The licensee further stated that, to confirm the replacement OTSG design loads, primary piping stresses, primary nozzle and lug loads, and equipment support loads, the FRA-ANP analysis loads at various locations are compared to existing (original) design loads to ensure that the design loads given in the replacement OTSG CDS (Reference 3) envelop the actual loads, which result from the loading analysis. Where the FRA-ANP analysis loads are higher, stress and fatigue analyses are performed in accordance with the original stress reports and the original design codes. The stress report summaries developed for the Reactor Coolant System piping, Reactor Vessel, CRDM, RCP, OTSG and pressurizer will be updated to reflect the final condition of the Reactor Coolant System.

The staff has reviewed the information provided and determined that the licensee's approach and methodology are reasonable to ensure that the design basis requirements for the reactor coolant piping, components, and supports are still met with the replacement OTSGs in the Reactor Coolant System.

3.0 LEAK BEFORE BREAK APPLICATION

The licensee's input transients for the reactor coolant system hydraulics re-analysis did not include breaks in the large bore primary system piping because the NRC has accepted the leak before break (LBB) application to the primary system piping at ONS. This was approved in the December 12, 1985, NRC letter to Babcock and Wilcox (B&W) Owners Group, which approved the use of topical report BAW-1847, Revision 1, "Leak-Before-Break Evaluations of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS."

Since the replacement OTSGs might alter the pipe loading, and aging might degrade piping material properties due to the number of years of operation, the licensee needed to demonstrate that the LBB analysis and results described in BAW-1847, Revision 1, continue to be bounding for the Oconee primary system piping for the period of extended operation. To resolve the concern on material aging, the licensee confirmed per Reference 2 that the results of its flaw stability analysis had previously been obtained using the appropriate lower-bound cast austenitic stainless steel fracture toughness curves from NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," and indicated that the LBB application is still valid for the period of extended operation. Additionally, to resolve the concern related to changes to pipe loading, the licensee made the following commitment in Attachment 1: "Changes in the Leak Before Break (LBB) loadings, as a result of the current loop re-analysis with the replacement steam generator and/or changes in piping/weldment materials that will be utilized, will be evaluated for Oconee Units 1, 2, and 3. If the results are not bounded by the current analysis, an LBB submittal will be made that will summarize the results of this evaluation with comparison to the results of BAW-1847, Rev. 1. ...It is expected that this evaluation may be available by the end of 2001." With this commitment, the staff concludes that the RCS hydraulics re-analysis is appropriate with the open item that the licensee shall demonstrate prior to finalizing the piping design that the LBB application is valid for the extended period of operation. The staff requests that a summary of this evaluation and conclusion(s) be provided when the analysis is completed.

4.0 CONCLUSION

Based on its review of the licensee's submittals dated August 28, 2000 (Reference 1), and July 26, 2001 (Reference 2), the staff finds the licensee's approach and methodology used for the reactor coolant loop re-analysis for the Oconee Nuclear Station, Units 1, 2, and 3, in support of the replacement of the steam generators is reasonable and, therefore, acceptable. One basis for this conclusion is that, prior to finalizing the piping design, the licensee will demonstrate that the LBB application is valid for the extended period of operation and that a summary of this evaluation and conclusion(s) be provided when the analysis is completed.

5.0 REFERENCES

1. Letter from W. R. McCollum, Jr., Duke Energy Corporation, dated August 28, 2000. Subject: Oconee Nuclear Station, Reactor Coolant Loop Re-Analysis.

2. Letter from W. R. McCollum, Jr., Duke Energy Corporation, dated July 26, 2001. Subject: Oconee Nuclear Station, Units 1, 2, and 3, Responses to Request for Additional Information, Reactor Coolant Loop Re-Analysis.

3. Duke Power Company Specification OSS-0279.00-001, Revision 0, "Bid Specification for Replacement Steam Generators."

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Date: September 6, 2001

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