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Regulatory Commission
Washington, DC 20555

Subject: Duke Energy Corporation
Oconee Nuclear Station, Units 1, 2, and 3,
Docket Nos. 50-269, 50-270, 50-287
Leak-Before-Break Evaluation for Steam Generator Replacement

By letter dated September 6, 2001, the NRC approved Duke Energy Corporation's (Duke) proposed methodology for the analysis of the reactor coolant loop (RCL) in support of steam generator replacement at Oconee Nuclear Station, Units 1, 2, and 3. As noted in that letter, one basis of the Staff's conclusion was that prior to finalizing the piping design, that Duke demonstrate that the leak-before-break (LBB) application previously approved by the NRC is valid for the extended period of operation. By letter dated February 19, 2003, Duke provided a summary of the evaluations that were done by Babcock & Wilcox Canada (BWC) to address the changes due to steam generator replacement per the NRC's September 6, 2001 letter.

Duke's February 19, 2003 letter indicated the hot leg elbows (which included a short spool piece) would be replaced with elbows meeting the original specification. The original specification for the elbow was SA-516 Gr 70 rolled plate and SA-106 seamless pipe for a short spool piece connecting the elbow to the SG hotleg inlet nozzle. Due to the availability of materials, the short spool piece between the elbow and the steam generator hotleg inlet nozzle will instead be fabricated from SA-516 Gr 70 rolled plate, the same as the elbow.

A follow-up evaluation was performed by BWC to address any impacts to the previous LBB evaluation because of the change in spool piece material. The evaluation results show that the new hot leg elbows and spool pieces remain bounded by the B&W Owners Group Topical Report BAW 1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS", September 1985. Attachment 1 provides a revision of the prior evaluation summary with the above described changes denoted by a change bar.

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Please direct any questions to Robert Sharpe at (704) 805-2007 or Robert Douglas at (864) 885-3073.

Very truly yours,

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**Attachment 1**

**Oconee Nuclear Station**

**Leak-Before-Break Re-Evaluation For Steam Generator Replacement Revision 1**

Background

In 1985, the B&W Owners Group (B&WOG) performed a generic Leak-Before-Break (LBB) evaluation for B&W designed plants (Reference 1). This evaluation reviewed the base metal and weld data for B&WOG plants and established a reasonable lower bound of material properties to be used in the evaluation. By letter dated December 12, 1985 (Reference 2), the NRC transmitted their Safety Evaluation Report and acceptance of the B&WOG LBB approach. This SER was applicable to Oconee.

By letters dated August 28, 2000 and July 26, 2001, Duke requested NRC's approval of the Framatome-ANP methodology that would be used for re-analysis of the reactor coolant loop in support of steam generator replacement at Oconee. This re-analysis did not include breaks in the large bore primary system piping because of the NRC's previous approval of LBB for Oconee. In reviewing this request, the NRC requested that Duke provide information to demonstrate that the analyses and results of the previous LBB evaluation still bound the plant-specific applications at Oconee after steam generator replacement.

The Once-Through Steam Generators (OTSGs) will be replaced in Fall 2003, Spring 2004, and Fall 2004, for Oconee Units 1, 2, and 3, respectively. Replacement of the OTSGs will involve changes to the piping material and welds at specific locations:

- A short (22½°) section of cold leg piping attached to each of the two OTSG outlet nozzles will be replaced with a one piece SA-508 Cl. 3a forging. The original cold leg elbow was fabricated from formed and welded SA-516 Gr. 70 plate.

- The hot leg elbow attached to the inlet nozzle of the OTSG will be replaced with identical specification elbows. This does not represent a material change from Reference 1. The short spool piece that connects the hotleg elbow to the OTSG hotleg nozzle will be replaced with a rolled and welded SA-516 Gr. 70 plate. The original was a seamless SA 106 spool piece.

- The original installation welds for the OTSG used shielded metal arc welds (SMAW) or submerged arc welds (SAW). For steam generator replacement, narrow groove gas tungsten metal arc welding (GTAW) will be used.

## Re-evaluation

Babcock & Wilcox Canada was contracted to evaluate the changes in the Oconee Units 1, 2, and 3 reactor coolant systems due to steam generator replacement and plant life extension, and to determine if the B&WOG LBB Topical Report (Reference 1) needed to be updated for Oconee. This evaluation increased the number of heatup and cooldown cycles from 240 to 360 cycles to be consistent with the current Oconee licensing bases. This evaluation resulted in the following findings:

### Weld Evaluation

The replacement steam generators will be welded into the reactor coolant loop with narrow groove GTAW with ER 70S-6 filler material. An assessment was performed to determine the full fracture toughness ($J_{1c}$) and tensile (true stress/strain) properties at the operating temperature. The new weldment and weld filler materials were shown to have fracture properties equal to or better than those of the materials covered in Reference 1.

### Cold Leg Elbow Evaluation

The 22½° section of the cold leg elbow will be replaced with a one piece SA-508 Cl. 3a forging. The new cold leg elbow section was fabricated as an integral part of the replacement steam generator. Due to differences in strength, P-number, and product form, a program was initiated to determine the full fracture toughness ($J_{1c}$) and tensile (true stress/strain) properties of the replacement SA 508 Cl. 3a material at the operating temperature. The partial cold leg elbow material was shown to have fracture properties equal to or better than those of the materials covered in Reference 1.

### Cast Stainless Steel Aging Assessment

Cast Austenitic stainless steel (CASS) is used for the reactor coolant pump casing. An assessment of the thermal embrittlement of CASS for the extended operating lifetime was performed. Using data from the literature, the cast stainless steel material of the primary pump housings was shown to have a fracture toughness $J_{1c}$ in excess of the requirement at the most highly loaded location in the cold leg piping.

### Hot Leg Elbow Replacement Assessment

As previously noted, the hot leg elbows will be replaced. The elbow material will be the same as evaluated in Reference 1. Consideration was also given to the hot leg cut location. The hot leg elbow will be cut from the OTSG at the inlet nozzle. The second cut will be made at a cut location 180 degrees away or at a lesser angle, based on the final fit-up plan.

A new hot leg elbow section will be made from the same carbon steel material as the original piping material. Therefore, the use of the bounding properties of Reference 1 are adequate to cover this change. There is no effect if the new weld location is not at the original 180° weld location because the Reference 1 analysis used the bounding weld properties in conjunction with the worst stress at any location and thus conservatively covered any new weld location.

Reconciliation of Piping Loads

Piping moments, with the replacement steam generators installed, were determined by separate studies. These results were compared to the piping loads in Reference 1. The piping moment loads were bounded by those of Reference 1 with one exception where the flaw size needed to obtain 10 gpm leakage flow increased relative to Reference 1. Re-analysis showed that the fracture and limit load criteria were still satisfied.

Reconciliation of Fracture Mechanics Analysis

The required flaw sizes that result in 10-gpm leakage flow were determined for the hot and cold leg pipes under pressure plus normal operating moment loads and compared to Reference 1. The fracture and limit load evaluation of the piping concluded that all locations satisfy the allowables.

Reconciliation of Crack Growth Analysis

The crack growth analyses performed in Reference 1 were redone using the number of cyclic events anticipated for the extended plant life of 60 years.

Crack growth was shown in Reference 1 to occur predominately in the radial direction. This conclusion from Reference 1 is not affected by the planned changes.

The evaluation of crack growth due to the application of cyclic loads concluded that a substantial initial flaw size is required for the crack to grow through-wall over the 60-year plant life. As this size considerably exceeds the maximum that would be permitted to remain in the piping under the construction code, the presence of such large flaws is extremely unlikely.

## Conclusion

The reconciliation of the changes that result from steam generator replacement and life extension performed by Babcock & Wilcox Canada in Reference 3 demonstrated that the Leak-Before-Break evaluation previously performed by the B&W Owners Group in Reference 1 remains conservative for Oconee Units 1, 2, and 3.

**References**

1. B&W Owners Group Topical Report BAW 1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS", September 1985

2. Letter from D. E. LaBarge, NRC, to W. R. McCollum, Jr. dated September 6, 2001

3. Oconee Unit 1,2,3 Reconciliation of RCS Piping Leak Before Break Evaluation for Steam Generator Replacement, B&W Report No. 006K-SR-05, Revision 1, Proprietary (Oconee Calculation No. OSC-7844, Revision 3)