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SOUTHERN COMPANY
Energy to Serve Your World℠

September 15, 2006

Docket Nos.: 50-348 50-424
50-425

NL-06-2088

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

Joseph M. Farley Nuclear Plant – Unit 1
Vogtle Electric Generating Plant – Units 1 & 2
Revised Page to Enclosure 2 in Response to NRC Request for Additional Information
Regarding Inservice Inspection Alternatives for Reactor Pressure Vessel Examinations

Ladies and Gentlemen:

By letter NL-06-1305, dated June 29, 2006, Southern Nuclear Operating Company (SNC) requested two Inservice Inspection (ISI) Alternatives (ISI-GEN-ALT-06-01 and -02) for the Reactor Pressure Vessel (RPV) Examinations scheduled for the Farley Nuclear Plant (FNP) 3rd ISI Interval extending from December 1, 1997 through November 30, 2007 and for the Vogtle Electric Generating Plant (VEGP) 2nd ISI Interval extending from May 31, 1997 through May 30, 2007. One alternative is related to the RPV shell to flange weld, for which SNC proposed that this weld be examined using Appendix VIII instead of Article 4 of ASME Section V. The second alternative is related to the examination of the safe-end welds in the reactor coolant piping, for which SNC proposed using Code Case N-696.

On August 7, 2006, SNC received an e-mail from the NRC (Robert E. Martin, NRC to B. Doug McKinney, SNC) containing a Request for Additional Information (RAI) related to the proposed alternative request ISI-GEN-ALT-06-02 using Code Case N-696. Subsequently, SNC received the RAI by NRC letter (Robert E. Martin, NRC to D. E. Grissette, SNC) dated August 15, 2006. By letter NL-06-1832, dated September 8, 2006, SNC provided responses to the RAI as Enclosure 1 and provided applicable pages from NL-06-1305 as Enclosure 2 that were revised to incorporate the RAI response information.

During subsequent review of NL-06-1832, it was discovered that Page 2 of Enclosure 2 contained editorial errors in the first paragraph of the "Proposed Alternative and Basis for Use" section. This letter provides a revised Page 2 to Enclosure 2 for letter NL-06-1832. SNC regrets any inconvenience this may have caused.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

H. L. Sumner, Jr.

HLS/LPH/daj

Enclosure: Revised Page 2 to Enclosure 2 of SNC letter NL-06-1832

cc: Southern Nuclear Operating Company
Mr. D. E. Grissette, Vice President – Plant Vogtle
Mr. J. R. Johnson, General Manager – Plant Farley
Mr. T. E. Tynan, General Manager – Plant Vogtle
RType: CFA04.054; CVC7000; LC# 14484

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. C. A. Patterson, Senior Resident Inspector – Farley
Mr. G. J. McCoy, Senior Resident Inspector - Vogtle

ENCLOSURE

Joseph M. Farley Nuclear Plant – Unit 1
Vogtle Electric Generating Plant – Units 1 & 2
Revised Page to Enclosure 2 in Response to NRC Request for Additional Information
Regarding Inservice Inspection Alternatives for Reactor Pressure Vessel Examinations

Revised Page 2 to Enclosure 2 of SNC letter NL-06-1832

Reason for Request:

SNC intends to perform examinations of the Reactor Pressure Vessel nozzle to safe-end dissimilar metal (DSM) welds from the inside diameter (ID) surface. Additionally, the adjacent austenitic weld may be examined at the same time because of its close proximity to the DSM weld. If only the DSM weld is examined, SNC will use the requirements of NRC approved Code Case N-695 for qualification requirements; however, if both welds are examined, SNC proposes using Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface," to perform a combined qualification. Code Case N-696 has been approved by ASME but has not been approved by the NRC. The "Reply" of Code Case N-696 states that when performing examinations from the ID surface, ". . . the following requirements may be used to expand successful Supplement 10 qualifications in conjunction with selected aspects of Supplements 2 and 3." (Note: Code Case N-696 also covers Supplement 3 ferritic piping welds which are not applicable for this alternative.)

SNC also proposes using an alternative RMS error sizing criteria as compared to the value stated in Code Cases N-695 and N-696. Each specifies that the RMS error of the flaw depth measurement, as compared to the true flaw depths, does not exceed 0.125-inch RMS. As stated in the Diablo Canyon DSM RAI response, dated July 7, 2005, "Per communication with the EPRI PDI administrator, no vendor has successfully complied with the Code-required RMS values of 0.125-inches for qualification tests from the reactor vessel inner diameter." The difficulty of achieving the required 0.125-inch RMS error results *partly* from ID geometry interferences (e.g., root, counterbore) when using a remote sled apparatus. To accurately size thru-wall indications, the transducers ability to scan over the flaw must not be impeded by these interferences. The qualification mockups for both Supplements 2 and 10 were fabricated from field supplied data, representing the expected worse case scenarios. In most cases, the austenitic joint (Supplement 2) is field welded with root and counterbore restrictions occurring. To add to the difficulty, one side of the Supplement 2 examination consists of Cast Stainless Steel material. Figure 1 is provided for illustrative purposes.

For the combined Supplements 2 and 10 qualification (as addressed in Code Case N-696), the SNC vendor was able to achieve a 0.245-inch RMS error. The Supplement 2 qualification was an "add on" module to the Supplement 10 qualifications. For the stand alone Supplement 10 qualification (as addressed in Code Case N-695), the SNC vendor was able to achieve a 0.189-inch RMS error.

Proposed Alternative and Basis for Use:

Proposed Alternative:

SNC proposes to use Code Case N-696 to perform a combined Supplements 2 and 10 qualification when examining the DSM and austenitic welds. The difference between the 0.245-inch RMS error and the Code Case N-696 required 0.125-inch RMS error will be applied to the flaw depths determined during actual sizing of flaws. SNC also proposes that if only the dissimilar metal welds are examined, the difference between the 0.189-inch RMS error and the Code Case N-695 required 0.125-inch RMS error will be applied to the flaw depths determined during actual sizing of flaws.