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SOUTHERN COMPANY
Energy to Serve Your World℠

September 8, 2006

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

NL-06-1832

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
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. In addition, a telephone conference was held on August 17, 2006 between SNC and the NRC to discuss the RAI questions.

Enclosed with this letter are the following attachments: Enclosure 1 provides the SNC responses to the RAI, and Enclosure 2 provides applicable pages from NL-06-1305 that have been revised to incorporate the RAI response information.

Approval is requested by September 15, 2006 to support scheduled examinations performed during the planned Unit 1 outage at VEGP beginning September 2006, the Unit 2 outage at VEGP beginning March 2007, and the Unit 1 outage at FNP beginning October 2007.

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

Sincerely,

Jeffrey T. Gasser

JTG/DRG/daj

Enclosures:
1. SNC Response to NRC RAI Related to Request ISI-GEN-ALT-06-02
2. Revised Pages from SNC Letter NL-06-1305

cc: Southern Nuclear Operating Company
Mr. H. L. Sumner, Vice President – Plant Farley
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# 14477

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 1

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

SNC Response to NRC RAI Related to Request ISI-GEN-ALT-06-02

REQUEST FOR ADDITIONAL INFORMATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO REQUEST ISI-GEN-ALT-06-02
JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1
VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2
SOUTHERN NUCLEAR OPERATING COMPANY
DOCKET NOS. 50-348, 50-424, AND 50-425

The licensee requested to use ASME Code Case N-696 (N-696) for the examination of selected dissimilar metal welds. N-696, Paragraph 3.3(d) states that Supplement 2 or Supplement 3 procedures, equipment, and personnel are qualified for depth-sizing when the flaw depths estimated by ultrasonics, as compared with the true depths, do not exceed 0.125 in. (3mm) [root mean square] RMS, when they are combined with a successful Supplement 10 qualification. The proposed alternative is specifically for the use of N-696. In the section identified as the "Basis for Use," there is an implied alternative for handling performance demonstrated depth-sizing values that are greater than 0.125 in. RMS.

In the submittal section "Basis for Use," a discussion on depth sizing values greater than 0.125 in. RMS was presented. The purpose of the discussion is not clear because it is suggesting another alternative separate from the submittal section "Proposed Alternative." If a depth sizing value greater than 0.125 in. is part of the proposed alternative, provide the alternative depth-sizing value. The basis for the acceptability of a RMS value greater than the ASME Code required value is that any required flaw evaluation adds to the measured flaw size, the difference between the ASME Code 0.125" RMS performance demonstration acceptance criterion and the larger RMS value from the vendor's procedure performance demonstration.

The staff considers the licensee's approach to address the current inability to meet the ASME Code-required UT depth-sizing RMS requirement as a short-term solution. Vendors continue to make improvements to their UT sizing capabilities. These improvements may reduce the difference between the depth-sizing RMS Code-requirement and currently achievable RMS values. Discuss options Farley and Vogtle have considered for the examination of the subject welds. Explain why the ASME Code RMS value is not achievable for the selected vendor.

Item 1
Provide the alternative depth sizing value.

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

Both Farley and Vogtle expect to experience some difficulty with the inside diameter (ID) geometry for the Supplement 2 joints, being field welds. For the Supplements 2 and 10 welds, the difference between the 0.245-inch RMS error and the Code required 0.125-inch RMS error will be applied to the flaw depths determined during actual sizing of flaws. If only the Supplement 10 welds are inspected, the difference between the 0.189-inch RMS error and the Code required 0.125-inch RMS error will be applied.

<u>Item 2</u>
Discuss options Farley and Vogtle have considered for the examination of the subject welds.

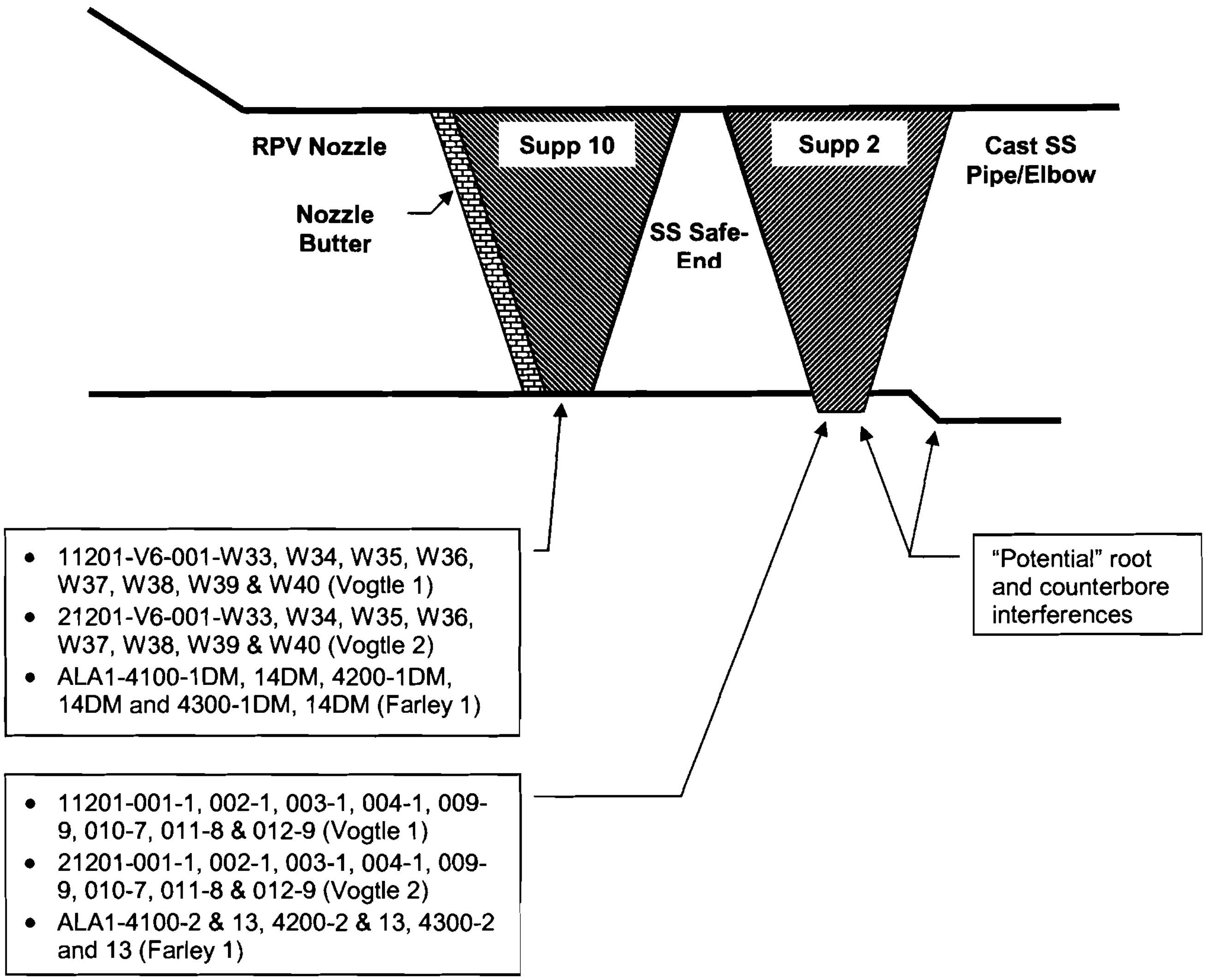
The vendor chosen for Farley and Vogtle is also SNC's NSSS supplier and has vast experience in the nuclear industry. They have performed all inservice RPV vessel examinations to date at Farley and Vogtle.

<u>Item 3</u>
Explain why the ASME Code RMS value is not achievable for the selected vendor.

The difficulty of achieving the required 0.125-inch RMS error factor results from ID geometry interferences (e.g., root, counterbore). 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. For most 3 and 4 loop Westinghouse designs, the dissimilar metal (DSM) joint (Supplement 10) has been shop machined, resulting in little or no geometric interferences. Both Farley and Vogtle Supplement 10 configurations are expected to be unimpeded. Therefore, it is expected that the proposed alternative 0.189-inch RMS error factor would be a conservative alternative compared to the qualification specimens. Part of the examination process for remote mechanized inspections is the gathering of profile data. This information is used for coverage determinations. The data is also being collected to ensure qualification mockups represent the full alignment of field conditions within the industry. Surface scan access and/or cast material difficulties restrict the examinations for the Farley and Vogtle configurations from the outside diameter (OD). A figure is provided for illustrative purposes.

As stated in the Diablo Canyon DSM RAI response, dated July 7, 2005, no vendor has demonstrated the ability to achieve a 0.125-inch RMS error value for the Supplement 2 or 10 qualifications when conducting examinations from the ID. This information is still accurate as of August 2006. Studies and projects are currently underway for the examination of cast material. Industry efforts with "phased array" examinations may be able to achieve greater coverage from the OD in the future, which may also aide in thru-wall sizing.

FIGURE 1
(Illustrative Purposes Only - not to scale)



ENCLOSURE 2

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

Revised Pages to SNC Letter NL-06-1305

Plant Site-Unit: Vogtle Electric Generating Plant (VEGP) Units 1 and 2 and Joseph M. Farley Nuclear Plant (FNP) - Unit 1.

Interval Dates: VEGP 1 & 2
2nd Inservice Inspection (ISI) Interval extending from May 31, 1997 through May 30, 2007 and

FNP 1
3rd ISI Interval extending from December 1, 1997 through November 30, 2007.

Requested Date for Approval : Approval is requested by September 15, 2006 to support scheduled examinations performed during VEGP 1R13 (September 2006), VEGP 2R12 (March 2007) and FNP 1R21 (October 2007).

ASME Code Components Affected: Category R-A, Item R1.15; Reactor Pressure Vessel (RPV) nozzle to safe-end dissimilar metal (DSM) butt welds and the adjacent Item R1.11 austenitic safe-end welds. (Farley and Vogtle have both implemented risk-informed ISI).

A list of welds is provided in Figure 1.

Applicable Code Edition and Addenda: The Vogtle and Farley Units are in their 2nd and 3rd inspection intervals, respectively. The applicable Code edition and addenda is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant components," 1989 Edition with no addenda. In addition, as required by 10 CFR 50.55a, ASME Section XI, 1995 Edition through 1996 Addenda is used for Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems."

Applicable Code Requirements: Category R-A, Item R1.15; RPV nozzle to safe-end DSM butt welds and the adjacent Item R1.11 austenitic safe-end welds.

Per the requirements of 10 CFR 50.55a, these examinations are required to be conducted in accordance with ASME Section XI, Appendix VIII, 1995 Edition with 1996 addenda. The applicable supplements for these examinations include Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds" and Supplement 2, "Qualification Requirements for Wrought Austenitic Piping Welds" for the Item R1.15 nozzle to safe-end welds and the Item R1.11 austenitic safe-end welds, respectively.

Additionally, Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds", has been approved by the NRC in Revision 14 of Regulatory Guide 1.147. This Code Case is an alternative to the Appendix VIII, Supplement 10 requirements for qualification requirements for dissimilar metal welds. Paragraph 3.3(c) indicates examination procedures, equipment, and personnel are qualified for depth-sizing when the RMS (Root Mean Square) error of the flaw depth measurements, as compared to the true depth flaws, does not exceed 0.125 inches.

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-695 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-695 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-696 required 0.125-inch RMS error will be applied to the flaw depths determined during actual sizing of flaws.

Basis for Use:

The vendor chosen for Farley and Vogtle is also SNC's NSSS supplier and has vast experience in the nuclear industry. They have performed all inservice RPV vessel examinations to date at Farley and Vogtle. As stated, no vendor has demonstrated the ability to achieve the required 0.125-inch RMS error value. For most 3 and 4 loop Westinghouse designs, the DSM joint (Supplement 10) has been shop machined, resulting in less (or no) ID geometric interferences. Both Farley and Vogtle Supplement 10 configurations are expected to be unimpeded. Therefore, it is expected that the proposed alternative 0.189-inch RMS error (or the 0.245-inch RMS error for the combined) would be a conservative alternative compared to the results achieved with the qualification specimens. Surface scan access and/or cast stainless material difficulties restrict the examinations for the Farley and Vogtle configurations from the outside diameter (OD). Studies and projects are currently underway for the examination of cast material. Industry efforts with "phased array" examinations may be able to achieve greater coverage from the OD in the future, which may also aid in thru-wall sizing.

In addition, part of the examination process for remote mechanized inspections is the gathering of profile data. This information is used for coverage determinations. The data will also be beneficial to ensure qualification mockups represent the full alignment of field conditions within the industry.

Use of Code Case N-695 and N-696, with the added compensation for depth sizing tolerance, provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i).

Duration of Proposed Alternative: The proposed alternative is applicable for the remaining 2nd Inservice Inspection Interval for VEGP and remaining 3rd Inservice Inspection Interval for FNP Unit 1.

Precedents: Use of the combined qualification requirements for Supplements 2 and 10 prior to availability of Code Case N-696, and the concept of adding the difference between the required RMS error value and the demonstrated RMS error value to the measured indication depth, were separately approved for V.C. Summer Station by NRC letter dated February 3, 2004.

This alternative is similar to and closely follows the content and statements made in the Diablo Canyon request submitted initially in a letter to the NRC dated April 1, 2005 and approved by the staff in a letter dated October 26, 2005.

References: None

Status: Awaiting NRC approval.

FIGURE 1
(Illustrative Purposes Only - not to scale)