Southern Nuclear
Operating Company, Inc.

SOUTHERN COMPANY
Energy to Serve Your World

October 12, 2009

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50-425

NL-09-1570

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

Vogtle Electric Generating Plant – Units 1& 2
Response to Request for Additional Information Related to
Risk-Informed/Safety Based Inservice Inspection
Alternative for Class 1 And 2 Piping
VEGP-ISI-ALT-02, Version 1

Ladies and Gentlemen:

On April 15, 2009, Southern Nuclear Operating Company (SNC) submitted Alternative VEGP-ISI-ALT-02, requesting authorization to implement Risk-Informed/Safety Based Inservice Inspection (RIS_B ISI). This alternative will be used in lieu of the existing ASME Section XI Code Category B-F, B-J, C-F-1, and C-F-2 requirements for examination of Class 1 and 2 piping welds. This alternative, which is described in Enclosure I of April 15, 2009 letter, has been developed in accordance with Code Case N-716, "Alternative Piping Classification and Examination Requirements."

On August 6, 2009, SNC received a request for additional information (RAI) letter, which contained five (5) questions.

SNC responses to RAI questions 1-5 from the August 6, 2009 letter are contained in Enclosure 1.

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

Sincerely,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Manager

MJA/TAH/lac

Enclosures: 1. Response to Request for Additional Information

cc: Southern Nuclear Operating Company
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Mr. T. E. Tynan, Vice President – Vogtle
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U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
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Enclosure 1

Response to Request for Additional Information

ENCLOSURE 1
VOGTLE RISK INFORMED/ SAFETY BASED
ISI ALTERNATIVE VEGP-ISI-ALT-02
RAI RESPONSES

Item 1: Table IWB-2500-1 of ASME Code, Section XI, 2001 Edition with 2003 Addenda requires volumetric and/or surface examination of all Category B-F or B-J Pressure Retaining Dissimilar Metal Welds greater than nominal pipe size (NPS) 1. Based on recent findings of primary water stress corrosion cracking in Alloy 82/182 dissimilar metal welds the NRC staff would like more information on your inspection plans for these welds in the third Interval ISI Plan for VEGP.

The submittal dated April 15, 2009, states that the plant augmented inspection program in response to MRP-139, Materials Reliability Program: *Primary System Piping Butt Weld Inspection and Evaluation Guideline* supplements the RIS-B program selection process. Describe the inspection plan of Alloy 82/182 dissimilar metal welds greater than NPS 1 in the third Interval ISI plan for VEGP (e.g., are these welds included in the number of welds selected for examination in the risk-informed inservice inspection (RI-ISI) program, how many of these welds are selected for examination, what examination method(s) are being employed, what is the frequency of examination, how is disposition of limited coverage (less than 90 percent) examinations handled, etc.).

Item 1 Response: Unmitigated Alloy 82/182 Category B-F dissimilar metal welds greater than NPS 1 that are subject to primary water stress corrosion cracking (PWSCC) are the eight RPV hot and cold leg nozzle to safe-end welds. The Steam Generator dissimilar metal welds are not subject to PWSCC because the welds are stainless steel, and all of the pressurizer dissimilar metal welds (and the adjacent stainless steel welds) greater than 1" NPS have been overlaid with Full Structural Weld Overlays (FSWOL). Per the Vogtle FSWOL approved alternative (ISI-GEN-ALT-07-01) and NRC SER (ML-080580291), all of the overlaid welds have been removed from the risk-informed program and will be examined in accordance with the proposed alternative.

In summary, there are eight Alloy 82/182 RPV hot and cold leg nozzle welds (per each VEGP unit), all of which are designated as highly safety significant (HSS) welds. Even though Code Case N-716 only considers the RPV hot leg nozzle Alloy 82/182 weld locations to be susceptible to PWSCC, SNC has selected all eight welds to be ultrasonically examined for PWSCC within the scope of Code Case N-716. Code Case N-716 requires the examination of these welds every ten years. However, the examination frequency for these eight welds is currently based on the frequencies established by the requirements of Materials Reliability Program (MRP)-139, Revision 1. MRP-139 currently requires that the unmitigated hot legs be examined on a five year frequency and the unmitigated cold legs be examined on a six year frequency. These frequencies are subject to change based on factors such as industry experience and potential issuance of NRC rulemaking.

Per Code Case N-716 (Table 1, Item No. 1.15, "Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC)"), selected butt welds are subject to volumetric examination. Per Note 3 of this Table, the exam includes essentially 100% of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Areas with acceptable limited examinations (coverage less than 90%), and their bases, shall be documented and submitted for relief per the requirements of 10 CFR 50.55a(g)(5)(iv).

Item 2: On page E1-17, Section 3.3.4, "Program Relief Requests," the licensee provides guidance for program RRs. The licensee states the process outlined in Title 10 of the *Code of Federal Regulations*, 50.55a, will be used for RRs. Please discuss how incomplete examinations' (i.e. where coverage greater than 90 percent is not obtained) effect on risk will be assessed.

Item 2 Response: Per footnote 3 of Table 1 of Code Case N-716, when the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the change-in-risk evaluation (paragraph 5 of Code Case N-716). The change-in-risk evaluation of Code Case N-716 is consistent with previous risk-informed ISI applications (e.g., EPRI TR-112657) and meets Regulatory Guide 1.174 change-in-risk acceptance criteria. Areas with acceptable limited examinations, and their bases, shall be documented.

Consistent with previously approved RI-ISI submittals [e.g., Waterford 3 Safety Evaluation (Reference 1)], Southern Nuclear will calculate coverage and use additional examinations and techniques in the same manner it has for traditional Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until that time. Relief requests will be submitted as necessary per the guidance of 10 CFR 50.55a(g)(5)(iv).

Item 3: Please provide a description of significant issues identified by the independent external contractor's evaluation of Vogtle's probabilistic risk assessment (PRA) flooding model and describe how these issues have been addressed during the development of your RI-ISI program.

Item 3 Response: There were no significant issues. However, comments made by the self-assessment reviewer for three of the ASME PRA Standard RA-Sb 2005 Supporting Requirements (SR) are addressed in attached Table 3-1. As shown in the table, all of the comments have been resolved.

Item 4: Attachment A (page A-3) states that the VEGP internal flooding PRA was re-performed in order to meet American Nuclear Society PRA standard Capability Category II (CCII). The NRC *staff* has concluded that additional work may be needed beyond CCII in order for the PRA technical adequacy to be consistent with that determined to be acceptable for PRAs that supported the Electric Power Research Institute TR-112657 RI-ISI process. Please explain how the following issues are addressed.

a. The supporting requirement (SR), IF-C3 (IFSN-A8), in ASME PRA Standard

RA-Sb-2005 identifies the failure mechanisms that shall be evaluated to determine the susceptibility of each safety-related structure, system, and component (SSC) in a flood area to flood-induced failures. CCII identifies failure by submergence and spray as requiring detailed analysis. Capability Category III includes jet impingement, pipe whip, and humidity, condensation, and temperature concerns. RI-ISI requires that all SSC failures induced by a pipe break be considered. Please demonstrate that all SSC failures that are induced by a pipe break are adequately addressed in your analysis (i.e., meets capability Category III for this SR).

Item 4a Response: The updated VEGP internal flooding analysis considers submergence, spray, jet impingement, pipe whip, humidity, condensation, and temperature effects in determining the flooding effects on equipment.

b. The SR, IF-C6 (IFSN-A14) and IF-C8 (IFSN-A16), permit screening out of flood areas based on, in part, the success of human actions to isolate and terminate the flood to meet CCII. The endorsed RI-ISI methods require determination of the flood scenario with and without human intervention which corresponds to the capability Category III (i.e., scenarios are not screened out based on human actions). Therefore, a capability Category III analysis is consistent with approved RI-ISI methods. To provide confidence that scenarios that might exceed the quantitative core damage frequency and large early release frequency guidelines are identified, please describe how credit is given to human actions and how the analysis meets capability Category III for these SRs.

Item 4b Response: The updated VEGP internal flooding analysis did not take credit for any isolation by human actions. Equipment in a room was assumed to be damaged due to flooding when a pipe break occurs in that room. Finally, no credit was given to manual isolation of flooding scenarios in developing or evaluating flooding scenarios for unscreened flooding events. Therefore, the use of the updated VEGP internal flooding PRA model in RI-ISI update is conservative.

c. SR IF-D3a (IFEV-A3) Category II permits grouping or subsuming flood initiating scenarios with existing plant initiating event groups. Capability Category III does not permit grouping, which is more consistent with the approved RI-ISI methods. If grouping of flood scenarios with other initiating events groups was done, please confirm that the subsumed flooding scenarios were identified during the flooding analysis and extracted during the RI-ISI analysis in order to ensure that their contribution to the RI-ISI analysis was properly included (i.e., meets capability Category III for this SR).

Item 4c Response: The VEGP updated internal flooding analysis did not group or subsume the flooding scenarios. For each identified flood scenario, conditional core damage probability (CCDP) given a flooding event was calculated by conditioning the internal PRA model appropriately (revised to fail affected components) and then core damage frequency for the flooding scenario was calculated by multiplying the calculated CCDP with the associated flooding frequency.

Item 5: Were new examination locations identified? If so, using an upper-bound estimate for new locations would overestimate the risk decrease and therefore be non-conservative. Please demonstrate that this non-conservative approach, if corrected in the evaluation of your proposed RI-ISI program, would not cause the delta risk guidelines to be exceeded.

Item 5 Response: New examination locations were identified and included in the change-in-risk estimate. The inspection selections for the original ASME Section XI program, the proposed N-716 program and the difference between those selections are contained in Tables 3.4-1a and 3.4-1b of the template under the columns entitled SXI, RIS_B and Delta respectively. The risk impact analysis included changes made to the original ASME Section XI inspections as a result of implementing N-716 and the results are displayed in the Delta column of Tables 3.4-1a and 3.4-1b as either no change (represented by 0), an increase (represented by a positive number) or a decrease (represented by a negative number).

A risk impact calculation was also performed, with estimated conditional core damage probabilities; the results were similar to upper bound and meet acceptance criteria. Also, Tables 3.4-1a and 3.4-1b were reviewed for cases where the RIS_B selections exceeded SXI selections (represented by a positive number in the delta column) and it was determined that even if this delta was reduced to zero, Code Case N-716 acceptance criteria is still met.

REFERENCE

1. Letter from Mr. T. G. Hiltz (NRC) to Vice President, Operations (Entergy Operations) dated April 28, 2008, Waterford Steam Electric Station, Unit 3 - Request for Alternative W3-ISI-005, Request to use ASME Code Case N-716

TABLE 3-1

ASME PRA Standard SR	Self-Assessment Comments	SNC Responses
IF-C3c	Design-related calculations were referenced to support flood rates and water heights in Section 9. However, flood rates in excess of 2,000 gpm should be considered since design calculations limited the break size to only an area equal to ½ the pipe diameter times ½ the wall thickness. Also, the duration of flooding was generally assumed to be 30 minutes to allow adequate time for operator intervention. However, larger flow rates may require a shorter duration for flood isolation/mitigation to prevent damage to structures, systems, and components (SSCs).	Results (timing, flood level, etc.) from the design-related calculations were not used in the evaluation and references to the design-related results with regard to flood level have been removed from the text. When a pipe break occurs in a room, equipment in that room is assumed to be damaged due to the flood. No credit was taken for operator action to isolate the pipe break in the evaluation of the flooding scenarios.
IF-D7	None of the flooding scenario frequencies listed in the tables of Section 9 would satisfy the 1E-7 /yr criterion found in IE-C4. In Section 9.4, a different screening criterion was used that utilized the failure frequency of equipment due to flooding being <0.1% compared to other causes (non-flooding). Although this may have	The analysis has been revised to use the screening criteria provided in SR IF-D7(b).

TABLE 3-1

ASME PRA Standard SR	Self-Assessment Comments	SNC Responses
	been an effective screening method, it was different from that specified by this SR. Therefore, if a different criterion is used, it needs to be reconciled with the specifics of this SR to show equivalency.	
IF-E3a	The Category I / II screening criterion of 1E-9/yr was not specifically used. Instead, a value of 1.0E-08 was cited in Section 9.4. Quantitative screening should be based on the specified value of 1E-09, even if the CDF values that were calculated were low enough such that they still would have met the IF-E3a criteria of <1E-9/yr.	The screening criterion used has been revised to 1.0E-09 per reactor year. However, flooding scenarios with CDF contributions less than this criterion are retained in the report.