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

Vogtle Electric Generating Plant
Additional Information Concerning GL 96-06,
Assurance of Equipment Operability and Containment
Integrity During Design Basis Accident Conditions

Ladies and Gentlemen:

By letter LCV-0897H dated December 2, 2002, Southern Nuclear Operating Company (SNC) provided additional information concerning GL 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions. Additional information was requested on February 7, 2003, during a teleconference between SNC personnel, Mr. Jim Tatum of the NRC Plant Systems Branch, and Mr. Walt Jensen of the NRC Reactor Systems Branch. The requested information is provided in the attachment to this letter.

Sincerely,

Jeffrey T. Gasser

JTG/kgl/daj

Attachment

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Vogtle
Mr. M. Sheibani, Engineering Supervisor – Plant Vogtle
Document Services RTYPE· CVC7000

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. F. Rinaldi, NRR Project Manager – Vogtle
Mr. J. Zeiler, Senior Resident Inspector – Vogtle

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A teleconference was held on February 7, 2003, between Southern Nuclear Operating Company (SNC) personnel, Jim Tatum of the NRC Plant Systems Branch, and Walt Jensen of the NRC Reactor Systems Branch. The subject of the phone call was the Vogtle Electric Generating Plant (VEGP) response (Ref. 1) to Generic Letter (GL) 96-06 Request for Additional Information. As a result of the phone call, the NRC representatives requested that SNC provide responses to four questions concerning the analyses described in Reference 1. Responses to these questions are provided as follows.

1. NRC Request

   *Provide a comparison of the HSTA calculated peak pressure with the pressure calculated using the Joukowski equation based on closing velocity*

   SNC Response

   Using Nuclear Service Cooling Water (NSCW) Unit 2 train A as an example, the peak closing velocity in the auxiliary containment air cooler return line is predicted to be 26.5 ft/sec. Using this value along with a density of 62.4 lbm/ft$^3$ and a (pipe softened) sonic velocity of 4080 ft/sec, the Joukowski equation (equation 5-3 in the EPRI User's Manual – Ref. 2) gives a surge pressure magnitude of 728 psi for water on water impact. For this same impact, the HSTA plotted output indicates a peak pressure head (hydraulic grade line, HGL, i.e., pressure head plus elevation) of 2041.7 ft absolute. In order to compare with the Joukowski result, the pressure increase predicted by HSTA should first be calculated by subtracting the node elevation head plus fluid vapor head (347.3 ft) and converting from units of head to pressure (i.e., divide by 2.31). Doing this yields an HSTA predicted surge pressure magnitude of 734 psi. Similarly, for the other three NSCW trains (remaining train on Unit 2 and the two Unit 1 trains), the HSTA predicted surge pressure at the location of peak closing velocity is either essentially the same or somewhat higher than the Joukowski calculated pressure.

2. NRC Request

   *What was used for the speed of sound?*

   SNC Response

   A value of 4080 ft/sec was used as the sonic velocity in the VEGP analyses. This value was calculated per equation 5-2 of the EPRI User's Manual and accounts for the effects of thin walled pipe deformation on the speed of sound.

3. NRC Request

*What adjustment was used for cushioning? What nomographs were used from the EPRI User's Manual?*

SNC Response

For the void closure at peak velocity that occurs in the auxiliary containment air cooler return piping, an adjustment of Vcushion/Vinitial = 0.82 was used in the VEGP analyses. This value was extracted from Fig. A-45 of the EPRI User's Manual. For void closure in the auxiliary containment air cooler supply piping, a value of Vcushion/Vinitial = 0.86 was conservatively selected based on Fig. A-44 of the EPRI User's Manual.

4. NRC Request

*What were the maximum stresses in the critical components and the margin to failure? What are the combinations of loads in the design basis (FSAR)?*

SNC Response

As indicated in Reference 1, engineering assessments were performed to verify that the pressure boundary integrity of the auxiliary containment air cooler piping loops will be maintained following the waterhammer events. Final documentation of these evaluations has not yet been completed.

The assessments indicate that pipe stresses meet ASME Section III code requirements for faulted conditions. The critical piping components were found to be small bore, cantilevered lines that connect to the main process piping. These small bore lines are used as vents, drains, test connections, and for overpressure protection. The connection points of these small bore lines to the main piping can experience high stresses during the waterhammer events, indicating that a nonlinear type stress analysis would be appropriate to use at these locations. Consequently, it was determined that these lines could be qualified per Appendix F of the ASME Section III code, paragraph F-1341.3 using limit analysis collapse load acceptance criteria. Paragraph F-1341.3 is based on an equivalent static load not exceeding 90% of the limit analysis collapse load. In view of the fact that code requirements are satisfied, the safety margin of the critical piping is consistent with that of other code components under faulted service conditions.

The initial engineering assessments for pipe supports indicated that two mechanical snubbers for one of the NSCW trains should be conservatively postulated to fail. Therefore, these two supports were analytically removed

from the associated piping stress analysis (i.e., no credit taken for these supports). The remaining critical pipe supports were assessed for structural integrity using the guidance of the ASME code, Section III, Subsection NF for components within its jurisdictional boundary. For pipe support components outside the jurisdiction of the ASME code, allowables were employed that are consistent with AISC (Ref. 4), Appendix P of EPRI NP-6041-SL (Ref. 5), and other industry approaches commonly adopted for margin assessment.

The results of the engineering assessments indicated that the critical pipe support components within the jurisdictional boundary of the ASME code were found to be within the stress allowable limits for the faulted service condition. Therefore, the safety margins of these pipe support components are consistent with other ASME code components under the faulted service condition.

For pipe support components outside the ASME code boundary, the results of the engineering assessments showed that most pipe supports meet the allowable stress limits set forth in the VEGP design basis. However, some exceptions were identified for further evaluations. For example, a fillet weld joining a section of tube steel to the web of a structural wide flange member was calculated to have a weld stress of about 41,200 psi, which is 1.15 of the AISC allowable stress of 35,700 psi. Alternatively, a more realistic weld capacity as described in Appendix P (capacity of fillet-welded connections) of Reference 5 was adopted for the purpose of demonstrating structural integrity. The results of this assessment indicated that the weld will have sufficient margin against failure. In another case, high loads were calculated to exist at undercut concrete anchor bolts on supports at the reactor cavity cooler manifold. In this case, the anchor bolts were found to have a safety factor of 2 against failure, thus providing sufficient safety margin for continued plant operation.

Design basis load combinations for class 2 and 3 components and component supports during faulted conditions are shown in FSAR Table 3.9.B.3-1 sheet 2. For components, these combinations include loading effects due to operating pressure, deadweight & liveweight, and transient dynamic events (PO + DW + DF). For component supports, these combinations include loading effects due to deadweight, liveweight, thermal expansion, transient dynamic events, and building settlement (DW + TH + DF + BS).

**References**

1. SNC letter to NRC, LCV-0897-H, Request for Additional Information Concerning GL 96-06, December 2, 2002.

2. EPRI Report No. 1006456, Generic Letter 96-06 Waterhammer Issues Resolution, User's Manual – Proprietary, April 2002.

3. VEGP Calculations X4C1202V56 Rev. 2, X4C1202V57 Rev. 2, X4C1202V60 Rev. 1, and X4C1202V61 Rev. 1.

4. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Part 2, 7th Edition, 1969.

5. EPRI Report NP-6041-SL Revision 1, A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1), August 1991.