Nuclear Operating Company

South Texas Project Electric Generating Station 4000 Avenue F – Suite A Bay City, Texas 77414

January 6, 2010
U7-C-STP-NRC-100003

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville MD 20852-2738

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Supplemental Response to Request for Additional Information

Reference: Letter, Scott Head to Document Control Desk, "Supplemental Response to Request for Additional Information" dated November 10, 2009. (U7-C-STP-NRC-090200)(ML093170203)

Attached is the second supplemental response to an NRC staff question included in Request for Additional Information (RAI) letter number 224 related to Combined License Application (COLA) Part 2, Tier 2, Section 5.3.

Subsequent to submittal of the supplemental response referenced above, the NRC Staff expressed a further concern that the methodology for performing the finite element analysis for determining through-wall thermal and pressure stress distributions for the STP 3&4 RPVs, which is documented in the STP 3 & 4 PTLR, is not an NRC-approved methodology. This submittal provides additional information regarding the methodology for calculating bending and membrane stresses using computer code finite element analysis.

The attachment provides the supplemental response to the RAI question listed below:

RAI 05.03.02-2

No changes to the COLA are required.

There are no commitments in this letter.

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NRO

STI 32595651

If you have any questions, please contact me at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 1/6/10

Scott Head
Manager, Regulatory Affairs
South Texas Project Units 3 & 4

gsc

Attachments:

Question 05.03.02-2, Supplement 2

cc: w/o attachment except*
(paper copy)

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RAI 05.03.02-2

QUESTION

During a telephone conference with the NRC Staff on November 30, 2009, the NRC Staff expressed a concern that the methodology for performing the finite element analysis for determining the through-wall thermal and pressure stress distributions for the STP 3&4 reactor pressure vessels (RPVs), which is documented in the STP 3 & 4 Pressure-Temperature Limits Report (PTLR), Rev. 0, is not an NRC-approved methodology. STPNOC agreed to supplement the initial and supplemental responses to RAI 05.03.02-2 with additional information regarding the methodology for calculating bending and membrane stresses using computer code finite element analysis, including:

a Identification of the computer code(s) that were used in the finite element stress analysis.

b. For any computer codes used, a description of how the code(s) were verified or benchmarked. Computer code verification shall be in accordance with a qualified 10 CFR 50, Appendix B Quality Assurance Program. As a part of computer code verificaton, benchmarking consistent with NRC GL 83-11, Supplement 1 shall be included.

c. Identification of the assumptions and the inputs to the finite element analysis. Necessary inputs to the analysis include any or all of the following:

- A description of plant operating conditions used (e.g., pressure and temperature). The conditions used must represent current plant operating conditions.
- A description of the heat transfer coefficients used and the methodology used to calculate them.
- A description of the model developed, including materials, material properties, finite element mesh pattern, and geometry.

STPNOC also agreed to provide the schedule for revising the STP 3 & 4 PTLR to include the methodology for calculating bending and membrane stresses using computer code finite element analysis and to submit any necessary updates to the PT curves provided in Revision 0 based on the analysis of the STP 3 reactor pressure vessel.

RESPONSE:

STPNOC will supplement the STP 3 & 4 Pressure-Temperature Limits Report (PTLR), Rev. 0 by May 31, 2010 to include additional information regarding the methodology used to perform the finite element analysis for determining the through-wall thermal and pressure stress distributions for the STP 3 & 4 RPVs using the ANSYS code. A markup of the STP 3 & 4 Pressure-

Temperature Limits Report (PTLR), Rev. 0, showing the anticipated revision with placeholders identified in square brackets [ ] where information is currently not available, is attached.

STPNOC will further supplement the STP 3 & 4 Pressure-Temperature Limits Report to include updated curves, if the ANSYS results show that it is necessary, by July 31, 2010.

STP 3 & 4 Pressure-Temperature Limits Report (PTLR), Rev. 0

STP Units 3 & 4 PTLR
Revision 0
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5.0 Discussion (final paragraph)

~~The stress distributions are based on the stress analysis data of a typical Toshiba ABWR plant. These data is calculated by two dimensional, axisymmetric finite element model analysis. Thermal stresses are evaluated based on the temperature transitions and the temperature distributions calculated for thermal transient conditions.~~

The only computer code used in the determination of the STP Units 3 and 4 pressure/temperature curves was the ANSYS (Release 11.0) finite element computer program for the [feedwater nozzle or shutdown cooling outlet nozzle] (non-beltline) stresses. This analysis was performed to determine through-wall thermal and pressure stress distributions for the STP Units 3 and 4 RPV nozzles due to a thermal transient. The ANSYS program is controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 (Reference 6.8) was performed. The verification and validation process compared ANSYS verification problem results provided by ANSYS Inc. with the results of the same problems run by the performer of the STP 3&4 stress calculation. No significant differences were found between the results provided by ANSYS and the results obtained by the STP 3&4 stress calculation performer. The documentation for the verification and validation process for Version 11.0 of ANSYS can be found in [ ].

The following inputs were used for the finite element analysis:

- With respect to operating conditions, stress distributions were developed based on the stress calculation result considering the thermal transient conditions of service level A and B (normal and upset conditions) for the [feedwater nozzle or shutdown cooling outlet nozzle] which is determined to exhibit the most limiting stresses among the RPV nozzles under these conditions. Based on the previous ABWR RPV stress analysis experience, the event where the maximum stress is assumed to occur for the nozzles is just after the reactor coolant temperature drops from 552°F to 376°F in 10 minutes at a system pressure of 1020 psig during the turbine bypass transient.

- Heat transfer coefficients were calculated in the design basis stress report for the STP Units 3 & 4 [feedwater nozzle or shutdown cooling outlet nozzle] and from a model of the heat transfer coefficient as a function of flow rate. The heat transfer coefficients were evaluated at flow rates that bound the actual operating conditions in the [feedwater nozzle or shutdown cooling outlet nozzle].

- A two-dimensional, axisymmetric finite element model of the [feedwater nozzle or shutdown cooling outlet nozzle] was constructed using the same modeling techniques that were employed to evaluate the nozzles in the design basis stress report. In order to properly model the nozzles, the analysis was performed as a penetration in a sphere and not in a cylinder. To make up for this difference in geometry, a conversion factor of [ ] times the cylinder radius was used to model the sphere. Material properties were evaluated according to the temperatures during the transient event condition.