Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

May 26, 2010
U7-C-STP-NRC-100118

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
<u>Supplemental Response to Request for Additional Information</u>

Reference: Letter, Scott Head to Document Control Desk, "Supplemental Response to Request for Additional Information" dated January 6, 2010 (U7-C-STP-NRC-100003) (ML101200211)

Attached is a 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 5.3.2 – Pressure/Temperature Limits.

The attachment provides a supplement to the responses to the RAI question listed below:

05.03.02-2, Supplement 3

No COLA changes are required as a result of this response; however, the changes shown in the attachment will be incorporated into the STP 3&4 Pressure-Temperature Limits Report by July 31, 2010.

There are no commitments in this letter.

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

D091
NRD

STI 32682114

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

Executed on 5/26/10

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

gsc

Attachment: RAI 05.03.02-2, Supplement 3

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

Director, Office of New Reactors
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Kathy C. Perkins, RN, MBA
Assistant Commissioner
Division for Regulatory Services
P. O. Box 149347
Austin, Texas 78714-9347

Alice Hamilton Rogers, P.E.
Inspections Unit Manager
Texas Department of Health Services
P. O. Box 149347
Austin, Texas 78714-9347

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

*Steven P. Frantz, Esquire
A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Ave. NW
Washington D.C. 20004

*Tekia Govan
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

(electronic copy)

*George F. Wunder
*Tekia Govan
Loren R. Plisco
U. S. Nuclear Regulatory Commission

Steve Winn
Joseph Kiwak
Eli Smith
Nuclear Innovation North America

Jon C. Wood, Esquire
Cox Smith Matthews

Richard Peña
Kevin Pollo
L. D. Blaylock
CPS Energy

RAI 05.03.02-2, Supplement 3

SUPPLEMENTAL 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 in its second supplemental response (U7-C-STP-NRC-100003 dated January 6, 2010) to revise the STP 3 & 4 PTLR, Rev. 0 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 for providing 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 revise the STP 3 & 4 Pressure-Temperature Limits Report (PTLR), Rev. 0 by July 31, 2010 with the attached change to Section 5.0, "Discussion" addressing 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.

STPNOC will further revise the STP 3 & 4 PTLR, Rev. 0 to include curves based on these ANSYS results by July 31, 2010. These revised curves will be provided as a function of coolant temperature rather than metal temperature as agreed in the response to RAI 05.03.02-5 (U7-C-STP-NRC-100109 dated May 12, 2010).

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

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 (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 RPV Vendor Document Program Test (Validation) Report ANSYS 11.0, document No. 32820500-C80L500, Rev. 0.

The following inputs were used for the finite element analysis.

- With respect to operating conditions, stress distributions were developed based on the stress analysis results, considering the thermal transient conditions of service level A and B (normal and upset conditions) for the feedwater nozzle, which is determined to exhibit the most limiting stresses among the RPV nozzles under these conditions. Based on the stress analysis results, the event where the maximum stress occurs for the feedwater nozzle is after the reactor coolant temperature drops from 552°F to 376°F in 10 minutes during the turbine bypass transient.

- Heat transfer coefficients were calculated in the design basis stress report for the STP Units 3 & 4 feedwater 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.

- A two-dimensional, axisymmetric finite element model of the feedwater nozzle was constructed using the same modeling techniques that were employed to evaluate the nozzles in the design basis stress report. 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 2.0 times the cylinder radius was used to model the sphere. Material properties were evaluated during the transient event condition.