Nuclear Operating Company

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

October 25, 2010
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Responses to Request for Additional Information and Audit Questions

Attached is a response to an NRC staff question included in Request for Additional Information (RAI) letter number 364 related to Combined License Application (COLA) Part 2, Tier 2, Appendix 6C. The response to the RAI question shown below is provided in Attachment 1.

RAI 06.02.02-28

In addition to this RAI response, Attachment 2 provides responses to five audit questions from a September 28, 2010 NRC audit of the STP 3 & 4 Fuel Assembly Cooling Calculation at the Westinghouse Offices in Rockville, Maryland.

There are no commitments in this letter.

If you have any questions regarding these answers, 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 10/25/10

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

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Attachments:

1. Question 06.02.02-28 Response
2. Response to Audit Questions

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NRD

STI 32766465

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

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RAI 06.02.02-28

QUESTION:

The June 10, 2010, response, to RAI 06.02.02-27 states that the sodium pentaborate from the Standby Liquid Control (SLC) System will be initiated during a LOCA in order to control the suppression pool pH. Please provide the calculated post-LOCA 30-day pH profile and describe the administrative controls that will be in place to ensure initiation of the SLC System injection.

RESPONSE:

The 30-day pH profiles for the STP 3&4 suppression pool following a LOCA are provided in the proprietary Toshiba Report No. SCO-2010-000050, dated September 14, 2010. This report is currently available for NRC review.

The Toshiba report evaluates the post-LOCA suppression pool pH both for Alternate Source Term (AST) and Design Basis Event (DBE) cases. The DBE cases do not assume fuel damage, and therefore include only production of nitric acid in the RPV as a contributor to changes in suppression pool pH. The AST cases, which assume that a severe accident results in release of radioactive inventory to the suppression pool, are not currently part of the STP 3&4 licensing basis, and would not apply to an assessment of the capability of the ECCS to prevent core damage.

For the DBE cases, the Toshiba report shows that suppression pool pH would gradually trend downward due to postulated nitric acid formation in the reactor pressure vessel, if sodium pentaborate is not injected via the Standby Liquid Control (SLC) system. For the AST (non-licensing basis) cases, the Toshiba report shows that the suppression pool pH would begin to drop much more quickly if sodium pentaborate is not injected via the SLC system. For both the DBE and AST cases, if sodium pentaborate is injected, the suppression pool pH would remain at about 8.4-8.6 for the 30-day post-LOCA period.

STP 3&4 will have administrative controls in place to ensure that, if the pH starts to trend downward, the SLC system is initiated to inject sodium pentaborate and maintain the suppression pool pH above 7. These controls will include suppression pool pH monitoring and initiation of the SLC, as necessary. Because this manual action will not be required until hours after the LOCA, these administrative controls will be included as part of the Technical Support Center (TSC) post-LOCA requirements for STP 3&4.

No COLA change is required as a result of this response.

QUESTION No. 1

Explain the relationship of the void fraction criteria of 0.95 to MCPR.

RESPONSE NO. 1

In accordance with 10CFR50 Appendix K, once there is departure from nucleate boiling, the GOBLIN code uses film boiling heat transfer coefficients for the remainder of the run. The void fraction criterion of 0.95 is below the GOBLIN limit for entering steam cooling and assures a significant liquid mass flow is present. Therefore, maintaining an exit void fraction below 0.95 ensures a two-phase mixture in the core and saturated coolant conditions. This criterion assures that there is no significant clad heat up. The plots of clad temperature confirm this behavior.

QUESTION NO. 2

Discuss the reasons why the fuel tests would use subcooled rather than saturated water.

RESPONSE NO. 2

Subcooled water is typically used when testing the hydraulic performance of fuel assemblies. The GSI-191 fuel assembly testing performed by the Pressurized Water Reactor Owner's Group (PWROG) to address downstream effects of debris on fuel used subcooled water (reference WCAP-16793-NP, Rev. 1).

Because of the nature of the test, which allows for flow recirculation in a closed loop and debris addition to a mixing vat at selected intervals, using saturated water would require more complicated test apparatus. Heating the water to saturated conditions or performing a heated test would require a pressure boundary that would be breached when adding debris.

The use of subcooled water provides an adequate test because the acceptance criterion is based upon the ratios of initial and final pressure drops and flow rates, which account for the impact of coolant temperature and density.

QUESTION NO. 3

Provide more discussion on how the reduction factor of 4 was selected and how this bounds the application to the fuel used in the certified design.

RESPONSE NO. 3

The blocked inlet loss factor reduction of 4 was selected as a conservative engineering margin to bound fuel design differences and the uncertainties associated with extrapolating test conditions to design conditions. Because the loss coefficient varies with area squared, the factor of 4 reduction in loss coefficient is equivalent to a factor of 2 increase in allowable flow area. As discussed in the response to Question 5 below, any hydraulic differences between the DCD fuel and Optima2 fuel are small, and the margin provided by this reduction factor will bound any small differences in fuel hydraulic performance. The comparisons in appendix D.4 of Reference 1 indicate that the difference in the hydraulic characteristics between different fuel assembly designs is very small and certainly bounded by the factor of 4.

QUESTION NO. 4

Explain how references 4, 6, and 11 are related and ensure that approved versions of the codes were used in the analysis.

RESPONSE NO. 4

Reference 11 (WCAP 16078, Rev. 0) establishes the use of GOBLIN version 3.12 and CHACHA version 3D.6 in the emergency core cooling system (ECCS) evaluation model (EM) for boiling water reactors (BWRs). References 4 and 6 are release notices for GOBLIN version 3.12.4 and CHACHA version 3D.6.2, respectively. These release notices document minor updates and error corrections made to GOBLIN version 3.12 and CHACHA version 3D.6. These changes have been evaluated to have no effect on predicted results and are consistent with the approved evaluation model requirements.

QUESTION NO. 5

Describe whether reference 7 provides a benchmark of GE fuel to OPTIMA2 fuel. If not, provide references that include this comparison.

RESPONSE NO. 5

Reference 7 is a BWR LOCA procedure for running CHACHA and does not consider specific fuel designs.

Reference 1 below provides the NRC approved methodology used by Westinghouse in providing replacement fuel for a BWR. This methodology has been applied for reload applications at a number of BWRs in the US supporting the Westinghouse SVEA-96 OPTIMA2 fuel design. This report demonstrates the application of the methodology by comparing several fuel designs, including the 8X8-2 design similar to the GE-7 fuel in the DCD, a 9X9-9 design, and the Westinghouse SVEA-96 (10X10, Watercross) design which is similar to Optima2. Sections 5.2.2, 5.3 and Appendix D.4 of this report show that the hydraulic characteristics of the various fuel designs are very similar, which is expected and required in a mixed core. The Westinghouse methodology is designed to assure the hydraulic similarity of the Westinghouse BWR fuel to the fuel specifications for a given BWR.

The use of SVEA-96 OPTIMA2 fuel has been approved in reload applications for several US BWRs using this methodology. The same methodology is being applied to the ABWR to assure the hydraulic performance of the Westinghouse fuel is consistent with the ABWR design.

References

1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel", ABB Combustion Engineering Nuclear Operations, July 1996.