UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 4, 2012

Mr. Kevin D. Richards
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 – NRC STAFF COMMENTS AND QUESTIONS RELATED TO REVIEW OF REPORT RE: RISK-INFORMED RESOLUTION OF GENERIC SAFETY ISSUE (GSI)-191 AT SOUTH TEXAS PROJECT, UNITS 1 AND 2 (TAC NOS. ME7735 AND ME7736)

Dear Mr. Richards:

The U.S. Nuclear Regulatory Commission (NRC) staff and representatives of STP Nuclear Operating Company (STPNOC, the licensee) had a number of pre-licensing meetings since February 2011 to discuss the risk-informed approach to resolution of Generic Safety Issue (GSI)-191 for South Texas Project (STP), Units 1 and 2. STPNOC documented the risk-informed approach and initial results of the assessments in a report titled, "Risk-Informed Resolution of GSI-191 at South Texas Project," and transmitted via e-mail on February 22, 2012, for discussion during the public meeting on March 1, 2012 (Agencywide Documents Access and Management System (ADAMS Accession No. ML120540667). STP, Units 1 and 2, are the lead plants for a risk-informed approach to resolution of GSI-191.

Based on the initial review of the report, the NRC staff has developed the enclosed comments and questions. Please note that these comments may not be all inclusive and the staff may have additional comments as the report is updated in the future. These comments and questions should not be treated as a request for additional information. The purpose of transmitting these comments and questions is to ensure that they are addressed in the future report updates.

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.singal@nrc.gov.

Sincerely,

Alan Wang for

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
As stated

cc w/encl: Distribution via Listserv

NRC STAFF COMMENTS AND QUESTIONS RELATED TO REVIEW OF

REPORT RE: RISK-INFORMED RESOLUTION OF GENERIC SAFETY ISSUE (GSI)-191

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

The U.S. Nuclear Regulatory Commission (NRC) staff and representatives of STP Nuclear Operating Company (STPNOC, the licensee) had a number of pre-licensing meetings since February 2011 to discuss the risk-informed approach to resolution of Generic Safety Issue (GSI)-191 for South Texas Project (STP), Units 1 and 2. STPNOC documented the risk-informed approach and initial results of the assessments in a report titled, "Risk-Informed Resolution of GSI-191 at South Texas Project," and transmitted via e-mail on February 22, 2012, for discussion during the public meeting on March 1, 2012 (Agencywide Documents Access and Management System (ADAMS Accession No. ML120540667). STP, Units 1 and 2, are the lead plants for a risk-informed approach to resolution of GSI-191. The NRC staff reviewed the report after the meeting and provided the licensee with an initial set of comments.

Based on the initial review of the report, the NRC staff has the following comments and questions. Please note that these comments may not be all inclusive and the staff may have additional comments as the report is updated in the future. Also, these comments and questions may not be treated as a request for additional information. The purpose of transmitting these comments and questions is to ensure that the NRC staff's concerns are addressed in the future report updates. The staff understands that several of these items are already being refined as STP develops its licensing application.

NRC Staff Comments/Questions

1. Page 6 of the report states, in part, that

> Key uncertainties that currently dominate initial quantification of CDF [core damage frequency] and LERF [large early release frequency] contributions from recirculation include: (1) the size of the damage zone surrounding a pipe break (commonly referred to as the Zone of Influence, or ZOI), (2) the amount of debris that will pass through a STP strainer, and (3) the formation of chemical products in STP sump conditions (if any) and their degree of impact on head-loss at the strainer and within fuel assemblies.

This appears to imply that transport of debris (including the relationship between size and percentage transported) to the screen is not a key uncertainty. Please describe how the transport of debris from the break area to the sump screens is modeled such that it is not a key source of uncertainty.

Enclosure

2. The conservatisms listed on page 7 of the report would appear to provide limited benefit if in-vessel limits are low and plant-specific testing planned for 2012 is not successful. The NRC staff has concerns that the assertions regarding conservatisms may not be realistic.

3. Page 8 of the report states that "additional refinement of both the STP PRA [probabilistic risk assessment] and CASA [containment accident sequence stochastic analysis] Grande will be needed to ensure a self-consistent treatment of time-dependent accident scenarios with multiple operator recovery actions." Several other places in the report indicate that "fidelity" improvements are anticipated in the loss-of-coolant accident (LOCA) scenarios. However, the discussion on pages 61 and 62 of the report indicates minimal changes have been made to the PRA (e.g. setting sump strainer blockage to zero and adding a failure event that in-vessel effects cause a core melt). In contrast, starting on page 71 of the report, Scenarios for 2012, discusses in detail LOCA scenarios.

   Please explain the following:

   a. Are these operator actions already modeled in the PRA?

   b. Are scenarios that require recirculation after 24 hours also included in the PRA? Are all in-vessel core damage scenarios less than 24-hour scenarios?

   c. During the meeting on June 2, 2011, the licensee stated that it intends to submit a license amendment request (LAR) in 2012 to establish the new design basis resulting from changes to the PRA. Does STP envision that there will be a peer review of the new models and supporting evaluations against the industrial standards endorsed in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"?

   d. The scenarios seem to include termination of containment sprays some few hours into the event. Sometimes containment spray is used (even after non-LOCA sequences) for long-term containment heat removal, pressure control, and radionuclide removal. Please describe the impacts of long-term containment spray operation to produce additional physical and chemical debris that could block the screens or contribute to additional in-vessel damage?

4. Page 11 of the report states that the model determines damage to insulation targets exactly as would occur in the plant; however, this may not be true if spherical or hemi-spherical assumptions are used to simplify ZOI damage assessment. Simplified ZOIs have been accepted by the NRC staff, but were not generally thought to match actual debris generation.

5. Page 11 of the report states, in part, that "only the break frequency size distributions are permitted to have substantial ranges. Every other parameter is defined very close to commonly held regulatory assumptions to facilitate interpretation and communication of results." Page 12 of the report specifies that a uniform distribution is used for physical

variability parameters. Tight uniform distributions, coupled with the "probability of exceeding threshold" steps as illustrated in Figure 6 of the report, would seem to enable the occurrence of numerous zero exceedance probabilities, if there is any reasonable margin in the expected performance. The licensee will need to fully describe and justify the parameter distributions in the proposed LAR.

6. The bypass correlation used by CASA Grande has not been accepted by the NRC staff. Bypass is likely more difficult to estimate than head loss. The concentrations in the testing were relatively high which may be non-conservative. The effect of larger strainer arrays does not appear to have been studied. Uncertainties, as noted on Pages 23 through 25 of the report, should be accounted for in future quantifications.

7. The report states that the spread of the confidence bands for bypass becomes greater as approach velocity increases. Please describe how this affects the STP evaluation. In the range of STP expected velocities, the two data points are near the top of the uncertainty band. It appears that the band used for STP may be non-conservative.

8. Please confirm whether the use of a revised break frequency/size methodology will be equivalent to switching out the values in Table 1 of the report and that the other methods and assumptions in the report will remain the same. Please provide the new changes in CDF and LERF estimates using the new break frequency/size methodology.

9. Page 26 of the report states that a 1/8-inch bed is assumed to result in complete strainer coverage. This is double as compared to the NRC staff guidance. Additionally, for a strainer with uniform approach velocities similar to the one used at STP, lower values of strainer coverage would be expected to result in a filtering bed when compared to the strainers with non-uniform approach velocities. Please explain the impact of using a higher theoretical bed thickness.

10. Please explain the use of a chemical bump-up of 2X, based on a single strainer test. The bump-up could be significantly different for different debris quantities or similar debris amounts but with a different mix of debris bed constituents.

11. The use of the NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," correlation with a chemical bump-up factor to calculate strainer head loss has not been accepted by the NRC staff. A large number of tests may be required to allow a CASA Grande correlation to accurately predict strainer head loss for STP. Application to other plants will most likely require additional testing.

12. The acceptable in-vessel head loss for the cold-leg break is stated to be 4.2 feet. Does this account for a break that would require steam to flow through the loop seal upstream of the reactor coolant pump (RCP) prior to being released from the break in the RCS?

13. Page 27 of the report states that chemical effects would not be significant factor during cold-leg injection. The NRC staff has not accepted that chemical effects would not be a significant factor during cold-leg injection. Please provide the basis for this statement.

14. The assumed fibrous debris loads of 75 g/FA for the cold-leg break and 150 g/FA for the hot-leg break are much higher than currently accepted limits. There does not appear to have been any testing to validate these assumptions. The tests used to justify these limits appear to have been blanked out. It is not apparent that the tests were based on limiting conditions. Please describe the tests used as the basis for these values and the underlying analytical assumptions.

15. Page 34 of the report states that, for hot-leg injection, there are no locations where significant blockage within the core would occur. This does not appear to have been demonstrated. The report does not discuss cases where any of the three safety injection trains is unavailable.

16. The report states that 500 g/FA is acceptable for a cold-leg break during cold-leg injection. Please describe any testing that would validate this value.

17. Please discuss when structural, flashing, and deaeration issues will be addressed.

18. The amount of debris assumed to transport to inactive regions does not appear to have been stated. Please explain the assumptions regarding inactive regions.

19. The report discusses a user-specified window of principal penetration on page 47 of the report and Tbypass on page 49. It is not apparent what these terms mean. Please define these terms and describe their use in documentation and future quantifications.

20. The report discusses future plans for a more generic transport diagram that will handle more debris sizes and transport fractionated steps. The NRC staff has concerns that implementation of such a scheme may be difficult to justify. This is not an issue for the initial quantification but it could become an issue if incorporated into future quantifications.

21. Page 49 of the report identifies several physical assumptions. A number of these assumptions do not appear to have been accepted by the NRC staff (e.g. 1/8-inch debris results in strainer coverage, in-vessel limits, and chemical bump-up), as described in the previous comments.

22. Please explain why certain debris (e.g. equipment tags, etc., that reduce strainer area) is ignored. The report did not provide justification.

23. Page 50 of the report uses the term, "debris-type histories." Please define this term and describe how it is determined.

24. Please provide the conditions for the curves in Figure 18 of the report.

25. RCP-related accidents, such as seal failure, are not included. Please describe how this affects the PRA comparison.

26. Page 62 of the report quotes failure rates for strainers due to blockage. Please describe the basis for choosing the 1E-5 value since Westinghouse recommended using a

different value for breaks with more debris and larger breaks. What impact will it have when RCP seal LOCAs are modeled in CASA Grande?

27. Page 65 of the report states, in part, that "it is not possible to have sufficient material present to cause sump strainer blockage resulting in loss of NPSH [net positive suction head]." Current test results indicate that the sump strainer will block following some large pipe ruptures. "Not possible" is substantively different than very unlikely. Please explain the differences between the assumptions underlying the test results indicating certain blockage versus the probabilistic analysis that lead to a zero probability of sump strainer blockage.

28. Options for alternative cooling are discussed on page 78 of the report. Please describe how long it would take to refill the refueling water storage tank (RWST). Does it need to have boric acid added? How long would it take to complete this evaluation, if required? Please describe the capabilities, procedures, and training for refilling the RWST, including the necessary operator actions and timing for refill.

29. Page 81 of the report provides Table 9, "Thermal-hydraulics case execution matrix" to identify the cases of interest for the Thermal-Hydraulic Analysis. Please explain why the concentration was on cold-leg breaks. Also, please explain why the breaks are assumed to occur at the bottom of the pipe, as stated on Page 83 of the report.

30. Please explain why the report does not address defense in depth.

31. Please provide justification for the assumption that unqualified coatings fail at 24 hours.

32. Please explain why the potential for breaks at locations other than welds was not addressed.

33. The transport assumptions appear to be conservative for strainer head loss, but are not described in sufficient detail. Please discuss the transport assumptions in detail.

34. The licensee stated that MELCOR simulations may soon be available to provide break-specific containment pressure and pool temperature histories. Please describe the use of the MELCOR simulations. The initial quantification does not credit containment accident pressure. Crediting containment accident pressure requires very detailed modeling of containment heat sinks that will result in substantial additional NRC staff review and it may weaken the defense-in-depth demonstration.

35. In cases where the strainer is fully covered, the calculated head loss was doubled to account for chemical effects. However, since chemical precipitation is not expected to occur until later in the event, the chemical effects bump-up was only applied for head loss values after 24 hours as stated on page 26 of the report. The NRC staff has concern that the chemical effects bump-up has not been modeled adequately.

36. Please explain how the chemical effect distributions in CASA will account for environmental conditions that are outside the current information base.

37. Please explain why the potential effects of boric acid precipitation are not addressed. What effect does this have on the initial quantification?

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.singal@nrc.gov.

Sincerely,

/RA by Alan Wang for/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
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