U.S. Nuclear Regulatory Commission Staff Audit of the South Texas Project, Units 3 and 4, Source Term Assessment Supporting Chapter 12 of the South Texas Projects Combined License Application

Audit Report

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit of the South Texas Project (STP), Units 3 and 4. The purpose of the audit was to verify that the applicant had adequately addressed the Advanced Boiling Water Reactor (ABWR) Design Control Document (DCD) Chapter 12 Combined License (COL) information item concerning the radiation sources for spent fuel storage in the STP Final Safety Analysis Report (FSAR). The audit was conducted at the Nuclear Licensing Center in Rockville, Maryland on January 13, 2011. The NRC staff conducted the audit in accordance with the Office of New Reactors (NRO) Office Instruction NRO-REG-108 [1]. The purpose and objectives of the audit are contained in the NRC staff's audit plan documented in the Agencywide Documents Access and Management System under accession number ML110871600 [2].

2.0 PARTICIPANTS

The following NRC staff members from the NRO, Division of Construction Inspection and Operational Programs (DCIP) and Division of New Reactor Licensing (DNRL) participated in the audit:

- Charles Hinson (Audit Team Leader, DCIP)
- Edward Stutzcage (General Physicist, DCIP)
- Edward Roach (Chief for Health Physics Branch, DCIP)
- Mark Tonacci (Chief for BWR Projects Branch, DNRL)
- George Wunder (Lead Project Manager, DNRL)
- Raj Anand (Chapter Project Manager, DNRL)

The following key individuals from the applicant's organization participated in the audit:

- Scott Head (STP/NINA)
- Scott Stephens (STP/NINA)
- Stan Anderson (Westinghouse)
- Brad Maurer (Westinghouse)
- Sylvia Want (Westinghouse)
- Tom Veitch (Tane)

3.0 BACKGROUND

Section 12.2 of the ABWR DCD states that the radiation sources for spent fuel storage will be provided by the COL applicant. During the review of the STP FSAR, the NRC staff determined that the radiation source information for the spent fuel assemblies provided in the STP FSAR was insufficient for the NRC staff to perform confirmatory shield calculations in accordance with the NRC staff guidance provided in Sections 9.1.2 and 12.3-12.4 of the Standard Review Plan (NUREG-0800). This guidance is intended to verify that dose rates to an operator from a raised spent fuel assembly in the spent fuel pool (SFP) are ALARA during refueling operations. To obtain the information needed to perform these confirmatory shielding calculations, the NRC staff issued Request for Additional Information (RAI) number 12.02-20 to the applicant on September 22, 2010. STP provided their response and draft supplemental response to this RAI on January 3 and 6, 2011, respectively. After review, the NRC staff determined that the applicant had not provided the data necessary for the NRC staff to perform its confirmatory shielding calculations. The NRC staff proposed conducting an audit of the COL applicant to be conducted at the Twinbrook office in Rockville, MD, in order to gather the information needed to perform the confirmatory calculations, and to review the applicant's calculation methodology and assumptions.

4.0 AUDIT ACTIVITIES

The NRC staff notified the applicant that the NRC staff would be performing an audit to review the following information related to RAI 12.02-20 during the course of the audit.

- Applicant's tabulation of the source term data for 1) the 2365 fuel bundles (270% of the core) stored in the SFP, 2) the single maximally irradiated fuel element in the SFP (the fuel element source term used by the applicant as the basis to calculate the dose rate to the operator on the refueling bridge), and 3) the source term of the radionuclides in the SFP water.

- Calculations and assumptions demonstrating that dose rates to personnel from fuel handling activities are maintained ALARA and less than 2.5 mr/hr from a raised fuel assembly in the SFP during refueling operations.

- A description of how the information contained in ABWR DCD Table 12.2-3b (Gamma Ray Source Energy Spectra-Post-Operation Gamma Sources in the Core), is used by the applicant, which is incorporated by reference by Section 12.2 of the STP FSAR.

The audit was conducted by a team of NRC staff knowledgeable in the ABWR. The findings of the audit are addressed in the NRC staff's safety evaluation report.

A summary of the applicant's audit presentation to address the NRC staff's concerns is provided below.

4.1 Spent Fuel Rack Source Term

In the applicant's initial response to RAI number 12.02-20 (January 3, 2011), the applicant stated that they would amend Chapter 12 of the STP FSAR to add an Appendix 12B which would contain site-specific information regarding the STP SFP geometry and methodology used to generate SFP radiation sources and dose rate assessments. As part of the applicant's audit presentation, the applicant described how the new source term information contained in STP FSAR Appendix B was derived.

The applicant described how they derived a new core source term. The STP, Units 3 and 4 core rated power is 3926 MWt. Using a core load of 872 fuel assemblies with an average enrichment of 4.2%, cycle length of 704 days, and reload batch size of 320 fuel assemblies, the applicant used the code ORIGEN-S of the SCALE system to calculate the equilibrium core gamma source. The applicant stated that, because ORIGEN-S calculations included the mass of supporting structural material in the core, in addition to fuel elements, the applicant added a 10% safety margin or bias to the calculated source term to bound the uncertainties in structural material specifications. The applicant provided a source term table comparison showing that this bias bound the DCD design core source term values.

The applicant described how they used the ORIGEN-S code to determine the radiation source term for the SFP filled to maximum capacity of 1493 spent fuel assemblies. This maximum SFP radiation source term consists of equilibrium core fuel assemblies with decay times ranging from one-day to ten years post shutdown. Similar to the radiation source term for the core, the applicant added an additional 10% safety margin to the calculated results for the spent fuel rack source term.

4.2 Spent Fuel Pool Shielding Analysis

The applicant described their assumptions used to model the dimensions of the SFP to calculate the estimated radiation levels in the neighboring areas surrounding the SFP. The applicant modeled the SFP as a rectangular parallelepiped with walls and floor 2 meters thick and assumed that the SFP peak radiation source was homogenized over the bottom 3.8 meters of the SFP (to represent the active fuel length of a typical BWR fuel assembly). Inputting this data into the point-kernel shielding code QAD-CGGP-A, the applicant determined that the resulting dose rates in all areas surrounding the SFP meet the reactor building radiation zone limits, as described in the ABWR DCD.

4.3 Maximum Assembly Source Term

The applicant stated that the average fuel assembly burnup for a core rated at 3926 MWt with 872 fuel bundles is 4.502 MWt/bundle. This average consists of once-, twice-, and thrice-burned bundles.

The applicant stated that the fuel bundle with the highest dose rate would be the bundle having the highest activity. The applicant determined that the activity of a once-burned batch of fuel would have be higher than the activities for either twice- or thrice-burned fuel bundles and, therefore, the maximum assembly source term would be from a once-burned fuel assembly.

4.4 Spent Fuel Handling Shielding Analysis

The applicant assumed that the fuel assembly was raised to the maximum up-position height of 8.5 feet below the spent fuel pool water surface. The applicant then described how they used the QAD-CGGP-A code to model the raised fuel assembly as three separate radiation sources; the active fuel region, the top fuel node where the power density is lower than the core average, and the assembly handle where it is assumed that majority of the activation products accumulate. Assuming that the operator on the refueling machine trolley platform is at a minimum of 8.8 ft above the water surface, the applicant calculated the resulting peak dose rate to the operator from the raised fuel assembly to be approximately 1.0 mrem/hr (a 10% dose rate conservatism was added to the calculated dose rate to obtain this value). (Note that in a revised response to RAI 12.02-20, dated April 26, 2011, under accession number ML11119A076, the applicant added a 30% dose rate conservatism to arrive at an estimated dose rate to the refueling operator of 1.2 mrem/hr from a raised fuel assembly). The applicant confirmed that this calculated peak dose rate to an operator is less than the 2.5 mrem/hr operator dose criteria stated in ANSI/ANS-57.1-1992.

The NRC staff asked the applicant to provide an estimate of the dose rate that an operator would receive from radionuclides in the SFP water. Using representative data from an operating, current generation, ABWR, the applicant estimated that the maximum dose rate at 1.2 meters (4 ft) above the refueling floor would be expected to be 0.7 mrem/hr during plant outages with fuel assemblies fully seated in the storage racks and the maximum levels of measured radionuclides in the SFP water. The applicant stated that the dose rates from radionuclides in the SFP water would be expected to be less than this for normal operations of the Fuel Pool Cooling and Cleanup system.

4.5 Source Term Data Contained in ABWR DCD Table 12.2-3b

As part of the NRC staff's evaluation of the applicant's initial response to RAI 12.02-20, the NRC staff used the source term data contained in ABWR DCD Table 12.2-3b ("Gamma Ray Source Energy Spectra-Post Operation Gamma Sources in the Core") to calculate the estimated dose rate from a raised fuel assembly in the SFP. When the NRC staff questioned the applicant on the reason for the large discrepancy between the NRC staff's calculated dose rate and the applicant's calculated dose rate from a raised fuel assembly, the applicant confirmed that there was an apparent error in Table 12.2-3b which resulted in the table values being low by six orders of magnitude.

One of the objectives of this audit was to verify if the applicant used the data contained in Table 12.2-3b to determine the shutdown source term for a single fuel element.

If the data in this DCD table was not used to determine the shutdown source term for a single fuel element, then the applicant was requested to describe how this source term was generated.

During the audit, the applicant stated that the information in DCD Table 12.2-3b was not used to generate the shutdown source term for a single fuel element. The applicant said that the bases for determining the source term for a single fuel element would be included in a new appendix to the STP FSAR (Appendix 12B, Spent Fuel Pool Geometry and Dose Rate Calculation) which would be provided in a revised response to RAI 12.02-20.

5.0 SUMMARY OF EXIT MEETING

The NRC staff conducted an exit meeting with the applicant and other meeting participants identified in Section 2.0 above on January 13, 2011.

In this meeting, the NRC staff summarized its activities during the audit and identified the following additional issues that the NRC staff would like the applicant to address in a revised response to RAI 12.02-20:

- Provide the assumptions used in determining the peaking factor for the maximum activity spent fuel element
- Correct the apparent discrepancies contained in the applicant's initial response to RAI 12.02-20 regarding dimensions of the SFP and the distances to measured dose points
- Include in FSAR Appendix 12B a graph showing the dose rate to the refueling operator from the various modeled components of the raised spent fuel assembly in the SFP
- Include in FSAR Appendix 12B a source term table for the maximum activity spent fuel element
- Address the apparent errors in Table 12.2-3b
- Modify the SFP figure in the response to RAI 12.02-20 to include the dimensions of a fuel assembly
- Provide references for the applicant's calculation packages used to determine the SFP radiation source, the SFP dose rates, and the dose rates from a single fuel assembly in the SFP
- Provide a basis for the SFP radionuclide concentration values listed in the response to RAI 12.02-20

In response to the additional NRC staff concerns listed above, the applicant stated that they would review their initial response to RAI 12.02-20, dated January 3, 2011, and submit a revised response to RAI 12.02-20 to the NRC staff within 30 days of this audit.

6.0 References

1. U.S. Nuclear Regulatory Commission Office of New Reactors (NRO) Office Instruction, “Regulatory Audits,” NRO-REG-108, USNRC, April 2, 2009.

2. Draft memorandum from Edward Roach, Branch Chief, CHPB, DCIP, to Mark Tonacci, Branch Chief, NGE, DNRL entitled, “Audit Plan to Gather Additional Information Regarding STP Nuclear Operating Company’s Calculations Used to Determine the Dose Rate to an Operator from a Raised Spent Fuel Element in the Spent Fuel Pool,” January 11, 2011.