December 23, 2009

Mr. Mark McBurnett, Vice President
Regulatory Affairs
South Texas Project Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: REGULATORY AUDIT SUMMARY OF SOUTH TEXAS PROJECT, UNITS 3 AND 4 COMBINED LICENSE APPLICATION REVISION 3 – PROBABILITY RISK ASSESSMENT

Dear Mr. McBurnett:

By letter dated September 20, 2007, South Texas Project Nuclear Operating Company (STPNOC) submitted to the U.S. Nuclear Regulatory Commission (NRC) a Combined License (COL) application to construct and operate two reactor units (Units 3 and 4) based on the U.S. Advanced Boiling Water Reactor (ABWR) Design Certification at the South Texas Project Nuclear Power Plant. The NRC Office of New Reactors (NRO) is reviewing the South Texas Project (STP) COL application that incorporates by reference the ABWR Design Control Document (DCD). As part of this review, the NRO Probabilistic Risk Assessment (PRA) and Severe Accidents Branch conducted an audit of documentation supporting the STP COL application on September 22 and 23, 2009, at the Nuclear Energy Institute (NEI) office in Rockville, Maryland. The NRC staff followed the guidance in NRO Office Instruction NRO-REG-108, "Regulatory Audits," in performing this audit. We have enclosed the detailed results of the audit.

Please contact Michael Eudy at (301) 415-3104 or Michael.Eudy@nrc.gov if you have any questions related to the audit.

Sincerely,

/RA - R. Anand for/

Mark Tonacci, Chief
ESBWR/ABWR Projects Branch 2
Division of New Reactor Licensing
Office of New Reactors

Docket Nos.: 52-012
52-013

Enclosure:
"Nuclear Regulatory Commission Staff Audit of South Texas Projects (STP), Units 3 and 4, Probabilistic Risk Assessment Supporting Chapter 19 of the STP COL Application"

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ACCESSION NO. ML **NRO-002**

OFFICE	NRO/DNRL/NGE2	NRO/DNRL/NGE2	NRO/DSRA/SPRA	NRO/DSRA/SPRA	NRO/DNRL/NGE2
NAME	MEudy	BAbeywickrama	THilsmeier	LMrowca	MTonacci
DATE	12-22-09	12-23-09	12-23-09	12-23-09	12-23-09 (RAnand)

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Nuclear Regulatory Commission Staff Audit of South Texas Projects, Units 3 and 4, Probabilistic Risk Assessment Supporting Chapter 19 of the South Texas Project Combined License Application

Audit Report

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff conducted an audit of the South Texas Projects (STP), Units 3 and 4 (3&4), probabilistic risk assessment (PRA), which supports Chapter 19 of the STP 3&4 Final Safety Analysis Report (FSAR) [1]. The audit was conducted at the Nuclear Energy Institute (NEI) office in Rockville, Maryland during the period of September 22-23, 2009. The staff conducted the audit in accordance with NRC Office of New Reactors (NRO) Office Instruction NRO-REG-108 [2]. The plan for this audit was distributed to the applicant in an e-mail dated September 17, 2009 [3].

2.0 PARTICIPANTS

The following NRC staff members from the NRO, Division of Safety Systems and Risk Assessment (DSRA) and Division of New Reactor Licensing (DNRL) participated in the audit:

- John Lai (Audit Team Leader)
- Donald Dube (DSRA Senior Level Advisor for PRA)
- Edward Fuller (Senior Risk & Reliability Engineer)
- Lynn Mrowca (Chief for PRA and Severe Accidents Branch)
- Todd Hilsmeier (Risk & Reliability Engineer)
- Keith Tetter (Risk & Reliability Engineer)
- Michael Eudy (Project Manager)
- Marie Pohida (Senior Risk & Reliability Engineer)

The following NRC contractors from Energy Research Incorporated participated in the audit:

- Roy Karimi
- Zhe Yuan
- Michael Zavisca

The following key individuals from the applicant's organization participated in the audit on a regular basis:

- D. W. (Bill) Stillwell (STP PRA Supervisor)
- Scott Bannet (STP PRA Analyst)
- Moshin Khan (PRA Contractor)
- Gene Hughes (PRA Contractor)
- Cheryl Eddy (PRA Contractor)
- Ricky Summit (PRA Contractor)
- Alan Toblin (PRA Contractor)

Enclosure

3.0 AUDIT ACTIVITIES

The audit was conducted by a team of NRC staff and contractor personnel knowledgeable in the Advanced Boiling Water Reactor (ABWR) and STP PRA. The audit covered the full range of topics addressed in the staff's safety evaluation report. A summary for each of the topics covered is provided below.

3.1 Accident Sequence Analysis

Prior to the audit of September 22-23, 2009, staff reviewed the accident sequence analysis in the ABWR Standard Safety Analysis Report (SSAR) [4], including selected event trees as provided in Section 19D of this report. Staff also reviewed Section 19.3.1.3 of the STP 3&4 FSAR, Revision 2, for departures. Based on this review, staff chose the following two at-power internal event trees in the SSAR for comparison against the reconstituted STP Computer Aided Fault Tree Analysis(CAFTA) model (this reconstituted model is represented by REC) during the audit:

- Large break loss-of-coolant accident
- Inadvertent opening of relief valve (IORV)

The REC model event trees were found to be functionally identical to those in the SSAR. No top events in the level-1 event trees were found for control rod drive (CRD) flow, containment overpressure protection system (COPS), and firewater addition in either the SSAR or the REC models. Staff further verified that CRD flow and firewater addition are not explicitly modeled in the pertinent STP fault trees.

3.2 Success Criteria

Prior to the audit visit, staff reviewed the success criteria as described in Section 19.3.1.3.1 of the SSAR and tabulated in Table 19.3-2 of the SSAR. The staff also reviewed the changes to the success criteria as described in Table 19.3-2 of the STP 3&4 FSAR. The only departure in the STP success criteria table is that of requiring the addition of a condensate booster pump wherever a condensate pump appears in the corresponding SSAR table.

Staff requested verification that the discharge pressure of the condensate booster pump would be sufficient to overcome reactor pressure vessel backpressure for the credited initiating events. Staff confirmed that the discharge pressure of the condensate booster pump was equivalent to that of the original condensate pump as described in the SSAR and adequate to provide injection as specified in the success criteria of Table 19.3-2 of the STP 3&4 FSAR.

Staff noted that Table 19.3-2 in both the SSAR and STP 3&4 FSAR show success of ADS8 + 1 firewater addition system pump for all transients. However, this is an apparent contradiction to Section 19.1.2 of the STP 3&4 FSAR that states:

"Although an AC-independent Firewater Addition System is incorporated in the design, no credit is taken for it in the calculation of core damage frequency."

As discussed under Section 3.1 of this audit report, staff verified via review of pertinent fault trees in the REC model and sampling of the event trees that no credit for firewater addition to the reactor vessel is taken in the level-1 internal events analysis. Therefore, the above statement in Section 19.1.2 of the STP FSAR is correct. For clarification purposes, the staff will request confirmation that no credit is taken for firewater addition to the reactor vessel in the calculation of the baseline core damage frequency (CDF).

STP personnel and their contractors substantially reconstituted the level-1 internal events PRA from the SSAR. Sequence by sequence comparisons were made between the REC model and the SSAR PRA. A number of significant discrepancies arose when no credit was taken for CRD flow and COPS (as well as residual heat removal (RHR) recovery actions prior to containment failure and core damage) in the level-1 REC model. These differences could be substantially reconciled when credit for CRD and COPS (and, apparently, recovery of RHR) was taken via post-processing of the relevant accident sequence frequencies. Specifically, it is estimated that without credit for CRD flow (or credit for recovery of some other high pressure injection system) in the REC model a number of sequences could be as much as an order of magnitude higher in frequency than the corresponding SSAR PRA results. Integrated over all sequences, credit for CRD flow reduces CDF by about 3%. Likewise, credit for COPS (and, apparently, RHR recovery) reduces the estimated internal events CDF by about a factor of 3 to 4.

While CRD flow is not explicitly described as part of the success criteria in Table 19.3-2, CRD flow (or recovery of some other high-pressure injection system) may be credited for several events in the reconstituted PRA model. Therefore, the staff will request clarification regarding the following statement in Section 19.3.1.3 of the STP 3&4 FSAR:

"The Control Rod Drive (CRD) pumps which have limited capacity have not been included in the success criteria."

Staff review of the SSAR also identified that while credit for COPS is not explicitly modeled in the level-1 PRA event trees, credit can be found in the containment event trees (CETs). For example, Figure 19D.5-10 of the SSAR (e.g., Amendment 33) shows the CET for the Class II plant damage state and corresponding sequences. COPS rupture disk opening (RD) for the branch path with no RHR recovery leads to successful core cooling and no core damage. Thus, the staff will request clarification regarding the extent to which credit is taken for COPS for relevant events.

Based on the above observation, a request for additional information (RAI) was issued to ask the applicant to clarify the level-1 modeling and related statements in FSAR.

3.3 Data Analysis

As a result of STP's design departures, additional basic events were added to the STP 3&4 plant-specific PRA model that were not included in the SSAR (i.e., Section 19D.6 of the SSAR) (e.g., fans were added to the reactor service water system ultimate heat sink (UHS) electrical components were added to the 4.16 kV medium voltage electrical system). The applicant stated that the failure data associated with these basic events exist in Electric Power Research Institute (EPRI) documents and other sources. The staff issued an RAI to formerly request these data sources.

3.4 System Modeling

Staff reviewed the impact of several departures on the STP 3&4 plant-specific PRA results. Departure STD DEP 19.3-1 evaluated common cause failures (CCF). From the STP 3&4 plant-specific PRA, the top gate of the RSW/UHS fault tree generated a failure probability of 7.1E-4 that includes CCF modeling. While the same top gate from the REC model generated a failure probability of 3.2E-4 that does not include CCF modeling. Based on the cutset results, the major difference in the results between the STP 3&4 plant-specific PRA and the REC model is due to CCF modeling. For the reactor core isolation cooling system (RCIC) departure described in STD DEP T1 2.4-3, the top gate of the RCIC fault tree in the STP 3&4 plant-specific PRA generated a failure probability of 5.77E-2 versus 5.82E-2 generated from the REC model. The minor difference confirmed that the RCIC departure had a minor effect on the PRA results.

Departure STD DEP 10.4-5 provides for four (4) variable speed, reactor feedwater (FW) pumps and four (4) condensate booster pumps. Normal rated power operation is with three (3) reactor FW pumps operating and one in auto standby. If one operating reactor FW pump trips and the reactor FW pump in auto start does not successfully start, then automatic power reduction (by recirculation runback) occurs to avoid a reactor scram. The staff requested that the applicant describe how this departure was modeled in the STP 3&4 plant-specific PRA. The applicant indicated that a point estimate for human error probability was modeled for the feedwater unavailability in the PRA, and that this point estimate does not change for this departure. The staff issued an RAI to request the applicant to describe how the point estimate was obtained for the feedwater system departure.

Departure STP DEP 1.1-2 provides for dual unit site with common fire protection system. The applicant stated that the change from a single unit to a dual unit for the fire protection system has no effect on the PRA. However, the staff determined that this departure would have a minimal or negligible effect on the STP 3&4 plant-specific internal events and fire PRA as opposed to no effect. Pending the results of the discussion provided in Section 3.7 of this report, the applicant will revise the STP 3&4 FSAR to state that this departure has a minimal or negligible effect on the STP 3&4 plant-specific PRA.

3.5 Initiating Event Frequency (Loss of Offsite Power)

The staff reviewed STP's detailed quantitative calculation used to determine that the risk impact of loss of offsite power (LOOP) events at STP is bounded by the analysis in Section 19D of the SSAR. This detailed calculation included a sensitivity analysis comparing the LOOP PRA results of the SSAR, including LOOP frequency and recovery data, to similar area specific data using the

Energy Reliability Council of Texas (ERCOT) regional information in Table 3-6 of NUREG/CR-6890 [6]. The power recovery distribution for STP 3&4 is consistent with that used in the SSAR. The detailed calculation showed that there is a decrease in core damage frequency from loss of offsite power initiating events for STP 3&4, which confirms that the frequency estimates for the loss of offsite power events used in SSAR Section 19D.3.1.2.4 are bounding for the STP 3&4.

The staff determined that the applicant did not actually use the ERCOT regional LOOP frequency (i.e., 0.0262/reactor-critical-year), but rather used the plant-level, industry average LOOP frequency in Table 3-1 of NUREG/CR-6890 (i.e., 0.0359/reactor-critical-year). This discrepancy, however, does not change the conclusions that the frequency estimates for the loss of offsite power events used in SSAR Subsection 19D.3.1.2.4 are bounding for the STP 3&4. Based on the above observation, the applicant agreed to revise the detailed calculation using the ERCOT data and resubmit their response to RAI 19.01-1.

3.6 External Events

3.6.1 External Flooding

To meet 10CFR Part 52.79 (d)(1), the combined license (COL) applicant referencing a standard design certification must update the plant-specific PRA information to account for site-specific design information and any design departures. Thus, the staff reviewed the quantitative evaluation for external floods involving breach of the Main Cooling Reservoir (MCR). This evaluation was found in Engineering/Licensing Evaluation titled "External Flooding Event, Breach of the Main Cooling Reservoir" (dated April 20, 2009). The staff raised questions regarding the justification for the site-specific MCR dam break frequency of 1.03E-6 per year. The staff needs additional justification on the reduction factors used to obtain this frequency from the generic dam failure frequency of 1E-4 per year. The staff also needs additional information on the probability (basic event - OCD) of the operator failing to close the single, normally open, watertight access door between the service building and the control building. Since the recently submitted STP 3&4 FSAR, Revision 3, presents new information related to MCR external flooding, the questions related to MCR external flooding described above will be addressed through the RAI process during the staff's review of the STP 3&4 FSAR, Revision 3.

3.6.2 Internal Fire Risk Assessment

The staff discussed with the applicant the evaluations for determining the impact of departures on fire risk. The applicant described their internal process for evaluating, validating, and documenting the fire impact. Also, the applicant will revise their response to RAI 19.01-23 to address the impact of turbine building modifications on the fire risk assessment.

3.7 Shutdown Risk Analysis

The staff discussed four shutdown RAIs with the applicant. The first RAI (19-16) pertains to a design departure involving the reactor vessel reference leg water level instrument lines being flushed by CRD water. These lines are continually flushed by CRD water to resolve concerns with non-condensable gas buildup. At the audit, the applicant stated that STP will resubmit their RAI response to provide information on: (1) the length of time the reactor vessel reference leg water level instrument lines can support accurate reactor vessel level readings, once both CRD pumps stop running, and (2) whether a commitment is needed in Section 19.9 of the FSAR that addresses the need to flush these instrument lines.

The second and third RAIs (19-18 and 19-20) pertain to departure STD DEP 1.1-2 which seeks additional risk information on the shared fire water system and shutdown hurricane risk. The STP 3&4 FSAR describes a dual unit site compared with the ABWR DCD [5] which describes a single unit site. The applicant provided a draft quantitative evaluation of the site hurricane shutdown risk given the shared fire water system. The CDF was estimated to be higher than the tornado-induced frequency 1.1E-8 per year. The staff will merge RAIs 19-18 and 19-20 into a single RAI requesting the following information: (1) the site shutdown hurricane CDF and large release frequency (LRF) given the shared fire water system (2) the top one hundred cutsets for CDF and LRF, (3) the Fussell-Vesely and Risk Achievement Worth for component failure events, human error probabilities, and common cause failures, and (4) the list of Structure, Systems, and Components that were identified as risk significant for the Reliability Assurance Program.

The fourth RAI (19-21) pertains to the risk of external floods at shutdown. As described in Section 3.6.1 for external flooding, the staff reviewed the quantitative evaluation for external floods involving breach of the MCR. The staff will reissue RAI 19-21 to request: (1) a discussion of procedural controls to prevent and minimize the effect of a MCR breach at shutdown, including the impact of breached flooding barriers to facilitate maintenance and (2) a discussion of any temporary openings or penetrations that could impact the effect of a MCR breach at shutdown.

3.8 Level-2 PRA and Severe Accident Evaluation

The applicant presented the results of Modular Accident Analysis Program (MAAP) 4.0.7 simulations of the representative severe accident scenarios analyzed by General Electric (GE) using a version of MAAP 3.0B made especially for the ABWR. The GE simulation results were documented in Chapter 19 of the ABWR SSAR. The staff discussed some of the details of the MAAP 4.0.7 results with the applicant, and requested additional information (i.e., RAI 19-28) that is needed to complete confirmatory assessment activities.

The Severe Accident Intergration Designer Alternative (SAMDA) analyses provided in the response to RAI 19-4 were discussed. In particular, the conservative approach taken for low-volatility fission product releases was explained by the applicant. The staff agrees that the approach is conservative.

4.0 Summary of Exit Meeting

The staff conducted an exit meeting with the applicant on September 23, 2009. The following individuals, in addition to those listed in Section 2.0, participated in the meeting:

- Mohammed Shuaibi, Acting Deputy Director, NRO/DSRA, NRC.
- Scott Head, Regulatory Affairs Manager, STPNOC (briefed after the exit meeting)

The staff summarized its activities during the audit and discussed the following key preliminary results:

- Several new RAIs will be issued as discussed in Section 3.0 and listed in Section 5.0 of this report.

- Several revised responses will be issued by STPNOC as discussed in Section 3.0 of this report.

- STPNOC will issue a white paper (or provide information) that describes the engineering process for evaluating the impact of departures on the STP 3&4 plant-specific PRA.

- STPNOC will provide a copy of the STP 3&4 plant-specific PRA description and results at the Westinghouse Twinbrook office for the staff's review.

5.0 RAIs Issued

The staff has issued to the applicant the following RAIs based on the results of the audit:

- RAI 19.01-28: Requesting information on how the feedwater unavailability was estimated for the STP 3&4 design.

- RAI 19.01-29: Requesting information on how the failure data was generated for new basic events added to the STP 3&4 plant-specific PRA.

- RAI 19.01-30: Requesting confirmation that CRD, COPS and RHR recovery are credited in both the STP 3&4 plant-specific PRA and the REC model.

- RAI 19-28: Requesting MAAP 4.0.7 input file and output results to simulate the accident scenario LCHP.

6.0 References

1. STP Nuclear Operating Company, "South Texas Projects, Units 3 and 4, Final Safety Analysis Report," Revision 2.

2. Nuclear Regulatory Commission Office of New Reactors (NRO) Office Instruction, "Regulatory Audits," NRO-REG-108, USNRC, April 2, 2009.

3. E-mail from Michael Eudy (USNRC) to D.W. Stillwell (STPNOC), “Final STP Audit Agenda (9/22-9/24),” September 17, 2009.

4. GE Nuclear Energy, “ABWR Standard Safety Analysis Report,” Chapter 19, Revision 9.

5. GE Nuclear Energy, "ABWR Design Control Document," Revision 4, March 1997.

6. NUREG/CR-6890 (INL/EXT-05-00501), “Reevaluation of Station Blackout Risk at Nuclear Power Plants - Analysis of Loss of Offsite Power Events: 1986-2004,” Volume 1, December 2005.