June 17, 2009

Mr. Scott Head, Manager
Regulatory Affairs
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 124 RELATED TO SRP SECTION 19 FOR THE SOUTH TEXAS PROJECT COMBINED LICENSE APPLICATION

Dear Mr. Head

By letter dated September 20, 2007, STP Nuclear Operating Company (STP) submitted for approval a combined license application pursuant to 10 CFR Part 52. The U. S. Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed application.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter.

To support the review schedule, you are requested to respond within **30** days of the date of this letter. For RAI 2801, you have requested a 45-day response time and is acceptable. If changes are needed to the safety analysis report, the staff requests that the RAI response include the proposed wording changes.

If you have any questions or comments concerning this matter, I can be reached at 301-415-8484 or by e-mail at Tom.Tai@nrc.gov or you may contact George Wunder at 301-415-1494 or George.Wunder@nrc.gov.

Sincerely,

/RA/

Tom M. Tai, Senior Project Manager
ABWR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Docket Nos. 52-012
52-013

eRAI Tracking No. 2624, 2712, 2754, and 2801

Enclosures:
Request for Additional Information

cc: William Mookhoek
Bill Stillwell

If you have any questions or comments concerning this matter, I can be reached at 301-415-8484 or by e-mail at Tom.Tai@nrc.gov or you may contact George Wunder at 301-415-1494 or George.Wunder@nrc.gov.

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***Approval captured electronically in the electronic RAI system.**

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Request for Additional Information No. 2624 Revision 2

South Texas Project Units 3 and 4
South Texas Project Nuclear Operating Co.
Docket Nos. 52-012 and 52-013
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19.0

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19-7

Table 19.2-2 of the STP COLA, Revision 2, describes Dual Units at STP 3 and 4 (STP DEP 1.1-2). This departure changes from a single fire protection system for a single unit to a single fire protection system for a dual unit. Please explain whether manual switchover from one unit to the other unit was modeled and, if so, its impact on CDF due to a Fire event? Describe the impact of this single fire protection system for two units on the PRA results due to an initiating event that can simultaneously affect both units (i.e. LOOP).

19-8

Table 19.2-2 of the STP COLA, Revision 2, describes Residual Heat Removal Flow and Heat Capacity Analysis (STD DEP 5B-1). In Table 19.2-2, the increased RHR heat removal rate (0.0427 MW/ºC) does not match that listed in the Departures Report (4.27×10^5 W/ ºC or 0.427 Mw/ºC). Please clarify what the correct value is. Explain whether PRA results are impacted by this change in the RHR heat exchanger heat removal capacity.

19-9

Table 19.2-2 of the STP COLA, Revision 2 describes ADS Manual Control (STD DEP 7.3-7). The Departures Report states that key lock switches are replaced with normal manual pushbutton switches. Please explain if this component is modeled in the PRA and, if so, what was the impact on PRA results. In Table 19.2-2, it states that there is a potential beneficial effect for plant-specific PRA. Please explain why this is beneficial with respect to the PRA.

19-10

Table 19.2-2 of the STP COLA, Revision 2, describes ESF Logic and Control System (ELCS) Mode (STD DEP 7.3-10). Table 19.2-2 states that this change is a clarification to text but the Departures Report states that this is a design change. Please clarify and explain how the PRA results are affected due to this design change of the ESF Logic and Control System (ELCS) Mode.

19-11

Table 19.2-2 of the STP COLA, Revision 2, describes Containment Spray Logic Change (STD DEP 7.3-13). Table 19.2-2 states that this change is a clarification to text but the Departures Report states that this is a design change. Please clarify and explain how the PRA results are affected due to the Containment Spray Logic Change.

19-12

Table 19.2-2 of the STP COLA, Revision 2, describes Residual Heat Removal Suppression Pool Cooling modification (STD DEP 7.3-14). Table 19.2-2 states that this change is a clarification to text but the Departures Report states that this is a design change. Please clarify and explain how the PRA results are affected due to the design change of the SPC manual initiation switch.

19-13

Table 19.2-2 and the Departures Report of the STP COLA, Revision 2, describe changes in testing of Safety Relief Valve Solenoid Valves (STD DEP 7.3-16). Please explain if these components are modeled in the PRA and, if so, what was the impact of these changes on PRA results. In Table 19.2-2, it states that there is a potential beneficial effect for plant-specific PRA. Please explain why this is beneficial with respect to the PRA.

19-14

Table 19.2-2 of the STP COLA, Revision 2, describes updated Reactor Building Cooling Water System (STD DEP 9.2-1). Table 19.2-2 states that this change is a clarification to text but the Departures Report states that this is a design capacity change. Please clarify and explain how the PRA results are affected due to the design capacity change of the Reactor Building Cooling Water System.

19-15

Section 19.9.2 of the STP COLA, Revision 2, states: “An evaluation of CUW operation in the heat removal mode will be completed and PRA will be updated prior to fuel load in accordance with 10 CFR 50.71 (h)(1).” This appears to be a commitment just as the “emergency operating procedure to operate the CUW in heat exchanger bypass mode will be developed and implemented prior to fuel loading” is annotated as Commitment 19.9-2. Please explain how this activity will be tracked for future implementation.

Request for Additional Information No. 2712 Revision 2

South Texas Project Units 3 and 4
South Texas Project Nuclear Operating Co.
Docket Nos. 52-012 and 52-013
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: Chapter 19

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19-16

STD DEP 7.7-1, RPV Water Level Instrumentation

On RPV water level instrumentation, the ABWR DCD mentioned that all instrument lines are flushed even when they do not need to be. The STP design addresses condensable gas build up in the RV reference leg water level instrumentation by using CRD to continually flush the instrument lines. The staff recognizes that the CRD system may not be operating in Modes 4 and 5 since it is not required to operate in Modes 4 and 5 according to Technical Specifications. Therefore, the staff requests STP to address how STP intends to flush the instrument lines during Modes 4 and 5 and how this action will be controlled.

19-17

STD DEP 10.4-5 Condensate and Feedwater System

In Section 19.L.7.2 of the STP FSAR, a list of core cooling systems that satisfy the core cooling system success criteria are listed. However, this list only contains pumps with the capability to keep the core covered. The core heat removal path is not listed such as (1) the number of SRVs that need to be opened to remove heat from the vessel or (2) where the core heat is to be discharged such as the suppression pool given an extended loss of DHR. The success criteria needs to be augmented to include all SSCs in the heat removal path, not just the list of injection paths.

19-18

STD DEP 1.1-2 Dual Units at STP 3 and 4

The STP FSAR describes a dual unit site compared with the ABWR DCD which describes a single unit site. In the FSAR, STP stated that the shared fire water system between STP 3 and 4 is not expected to result in any changes to the assessed risk associated with shutdown since the frequency for both units being in a shutdown condition and requiring backup cooling is extremely small. Since (1) there are currently no administrative controls precluding both units entering into a refueling outage or entering a forced outage simultaneously and (2) the Abnormal Procedures for STP 1 and 2 require a plant shutdown prior to the occurrence of a hurricane, the staff needs additional information to conclude that the shared fire water system does not result in any change to shutdown risk. The staff requests STP to evaluate quantitatively the core damage frequency resulting from a postulated dual unit SBO event given a grid-related or severe weather loss of offsite power (includes hurricanes and tornadoes) during Modes 4 and 5.

Request for Additional Information No. 2754 Revision 2

South Texas Project Units 3 and 4
South Texas Project Nuclear Operating Co
Docket No. 52-012 and 52-013
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: Chapter 19

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19-1

In ABWR DCD Section 19.3.2.1, it is stated that " …The accident event progression for each of the accident classes was analyzed using the MAAP code…". Please clarify if the same version of the MAAP code was utilized by the STP to produce the accident progression results for STP 3 and 4. If a different version of the MAAP code was used:

(a) Please explain the implication of these differences on the ABWR containment failure probability and the large release frequency.

(b) Please provide comparisons of key event times (e.g. core uncovering, core relocation to lower head, vessel failure, drywell flooder activation, containment failure), and additional figures-of-merits (e.g., melt mass, melt composition, melt temperature, water mass in lower drywell, containment pressure and temperature, etc.) at the times of these key events. Discuss these differences in terms of their quantitative impacts on the calculated loads and the probability of containment failure due to various severe accident challenges, including direct containment heating, ex-vessel steam explosions, and basemat/pedestal penetration. Provide a discussion of the MAAP results for STP 3 and 4 for radial ablation of the pedestal, and whether or not they affect the conclusions previously drawn from the MAAP3B-ABWR code results. In particular, please verify that the MAAP results for STP 3 and 4 continue to suggest that using a 1.5 m layer of basaltic concrete as sacrificial material would avert containment liner failure for approximately 24 hours after core damage for the more likely severe accident challenges. Finally, discuss any implications pertaining to developing the technical basis for accident management procedures.

19-2

The submittal incorporates by reference the information in Appendix 19D of the reference DCD (which in turn points to the SSAR) with no departures or supplements. The SSAR section contains quantified CETs (as of Amendment 33) with numerical estimates of phenomenology-based event outcomes. This latter document was developed circa 1990s. To assure that there are no significant increases in large release frequency (LRF) or conditional containment failure probability (CCFP), on the basis of the differences in the results calculated by STP, please describe any significant changes and report any differences in the estimates of the LRF or CCFP. If there are indeed significant changes, please discuss them in the context of any phenomenological modeling improvements in the version of the MAAP computer code utilized by STP. In addition, please provide the results of CET quantification.

19-3

Contributions to LRF and CCFP from severe accidents during low power or shutdown operations were not included in the ABWR SSAR or in the STP 3 and 4 FSAR. More recent design certification PRAs have shown that such scenarios are significant and sometimes dominant contributors to LRF and CCFP. Please discuss the impacts on LRF and the overall CCFP from low power and shutdown scenarios for STP 3 and 4. In addition, please explain whether or not the deletion of the Flammability Control System, including the recombiners, from the STP 3 and 4 design, affects the consideration of hydrogen combustion during the startup/shutdown periods when the containment may not be inerted.

19-4

FSAR Section 19E.3 provides the consequence analysis results, incorporating by reference the information contained in the ABWR DCD Section 19E.3. In this section, the accident consequences for nine cases were re-evaluated based on the STP site-specific parameters. However, the listed source terms are limited to three release groups (i.e., noble gas, iodine and cesium). The applicant indicated that, the remaining five groups have negligible releases. However, the assessment of severe accident mitigation alternatives (SAMA) requires consideration of the impacts of these releases. Please provide a complete list of release fractions for all cases evaluated. In addition, please discuss how the addition of the missing released nuclides would affect the SAMA evaluations in the Environmental Report.

19-5

In developing the technical basis for accident management procedures for STP 3 and 4, it will be necessary to identify departures from Revision 2 of the BWROG Accident Management Guidelines and capture the severe accident-related insights from the ABWR SSAR and the STP PRA. This is necessary to address potential changes in the emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs), particularly with respect to strategies for flooding the containment by using the drywell flooder, or from the AC-independent water addition (ACIWA) system sprays. It will be necessary, for example, to avoid inadvertent operation of these features in order to assure that there is not a pool of water in the lower drywell into which molten core debris could pour during a severe accident and cause a large steam explosion. Please describe the necessary changes to the BWROG EPGs and SAGs, as applied to the STP 3 and 4 ABWRs, to ensure sound severe accident mitigation strategies and procedures.

19-6

The staff requests the following information to support its confirmatory assessment activities:

(a) The parameter file for the MAAP version used by the STP for units 3 and 4, and the scenario-specific input files prepared by the STP.

(b) The site-specific MACCS input deck, including the ABWR-specific core inventory.

Request for Additional Information No. 2801 Revision 2

South Texas Project Units 3 and 4
South Texas Project Nuclear Operating Co.
Docket Nos. 52-012 and 52-013
SRP Section: 19.01 - Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed
Application Section: 19.0

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19.01-1

Section 19.3.1.1 of the STP FSAR, Revision 2, in support of meeting the requirement of 10 CFR 52.79(a)(46) pertaining to the plant-specific probabilistic risk assessment (PRA), states the following: “In order to verify that the Subsection 19D.3 remains bounding for the STP 3 and 4, loss of offsite power and power recovery data from NUREG/CR-6890 (Reference 19.3-8) was also evaluated. Industry composite data in NUREG/CR 6890 was used, which conservatively bounds the experience for the STP site. This evaluation verified that the overall risk impact of grid events at STP is bounded by the original Subsection 19D analysis.”

The staff requests that the applicant describe the quantitative information used to determine that the risk impact of loss of offsite power events at STP is bounded by the analysis in Subsection 19D of the referenced Design Control Document (DCD). Also, describe the impact of the plant-specific loss of offsite power and power recovery data on the DCD PRA results and insights.

19.01-2

In Section 19R.4.4 ("Control Building") of the STP FSAR, Revision 2, the set of assumptions for the "worst case" control building flood is presented twice with each having a different assumption for pipe length between the ultimate heat sink and the RCW/RSW room. The staff requests that the applicant clarify in the STP FSAR the assumptions for the "worst case" control building flood.

19.01-3

Section 19R.4.6 ("RSW Pump House") of the STP FSAR, Revision 2, states "The results of this 'worst case' RSW pump house flood are: (1) A leak occurs in the RSW pump room and the lower level sensor alarms at 0.4 meters." The results for this "worst case" RSW pump house flood appears to be incomplete. The staff requests that the applicant provide additional information on the results for this "worst case" RSW pump house flood (e.g., With no operator action, the water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped and redundant supply isolation valves are automatically isolated at 1.5 meters.).

19.01-4

The departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the plant-specific probabilistic risk assessment (PRA) for reactor building flooding in Section 19R.5 ("Probabilistic Flood Assessment") of the STP FSAR, Revision 2. For example, these departures can impact the failure probabilities associated with the top events for bringing reactor to safe shutdown condition in the reactor building flooding event trees (refer to Figures 19R-11 to 19R-13 in the ABWR Standard Safety Analysis Report). Also, Section 19R.5 did not provide the plant-specific core damage frequency (CDF) for reactor building flooding. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the plant-specific CDF value for internal flooding of reactor building, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

19.01-5

Departure STD DEP 10.4-2 increased the number of circulating water pumps to four. This can impact the plant-specific probabilistic risk assessment (PRA) for turbine building flooding in Section 19R.5.3 ("Turbine Building") of the STP FSAR, Revision 2. For example, this departure can impact the failure probabilities associated with top events "PTRIP" and "VCLOSE" in the turbine building flooding event tree (refer to Figure 19R-8 "Turbine Building Flooding, High PCHS" in the ABWR Standard Safety Analysis Report). In addition, the departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the failure probabilities associated with the top event for bringing reactor to safe shutdown condition in the turbine building flooding event tree. Also, Section 19R.5.3 did not provide the plant-specific core damage frequency (CDF) for turbine building flooding. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the plant-specific CDF value for internal flooding of turbine building, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

19.01-6

Departure STD DEP 9.2-5 increased the RSW flow rate per pump from 1800 m^3/h to 3290 m^3/h and increased RSW pipe sizes. This can impact the plant-specific probabilistic risk assessment (PRA) for control building flooding in Section 19R.5.4 ("Control Building") of the STP FSAR, Revision 2. For example, this departure can impact the timing associated with operator actions in top events "OPACT1", "OPACT2" and "OPACT3" in the event tree for control building flooding due to RSW line break (refer to Figure 19R-9 "RSW Control Building Flood" in the ABWR Standard Safety Analysis Report). In addition, the departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the failure probabilities associated with the top events for bringing reactor to safe shutdown condition in the control building flooding event trees. Also, Section 19R.5.4 did not provide the plant-specific core damage frequency (CDF) for control building flooding due to RSW line and fire water system breaks. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the plant-specific CDF values for internal flooding of control building due to RSW line and fire water system breaks, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

19.01-7

Section 19R.5.6 ("RSW Pump House") of the STP FSAR, Revision 2, states "Unisolated breaks in the fire water system could cause inter-divisional flooding since the RSW divisional separation splits the RSW pump house into three, watertight compartments." However, Section 19R of the STP FSAR does not provide or describe a probabilistic risk assessment (PRA) for internal flooding due to unisolated breaks in the fire water system in the RSW pump house. The staff requests the applicant describe in the STP FSAR the PRA internal flooding analysis for this scenario, or justify why it should not be included.

19.01-8

Section 19R.5.6.1 ("RSW Line Breaks") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for internal flooding due to reactor service water (RSW) line breaks in the RSW pump house. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for internal flooding due to RSW line breaks in the RSW pump house:

- Total CDF for this internal flooding event,
- PRA significant accident sequences and their mean CDFs,
- Initiating event frequency estimation and its basis, and
- Top event failure probabilities and their basis.

Also, for this internal flooding event, explain where the PRA assumes the worst case pipe break to occur (e.g., pipe break occurs downstream of the RSW pump discharge motor-operated valve, pipe break occurs upstream of the RSW pump discharge motor-operated valve).

19.01-9

The reference numbers provided in Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, do not correspond to the references in Section 19R.8 ("References"). The staff requests the applicant correct this inconsistency in the STP FSAR.

19.01-10

Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for external flooding due to Main Cooling Reservoir (MCR) breach. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for external flooding due to MCR breach:

- Total CDF for this external flooding event,
- PRA significant accident sequences and their mean CDFs,
- Initiating event frequency for MCR failures that could impact STP Units 3 and 4 and the basis for this frequency,
- Top event failure probabilities and their basis, and
- Failure probability for operator action to close control room watertight access door and the basis for this failure probability.

Also, confirm the staffs interpretation that the watertight control room access door is normally closed (except for intermittent ingress and egress), but the MCR external flooding PRA conservatively assumes this door to be open prior to MCR breach.

19.01-11

Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for external flooding due to multiple, concurrent upstream dam (MCUD) failures. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for external flooding due to MCUD failures:

- Total CDF for this external flooding event,
- PRA significant accident sequences and their mean CDFs,
- Initiating event frequency for MCUD failures and the basis for this frequency, and
- Top event failure probabilities and their basis.

Also, Section 19R.7.3 ("Quantification of External Flooding Initiating Event Frequency") of the STP FSAR, Revision 2, states "The frequency of multiple, concurrent upstream dam breaks considers the failure of three dams, the S. W. Freese, Buchanan, and Mansfield Dams. … Downstream of the S. W. Freese Dam is the Buchanan Dam. It is assumed that failure of the Buchanan Dam is dependent on the failure of the S. W. Freese Dam. Table 19R-4 gives values for common cause factors. Although not considered a common cause failure in the traditional sense, the second and third dam failures are analyzed using the common cause factors from Table 19R-4. Using the Beta factor from Table 19-4, failure of the Buchanan Dam, given failure of the S. W. Freese Dam is calculated. Failure of the third dam, the Mansfield Dam, given failure of the first two dams, is calculated using the Gamma factor given in Table 19R-4. The frequency of multiple concurrent dam failures considered as external flooding initiating events is calculated to be very low." The uncertainty associated with the MCUD initiating event frequency could be large. It may be more appropriate to conservatively assume that the Beta or Gamma factors are 1.0. The staff requests the applicant provide the basis for the assumption of analyzing the second and third dam failures using the common cause factors from Table 19R-4.

19.01-12

The third paragraph of Section 19K.11.1 of the ABWR DCD (“Component Inspections and Maintenance”) states "Multiplexers which provide multiple signals to several systems are identified by the Level 1 analysis as high importance components. Safety system multiplexers have a built-in self test that checks circuits frequently. In addition, one of four multiplexers can be bypassed and tested during plant operation without loss of system function. …" Section 19K.11.1 of the STP FSAR, Revision 2, does not appear to address this statement for departure STD DEP T1 3.4-1 ("Safety-Related I&C Architecture"). The staff requests that the applicant address the above comment and revise Section 19K.11.1 of the STP FSAR as necessary.