



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

July 5, 2011
NOC-AE-11002687
10CFR54
STI: 32889793
File: G25

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2746

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information for the
South Texas Project License Renewal Application (TAC No. ME4938)

- Reference:
1. STPNOC Letter dated October 25, 2010, from G. T. Powell to NRC Document Control Desk, "License Renewal Application", (NOC-AE-10002607) (ML103010257)
 2. NRC letter dated May 31, 2011, "Request for Additional Information for the Review of the South Texas Project, License Renewal Application (ML11140A015)

By Reference 1, STP Nuclear Operating Company (STPNOC) submitted the License Renewal Application (LRA) for South Texas Project (STP) Units 1 and 2. By Reference 2, the NRC staff requested additional information for the review of the STP LRA. STPNOC's response to the request for additional information is included in the Enclosure to this letter.

There are no regulatory commitments in this letter.

Should you have any questions regarding this letter, please contact either Arden Aldridge, STP License Renewal Project Lead, at (361) 972-8243 or Ken Taplett, STP License Renewal Project regulatory point-of-contact, at (361) 972-8416.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 5, 2011
Date

G. T. Powell
Vice President,
Technical Support & Oversight

KJT

Enclosure: STPNOC Response to Request for Additional Information

A147
NRR

cc:
(paper copy)

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
612 East Lamar Blvd, Suite 400
Arlington, Texas 76011-4125

Balwant K. Singal
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North (MS 8B1)
11555 Rockville Pike
Rockville, MD 20852

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

John W. Daily
License Renewal Project Manager (Safety)
U.S. Nuclear Regulatory Commission
One White Flint North (MS O11-F1)
Washington, DC 20555-0001

Tam Tran
License Renewal Project Manager
(Environmental)
U. S. Nuclear Regulatory Commission
One White Flint North (MS O11F01)
Washington, DC 20555-0001

(electronic copy)

A. H. Gutterman, Esquire
Kathryn M. Sutton, Esquire
Morgan, Lewis & Bockius, LLP

John Ragan
Catherine Callaway
Jim von Suskil
NRG South Texas LP

Ed Alarcon
Kevin Pollo
Richard Pena
City Public Service

Peter Nemeth
Crain Caton & James, P.C.

C. Mele
City of Austin

Richard A. Ratliff
Alice Rogers
Texas Department of State Health Services

Balwant K. Singal
John W. Daily
Tam Tran
U. S. Nuclear Regulatory Commission

Enclosure

STPNOC Response to Request for Additional Information

STPNOC Response to Request for Additional Information

**SOUTH TEXAS PROJECT
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION
REGARDING THE ANALYSIS OF SEVERE
ACCIDENT MITIGATION ALTERNATIVES**

NRC Requested Information:

- 1. Provide the following information regarding the Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternative (SAMA) analysis:**
 - a. Environmental Report (ER) Section F.2 states that the current PRA model (STP_REV6) reflects the plant design configuration as of December 31, 2007. Confirm that this applies to the analysis of all initiating events, both internal and external. Describe any significant changes made to plant design or operation since that date and their impact on the SAMA analysis.**

STPNOC Response:

The review of STP_REV6 Initiating Events Notebook confirmed that it reflects the plant design configuration and operating history as of December 31, 2007, including internal and external events. Table 5.4 of the Initiating Events Notebook details the plant operating history updated in STP_REV6.

The review of plant design changes to be included in STP_REV7 (through December 31, 2010) identified one plant modification that will require a revision to the PRA model currently being updated. The steam generator power-operated relief valves (PORV) are now failed closed upon loss of alternating current (AC) power to the hydraulic pumps that charge the accumulators for providing motive power to the PORVs to comply with accident analysis assumptions. An operator action is now required to allow the accumulators to operate the PORVs upon loss of AC power to the SG PORV hydraulic pumps. Although a final analysis is not complete yet, a Human Reliability Analysis will be performed to evaluate adding this operator action to the upcoming PRA model revision. It is expected that this design change will not significantly impact the PRA model results or the SAMA results currently under review.

NRC Requested Information:

- b. The South Texas Project (STP) PRA appears to be a single unit model. Identify any significant design or operating differences between STP, Units 1 and 2, and, if there are shared systems between units, describe how these systems are**

modeled in the PRA. Provide an assessment of the impact of any significant differences between units or shared systems on the SAMA analysis.

STPNOC Response:

The STP PRA model is a single unit model because STP Units 1 and 2 are designed to be identical, and therefore, the STP PRA model results apply to both units. The common switchyard, the Main Cooling Reservoir, and the Essential Cooling Water Pond are shared between STP Units 1 and 2. The units were designed to be identical and have primarily remained so. There are currently two plant differences of significance.

The first difference was identified while reviewing a design change package prior to the STP_REV6 update. A design change to implement automatic load tap changers for the Engineered Safety Features (ESF) transformers will be installed in all three safety trains in both units over several years. During the STP_REV6 model development and update phase, a sensitivity study was performed to determine the impact to core damage frequency (CDF) and large early release frequency (LERF).

Table 1-1 below shows the overall model result differences. A change of 0.43% for CDF and 0.46% for LERF is small. The differences can be accounted for in loss of main transformer (LOMT) initiating event (~18%). The LOMT is a direct impact to the unit auxiliary transformer. The change for a general alignment scheme of Unit Auxiliary Transformer and Standby Transformer means that loss of the main transformer initiating event is slightly more important.

Table 1-1			
OVERALL ESF TRANSFORMER ALIGNMENT SENSITIVITY RESULTS			
CDF (events/year)			
BEFORE	AFTER	DELTA CDF	% Delta
1.15670E-05	1.16170E-05	5.00000E-08	0.43%
LERF			
BEFORE	AFTER	DELTA	%
7.0824E-07	7.1153E-07	3.29000E-09	0.46%

Notes

- BEFORE: Prior to ESF transformer automatic load tap changer design change
- AFTER: After installing an automatic load tap changer in one of three ESF transformers per unit (i.e. current condition of the design change)

Given that the sensitivity results were so low, it has been determined that no change the Probabilistic Risk Assessment (PRA) model is necessary during the implementation of this design change. Based on the above sensitivity results, it is concluded that these alignment changes will not have significant impact on the SAMA analysis.

The second difference between the units is a much shorter time frame difference. When implementing a design change (see response to RAI question 1.a) for the steam generator (SG) power-operated relief valves (PORV), it was identified that having control switches for

operating the PORVs in the control room would be beneficial. This additional change was identified too late for implementation in Unit 1 the refueling outage 1RE16. The majority of the design change was implemented in the 1RE16 (April 2011). The entire design change will be implemented in the Unit 2 refueling outage 2RE15 (November 2011) and the additional control switches in the control room will be implemented in Unit 1 during 1RE17 (October 2012). The design changes for the SG PORV will require a human reliability analysis (HRA) for the proceduralized operator action to manually control the SG PORV. Until the Unit 1 control switches are installed in the control room during 1RE17, the HRA will reflect only local actions in the field. The HRA will be updated accordingly when the additional control switches are installed. The above differences between Units 1 and 2 have minimal impact on the PRA and SAMA analysis.

NRC Requested Information:

- c. The listing and description of initiators for the external events in ER Table F.2-1 appear to indicate that there are other initiators evaluated but not listed. For example, three control room fire scenarios (i.e., 18, 23 and 10) are listed that address only three fire zones (i.e., 047, 071, and 147). Discuss the initiators not included in the table and their contribution to core damage frequency (CDF), and assess their potential impact on the SAMA assessment.**

STPNOC Response:

Environmental Report (ER) Table F.2-1 contains a complete listing of the external initiating events. Note that control room fires contain a prefix "FR" and fire scenarios for other areas are denoted with a prefix of "Z" followed by the fire zone number and a scenario letter. Hence the control room fires listed are fire areas FA10, FA18, and FA23. The fire scenario with the highest frequency CDF is named Z047 and models a fire in zone 047. The response to RAI 3.d. contains additional information regarding a fire scenario and its impact. Fire initiator Z047 is a fire in the cable spreading room train B.

NRC Requested Information:

- d. ER Section F.7.1 states that the CDF of 6.39E-06 per year is a mean value from the RISKMAN Monte Carlo quantification. Confirm that all the CDF and release category frequency values given are also mean values. If so, describe why it appears that the sum of the initiating event contributor's mean values reported in Table F.2-1 equal the mean of the total distribution.**

STPNOC Response:

The quantification of the Level 1 model results in over 62,000 sequences. Each sequence contains a complete set of failed, succeeded, and bypassed top events. The database cannot store all of the sequences. Thus a reduced set of sequences is used for Monte Carlo analysis to determine data uncertainty characteristics. However, due to the reduced set of saved sequences, the resulting Monte Carlo distribution is scaled so that the mean of

the distribution matched the mean of the CDF point estimate. The CDF of 6.39E-06 per year is a point estimate.

NRC Requested Information:

- e. Briefly describe the modeling of the planned and unplanned maintenance conditions assumed for the SAMA analysis. Specifically, indicate if the PRA results used in the SAMA analysis represent the results for the annual average unavailability of systems. If different than this, assess the impact of using the annual average maintenance and testing condition on the SAMA analysis.**

STPNOC Response:

The PRA results used in the SAMA analysis represent the results for the annual average unavailability of systems.

The planned and unplanned maintenance conditions modeled in STP_REV6 are described by the following information in the Planned Maintenance Event Tree (PMET) Notebook for STP_REV6:

For the purposes of modeling in the PRA there are two kinds of maintenance: planned and unplanned. Unplanned maintenance is assumed to occur randomly, except as prevented by Technical Specification rules which would require a shut down. Planned maintenance occurs according to the rolling maintenance schedule and is organized to account for dependencies among systems to minimize system/train outage time.

The general rule for modeling unavailability in STP_REV6 (and previous models) is that planned maintenance is modeled in the event tree module in the PMET. Unplanned maintenance is modeled in systems module as system maintenance/test alignments or into the system fault trees as basic events, so that the effects are included in the master frequency file split fractions.

The majority of the planned maintenance is modeled in the PMET, following the functional equipment groups (FEG) which are organized by work control to account for dependencies among systems to minimize system/train outage time and thereby minimizing risk. At the time that PMET was developed, there were several modeling size constraints. Because of these constraints, only a limited number of FEGs could be modeled in PMET. Any system that does not have its planned maintenance modeled in PMET has unavailability (planned and unplanned) modeled as unplanned maintenance in the systems module.

NRC Requested Information:

- f. **Provide a brief summary of the history of the STP Level 1 PRA that includes for each revision: the date released, the CDF contribution for internal events and each of the external event hazards [i.e., seismic, fire, tornado, and main cooling reservoir (MCR) breach], and the major changes in the revision that led to the change in the CDF, including identification of major changes or updates to the modeling for various initiator groups such as internal flooding, fire, and seismic. Also, identify the STP PRA revision reviewed in the 2002 Westinghouse Owners Group (WOG) peer review.**

STPNOC Response:

The summary will start with model STP 1999, which was the model peer reviewed by the Westinghouse Owners Group (WOG) in 2002. The model was released in October 2001.

Table 1-2			
STP_1999 Core Damage Frequency (CDF) Groupings (events/year)			
Total CDF	Internal Events Contribution	External Events Contribution	
1.17E-05	8.84E-06	Fires	1.40E-06
		Floods	1.41E-08
		Flood MCR	2.88E-07
		High Winds (i.e. tornados)	1.1E-06
		Seismic	7.29E-08
		Total External	2.87E-06

Model STP_REV4 was released in September 2003. The major changes that affected CDF and the containment response were:

- The plant specific data updated for train unavailability, initiating events update, and component failure data update
- Incorporation of the latest operator error modeling and improved loss of offsite power (LOOP) recovery modeling.
- Inclusion of Safety Injection accumulator modeling in Large and Medium loss of coolant accident (LOCA) event trees
- Inclusion of hot leg recirculation modeling in the Large LOCA event tree
- Removal of the 150-ton air conditioning chillers, and
- Improved modeling of support system initiating events.

Table 1-3			
STP_REV4 CDF Groupings (events/year)			
Total CDF	Internal Events Contribution	External Events Contribution	
1.17E-05	9.08E-06	Fires	1.0E-06
		Floods	1.48E-08
		Flood MCR	2.88E-07
		High Winds (i.e. tornados)	1.1E-06
		Seismic	7.26E-08
		Total External	2.48E-06

In November of 2004 STP_RV41 was released. The major changes that affected the CDF and containment response were:

- Addition of "Operator Depressurization" to the Small LOCA event,
- Correction of modeling errors in Medium LOCA Long-Term response model,
- Re-quantification of the initiating event frequency for Inadvertant Opening of One and Two Pressurizer Safety Valves to reflect the failure to reclose in the initiating events,
- Correction of the Conditional Split fraction definitions used in the model to correct errors in the Basic Event importance calculations
- Re-binning of several maintenance duration data variables to correct input problems with RISKMAN Revision 7, and,
- Splitting of fault tree basic events containing several components into individual basic events to prepare for mitigating systems performance indicator implementation and to remove undue conservatism in basic event importance calculations used in the Graded Quality Assurance process.

STP_RV41 reported a CDF 1.2% higher than STP_REV4. A model STP_RV42 was made to correct issues found during component risk ranking using STP_RV41. STP_RV42 was released in February of 2005 and showed a negligible increase (9E-08) from STP_RV41. The CDF was 9.28E-06.

STP_REV5 was released in September of 2005. This model incorporated plant modifications, procedure changes, and data update through 2004. In addition, modifications to the Class IE Vital AC system and the main steam isolation valves are modeled. The model incorporates a major change in the human reliability analysis (HRA) methods to use the Electric Power Research Institute (EPRI) HRA calculator. A Level 2 analysis update with a revision in the containment capability analysis was also included.

Table 1-4			
STP_REV5 CDF Groupings (events/year)			
Total CDF	Internal Events Contribution	External Events Contribution	
1.04E-05	7.67E-06	Fires	9.72E-07
		Floods	1.43E-08
		Flood MCR	2.88E-07
		High Winds (i.e. tornados)	1.1E-06
		Seismic	7.28E-08
		Total External	2.73E-06

A STP_RV51 was released in July, 2007 but did not result in any change to the CDF or the Large Early Release Frequency.

STP REV6 was released in July of 2008. The revision consisted primarily of data and groupings for planned maintenance and data variables for component failures and initiating events. The process for updating data variables describing component failures was augmented from that used in the past revisions. Consistent with previous revisions, the variables for which component failures occur were updated. Some selected, important variables for which no component failures had occurred were also updated.

Table 1-5			
STP_REV 6 CDF Groupings (events/year)			
Total CDF	Internal Events Contribution	External Events Contribution	
6.39E-06	3.89E-06	Fires	1.02E-06
		Floods	1.26E-08
		Flood MCR	2.90E-07
		High Winds (i.e. tornados)	1.11E-06
		Seismic	7.31E-08
		Total External	2.50E-06

NRC Requested Information:

- g. STP Nuclear Operating Company (STPNOC) risk managed technical specifications (RMTS) submittal of February 28, 2007, stated that there had been a follow on peer review of the human reliability analysis (HRA) of STP_REV5 PRA which had identified one Level A and nine Level B Facts and Observations (F&Os). Describe these F&Os, their resolution status, and the impact of their resolution on the SAMA analysis.**

STPNOC Response:

Level A F&O

HR-16

There is no evidence that an analyst reviewed the changes to the PRA model incorporated since the Individual Plant Examination (IPE) to decide if the current Human Failure Events (HFE) were adequate and that no new HFEs needed to be added.

Level B F&Os

HR-08

The methodology for analyzing the dependence of time-sensitive actions does not identify groups of actions that involve both dynamic actions modeled as part of the system initiator and subsequent dynamic actions in response to the initiator on the failure of actions considered in the system initiator.

HR-09

Table 7-1 (Post Initiator Human Error Probability (HEP) Summary) of the report does not identify some dynamic actions though they are used in the sequence model. Table 7-1 also does not identify which actions are dependent on others or under what conditions.

HR-10

In one case in the HFE model, the importance of the action to start the positive displacement pump without centrifugal charging pumps (HERA6) will not be correctly reflected in the basic event importance report.

HR-13

The scenarios defined for evaluation of risk significant events to trip the reactor coolant pumps (RCP) after loss of component cooling water (HERCP1) and the action to start the positive displacement pump and manually trip the RCPs (HERC6) appear inconsistent with the sequence models for which they are used.

HR-14

Some judgments made in the annunciator response model used for evaluation of the failure to place the standby Electrical Auxiliary Building (EAB) HVAC train into service (HEEAB1) are questionable.

HR-15

There is no obvious evidence (e.g. documentation) that a procedure requirement was performed to review model updates to ensure that any human errors that could be plant specific or industry specific are addressed.

HR-17

The step for verifying recovery actions for the failure to initiate residual heat removal cooling for a steam generator tube rupture (HEOC01) needs to be improved.

HR-18

The recovery execution for basic event modeling the failure to initiate bleed and feed cooling (HEOB02) appears to reference an incorrect procedure.

HR-19

The calculation used and the correct cognitive failures need to be identified for the failure to open doors for 2 of 3 EAB HVAC fan trains failed (HEOS01).

Resolution Status of the above F&Os

A formal resolution for these F&Os is not complete at this time. The F&O draft resolutions are expected to be complete by the end of 2011. Preliminary review of the F&Os has determined that they are not likely to have a significant impact on the STP PRA model. Based on this preliminary review of possible model impact, it is not expected that any resolution will impact the SAMA analysis.

NRC Requested Information:

- h. Table F.2-1 does not include any internal flood initiators. Discuss the modeling and disposition of internal flood events and their contribution to the CDF.**

STPNOC Response:

The internal flood initiators were screened during the Individual Plant Examination of External Events (IPEEE). STP is a relatively recent vintage plant with a high degree of separation between trains. This separation and the three train design is the reason that the flooding initiating events were able to be screened. A review of internal flood screening was performed in support of the Risk-Managed Technical Specification license amendment process and concluded that the previous IPEEE internal flood screening remains valid. This was accepted by the NRC in its original safety evaluation report (SER) (accession number 9201300172) for the STP PRA and in the SER (ML071780186) for Risk-Managed Technical Specifications (RMTS).

NRC Requested Information:

- i. Provide the contribution to CDF due to station blackout (SBO) and anticipated transients without scram (ATWS) events.**

STPNOC Response:

Group	CDF(events/year)	% Contribution
SBO	2.23E-06	35.0%
ATWS	2.75E-07	4.3%
Total CDF	6.39E-06	100%

NRC Requested Information:

- 2. Provide the following information relative to the Level 2 analysis:**

- a. Provide a summary description of the current Level 2 PRA including: (a) the Level 1 to Level 2 linking, the containment event trees, the binning of Level 2 sequences to the 15 end-states cited in ER Section F.3.6 and (b) the process used to assign the 15 end-states to the four major release categories.**

STPNOC Response:

A detailed description of the Level 2 PRA is provided in the following documents on file at STPEGS:

- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Analysis - Containment Event Tree" STP_REV6, August 18, 2009.
- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Accident Sequence Progression," STP_REV6, August 8, 2009.
- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Results" STP_REV6, September 14, 2010.
- STPNOC 2005 Level 2 PRA Update Reports (3 Final reports, Revision 0), ABSG Consulting Inc., "Level 2 Probabilistic Risk Assessment Update – 2005," "Containment Analysis Report - 2005," and "MAAP (Revision 4.05) Analysis Report," October 14, 2005.
- STPEGS Level 2 PSA and Individual Plant Examination, prepared by HL&P and PLG, Inc., August 1992.

The set of documents identified above were specifically developed to address peer review findings and to align the STPEGS Level 2 PRA with American Society of Mechanical Engineers (ASME) PRA Capability Category 2 (or higher) requirements.

The following excerpts from those documents address this RAI:

The entry to the containment event tree (CET) is characterized by the thermal-dynamic conditions in the reactor coolant system and containment at the time of severe core

damage, and the availability of both passive and active plant features that can terminate the accident or mitigate the release of radioactive materials.

The CET is different from the Level 1 event trees in that it deals primarily with severe accident phenomena rather than the success or failure of equipment and operator actions. This makes the progression of events and status of important physical parameters in the Level 2 analysis much more difficult to determine. The plant response to a severe accident consists of a sequence of complex interrelated physical phenomena that are not always adequately described by success or failure, but are sometimes a matter of degree or timing. The effects of these phenomena must be reduced to binary or multi-branch events in order to be evaluated by PRA methods.

Each accident sequence has a unique combination of top event successes and failures. Ideally, each accident sequence that results in core damage should be evaluated explicitly in terms of accident progression and the release of radioactive materials to the environment. However, because there can be millions of such sequences, it is impractical to perform such analyses for each one. Therefore the individual sequences must be categorized into bins by some set of parameters that define the group. Each bin collects all of those sequences for which the progression of core damage, the release of fission products from the fuel, the status of the containment and its systems, and the potential for mitigating source terms are similar. Detailed analysis is then focused on specific sequences selected to represent each of these bins.

All of the plant model information on the operability status of active systems that is important to the timing and magnitude of the release of radioactive materials must be passed into the CET. This requires that, in addition to representing the systems and functions that are important to core cooling, the Level 1 event trees also address active systems and functions important to containment isolation, containment heat removal, and the removal of radioactivity from the containment atmosphere. The containment spray system is a good example of such systems.

The concept of "plant damage states" used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," and discussed in the STP Individual Plant Examination (IPE) has been discarded. As far as RISKMAN[®] is concerned, the CET is another tree just like the ones in the Level 1 part of the analysis. Therefore the code passes the status of all the top events previously evaluated on to the CET as well as the values of all the macros. There is no need to define transitional states as the CET can directly determine the status of the plant through the application of split fraction rules.

The CET considers the influence of physical and chemical processes on the integrity of the containment and on the release of fission products once core damage has occurred. The considerations that influence the progression of core damage, the time and mode of containment failure, and the release of radioactive materials to the environment fall into two categories:

- the physical conditions in the reactor coolant system (RCS) and containment at the time of core damage, fission product release and vessel breach, and
- the status and availability of containment systems for mitigating fission product release and removing decay heat.

The considerations of physical conditions in the RCS and containment that are included in the CET are as follows:

- the pressure inside the reactor vessel at the onset of core damage and at vessel breach,
- the availability of cooling on the secondary side of the steam generators, and
- whether or not the reactor cavity is flooded at the time of vessel melt-through.

The pressure inside the RCS at the onset of core damage is an important parameter because it influences the pressure at the time of vessel breach. The pressure at vessel breach is also a strong function of the core degradation process. If the reactor vessel fails by rupture of the bottom head at high RCS pressure, core debris will be dispersed beyond the confines of the reactor cavity. The pressure at vessel breach depends on the core damage process and whether or not any natural or operator induced mechanisms reduce RCS pressure as the accident sequence progresses. If the pressure at vessel breach exceeds approximately 200 psia, there exists a potential for ejection of dispersed core debris into the containment atmosphere, thereby increasing the containment loading at the time of vessel failure. High RCS pressures (particularly at the system set point) can be conducive to significant natural circulation in the RCS.

While natural circulation has beneficial effects for most transients; in a severe accident, natural circulation can transport hot gases from the core into the steam generators inducing tube rupture and a possible direct pathway to the environment. The availability of steam generator secondary side cooling will determine whether or not the steam generator tubes will be subject to high temperatures and potential failure if combined with high RCS pressure.

The presence of water in the reactor cavity at the time of reactor vessel melt-through is important to containment response because the interaction of this water with hot core debris can:

- fragment and disperse the core debris from the reactor cavity into other regions of the containment,
- cause the containment pressure to increase by vaporization of the water (i.e., steam spikes) and direct heating of the containment atmosphere, and
- enhance the release of fission products from the core debris due to oxidation of the particulates.

Functional containment status considerations included in the CET are:

- The state of the containment itself (intact, bypassed or failed) at the time when severe core damage starts (i.e., when the CET is entered). This distinction not only includes containment isolation failure and bypass considerations, but is also of particular importance for external events that can cause containment failure prior to core damage (e.g., earthquakes, severe storms, or external missiles).
- The availability of containment engineered safety features (such as containment sprays and fan coolers) for cooling the containment atmosphere and fission product removal before and after failure of the reactor vessel.
- The potential availability of filtration and/or other mechanisms for fission product removal in the containment leakage path (such as auxiliary building filters for

interfacing systems loss-of-coolant accident (LOCA) or purge filters for sequences involving isolation failure) if the containment is failed at the time core damage is initiated.

Based on a review of the STP design and reference plant documents, the following specific items are considered for entry into the containment event tree:

RCS Pressure. The following four ranges of RCS pressure (P) at the time of core damage have been defined:

- $P < 200$ psia
- $200 \leq P < 600$ psia
- $600 \leq P < 2000$ psia
- $P \geq 2000$ psia (i.e., "pegged" at the system set point)

The primary system pressure is assumed to be a function of the following parameters that are determined in the Level 1 analysis.

- Availability of steam generator cooling.
- Type and size of LOCA.
- Position of pressurizer power-operated relief valves (PORV).
- Status of high head safety injection (HHSI) system.
- Success of RCS cooldown.

The assumed relationship between these parameters and RCS pressure at the time of core damage is shown in Table 2-1 on the following page.

Table 2-1					
Assignment of RCS Pressure at the Time of Core Damage for Sequences with Reactor Trip Successful					
Steam Generator Cooling Available	LOCA Size	Two Pressurizer PORVs Open	HHSI Available	Secondary Depressurization /RCS Cooldown	RCS Pressure at Time of Core Damage
Yes	RCP Seals only	N/A	Yes	Yes	Medium
Yes	RCP Seals only	N/A	Yes	No	High
Yes	RCP Seals only	N/A	No	Yes	High
Yes	RCP Seals only	N/A	No	No	High
Yes	Pressurizer PORV or Ruptured Steam Generator ⁽¹⁾	N/A	Yes	Yes	Medium
Yes	Pressurizer PORV or Ruptured Steam Generator ⁽¹⁾	N/A	Yes	No	High
Yes	Pressurizer PORV or Ruptured Steam Generator ⁽¹⁾	N/A	No	Yes	High
Yes	Pressurizer PORV or Ruptured Steam Generator ⁽¹⁾	N/A	No	No	High
No	None	No ⁽²⁾	N/A	No ⁽³⁾	System Setpoint ⁽⁴⁾
No	RCP Seals only	No ⁽²⁾	N/A	No ⁽³⁾	System Setpoint
No	Pressurizer PORV or Ruptured Steam Generator ⁽¹⁾	No ⁽²⁾	N/A	No ⁽³⁾	High
No	N/A	Yes ⁽⁵⁾	Yes ⁽⁵⁾	No ⁽³⁾	High

(1) The ruptured steam generator is not isolated so that leakage of reactor coolant through the secondary side to the environment occurs. HHSI is considered unavailable in the long term for such circumstances if it is initially successful, but makeup to the reactor water storage tank (RWST) is not provided.

(2) Bleed and feed cooling unsuccessful.

(3) No secondary depressurization possible if steam generator cooling unsuccessful.

(4) At or above the pressurizer PORV setpoint.

(5) Bleed and feed cooling successful.

Steam Generator Heat Removal. Auxiliary feedwater is providing secondary side heat removal with or without secondary pressure control and the ability to cool down.

Water in Cavity Prior to Vessel Breach. This parameter addresses whether or not the contents of the RWST have been injected into the containment to "spill" into the reactor cavity.

Containment Isolation and Bypass Status. The following five situations are considered:

- Containment isolated and not bypassed.
- Containment not isolated or failed prior to core damage; leak area less than the equivalent of 3 inches in diameter.
- Containment not isolated with a leak area greater than or equal to the equivalent of 3 inches in diameter. The specific event causing this is a failure of the supplementary containment purge valve to close.
- Small containment bypass (i.e. a steam generator tube rupture (SGTR) or letdown isolation failure).
- Large containment bypass. This is an interfacing LOCA through the residual heat removal (RHR) or low head safety injection (LHSI) system.

Containment Spray Operation. Three combinations of containment spray (injection and recirculation) have been considered:

- Containment Spray Injection (CSI) and Containment Spray Recirculation (CSR) are available.
- Only CSI is available.
- Neither CSI nor CSR is available.

Containment Heat Removal. Heat is removed from the containment via the RHR heat exchangers in conjunction with LHSI or via the containment fan coolers.

Initiating Events

In addition to the phenomena related "thresholds" discussed above, the classification of pressurized-water reactor (PWR) accident sequences into RCS pressure ranges at the onset of core damage has, in past PRA studies, been correlated to initiating event type, as shown in Table 2-2 below:

Table 2-2 Initiating Event Type	RCS Pressure
Transients without vessel breach	PORV Setpoint
Transients with vessel breach (e.g., Stuck open PORVs, RCP seal LOCA)	High
Small LOCAs	High
Medium LOCAs	Low
Large LOCAs	Low

The availability of steam generator secondary side cooling will determine whether or not the steam generator tubes will be subject to high temperatures and potential failure if combined with high RCS pressure.

To insure consistency between the Level 1 and Level 2 Probabilistic Safety Assessment (PSA), the same initiator definitions are used with the same lists of events trees. This is accomplished in RISKMAN[®] by creating a new set of initiators for Level 2 that are identical to the Level 1 initiators, except that the CET is added to each Level 2 initiator event tree list.

The concept of a single containment event tree for all initiators is a difficult one to implement in RISKMAN[®] because there are limitations in the RISKMAN[®] program that make the transition from the many Level 1 trees to the one CET difficult. The reason is that each accident initiator evaluates its own particular series of event trees, which means that a particular top event may be evaluated for one initiator, but not for another initiator that takes a different path. However RISKMAN[®] requires that in a tree all the referenced top events, even if not used for the current initiator, be previously evaluated. Therefore the CET cannot reference top events that are not universally common to all sequences without producing RISKMAN[®] execution errors.

The way this problem is handled is to construct "linking" macros, whose purpose is to convert top events statuses into macros, which can be universally defined. Each macro may have different definitions in different trees depending on which initiators use that tree. For initiators that do not need a particular macro, that macro can be set to a dummy value just to satisfy RISKMAN[®]. As an alternative, the linking tree may contain dummy top events just so they can be referenced. The linking macros in the STP PRA model are in the plant damage state or "PDS" trees. These are the last trees in the Level 1 analysis and contain only the Level 1 success criteria and the links to the CET.

In the PDS trees, the SUCC "success" macro defines those sequences which do not lead to core damage. When the Level 1 analysis is used alone, this macro is used in the core damage binning rule, and when the Level 2 is added, the macro is used in the first top event of the CET to filter out non-core damage sequences.

The CET considers the progression of severe accidents beginning with the onset of core damage and ending when a stable state is achieved. It addresses the events and physical processes that are important in determining the time, cause, and mode of containment failure, and the resultant release (via release categories) of radioactive fission products into the environment. The entry conditions for CET analyses are the Level 1 core damage sequences.

The STP CET addresses events occurring prior to vessel breach (including the potential for in-vessel recovery of the damaged core); the phenomena associated with both in-vessel and ex-vessel progression of the accident; containment integrity challenges; and the potential for containment failure. If containment failure does occur, the timing and mode (i.e., a small, controlled leak or a large break and the location of such failures) of failure are also addressed.

A detailed description of release category assignment to major release group is provided in the Containment Event Tree Notebook, and is summarized in Table 2-3 below.

Table 2-3. Release Category Assignment Matrix

RELEASE CATEGORY	RCS PRESSURE @ VB				SPRAYS I=INJECTION R=RECIRC			EX- VESSEL DEBRIS COOLING		CONTAINMENT FAILURE MODE						Major Release Group		
	HI	MED	LO	NO VB	I+R	I	NONE	YES	NO	PRE- EXIST	EARLY		LATE		BYPASS		None	
										SMA	LAR	SMA	LAR	SMA	LAR			SMA
R05LU	X-	-X				X-	-X		X			X	X				II	
R05SLU	X-	-X				X-	-X		X	X			X				II	
R06	X-	-X			X			X				X					II	
R06S	X-	-X			X			X		X							II	
R06L	X-	-X			X			X				X	X				II	
R06SL	X-	-X			X			X		X			X				II	
R06U	X-	-X			X				X			X					II	
R06SU	X-	-X			X				X	X							II	
R06LU	X-	-X			X				X			X	X				II	
R06SLU	X-	-X			X				X	X			X				II	
R07			X			X-	-X	X				X					II	
R07S			X			X-	-X	X		X							II	
R07L			X			X-	-X	X				X	X				II	
R07SL			X			X-	-X	X		X			X				II	
R07U			X			X-	-X		X			X-					II	
R07SU			X			X-	-X		X	X							II	
R07LU			X			X-	-X		X			X	X				II	
R07SLU			X			X-	-X		X	X			X				II	

Table 2-3. Release Category Assignment Matrix

RELEASE CATEGORY	RCS PRESSURE @ VB				SPRAYS I=INJECTION R=RECIRC			EX- VESSEL DEBRIS COOLING		CONTAINMENT FAILURE MODE						Major Release Group		
	HI	MED	LO	NO VB	I+R	I	NONE	YES	NO	PRE-EXIST	EARLY		LATE		BYPASS		None	
										SMA	LAR	SMA	LAR	SMA	LAR			SMA
R08			X		X			X				X						II
R08S			X		X			X		X								II
R08L			X		X			X			X	X						II
R08SL			X		X			X		X			X					II
R08U			X		X				X		X							II
R08SU			X		X				X	X								II
R08LU			X		X				X		X	X						II
R08SLU			X		X				X	X			X					II
BYNCV	X-	-X-	-X													X		III
R09	X-	-X					X	X					X					IIIA
R09U	X-	-X					X		X				X					IIIA
R10	X-	-X			X-	-X		X					X					IIIA
R10U	X-	-X			X-	-X			X				X					IIIA
R11			X				X	X					X					IIIA
R11U			X				X		X				X					IIIA
R12			X		X-	-X		X					X					IIIA
R12U			X		X-	-X			X				X					IIIA
R13	X-	-X				X-	-X	X						X				IIIA
R13U	X-	-X				X-	-X		X					X				IIIA

Table 2-3. Release Category Assignment Matrix

RELEASE CATEGORY	RCS PRESSURE @ VB				SPRAYS I=INJECTION R=RECIRC			EX-VESEL DEBRIS COOLING		CONTAINMENT FAILURE MODE						Major Release Group		
	HI	MED	LO	NO VB	I+R	I	NONE	YES	NO	PRE-EXIST	EARLY		LATE		BYPASS		None	
										SMA	LAR	SMA	LAR	SMA	LAR			SMA
R14	X-	-X			X			X						X				IIIA
R14U	X-	-X			X				X					X				IIIA
R15			X			X-	-X	X						X				IIIA
R15U			X			X-	-X		X					X				IIIA
R16			X		X			X						X				IIIA
R16U			X		X				X					X				IIIA
INTACT1	X-	-X-	-X														X	IV
INTACT2				X	X-	-X-	-X										X	IV

NOTES to Table 2-3:

The symbols -X, -X-, and X- appear in the columns for RCS pressure and Sprays. The RCS pressure columns appear in two ways:

1. X- adjacent to -X with the other column blank. In the RCS pressure columns this means the pressure is between the two corresponding classes. For example R14 is at high-medium pressure at vessel breach. In the containment spray columns, this means either column is sufficient.
2. If all three symbols appear, then the variable (pressure or spray) is irrelevant.

NOTES to Table 2-3 (Continued):

3. The last column shown in Table 2-3 is the major release group. The major release group, which has been used as a very informative high level plant risk characterization index, classifies sets or bins of large-early releases (Group I), small-early releases (Group II), late releases (Group III), and the benign case of no containment failure (Group IV). Of the major release categories, Group 1, the measure of which is large early release frequency (LERF), is the most important, because it has the greatest potential for adverse health effects. Large releases are those associated with a breach of containment equivalent to a 3-inch line or larger. This includes failure of the supplementary purge valve, a V-sequence break through the LHSI lines, and all energetic mechanisms such as failure coincident with vessel breach or hydrogen detonation. In addition, induced steam generator tube ruptures are assumed to be large releases because of the large potential driving head to pump out fission products provided by system pressure. As stated before, early releases are those that occur within 4 hours of vessel breach. All top events connected with containment failure up through CE are classified as early.

A large number of Modular Accident Analysis Program (MAAP) cases were run in support of the STP Level 2 PSA and IPE. Sensitivity studies were performed that show how the plant responds to different reactor coolant pump seal LOCAs, a spectrum of LOCA break sizes, and various differing core debris spread areas and numbers of operational reactor containment fan coolers. These cases are documented in Section 4.6 of the IPE report. Within the bounds of the assumptions made, these analyses are still valid. What has changed in recent code revisions starting with the STP_1997 model is primarily associated with the frequency and importance of the various accident sequences as predicted by RISKMAN[®], rather than differences in the accident progressions as determined by MAAP.

The probabilistic portion of the STP Level 2 analysis for model STP_REV6 is documented in the Containment Event Tree Notebook. In that analysis, the RISKMAN[®] code is used to quantify the reference PRA Level 2 model and generate reports showing the end state frequencies and most important sequences.

The accident progressions leading to some of the highest frequency end states are verified with thermal-hydraulic analyses presented in the Accident Sequence Progression Notebook. At the same time, radioactive release fractions are generated for these cases. The end states selected, along with their frequencies, are shown in Table 2-4 below. The selected end states represent greater than greater than 80 percent of the frequencies of major release groups I, II, and III respectively.

Table 2-4		
Representative End States Analyzed in MAAP		
End State	Group	End State Description
ISGTR	I	Induced Steam Generator Tube Rupture
R05SU	II	RCS High Pressure, Pre-Existing Leak, No Debris Cooling
CICV	II	Isolation Failure, Damage Arrested In-Vessel
R07SU	II	RCS Low Pressure, Pre-Existing Leak, No Debris Cooling
R15U	III	No Cooling, RCS at Low Pressure, Small Reactor Containment Building (RCB) Failure
BYPASS	III	Small Containment Bypass
R13U	III	No Cooling, RCS at High Pressure, Small RCB Failure
R11U	III	No Cooling, RCS at Low Pressure, Large RCB Failure

Accident sequences that lead to these end states were chosen for analysis. According to the probabilistic analysis, external events and support system failures dominate risk of release. Many of these initiators result in the loss of Class 1E power so that they are effectively station blackouts. Therefore all the sequences selected for analysis, except one, are initiated in MAAP with loss of AC power. They are different in the timing of power recovery, mode of vessel failure, and mode and timing of containment failure. The one non-blackout sequence is the small containment bypass, which is defined as either failure to isolate primary letdown or the spontaneous rupture of a steam generator tube.

The release fractions for all the cases are summarized in Table 2-5 on the following page. The timing and size of release fractions from MAAP can be used to verify that each end state is placed in the correct major release group. In the IPE, the "small" containment bypass category was classified as a small early release, Category II. However MAAP shows that if a steam generator tube rupture or letdown isolation failure is allowed to proceed to core damage, the releases for many of the species are just as big as those in the large category. However, the small bypass scenarios take many hours to develop, providing time to implement protective measures. Therefore the small bypass end state is reclassified as a late release, Category III.

**Table 2-5
 Severe Accident Radioactive Release Fractions**

Species/ Case	Noble	Iodide s	TeO2	SrO	MoO2	CsOH	BaO	Lanthin	CeO2	Sb	Te2	U & Trans
R05SU	0.50	0.0060	0.031	3.0E-4	6.7E-4	0.0056	4.0E-4	7.6E-5	2.5E-4	0.014	0.012	5.0E-7
BYPAS S	1.0	0.50	4.4E-4	5.4E-4	.018	0.41	0.0049	2.7E-5	3.0E-5	0.14	7.0E-6	2.5E-9
CICV	0.29	0.0023	0	1.8E-6	6.0E-5	0.0021	1.7E-5	2.4E-7	2.4E-7	4.5E-4	0	0
R07SU	0.53	0.026	0.023	5.3E-4	0.0023	0.027	0.0011	1.0E-4	5.3E-4	0.030	0.0023	1.6E-6
ISGTR	0.25	0.13	0.0030	0.0023	0.040	0.090	0.020	0.0065	0.0066	0.11	0.0068	3.3E-8
R15U	0.55	2.0E-4	0.0025	8.2E-5	5.2E-5	1.3E-4	8.2E-5	7.2E-5	7.4E-5	0.0054	0.0041	2.2E-8
R13U	0.50	4.5E-4	0.0025	4.3E-6	1.5E-5	8.0E-4	1.4E-5	5.0E-6	7.1E-6	0.0030	0.0025	2.1E-8
R11U	1.0	0.0075	0.016	8.5E-4	2.8E-4	0.0043	5.8E-4	6.0E-4	6.1E-4	0.060	0.014	1.2E-7

NRC Requested Information:

- b. Describe any internal and external reviews of the complete update of the Level 2 model incorporated in STP_REV5. Describe any unresolved F&Os from these reviews, their resolution status, and the impact of their resolution on the SAMA analysis.**

STPNOC Response:

The Level 2 PRA update incorporated in STP_REV5 is documented in three reports on file at STPEGS:

- STPNOC 2005 Level 2 PRA Update Reports (3 Final reports, Revision 0),
- ABSG Consulting Inc., "Level 2 Probabilistic Risk Assessment Update – 2005," "Containment Analysis Report," and
- "MAAP (Revision 4.05) Analysis Report," October 14, 2005.

These reports were reviewed by , ABSG Consulting Inc., prior to submittal to STPNOC; then they were reviewed and approved by STPNOC, and applied in the STP_REV5 PRA Level 2 Analysis. The STPEGS STP_REV5 PRA, including the updated Level 2 Analysis, and subsequent STPEGS STP_REV6 PRA applied for SAMA analyses were reviewed and approved by STPNOC. Specifically, the review and approval of STPEGS PRA Level 2 Analyses is documented via the following three PRA notebooks:

- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Analysis - Containment Event Tree" STP_REV6, August 18, 2009.
- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Accident Sequence Progression," STP_REV6, August 8, 2009.
- STPEGS Probabilistic Risk Assessment Notebook, "Level 2 Results" STP_REV6, September 14, 2010.

It is important to note that the reports referenced above were developed to address then-recent industry/regulatory issues and advances in Level 2 PRA technology and to resolve Westinghouse Owners Group (WOG) Peer Certification containment performance technical elements that were graded less than 3 or contingent grade 3, and any associated Fact/Observations (F&O) with levels of significance of A or B. The specific F&O comments addressed are documented in the STP reports. The F&Os were resolved and incorporated into the STP_REV6 PRA model used for SAMA analysis. A summary of issues addressed is provided in the following table.

Table 2-6 Summary of F&Os		
Issue Number	WOG Peer Certification F&O Identifier	Issue Description
1	L2-01, Element 5, Sub-element 5	Early containment failure
2	L2-02, Element 2, Sub-element 5	Thermally induced steam generator tube rupture (SGTR)
3	L2-04, Element 2, Sub-element 5	Level 2 success criteria
4	L2-05, Element 2, Sub-element 11	Pressurizer PORV and fan cooler survivability
5	L2-06, Element 2, Sub-element 21	Pre-existing containment leakage
6	L2-06, Element 2, Sub-element 21	Assignment of spontaneous SGTR core damage frequency (CDF) sequences to the "late containment failure" category
7	L2-07, Element 2, Sub-element 23	Emergency action levels (EALs) not included in the evacuation model

NRC Requested Information:

- c. Provide a brief history of the Level 2 PRA and the major changes in modeling that impact the release category frequencies.**

STPNOC Response : The following summary table represents the changes in the containment response over the last several approved PRA models:

Table 2-7 Summary of Changes in Containment Response				
Model	Large Early Release	Small Early Release	Late Containment Failure	Containment Intact
IPE (1992)	9.89E-07	6.67E-06	1.08E-05	2.56E-05
STP_1996	1.37E-07	2.93E-06	8.28E-07	4.99E-06
STP_1997	6.20E-07	2.05E-06	2.40E-06	6.42E-06
STP_1999	5.76E-07	2.14E-06	3.22E-06	5.39E-06
STP_REV4	5.37E-07	1.34E-06	2.46E-06	4.71E-06
STP_RV42	5.12E-07	1.44E-06	2.18E-06	4.88E-06
STP_REV5	6.06E-07	1.90E-06	2.61E-06	4.98E-06
STP_REV6	5.01E-07	1.16E-06	1.48E-06	3.10E-06

These results are different from the Internal Plant Examination (IPE) due to several changes in the model, as documented in the STPEGS PRA Level 2 Analysis Notebooks for each PRA revision. The IPE showed the largest contribution to Large Early Release Frequency (LERF) to be the failure of the containment supplementary purge valve to close on demand. It has been found that containment purge is not in service near as much as previously assumed, and that the database variable used for the air-operated dampers was "valve failure to close", rather than the more appropriate "failure to return to fail-safe position". Correcting these parameters greatly reduced the large early isolation failure contribution.

A NUREG 1570, *Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture*, approach was used in previous models to evaluate ISGTR, which increased its importance. The NUREG 1570 approach was re-evaluated for STP_REV5, and is no longer considered "overly conservative."

The current results show LERF to be dominated by ISGTR (Induced Steam Generator Tube Rupture) and VSEQ (Interfacing systems LOCA (ISLOCA)). The path of concern for ISLOCA is through the RHR suction motor-operated valves (MOV), rupturing the RHR/component cooling water (CCW) heat exchanger tubes, exposing CCW to high pressure. Credit for isolation of the CCW containment isolation valves was removed in STP_REV6 because the operator action is not proceduralized and there is significant uncertainty regarding MOV isolation capability during the ensuing flow and pressure transient caused by the ruptured RHR tubes. An additional initiator with a large contribution to LERF, the loss of essential cooling water (ECW) from a tornado event (HWIND2), was not included in the IPE analysis, however was subsequently added to the PRA model.

NRC Requested Information:

- d. Identify the version of Modular Accident Analysis Program (MAAP) used to determine the release fractions.**

STPNOC Response:

MAAP Version 4.05 was used.

NRC Requested Information:

- e. ER Section F.3.6 describes the selection of the representative accident sequence/source terms for the major release categories. The one example discussed was that an accident sequence with a moderate frequency and severe release characteristics would be selected over an accident sequence with a relatively high frequency and a minor radionuclide release. From the information provided, none of the selected representative sequences (for those categories where multiple source term results are provided) follow this conservative example. For major Release Categories II and III, the selected sequences are not those with the most severe release characteristics. While the information provided in ER Table F.3-8 indicates that the representative sequences are appropriate for the base case, this is not necessarily true for a SAMA case where the Level 2 end-state distribution would be different from the base case. For**

example, if a SAMA primarily impacted sequences which have low reactor pressure vessel (RPV) failure pressure then the frequency of end-states R07SU and R11U would be reduced. Since these end-states have higher release fractions (and most likely, higher dose-risk and offsite economic cost risk per event) than the representative sequence chosen, the benefit could be larger than that assessed using the representative release fractions.

If the source term chosen for a release category is not the most severe of the significantly contributing end-states, the benefit could be underestimated for any SAMA which primarily impacts an end-state with a higher release fraction. For example, SAMA 4 impacts only the end-state VSEQ (interfacing system loss of coolant accident (LOCA)) portion of Release Category I. It is not clear that end-state VSEQ has a less severe release than the ISGTR end-state, which was chosen as representative for Release Category I. Release fractions for Inter-System LOCA (ISLOCA) are usually greater than that given for ISGTR. The STP IPE (Table 4.8.3-4) gives interfacing system Cs and I release fractions from 0.15 to 0.4 depending on the methodology. Similarly for SAMA 10, which impacts steam generator tube rupture (SGTR) sequences, the removal of these sequences from Release Category III will have a more significant impact since the release fractions for SGTR are three orders of magnitude greater than those for the representative sequence.

Provide further support for the selection of the representative sequences and their adequacy for the SAMA analysis.

STPNOC Response:

The intent of using a "representative source term" is to characterize the average nature of a major release category's numerous contributing sequences with a single term. The use of representative source terms can result in overestimating the impact of implementing some SAMAs while underestimating it for others; however, this simplified quantification process has previously been accepted in License Renewal applications.

The concern stated in this RAI is that STP's representative source terms are non-conservative and that they underestimated the impact of implementing some of the Phase II SAMAs. The implication is that some of the SAMAs that were determined not to be cost beneficial in the ER could have positive net values if the higher consequence sequences were mapped to their corresponding source terms rather than to a representative source term. In order to demonstrate that the representative source terms defined in the Environmental Report (ER) do not impact the conclusions of the analysis, a sensitivity case has been developed that shows none of the SAMAs would be classified as cost beneficial even when the most conservative source terms are assigned to each major Release Category.

Because the ER validated the use of the representative source terms for the baseline model, no changes are proposed to the SAMA identification process or Phase I screening analysis (no change to the MACR). The focus of the sensitivity analysis is on the Phase II

quantifications in which the changes to the sequence frequencies may not be properly mapped to dose and off-site economic consequences.

The approach for this sensitivity analysis is to select the most conservative, relevant available source term for each major Release Category and to use the results to update the Phase II quantifications. The following Table 2-8 provides a summary of the conditional dose and off-site economic cost values for each of the source terms that were analyzed for STP. These values, which were taken from the STP Level 3 model, can be obtained from Table F.3-8 of the ER by dividing the dose-risk and off-site economic cost-risk values by the corresponding frequencies (there will be slight differences given that the results in Table F.3-8 were rounded for presentation).

Table 2-8									
Summary of Conditional Doses and Off-site Economic Costs for Available STP Source Terms									
Release Category	Group I (ISGTR)	Group II (R05SU)	Group II (CICV)	Group II (R07SU)	Group III (R15U)	Group III (R13U)	Group III (R11U)	Group III (Bypass)	Group IV (Intact)
Dose (person-rem)	1.36E+06	5.12E+05	2.12E+05	7.50E+05	1.49E+05	2.85E+05	4.25E+05	2.22E+06	1.70E+04
Off-site Economic Cost (\$)	2.40E+09	3.44E+08	8.60E+07	1.07E+09	7.14E+06	1.54E+07	4.02E+08	2.81E+09	4.68E+04

For Group I, there is only one analyzed source term, but the ISLOCA sequence, which is part of Group I, is a "bypass" scenario. Group III includes a "bypass" source term, which has a larger conditional dose and off-site economic cost than the ISGTR source term from Group I. As a result, the Group III "bypass" source term is used for the Group I source term.

For Group II, source term R07SU has the largest conditional dose and offsite economic cost values of the small-early releases, so it has been assigned as the Group II source term.

For Group III, the "bypass" source term has the largest conditional dose and offsite economic cost values of the late releases, so it has been assigned as the Group III source term.

No alternate source terms are available for Group IV and no changes have been made to the source term for this major Release Category.

The same PRA results documented in the ER were used in conjunction with the source terms above to obtain the updated dose-risk and off-site economic cost-risk (OECR) values for each SAMA. Tables 2-9 and 2-10 summarize these results (note: the base case has also been updated to reflect the revised source terms).

Table 2-9 Revised Dose-Risk Results					
Population dose-risk, 0-50 mile (risk in person-rem/yr)	Group I (Bypass)	Group II (R07SU)	Group III (Bypass)	Group IV (Intact)	Total Dose-Risk
Baseline	1.11	0.87	3.29	0.05	5.32
SAMA 3b	1.11	0.85	3.26	0.05	5.27
SAMA 4	0.83	0.87	3.29	0.05	5.04
SAMA 10	1.10	0.88	2.99	0.05	5.02
SAMA 12	1.11	0.87	3.29	0.05	5.32
SAMA 13	1.11	0.87	3.26	0.05	5.29
SAMA 15	1.10	0.86	3.22	0.05	5.23

Table 2-10 Revised OECR Results					
Total economic cost-risk, 0-50 miles (risk in \$/yr)	Group I (Bypass)	Group II (R07SU)	Group III (Bypass)	Group IV (Intact)	Total OECR
Baseline	\$1,408	\$1,241	\$4,159	\$0	\$6,808
SAMA 3b	\$1,408	\$1,209	\$4,131	\$0	\$6,748
SAMA 4	\$1,057	\$1,241	\$4,159	\$0	\$6,457
SAMA 10	\$1,387	\$1,249	\$3,779	\$0	\$6,415
SAMA 12	\$1,401	\$1,241	\$4,159	\$0	\$6,801
SAMA 13	\$1,405	\$1,241	\$4,131	\$0	\$6,777
SAMA 15	\$1,394	\$1,220	\$4,075	\$0	\$6,689

The cost benefit analysis was updated using the above information and net values were recalculated for each SAMA. The tables below summarize the results for the baseline PRA results as well as for the 95th percentile PRA results.

Table 2-11			
Cost Benefit Results Using Revised Source Terms and Baseline PRA Results			
SAMA ID	Cost of Implementation	Total Averted Cost-Risk	Net Value
SAMA 3b	\$796,677	\$6,518	-\$790,159
SAMA 4	\$100,000	\$34,786	-\$65,214
SAMA 10	\$100,000	\$29,868	-\$70,132
SAMA 12	\$100,000	\$212	-\$99,788
SAMA 13	\$100,000	\$3,874	-\$96,126
SAMA 15	\$100,000	\$14,106	-\$85,894

Table 2-12			
Cost Benefit Results Using Revised Source Terms and 95th Percentile PRA Results			
SAMA ID	Cost of Implementation	Total Averted Cost-Risk	Net Value
SAMA 3b	\$796,677	\$10,404	-\$786,273
SAMA 4	\$100,000	\$55,527	-\$44,473
SAMA 10	\$100,000	\$47,677	-\$52,323
SAMA 12	\$100,000	\$338	-\$99,662
SAMA 13	\$100,000	\$6,184	-\$93,816
SAMA 15	\$100,000	\$22,517	-\$77,483

In some cases, the percent change in the averted cost-risk was substantial, but because the absolute change in the risk was low, the averted cost-risk values remained low relative to the costs of implementation and none of the SAMAs are cost beneficial, which is consistent with the conclusions stated in the ER.

NRC Requested Information:

3. Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

a. Provide a description of fire scenarios X, B, 18, BC and 23 as included ER Table F.2-1.

STPNOC Response

The fire scenarios included in ER Table F.2-1 are described below in the following order: 18, 23, X, B, and BC. Details of the initiating event frequency quantifications can be found in Section 3.4.2 of the IPE report:

- South Texas Project Electric Generating Station, "Level 2 Probabilistic Safety Assessment and Individual Plant Examination," prepared by HL&P and PLG, Inc., August 1992.

The supporting Spatial Interactions Analyses are documented in Section 3.4 of the IPE report.

FR18 – Control Room Fire (Scenario 18)

In this scenario, EAB and control room envelope (CRE) ventilation trains fail due to fire. The controls for all three trains are located on control room panel 22/4. The following assumptions are made:

- Failure of either supply fan or return fan of an HVAC loop fails that loop.
- Due to the close proximity of the CRE and the EAB circuits for each loop, it is assumed that the CRE and EAB ventilation trains fail together.
- Air supply, cleanup, and makeup equipment are not considered essential for the success of the ventilation system.
- The frequency of a hot short that spuriously closes a damper is considered to be much less than the frequency of any of the fans failing.

In Scenario 18, trains A, B, and C of both the EAB and CRE HVAC systems fail because of control room fire.

FR23 – Control Room Fire (Scenario 23)

The impact of this control room fire is failure of all auxiliary feedwater (AFW) trains.

Scenario 23 involves a fire on control room panel 6 that causes failure of four trains of steam generator controls.

For this scenario to occur, either the AFW isolation valve or the AFW pump control circuits must suffer an open circuit because of the fire in all four trains.

Z047X –Scenario X

This initiating event represents a fire in Cable Spreading Room Train B. Scenario impacts include AC Power Train B and C, direct current (DC) Power Trains B and C, Fan Coolers Train A, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A, MSIVs, the centrifugal charging pumps (CCP) and positive displacement pump (PDP), and CCW supply to the RCPs.

Z071X –Scenario X

This initiating event represents a fire in the Auxiliary Shutdown Area. Scenario impacts include AC Power Trains A, B and C, AFW Train D, Containment Isolation Trains A and C, and the PDP.

Z047B –Scenario B

This initiating event represents a fire in Cable Spreading Room Train B. Scenario impacts include AC Power Train B, DC Power train B, Fan Coolers train A, Recirculation Cooling Train A, RCP Seal Injection and PORV 656A.

Z47BC –Scenario BC

This initiating event represents a fire in Cable Spreading Room Train B. Scenario impacts include AC Power Train B and C, DC Power Trains B and C, Fan Coolers Train A, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A and the CCPs and PDP.

NRC Requested Information:

- b. In the May 9, 2007, STPNOC response to requests for additional information (RAIs) for RMTS, it was stated that a review of the fire frequency data presented in NUREG/CR-6850 was planned for a future reanalysis of fire hazards at STP. If the results of this review have not been incorporated in STP_REV6, assess the impact the fire frequency data on the SAMA assessment.**

STPNOC Response:

A review of the fire frequency data presented in NUREG/CR-6850 has not been incorporated in STP_REV6. Future updates to the Fire PRA and External Events PRA will be performed in accordance with the STP Risk Management Strategic Plan. STP intends to review the impact of NUREG/CR-6850 fire frequency data for incorporation into the STPEGS PRA in 2013.

NRC Requested Information:

- c. Identify the seismic hazard curves used to determine the seismic CDF in STP_REV6. If the seismic CDF is based on the Electric Power Research Institute (EPRI) hazard curve, provide the seismic CDF using the Lawrence Livermore National Laboratory (LLNL) hazard curve or the more recent USGS 2008 assessment and include a description of the dominant seismic CDF sequences. Discuss the impact of these results on the SAMA assessment.**

STPNOC Response

The seismic hazard curves used to determine the seismic CDF in STP_REV6 are the EPRI hazard curves. This same set of curves has been used since the original Internal Plant Examination of External Events (IPEEE). The NRC determined the use of these curves to be acceptable in approving the Risk Managed Technical Specifications for STP.

The NRC published Information Notice 2010-18 to inform Licensees about a August 2010 NRC document, "Safety/Risk Assessment Results for Generic Issue (GI) 199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," (ML100270582) The referenced NRC document stated that based on the assessment of the seismic performance of existing plants (using the seismic hazard information available at the time of the IPEEE), the NRC staff determined that the seismic designs of operating plants in the central and eastern United States still provide adequate safety margins. The regulatory assessment of GI 199 is ongoing including the proposal of a NRC Generic Letter. STP is monitoring the results of this assessment to determine the need to update their seismic CDF based on updated hazard curves. It should be noted that STP Units 1 and 2 reside in an area of very low seismic activity.

NRC Requested Information:

- d. Provide a brief description of the latest fire and other external events models incorporated in the STP PRA.**

STPNOC Response:

STP_REV5 was reviewed by the NRC for issuance of the STP SER (ML071780186) in support of Risk Managed Technical Specifications. STP_REV6 external events are unchanged from the NRC reviewed model.

An important class of common cause initiating events is due to *external events* that include events external to the plant and events such as fires, which can occur inside the plant but are external to the plant processes. External events analysis is a major task in the development of a risk model and separate sections have been reserved for its documentation (Individual Plant Examination (IPE) report, Section 3.4). As an integral part of this analysis, physical interactions that cause one or more initiating events and possible

additional damage to one or more plant systems are identified. The analysis of these interactions is specific to the types of external events considered, which include:

- Seismic Events
- Fires and Explosions
- Internal and External Flooding
- Aircraft Crash
- Wind and Wind-Generated Missiles
- Turbine Missile
- Hazardous Chemical Releases
- Transportation Accidents

A list of common cause initiating events for the above sources of physical interaction is identified in Section 3.4 of the IPE report – South Texas Project Electric Generating Station, “Level 2 Probabilistic Safety Assessment and Individual Plant Examination,” prepared by HL&P and PLG, Inc., August 1992.

The 19 external event categories that were selected for quantification in the PRA are listed in Table 3-1. The term *category* is used in the sense that many specific events can be identified for each category by breaking down the category into cause, failure mode, degree of severity, etc. The distinctions made by specifying different categories are those necessary to account for the influence of the initiating events on the development or unfolding of the accident sequences in the event sequence model and to isolate key factors of importance in quantifying accident sequence frequencies and damage levels. For this reason, discrete seismic hazard intensity levels, distinct fire locations and magnitudes, and distinct flood scenarios are counted as separate initiating event categories..

Each class of seismic events was quantified for point estimate results at different, discrete levels of earthquake ground acceleration. The particular ground acceleration levels were selected on the basis of the seismicity and fragility curves. This is discussed more fully in Section 3.4.4 of the IPE report.

The different fire cases consist of different degrees of damage done by fires in the control room area (see Section 3.4.2 of the IPE) and the effects of fire in the Train B cable spreading room and the auxiliary shutdown panel areas (see the STPEGS Fire PRA Update Report). The Fire PRA Update Report is ST-RL-HL-0563, “PLG-1015 Fire Analysis Update for the South Texas Project Electric Generating Station PSA for Selected Zones,” prepared by PLG, Inc., December 1994.

The external flooding events, described in IPE report Section 3.4.6, consist of the flood and flood-induced failure of equipment in different areas due to the flow of flood water across these areas.

The final external event category consists of severe winds (tornados) that disable the offsite grid (345kV and 138kV) and the Technical Support Center Diesel Generator (TSCDG), and dump sufficient debris in the vicinity of the essential cooling water (ECW) intake structure to cause plugging of the ECW pump traveling screens. This analysis is also presented in IPE report Section 3.4.7.

Table 3-1 below presents the external events, including fire scenarios, explicitly modeled in the current STPEGS PRA, STP_REV6.

Table 3-1		
External Event Categories Selected for Quantification in the Current South Texas Project Risk Model (STP_REV6)		
Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Seismic Initiating Events	0.1g Seismic Event	SEIS1
	0.2g Seismic Event	SEIS2
	0.4g Seismic Event	SEIS3
	0.6g Seismic Event	SEIS4
Plant Fires	Control Room - Loss of All Three Motor-Driven AFW Pumps	FR10
	Control Room - Loss of CR HVAC and EAB HVAC	FR18
	Control Room - Loss of All AFW Trains	FR23
	Zone 31Z047 - Cable Spreading Room Train B, Scenario B - Affects Train B AC and DC, RCFC A, Recirculation Cooling Train A, RCP Seal Injection, and PORV 656A	IZ047B
	Zone 31Z047 - Cable Spreading Room Train B, Scenario BC - Affects Train B and Train C AC and DC, RCFC A, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A, and the CCPs and PDP	IZ47BC
	Zone 31Z047 - Cable Spreading Room Train B Scenario X - Affects Train B and Train C AC and DC, RCFC a, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A, MSIVs, CCPs, PDP, and RCP CCW Supply	IZ047X
	Zone 07Z071 - Auxiliary Shutdown Area, Scenario X - Affects Train A, Train B, and Train C AC Power, AFW Train D, Containment Isolation (CI) Trains A and C, and the PDP	IZ071X
	Zone 03Z147 – MAB 41' Corridor and Changing Scenario O - Affects CCW A, CCW B, CCW C, LHSI A, HHSI A, CS A, CI Train A, ECH C, CCP A, and SI Recirculation Cooling Train A	IZ147O

Table 3-1		
External Event Categories Selected for Quantification in the Current South Texas Project Risk Model (STP_REV6)		
Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Plant Flooding - Breach of the Main Cooling Reservoir	LOOP, Technical Support Center DG (TSCDG) leads to loss of PDP, Balance of Plant Diesel Generator (BOPDG)	FL1
	LOOP, PDP and all Three Emergency Diesel Generators	FL26 (4 item included in Initiator FL26)
	LOOP, TSCDG, PDP, BOPDG, All CCW and One Train (B) of Essential Chilled Water Chillers	
	LOOP, TSCDG, PDP, BOPDG, and Loss of ECW Intake Structure	
	LOOSP, TSCDG, PDP, BOPDG, and One Train (B) of RCFCs	
	LOOP, TSCDG, PDP, BOPDG, and Plugging of the ECW Pump Traveling Screen by Debris	FLECW
Other External Events	Severe Wind (Tornado) - LOOP, TSCDG, BOPDG, and Plugging of the ECW Pump Traveling Screen by Debris	HWIND

NRC Requested Information:

4. Provide the following information relative to the Level 3 PRA analysis:

- a. ER Section F.3.2 states that two previously identified sector population, land fraction, and economic estimation program (SECPOP) errors were corrected in the SAMA analysis. Clarify whether a third known error for incorrect column formatting of the output file was also corrected.**

STPNOC Response:

The column formatting of the output file, referred to in this RAI, involved changing the regional economic data format to comply with MACCS2 input requirements, which has been completed.

NRC Requested Information:

- b. Provide a table of the sector population breakdown for the SECPOP rosette for the year 2000 and the projected rosette for the year 2050. ER Section 2.6.1 identifies a total population of 255,118 within the 50-mile radius. However, no population is provided for the year 2050. Provide the total population used for the year 2050 (and the year 2000 if different than ER Section 2.6.1, including explanation for the reason for the difference).**

STPNOC Response:

Tables 4.b.1i and 4.b.1ii show the year 2000 population breakdown by distance and direction (SECPop rosette) to 10 and 50 miles from the site. Tables 4.b.2i and 4.b.2ii show the corresponding 2050 population breakdown.

Table 4.b.1i YEAR 2000 POPULATION BY DISTANCE (TO 10 MILES) AND DIRECTION FROM STP

Distance (miles):	0-1	1-2	2-3	3-4	4-5	5-10
Direction						
N	0	0	15	0	0	32
NNE	0	0	0	0	498	168
NE	0	0	0	0	31	99
ENE	0	0	0	0	0	420
E	16	0	0	0	3	217
ESE	0	0	0	71	116	70
SE	0	0	0	3	100	1212
SSE	0	0	0	0	0	108
S	0	0	0	0	0	0
SSW	0	0	0	0	0	0
SW	0	0	1	0	0	76
WSW	0	0	0	4	6	116
W	0	0	0	5	0	114
WNW	0	0	0	0	4	2285
NW	0	0	0	19	30	219
NNW	0	0	0	0	0	34

Table 4.b.1ii YEAR 2000 POPULATION BY DISTANCE (TO 50 MILES) AND DIRECTION FROM STP

Distance (miles):	0-10	10-20	20-30	30-40	40-50	Total
Direction						
N	47	1237	536	14097	5445	21362
NNE	666	21441	1120	2540	10968	36735
NE	130	931	6687	11447	24758	43953
ENE	420	271	2480	16635	62994	82800
E	236	83	1243	87	46	1695
ESE	257	2	0	0	0	259
SE	1315	13	0	0	0	1328
SSE	108	117	0	0	0	225
S	0	0	0	0	0	0
SSW	0	1	0	0	0	1
SW	77	345	0	1111	628	2161
WSW	126	5671	1074	14758	3240	24869
W	119	261	829	1302	3614	6125
WNW	2289	1181	492	9669	1259	14890
NW	268	477	787	1455	222	3209
NNW	34	484	4469	11928	2211	19126

**Table 4.b.2i YEAR 2050 PROJECTED POPULATION BY DISTANCE (TO 10 MILES)
 AND DIRECTION FROM STP**

Distance (miles):	0-1	1-2	2-3	3-4	4-5	5-10
Direction						
N	0	0	20	0	0	44
NNE	0	0	0	0	677	228
NE	0	0	0	0	42	135
ENE	0	0	0	0	0	571
E	22	0	0	0	4	295
ESE	0	0	0	97	158	95
SE	0	0	0	4	136	1648
SSE	0	0	0	0	0	147
S	0	0	0	0	0	0
SSW	0	0	0	0	0	0
SW	0	0	1	0	0	103
WSW	0	0	0	5	8	158
W	0	0	0	7	0	155
WNW	0	0	0	0	5	3108
NW	0	0	0	26	41	298
NNW	0	0	0	0	0	46

**Table 4.b.2ii YEAR 2050 PROJECTED POPULATION BY DISTANCE (TO 50 MILES)
 AND DIRECTION FROM STP**

Distance (miles):	0-10	10-20	20-30	30-40	40-50	Total
Direction						
N	64	1681	706	19276	10482	32209
NNE	905	29160	1677	5277	29545	66564
NE	177	1266	12458	23466	51565	88932
ENE	571	369	4164	34102	129138	168344
E	321	113	1728	174	94	2430
ESE	350	3	0	0	0	353
SE	1788	18	0	0	0	1806
SSE	147	159	0	0	0	306
S	0	0	0	0	0	0
SSW	0	1	0	0	0	1
SW	104	469	0	1522	860	2955
WSW	171	7624	1446	20212	4503	33956
W	162	348	1078	1729	5512	8829
WNW	3113	1583	640	12570	1654	19560
NW	365	644	1030	1903	272	4214
NNW	46	653	5854	15626	2780	24959

The total 50-mile populations for years 2000 and 2050 are 258,738 and 455,418.

The total year 2000 population from Tables 4.b.1.ii is 1.4% greater than that in ER Section 2.6.1. This small difference is because the population in ER Section 2.6.1 was taken from SECPOP's data base location of the existing units. As noted in ER Section F.3.1 for the SAMA, the population distribution projections were taken from an analogous study for the potential construction and operation of two new units at the STP site. The small difference in populations resulting from the small difference in STP Units 1 and 2 versus planned STP Units 3 and 4 plant coordinates was not deemed to warrant reworking the complex population projection calculations.

NRC Requested Information:

- c. The Houston-Sugar Land-Baytown metropolitan area is just outside the STP 50-mile radius. However, the projected growth through year 2050 is expected to be high. Briefly explain how/whether the population studies addressed the potential for a step change in population within the 50-mile radius if/whether this metropolitan area expands to the southwest.**

STPNOC Response:

The population projections were based on projections developed by the individual counties. The multiplier developed for each sector was based on county-weighted population projections. Thus, those sectors near the Houston-Sugar Land-Baytown Metropolitan Statistical Area (MSA) would have already included any effect expected by local demographers from the high growth rate expected for this MSA.

NRC Requested Information:

- d. The evacuation study was performed for the year 2007. Provide the year 2007 transient and total population used in the study.**

STPNOC Response:

The 2007 evacuation time estimate was used only for calculation of the evacuation time parameter in MACCS2. As such, the 2007 populations were not relevant to the projected 2050 population. The total year 2007 population used to project the year 2007 evacuation speed to 2050 was based on the exponential growth rate from 2000 to 2050 of 0-10 mile population (the distribution of that population is shown in the response to RAI 4.b). The resulting 2007 population used in the SAMA study was 6,360. The transient component of this total 2007 0-10 mile population was not separately calculated. ER Table F.7.2-1 demonstrates that the baseline risk is insensitive to the evacuation speed (and thus the choice of 2007 population).

5. Provide the following information with regard to the selection and screening of Phase I SAMA candidates:

- a. In ER Section F.5.1.3.1, Wolf Creek SAMA 13, which provides for a gravity feed fuel oil supply, was screened from further consideration at STP based on an existing STP capability which requires a pump. The use of a pump has less capability than a gravity system. Provide further justification for the screening of this SAMA.**

STPNOC Response:

The current STP fuel oil transfer system already uses a gravity feed line between the fuel oil storage tank (FOST) and the standby diesel generator (SBDG). Each SBDG has a dedicated FOST with a seven-day fuel oil supply. The FOST fuel inventory exceeds the PRA mission time and would support running the diesels for even the most protracted loss of offsite power (LOOP) events at STP.

In the event that the inventory is low in any of the six FOSTs, fuel oil makeup can be provided from either the auxiliary fuel oil storage tank (AFOST), the truck fill line, or the emergency fill connection. Each unit has an auxiliary fuel oil filtration skid that can pump makeup fuel to the FOST from either the AFOST or the truck fill line. Because the skid is powered by a pump, it can take suction from multiple fuel oil sources rather than being limited to one that is located at a higher elevation. In this sense, the existing FOST makeup configuration is considered to provide more capability than an additional gravity feed line. In addition, a new gravity feed line for FOST makeup would require a new tank at STP because the AFOST is at a lower elevation than the FOSTs.

The normal gravity feed configuration of the fuel oil system provides a reliable, passive method of providing fuel from the FOSTs to the SBDGs. The capability to refill the FOSTs from multiple sources exists, if necessary, but the capacity of the tanks limits the need to do so. Providing an additional gravity feed fuel line to either the FOSTs or directly to the SBDGs to supplement the existing gravity feed lines would have very limited benefit. Finally, any potential benefit would be extremely difficult to measure for STP given that the fuel oil transfer system is within the SBDG component boundary and changes to the fuel oil transfer configuration would not be directly represented in the PRA model.

Based on the reasons presented above, Wolf Creek SAMA 13 is not required for STP and it has been screened from further analysis.

NRC Requested Information:

- b. SAMA 16 involves using a portable engine driven instrument air compressor. This SAMA was based on Prairie Island SAMA 22 which utilized nitrogen bottles rather than a portable compressor. STP SAMA 16 has an estimated implementation cost of \$1.2M while Prairie Island SAMA 22 had an estimated implementation cost of \$78K (\$39K per unit). The cost of nitrogen bottles appears to be considerably less than that of an air compressor. Consider a SAMA that utilizes nitrogen bottles to provide an alternate air source.**

STPNOC Response:

For STP, the Instrument Air system is modeled in the PRA, but loss of Instrument Air was not identified as a significant contributor, which is supported by the diversity of the system and the ability to power compressor 14 from the BOP diesel. The implication is that any SAMAs designed to improve Instrument Air would have an averted cost-risk below that corresponding to the review threshold of \$11,000 for the site.

In response to this question, the full importance list was reviewed for STP and there was only one Instrument Air split fraction with a Risk Reduction Worth value greater than 1.000, which was IAS14 at 1.016. That corresponds to an averted cost-risk of about \$8,100 for the site, which is about an order of magnitude below the Prairie Island cost estimate for using nitrogen bottles as a backup air source. Even if the nitrogen bottles are capable of mitigating all loss of Instrument Air events and the SAMA can be implemented for a cost of \$78,000, the enhancement would not be cost beneficial ($\$8,100 - \$78,000 = -\$69,900$). Because the cost of implementation is over 9 times the averted cost-risk, this SAMA would not be cost effective even if the 95th percentile PRA results were applied.

NRC Requested Information:

- c. **At Indian Point the final SAMA evaluation included three cost beneficial SAMAs not evaluated in STP Section F.5.3.1.2. These are: SAMA 9 - create a reactor cavity flooding system to reduce the impact of core-concrete interaction from molten core debris following core damage and vessel failure, SAMA 53 - keep both pressurizer PORV block valves open, and a gagging device for SGTR events that would provide a means of closing a stuck open SG relief valve. At Prairie Island the final SAMA evaluation included three cost beneficial SAMAs not evaluated in STP Section F.5.3.1.5. These are: SAMA 3 – provide alternate flow path from refueling water storage tank (RWST) to charging pump, SAMA 19a – provide a reliable backup water source for replenishing the RWST (for Unit 2), and a gagging device for SGTR events that would provide a means of closing a stuck open SG relief valve. Consider these SAMAs for STP.**

STPNOC Response:

The following tables provide the disposition of the SAMAs identified in this RAI:

Table 5-1			
Review of Additional Indian Point Cost Beneficial SAMAs			
Industry Site SAMA ID	SAMA Description	Discussion for STP	Disposition for STP SAMA List
9	Create a reactor cavity flooding system	The implementation cost of the reactor cavity flooding system is estimated to be \$3,714,000 per unit in Appendix E, Table E.2-2 of the Indian Point Energy Center Environmental Report [ENO 2007]. This cost, which is considered to be applicable to STP, is over 14 times larger than the base STP MACR of \$259,000 per unit. This SAMA could not be a cost beneficial for STP even if it eliminated all plant risk and the 95 th percentile PRA results are applied.	Cost of implementation exceeds plant MACR. Screened from further analysis.
53	Keep both pressurizer power-operated relief valve block valves open	STP normally operates with the PORV block valves open. If a PORV is leaking, the block valve may be closed to isolate the leak, but the intent of this SAMA is considered to be met.	Already implemented. Screened from further analysis.
NA	Develop a Main Steam Safety Valve Gagging Device	A hydraulic ram device could be used to close a stuck open SV, but because an open SV at STP would create a local steam environment, no significant installation work could be performed once a valve has opened. Hydraulic ram devices could be permanently installed so they could be used in an accident scenario, but such a device would be required on each of the SVs (20 per unit). Because a single hydraulic ram device is estimated to cost about \$75,000, implementation at the site would exceed \$3 million for the hardware alone. This is over 3.5 times greater than STP's 95 th percentile MACR of \$826,854.	Cost of implementation exceeds plant MACR. Screened from further analysis.

Table 5-2			
Review of Additional Prairie Island Cost Beneficial SAMAs			
Industry Site SAMA ID	SAMA Description	Discussion for STP	Disposition for STP SAMA List
3	Alternate flow path from RWST	Parallel paths are already installed in the RWST suction path for the STP Charging system. The two MOVs, which are powered from separate divisions of diesel backed power, provide an independent means of opening the RWST suction path apart from a common check valve. Successful operation of either MOV can provide a suction source for all three charging pumps. The intent of this SAMA is considered to be met for STP.	Already implemented. Screened from further analysis.
19a	Replenish RWST from large water source	Providing the capability to provide makeup to the RWST at a capacity that could support long term injection in an SGTR event is a significant hardware modification. Prairie Island estimated the implementation cost to be \$1.935 M per unit (\$3.87M for the site) [NMC 2008]. While the cost of implementation may be different than for Prairie Island, the scope of the change is considered to be consistent. Because the implementation cost is over 7 times greater than the STP site MACR of \$518,000, providing makeup capability to the RWST could not be cost beneficial even if it eliminated all plant risk and the 95 th percentile PRA results are used.	Cost of implementation exceeds plant MACR. Screened from further analysis
NA	Develop a Main Steam Safety Valve Gagging Device	Addressed above for Indian Point.	Addressed above for Indian Point.

REFERENCES:

- ENO 2007 ENO (Entergy Nuclear Operations, Inc.). 2007. Applicant's Environmental Report; Operating License Renewal Stage; *Indian Point Energy Center*. Attachment E – Severe Accident Mitigation Alternatives Analysis. April.
- NMC 2008 NMC (Nuclear Management Company, LLC). 2008. Application for Renewed Operating Licenses – Prairie Island Nuclear Generating Plant Units 1 and 2. Appendix F SAMA Analysis. March.

NRC Requested Information:

- d. ER Section F.5.1 states that the industry based SAMA list from NEI 05-01 was used to identify the types of SAMAs that might address a particular issue. There is no further discussion of the use of this list in the ER. Clarify whether any SAMAs were developed from considering this list.**

STPNOC Response:

One of the reasons that the NEI 05-01, *Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document*, guidance was developed was to move the industry toward a SAMA identification process that was based on plant specific risks. The development of the guidance was initiated after the NRC review of the H.B. Robinson SAMA analysis. During this review, the NRC explicitly stated that a review of a generic SAMA list was of limited benefit; the generic SAMAs had been analyzed by multiple plants and were consistently found not to be cost beneficial. The real benefit was considered to be in the development of SAMAs generated from plant specific risk insights. The STP SAMA identification process is consistent with this philosophy given that it is based on plant specific risk insights from the PRA models.

In addition, the generic SAMA list provided in NEI 05-01 may neither be complete nor applicable to STP. The list was derived from the body of SAMAs identified from previous SAMA submittals and other industry guidance (with duplicates deleted). There is no guarantee that the list of SAMAs is comprehensive or that it is relevant to any given plant beyond the fact that that it includes potential plant enhancements that may have been derived from similar plants.

As stated in Section F.5.1 of the ER, the generic SAMA list from NEI 05-01 was used as an idea source to generate SAMAs for the important contributors to STP risk. The process for developing the SAMAs is essentially the same as described above, but the SAMAs to be reviewed are derived from the PRA rather than using resources to disposition the entire contents of Table 14 of NEI 05-01.

Examples of SAMAs that were derived from the concepts included in Table 14 of NEI 05-01 are summarized in Table 5-3 below:

Table 5-3 SAMA Summary	
STP SAMA	SAMA Concept Sources (from Table 14 of NEI 05-01)
SAMA 1: Use Portable 480V AC Generator for Long Term AFW Support and Protect the Technical Support Center Diesel Generator to Support the Positive Displace Pump for Reactor Coolant Pump Seal Cooling	003: Add additional battery charger or portable, diesel-driven battery charger to existing DC system. 056: Install and independent reactor coolant pump seal injection system, without dedicated diesel.
SAMA 1a: Use Portable 480V AC Generator for Long Term Auxiliary Feedwater Support and Install Westinghouse Shutdown Seals to Preserve Primary Side Inventory	003: Add additional battery charger or portable, diesel-driven battery charger to existing DC system. 058: Install improved reactor coolant pump seals
SAMA 2: Install a Core Spray System	028: Add a diverse low pressure injection system.
SAMA 6: Install an Additional Diverse, High Head Safety Injection Pump	026: Provide an additional high pressure injection pump with independent diesel.
SAMA 7: Provide Portable Fans and Ductwork for Alternate Electrical Auxiliary Building Room Cooling	082: Stage backup fans in switchgear rooms
SAMA 8: Enhance Fire Barriers in Control Room Envelope Panel 22/4	143: Upgrade fire compartment barriers

NRC Requested Information:

- e. ER Section F.6 states that site-specific cost estimates were developed for several of the SAMAs. ER Table F.5-3 cites Reference STPNOC 2009a as the source of the site-specific cost estimates. This reference is an e-mail from Engineering and Research Incorporate (ERIN) on High Head Safety Injection (HHSI). Briefly describe the process and level of detail used to develop the cost estimates (i.e., the general cost categories considered). Clarify the level of involvement and expertise of STP staff and ERIN staff in the development of the site-specific cost estimates. Provide the detailed cost estimates for SAMAs 3b and 11.**

STPNOC Response:

The scope and definition of the SAMAs were initially developed by the PRA analysts (STP and ERIN personnel) and then reviewed/modified by the STP design staff to account for any plant specific issues that could interfere with or improve the SAMA designs. After finalizing the scope and definition for each of the SAMAs requiring cost estimates, the major cost

contributors for each were identified and their magnitudes were estimated by STP design engineers. The STP design engineers perform cost estimates as part of their normal job duties.

The cost estimates that were developed for SAMAs 3b and 11 are provided below:

Table 5-4				
SAMA 3b				
Install Fire Wrap on PDP Cables in Cable Spreading Room				
Description	Qty.	Unit	Unit \$	Total
Engineering	1	LS	\$250,000	\$250,000
Wrap B and C trays in Cable Spreading Room With Fire Protection Wrap	242	LF	\$350	\$84,700
Wrap Conduits in Cable Spreading Rm. With Fire Protection Wrap	23	LF	\$125	\$2,875
Sub-Total				\$337,575
Sub-Total per Unit X 2 Units	2	Units	\$337,575	\$675,150
18% Capital & Corporate Overheads				\$121,527
Total				\$796,677

Table 5-4 Notes:

- LS – lump sum
- LF – linear feet

Table 5-5				
SAMA 11				
Modify Fire Protection System to Supply Containment Spray (CS) Headers				
Description	Qty.	Unit	Unit \$	Total
Engineering	1	LS	\$50,000	\$50,000
Procedure Revisions	1	Ea	\$40,000	\$40,000
100 feet of 6 inch, 150 psi CS pipe with stainless steel (SS) Tee and double manual ASME SS valves at CS Pump discharge and CS tee and single non-safety valve at fire header	1	Ea	\$220,000	\$220,000
Operator Training	1	LS	\$50,000	\$50,000
Sub-Total				\$360,000
Sub-Total per Unit X 2 Units	2	Units	\$360,000	\$720,000
18% Capital & Corporate Overheads				\$129,600
Total				\$849,600

Table 5-5 Notes:

- LS – lump sum
- EA - each

NRC Requested Information:

f. **SAMA 17a, “install Westinghouse Reactor Coolant Pump (RCP) Shutdown Seals,” has an estimated implementation cost of \$7,611,000. This suggests that the single unit cost for this modification is estimated to be about \$3.8 million. A recent submittal by Tennessee Valley Authority (TVA) for Watts Bar Unit 2 (TVA letter to NRC of July 23, 2010, ML102100588) estimated the cost to install improved Westinghouse RCP seals to be \$1.1M per unit while this modification was estimated to cost \$1.05M per unit in the Vogtle license renewal application. Furthermore, the NRC staff is aware that the new seal package technology is**

being demonstrated at the Farley Nuclear Plant (TVA letter to NRC of January 31, 2011, ML110340040). Describe the difference between the “shutdown seals” assumed in SAMA 17a and the improved seals cited by TVA and Vogtle. Also, provide a more detailed description of the SAMA 17a modification and justification for the estimated cost to install the “shutdown seals” at STP.

STPNOC Response:

The “shutdown seals” identified in STP SAMA are not the same type of seal as the “improved seals” described by Watts Barr and Vogtle. STP utilizes Westinghouse Model 100 RCPs with a specific seal housing design. The specific pump seal design would result in an increased design and engineering cost that would be incurred by STP while other industry sites would share the design and engineering cost among a larger pool of contributors. It should be noted that at the time the SAMA analysis was developed, there was no finalized product or cost available from the vendor.

The following table provides the cost estimate developed by STP for installation of the “improved seals” for the RCPs:

Table 5-6 SAMA 17a: Install Westinghouse RCP Shutdown Seal				
Description	Qty.	Unit	Unit \$	Total
Engineering - RPE	1	LS	\$100,000	\$100,000
Procedure Revisions	1	Ea	\$25,000	\$25,000
Modified #1 Seal Housing	4	Ea	\$40,000	\$160,000
New Emergency RCP Seal	4		\$350,000	\$1,400,000
CEM Emergency RCP Seal + Housing	1		\$340,000	\$340,000
Installation	4		\$300,000	\$1,200,000
Sub-Total				\$3,225,000
Sub-Total per Unit X 2 Units	2	Units	\$3,225,000	\$6,450,000
18% Capital & Corporate Overheads				\$1,161,000
Total				\$7,611,000

Notes for Table 5-6 above:

- LS – lump sum
- EA – each

NRC Requested Information:

- g. The cost of \$4.5M given in ER Table F.5-3 for SAMA 14 seems very high given that an inter-unit cross-tie is already available. Provide a more detailed description of the modification and justification for the estimated cost. In the response, discuss the possibility of using existing breakers and buses to cross-tie buses in one unit under emergency conditions.**

STPNOC Response:

The original intent of SAMA 14 was to provide the capability to perform a cross-tie between emergency 4KV AC buses within a single unit rapidly enough to prevent an RCP seal LOCA. It was assumed that the most effective means of providing this capability was through a direct bus to bus connection, which does not exist at STP.

In response to this RAI, STP's 4KV AC cross-tie capabilities were investigated further and it was determined that a viable 4KV AC cross-tie hardware path already exists on the affected unit via a back feed through an emergency transformer; however, this alignment is part of the mitigating strategies B.5.b program and the details are not available to the public.

While the hardware to support the cross-tie capability exists within each single unit, incorporating the capability into plant procedures would be significantly more resource intensive than other procedure changes that have been investigated as SAMAs (e.g., SAMA 4). The primary reason is because performing the cross-tie introduces the potential for a single failure to disable multiple divisions of equipment. The cost of the analysis that would be required to support the cross-tie alone would potentially range from \$300,000 to \$400,000, the scope of which is beyond the more general analysis that supports the use of the cross-tie.

In addition to the cost of analysis, the cost of the relatively complex procedure changes associated with the cross-tie has been estimated to be an additional \$300,000, for a total of about \$600,000.

With regard to capability, it is estimated that the cross-tie could not be aligned in time to prevent a RCP seal LOCA, thus limiting the potential benefit of the SAMA to a small subset of LOOP scenarios in which equipment on the "powered" division has failed. These types of scenarios were not identified in the STP importance list review and a site specific AC cross-tie SAMA was not developed (it was identified through an industry review for a "similar" plant). The implication is that the benefit for the AC cross-tie SAMA would be below the \$11,000 review threshold established in Section F.5.1.1 of the ER and should be screened from review. Even if the entire LOOP contribution were assumed to be mitigated by this SAMA, the maximum potential averted cost-risk would only be \$205,060 when the 95th

percentile PRA results are applied ($\% \text{ contribution from LOOP} * 95^{\text{th}} \text{ percentile MACR} = \text{averted cost-risk}$, or $0.24 * 856,854 = 205,060$). This would result in a net value of $-\$394,940$ ($\$205,060 - \$600,000 = -\$394,940$).

Finally, the manipulation time for the cross-tie is estimated to be driven by event classification and navigation of procedures rather than the physical process of aligning the buses such that providing a direct bus to bus cross-tie would provide no additional benefit for preventing an RCP seal LOCA. As a result, the original SAMA 14 design would not be required for STP and both are screened from further review.

NRC Requested Information:

6. Provide the following information with regard to the Phase II cost-benefit evaluations:

- a. ER Section F.6.2 describes that SAMA 10 was modeled by reassigning the SGTR CDF contribution for Release Categories I ($7.48\text{E-}09$ per year) and III ($1.35\text{E-}07$ per year) to Release Categories II and IV, respectively. Neither of these contributions corresponds with any frequency values reported in ER Table F.3-5. Provide additional information on the source of each of these release frequency contributions and clarify that they represent a realistic assessment of the potential risk reduction for this SAMA.**

STPNOC Response:

Because SAMA 10 is dependent on the availability of secondary side makeup to ensure inventory in the steam generators is maintained above the tubes, only a fraction of the steam generator tube rupture (SGTR) scenarios are relevant to the SAMA 10 quantification. Specifically, the induced SGTR (ISGTR) contributors were excluded because the ISGTR classification implies lack of secondary side makeup capability in the STP model. This is one reason why the Release Category I frequency of $7.48\text{E-}08/\text{yr}$ and the Release Category III frequency of $1.35\text{E-}07$ per year do not directly correspond to the frequencies listed in Table F.3-5 of the ER. Another reason is that the SGTR contributors are binned among different end states with contributors from other initiating events such that no single end state exclusively captures all of the SGTR scenarios that would be impacted by steam generator flooding.

In order to identify the relevant frequencies for this SAMA, the PRA model results were examined to determine how the SGTR events were distributed among the STP Level 2 release categories. This information was provided explicitly by the PRA software and the results are reproduced below:

Group Name	Frequency	Description
RELI	7.4767E-009	RELEASE CATEGORY 1, LARGE EARLY RELEASE
RELII	7.6343E-008	RELEASE CATEGORY II, SMALL EARLY RELEASE
RELI	1.3498E-007	RELEASE CATEGORY III, LATE RELEASE
RELIV	2.0146E-007	RELEASE CATEGORY IV, NO RELEASE

It was assumed that all SGTR contributors had AFW available such that SAMA 10 could be credited with scrubbing the releases; however, because Release Category II is already a small release, no further reduction was assumed. Release Category IV represents the "no release" condition and those contributors were also assumed to not be impacted by SAMA 10.

In order to simulate the scrubbing effect of SAMA 10, the Large-Early releases were reduced to Small-Early releases and the late releases were assumed to result in "no release", as described in the ER. These assumptions are considered to adequately represent the impact of SAMA 10 implementation.

NRC Requested Information:

- b. ER Section F.6.3, 5th paragraph, explains that the evaluation of SAMA 12 did not consider the condition in which non-condensable gases such as hydrogen are present since this condition is not modeled in the PRA, but that this condition is conservatively treated in the PRA. If this SAMA impacts this condition then the estimated risk reduction is potentially underestimated. Also, this same section of the ER states that SBO sequences were excluded in the modeling of this SAMA because AC power is needed to start a reactor coolant pump (RCP). This also potentially underestimates the risk reduction benefit for this SAMA since it does not appear to include SBO scenarios in which AC power is recovered. Discuss these issues and their impact on the SAMA analysis.**

STPNOC Response:

Comments on the following paragraph in ER Section F.6.3 addresses Part 1 of RAI 6.b, regarding non-condensable gases.

"The presence of non-condensable gases such as hydrogen may occur, but is not typically modeled in nuclear plant PRA models. The use of discrete, success or failed events until explicitly recovered, which is the nature of PRA models is not sufficient to resolve periods of system failure followed by periods of later successful system response. The generation of hydrogen within the RCS that occurred following a temporary interruption of steam generator cooling and high pressure injection at TMI-2 is not modeled in the STP PRA. It is instead conservatively represented by sequences in which high pressure injection and AFW are initially lost and not restored."

Although the specific TMI scenario leading to hydrogen gas generation is not explicitly modeled in the STP PRA, it is, as noted, represented conservatively in induced SGTR scenarios, by sequences where high pressure injection (HPI) and AFW are initially lost and not restored resulting in core damage with the RCS at high pressure. These sequences are included in the assessment of the impact of this SAMA. The presence of non-condensable gases in the RCS at the high system pressures required for induced steam generator tube ruptures is not expected to significantly affect the heat transfer rate to the steam generator tubes relative to that of the hot legs. Therefore, the estimated risk reduction of SAMA 12 is not underestimated.

For Part 2 of RAI 6.b, regarding SBO scenarios:

The station blackout scenarios excluded from the assessed impacts of this SAMA (due to power being unavailable to the RCPs) is appropriate. Without offsite power, the RCPs could not be operated. For station blackout scenarios initially losing offsite power but with offsite power recovered in time to prevent core damage, such sequences are mapped to success and induced steam generator tube ruptures are not at issue.

For station blackout scenarios initially losing offsite power and with offsite power recovered only after core damage, the governing STP procedure would be 0POP05-E0-EC02, "Loss of All AC Power Recovery With SI Required". In this case, the critical function status trees are to be monitored for information only and the function restoration procedures are not to be implemented until after step 11 of the procedure. While RCP seal cooling is to be established in preparation for pump operation, the initial steps of this procedure are to manually load ECCS pumps including HHSI pumps for injection, and if steam generator level is low, then to initiate AFW pumps to establish AFW flow. This procedure does not instruct the operators to start the RCPs to reestablish primary flow thereby potentially affecting loop seal clearing. To the contrary, if RCP seal cooling is lost long enough, the operators are specifically directed not to start the affected RCP. Therefore, elimination of station blackout sequences in which recovery of offsite power occurs after core damage, from the impacts of SAMA 12 is also appropriate.

STP Function Restoration Guideline 0POP05-E0-FRC1 may have been entered prior to the time of core damage in some scenarios, but only for non-station blackout scenarios. These scenarios were considered in the benefit assessment of SAMA 12.

Therefore, the SAMA evaluation does not underestimate the impact of this SAMA due to conservatism in the assessment.

NRC Requested Information:

- c. **ER Section F.6.5, for SAMA 15, states that "common cause failures were added, after the common cause data was edited." Explain what is meant by this statement.**

STPNOC Response:

It is believed that this question is commenting on a statement describing changes made to the top event model to develop the PRA sensitivity case for this SAMA. In RISKMAN[®], total failure rate data (in this case for HVAC fans to start and run), are entered in the common cause group modeling area. In RISKMAN[®], changes to the CCF model or data require "removing" CCF, making the desired changes, and then "adding" CCF. The statement, "common cause failures were added, after the common cause data was edited," simply refers to the procedure for CCF changes performed in RISKMAN[®], and is correct. It is simply defining the step-by-step process required to implement the change in the failure rates. No changes to CCF group inventory or logic in the associated fault tree(s) are associated with the referenced comment in this case.

NRC Requested Information:

7. Provide the following information with regard to the sensitivity and uncertainty analyses:

- a. ER Section F.7.1.1.1 describes the PRA model changes made to evaluate SAMA 3b as deleting macros IZ47BC and IZ047X. Describe these macros.**

STPNOC Response:

The macro IZ47BC describes the fire initiating event Z47BC. The macro IZ047X describes the fire initiating event Z047X.

b. NRC Requested Information:

The ratio of the 95th percentile CDF to the mean value CDF was reported to be 1.6 in Section F.7.1 of the ER. While this is a "typical" result for internal event CDF, it seems quite low for the fire and seismic CDFs which generally have wider uncertainty bands than internal events. Provide support to the adequacy of this distribution result given the expected wider distribution for external events and considering the impact of more current seismic hazard curves such as the USGS 2008 assessment.

STPNOC Response:

With the exception of the frequencies for the seismic initiating events, all of the initiating event frequencies utilize data variables that are described by probability distributions. Because seismic CDF at STP is small, the use of point estimates for the peak acceleration frequencies has no visible effect on the data uncertainty curve. See response to RAI 3.c. regarding consideration of more current seismic hazard curves.

NRC Requested Information

8. For certain SAMAs considered in the ER there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, provide an evaluation of the following SAMAs:
- a. **SAMA 1, involving using a portable AC generator for long term auxiliary feedwater (AFW) support and protecting the Technical Support Center (TSC) emergency diesel generator (EDG) from tornado events, is identified as a means of mitigating a large number of important basic events. While the tornado protection is important for HWIND (i.e., Tornado Induced Failure of Switchyard) initiated sequences, many other sequences would be mitigated without the cost of the tornado protection. Consider such a SAMA.**

STPNOC Response:

The cost of protecting SAMA 1 such that it could function in a high wind event, while not negligible, is not a critical factor in the Phase I screening for SAMA 1. The SAMA 1 cost estimate has been modified to eliminate the costs associated with providing high wind protection.

All costs associated with constructing the enclosures for the TSCDG and Load Center 1W (i.e. the load center supplied by the TSCDG) were eliminated. In addition, the engineering costs were reduced by 50 percent to reflect the reduced scope of work. The revised cost of implementation for the site is \$2,419,000, which is over 4.5 times larger than the baseline MACR. Even if this modified version of SAMA 1 were assumed to eliminate all on-line risk and the 95th percentile PRA results were applied, it would not be cost effective.

The tables below provide the cost information for the baseline version of SAMA 1 and the version of SAMA 1 without high wind protection.

Table 8-1				
SAMA 1 Baseline Cost Estimate				
Description	Qty.	Unit	Unit \$	Total
Engineering	1	LS	\$500,000	\$500,000
Procedure Revisions	1	LS	\$100,000	\$100,000
480V DG set + fuel tank for 8 hours	1	LS	\$225,000	\$225,000
Conduit & Wire	500	LF	\$300	\$150,000
Switches/breakers to switch board	1	Ea	\$25,000	\$25,000
Construct concrete block enclosure for TSC DG	1	LS	\$50,000	\$50,000
Construct concrete block enclosure for Load Center 1W on EAB roof	1	LS	\$140,000	\$140,000
New test procedures & Emergency Operating Procedures	1	LS	\$175,000	\$175,000
Testing	1	LS	\$25,000	\$25,000
Operator Training Program	1	LS	\$75,000	\$75,000
Sub-Total				\$1,465,000
Sub-Total per Unit X 2 Units	2	Units	\$1,465,000	\$2,930,000
Capital & Corporate Overheads	18%			\$527,400
Total				\$3,457,400

Notes:

LS – lump sum
 LF – linear feet
 EA – each

Table 8-2				
SAMA 1 Cost Estimate Without High Wind Protection				
Description	Qty.	Unit	Unit \$	Total
Engineering	1	LS	\$250,000	\$250,000
Procedure Revisions	1	LS	\$100,000	\$100,000
480V DG set + fuel tank for 8 hours	1	LS	\$225,000	\$225,000
Conduit & Wire	500	LF	\$300	\$150,000
Switches/breakers to switch board	1	Ea	\$25,000	\$25,000
Construct concrete block enclosure for TSC DG	1	LS	\$0	\$0
Construct concrete block enclosure for Load Center 1W on EAB roof	1	LS	\$0	\$0
New test procedures & EOPs	1	LS	\$175,000	\$175,000
Testing	1	LS	\$25,000	\$25,000
Operator Training Program	1	LS	\$75,000	\$75,000
Sub-Total				\$1,025,000
Sub-Total per Unit X 2 Units	2	Units	\$1,025,000	\$2,050,000
18% Capital & Corporate Overheads				\$369,000
Total				\$2,419,000

Notes:

LS – lump sum
 LF – linear feet
 EA – each

NRC Requested Information

- b. With respect to RAI 8.a, discuss the possibility for the TSC EDG to supply the positive displacement pump (PDP) and support AFW operation. If feasible, consider such a SAMA.**

STPNOC Response:

Assuming that the Technical Support Center diesel generator (TSCDG) could support its normal loads, the PDPs, and the loads associated with supporting long term AFW operation, the SAMA 1 implementation cost could be further reduced by eliminating the costs associated with the portable 480V AC diesel generators. The cabling and analysis costs are retained as a connection between the non-safety TSCDG and the safety related bus supporting AFW would be required.

The table below provides the cost information for SAMA 1 assuming that the TSCDG can be used to support AFW instead of a separate generator and assuming that no high wind protection is installed for the TSCDG.

Table 8-3				
SAMA 1 Cost Estimate Using the TSC EDG for AFW Support and No TSC High Wind Protection				
Description	Qty.	Unit	Unit \$	Total
Engineering	1	LS	\$250,000	\$250,000
Procedure Revisions	1	LS	\$100,000	\$100,000
480V DG set + fuel tank for 8 hours	1	LS	\$0	\$0
Conduit & Wire	500	LF	\$300	\$150,000
Switches/breakers to switch board	1	Ea	\$25,000	\$25,000
Construct concrete block enclosure for TSC DG	1	LS	\$0	\$0
Construct concrete block enclosure for Load Center 1W on EAB roof	1	LS	\$0	\$0
New test procedures & EOPs	1	LS	\$175,000	\$175,000
Testing	1	LS	\$25,000	\$25,000
Operator Training Program	1	LS	\$75,000	\$75,000
Sub-Total				\$800,000
Sub-Total per Unit X 2 Units	2	Units	\$800,000	\$1,600,000
18% Capital & Corporate Overheads				\$288,000
Total				\$1,888,000

Notes:

LS – lump sum
 LF – linear feet
 EA – each

If the costs of the 480V AC generators are eliminated in conjunction with the TSC high wind protection measures, the cost of the SAMA is reduced to \$1,888,000. This is over two times as large as the 95th percentile maximum averted cost-risk of \$826,854 and the SAMA would not be cost effective even if it could eliminate all plant risk. Without the high wind protection measures, however, it should be noted that this version of SAMA 1 would not address the single largest contributor to plant risk and that the potential averted cost-risk would be significantly less than the maximum averted cost-risk

NRC Requested Information

- c. The HWIND initiating event is the largest single contributor to CDF. For mitigating the HWIND sequence, consider a SAMA to provide an alternate intake structure for the essential cooling water (ECW) either in the essential cooling water pond (ECP) or the MCR that would minimize the likelihood of debris preventing ECW cooling and/or the possibility of using a temporary/portable pump with a movable suction that could provide water to the ECW system.**

STPNOC Response:

The construction of an alternate intake structure or a modification to the existing structure that would not be susceptible to clogging in a high wind scenario constitutes a major plant modification. Based on a review of the existing intake structure design and the debris clogging failure mode, a modification to the intake structure is expected to be less resource intensive than the installation of an alternate intake structure. A potential modification would be to install a large surface area debris cage (136 feet x 25 feet x 16.5 feet) that would be less likely to be completely clogged in a high wind event. The modification would be seismically rated, but not safety related. STP has estimated the cost of the design, installation, and materials of the debris cage to be \$827,800, which exceeds the STP MACR of \$518,000 for the site. Even considering the 95th percentile Probabilistic Risk Assessment (PRA) results, this SAMA would not be cost beneficial.

The portable suction/pump SAMA would potentially be a less costly alternative; however, the same high wind event that caused failure of the intake structure may introduce Essential Cooling Pond access issues that would prevent alignment of a portable pump in time to prevent core damage. The time frame required for Essential Cooling Water (ECW) recovery is considered to be the time to core damage in a Reactor Coolant Pump (RCP) seal loss-of-coolant accident (LOCA) scenario, which may be around 40 to 60 minutes. Because recovery of the ECW system would restore emergency 4KV power, it is not necessary to prevent an RCP seal loss-of-coolant accident (LOCA), but 4KV power must be restored in time to provide injection to the reactor coolant system (RCS) to prevent core damage. For this evaluation, it is assumed that this is possible by using a "portable" ECW pump; however, the pump would have to be large enough to provide about 35,000 gallons per minute (total for both units for diesel cooling and decay heat removal), be self-powered, and be capable of being moved in conjunction with any required suction and discharge piping. These requirements imply a truck based pump. Systems with these capabilities are available, but the cost estimate for a 30,000 gallons per minute mobile pump, which was obtained from a vendor, is \$300,000. The cost of installing hard piping at the structure to direct flow to the appropriate ECW bays is estimated to cost an additional \$50,000, for a total of \$350,000. Even without including any costs associated with training and procedure updates to support this SAMA, the cost of implementation would exceed the potential averted cost-risk associated with the HWIND initiating event.

The HWIND initiator contributes 17.3% of core damage frequency (CDF) and is associated with events that have similar risk reduction worth (RRW) values in both the Level 1 and Level 2 importance lists. This would correlate to about 17.3% of the maximum averted cost-risk (MACR), which is \$89,614 ($\$518,000 * 0.173$). Even if the 95% PRA results are applied, the potential averted cost-risk is only \$143,046 ($\$826,854 * 0.173$).

Based on these factors, neither of these potential enhancements would be cost-effective for STP.