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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE PROBABILISTIC SAFETY ANALYSIS EVALUATION

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated April 14, 1989 (ST-HL-AE-3059), Houston Lighting & Power Company (HL&P, the licensee) submitted the South Texas Project (STP) Probabilistic Safety Assessment (PSA) Summary Report which included the results of the Level 1 Probabilistic Risk Analysis (PRA). In that letter, the licensee also informed the staff that it planned to provide a risk based analysis of the STP technical specifications (TS) based on the STP PRA model. The intent of the risk based analysis would be to propose changes in the areas of allowed outage times and surveillance intervals of the TSs based on the features of the STP three train design. The staff requested a complete copy of the PSA in order to evaluate it as an adequate basis for reviewing the expected TS changes.

The PSA was submitted by letter dated June 15, 1989 (ST-HL-AE-3137). The PSA was done by Pickard, Lowe and Garrick, Inc. (PLG), under contract to the licensee. After completion, the licensee took possession of the PSA and responsibility for its maintenance.

2.0 REVIEW PROCESS

Staff review of the PSA was supported through a contract to Sandia National Laboratories (SNL), and SEA Corporation to perform a preliminary review of the portions of the PSA internal events analyses and fire sequence analyses. As part of the staff's review, the staff, SNL, and SEA performed three separate site walk-downs and conducted four separate meetings on August 8, 1989, November 28, 1989, May 30, 1990, and October 15, 1990, with the plant staff to obtain additional information and responses to questions. The staff also issued a request for additional information (RAI) to HL&P on January 3, 1990. The licensee's responses to the RAI were received by letters dated January 25, 1990 (ST-HL-AE-3352), March 1, 1990 (ST-HL-AE-3380), and April 11, 1990

23

(ST-HL-AE-3414) and incorporated as appropriate into a Draft Technical Evaluation Report (TER) developed by SNL. This draft report was issued on April 10, 1990. A separate evaluation report on the STP fire sequences and an associated RAI were issued to the licensee on June 18, 1990 and August 30, 1990. On August 7, 1990, a separate RAI on the human reliability analysis (HRA) review was issued to the licensee. The licensee's comments on SNL's draft evaluation report, and responses to the RAIs on the HRA review and fire sequences review (HL&P's letters of April 11, 1990 (ST-HL-AE-3414), June 19, 1990, (ST-HL-AE-3478), August 26, 1990, (ST-HL-AE-3551), October 11, 1990, (ST-HL-AE-3590) and November 20, 1990 (ST-HL-AE-3636)) have been reviewed and incorporated into the final TER, "A Review of the South Texas Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning" (NUREG/CR-5606). Overall, the review process employed for the STP PSA was an interactive process with the licensee and its contractors. The staff's Safety Evaluation (SE) is based on its own review of the applicable portions of the PSA as well as its review of SNL's TER, which is attached to this SE. Review of external events (except fire) is still under staff review and will be reported in a future SE.

3.0 EVALUATION

The results of the internal events review are reported in Section 3.1 of this SE. Section 3.2 is the documentation of the fire analysis review.

3.1 Internal Events

3.1.1 Initiating Events

The licensee's analysis of initiating events is documented in Section 5.2 of the STP PSA and in Section 3.1 of the TER. Based on the review, the staff accepts the PSA findings related to categorizing, grouping, and screening of various events that could lead to a transient and/or a LOCA event. The staff accepts the licensee's responses to the staff's RAI in the areas related to the treatments of the steam-line break of the Auxiliary Feedwater (AFW) system steam driven train and the postulated core blockage event as applicable to the STP facility. The staff notes that the licensee has analyzed the failures of the support systems (such as the Instrument Air [IA] system, Main Control Room [MCR] HVAC, and Electrical Auxiliary Building [EAB] HVAC) as initiating events and categorized them accordingly for core damage frequency quantification purposes (Table 5.2-4 and Table 7.6-1 of the PSA).

The staff accepts the licensee's estimates of various initiating event frequencies. These estimates are provided in Table 3.4.2-2 of the TER. This table also compares their estimates with the published NUREG-1150 (Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants) results provided for similar initiating events. The staff finds a close agreement between the STP PSA results and the NUREG-1150 results except for loss of Main Feedwater (MFW), reactor trip, turbine trip, and steam generator tube rupture (SGTR) events. The staff accepts the licensee's estimates for all transients because the licensee's estimates are based on an extensive search (PSA subsections 5.2.1 thru 5.2.3) of operating experience and screening analyses for the applicability of generic data (Section 7.4) to the STP facility. The staff notes however that the number of plant trips at STP are different from the PSA estimates for the plant trip and turbine events. The licensee indicated that as more plant specific information is generated, it will be evaluated for use in future PSA updates. The staff also accepts the licensee's estimate for loss of offsite power events ((0.13 per year) which is based on data for the Central Power & Light grid system. The licensee's estimate for the SGTR event is based on single tube failure events which is common PRA practice.

Overall, the staff accepts the licensee's analyses in the area of initiating events with exceptions as noted above.

3.1.2 Accident Sequence Modeling

The licensee's discussions related to the development of accident sequences that could occur following a transient and/or a postulated LOCA event are provided in Section 5.4 of the PSA. Unlike the NUREG-1150 methods, the STP PSA has made use of very large event trees to develop accident sequences. In addition to modeling frontline core cooling systems as part of the event trees, the PSA has also accounted for the impact of failure and success of the support systems such as electrical and mechanical systems. The plant responses that could be expected during both the early and late stages following the initiating event, including operator recovery actions (per the STP emergency operating procedures), have also been modeled explicitly. The PSA has also developed various event sequence diagrams (ESD) to develop a thorough understanding of various methods of achieving core cooling following a transient, LOCA event, ATWS, or SGTR event. This information has been used in developing the longer event trees and documenting critical assumptions needed to scope the sequence modeling process. The PSA has also developed logic models (referred to quantitatively as "split fractions") for each top event modeled in the event tree, that reflect the impact of failure and success of prior top events. Further, a method of event tree linking has been used to characterize a given top event and to quantify it in order to estimate the frequencies of all sequences involving that particular top event. The details related to the event tree linking procedures have been documented in Section 4.3.5 of the PSA.

Section 5.4 of the PSA has provided ESDs and event trees for a general transient, ATWS, SGTR event, small LOCA event, medium LOCA event, and large LOCA event and has documented the graphical displays of potential accident sequences along with details on split fractions and critical assumptions. Based on the initial review findings, the staff accepted the PSA's modeling of accident sequences with the exception of interfacing LOCA sequences. In response to the staff's RAI, the licensee has further developed sequence modeling details for interfacing LOCA events (Appendix 3 of the TER). The staff has reviewed the response and finds it acceptable.

The staff also notes that the STP PSA transient event tree is more systematic than those used for other PRAs, but is too complex to manually trace a single sequence to reproduce results manually.

3.1.3 System Modeling

The staff, with the help of SNL, has reviewed the modeling adequacy of frontline systems and support systems as documented in Sections 3.2.2 through 3.2.5 of the STP PSA with respect to the methods used to analyze system failures and their combinations, critical assumptions made in the PRA, modeling adequacy of system dependence requirements, test and maintenance unavailabilities, treatment of common cause failures and human errors, and modeling adequacy of operator recovery actions. The frontline systems modeled are: (1) High Head Safety Injection (HHSI), (2) Low Head Safety Injection (LHSI), (3) Containment Spray System (CSS), (4) Reactor Containment Fan Cooler (RCFC), (5) Residual Heat Removal (RHR), (6) Containment Isolation System (CIS), (7) Auxiliary Feed water (AFW), (8) Chemical and Volume Control System (CVCS), (9) Reactor Coolant System (RCS), including steam generators, pressurizer and power operated relief valves (PORV), (10) Hotwell Condensate, (11) Main Feed water (MFW), and (12) Steam and Power Conversion. The support line systems modeled are: (1) Component Cooling Water (CCW), (2) Essential Cooling Water (ECW), (3) Essential Cooling Pond, (4) Vital and non-vital AC Power (4.16 KV, 480 V, and 125 V) buses, including motor control centers, Class IE diesel generators (DG), Technical Support Center (TSC) diesel generator, and inverters, (5) Vital and non-vital DC-power buses (125 V and 250 V buses), including batteries and chargers, (6) Compressed Air, (7) Reactor Protection, and (8) Heating, Ventilating and Air Conditioning (HVAC) for various buildings. Based on SNL's technical review findings on the systems modeling, the staff provides the following statements:

1. The lack of a need for Emergency Core Cooling System (ECCS) pump room cooling during a transient or a LOCA event was evaluated in detail along with the licensee's additional response to the staff's RAI (IE13 of Appendix 3 of the TER). The staff accepts the licensee's response.

2. The treatment of instrument air system failures was evaluated along with the licensee's additional response to the staff's RAI (IE14 of Appendix 3 of the TER). The staff accepts the licensee's response.

The staff's review also resulted in the following observations:

1. The STP diesel generators do not use dedicated batteries for field flashing. Instead, they receive DC power from Class 1E DC buses. However, this unique dependence of the diesel generators has been found to be an insignificant contributor to the overall core damage frequency.

2. The motor-driven AFW pump room requires ventilation during its operation. However, the turbine-driven AFW pump does not require room cooling. This eliminates the HVAC dependence during the station blackout scenario.

3. Operation of positive displacement charging pump (PDP) is found to be significant because it can be used to provide seal injection during a station blackout event (given that the isolation of letdown is accomplished), and it can receive power from the TSC diesel generator. It is self-cooled. Also, its room cooling is not needed during a station blackout scenario.

4. The loss of the instrument air system, its impact as an initiating event, and its impact on other frontline systems have been found to be small contributors to core damage frequency estimates.

5. Feed and bleed operation as a backup method of decay heat removal for the STP facility has been credited based on generic Westinghouse analyses which requires one train of the HHSI system and two PORVs.

6. Unlike the LHSI system at some other PWR's, the LHSI system at STP is not required for operation in the piggy-back mode with the HHSI system during high pressure recirculation. The use of the LHSI system following a small LOCA event requires depressurization of the reactor primary system.

7. STP has a separate LHSI system independent of the RHR system. This is a feature unique to the STP design.

3.1.4 Success Criteria

The licensee's discussions related to success criteria are provided in Section 5.4 of the PSA. SNL's technical findings are provided in Section 2.1 of the TER. Based on SNL's technical review findings on modeling adequacy of success criteria, the staff provides the following statements:

1. The adequacy of the steam generator boil-dry time estimated for various transients was evaluated since this time affects the operator recovery probabilities. The licensee's response to the staff's RAI on this issue, estimated a minimum time of 34 minutes to steam generator dryout following a loss of offsite power event. This is acceptable.

2. The assigned conditional probability (0.0001 per demand) of reactor vessel failure at the STP facility, given a pressurized thermal shock (PTS) event, has been evaluated and is found to be acceptable to the staff. This is based on the significantly lower content of certain elements at STP, such as copper (about 0.05 percent) and nickel (about 0.64 percent), than at other PWR reactor vessels. An acceptable limit for the contents of these elements is about 0.4 percent for copper and is about 1 percent for Nickel ("Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Rev. 2). It should be noted that these elements contribute to a reduction in the toughness of the vessel that could lead to vessel failure during a postulated PTS event.

3. The staff evaluated the impact on core damage frequency of the lack of accumulator injection for postulated medium and large LOCA events. Based on the licensee's response (IE3 of Appendix 3 of the TER) to the staff's RAI on this issue, the staff accepts the licensee's core damage frequency estimate of 4 E-7 per reactor-year for these events assuming that 2 accumulators will inject water to the reactor following a medium and/or large LOCA (greater than a 2-inch break) event. The staff also accepts the licensee's conclusion that, for break sizes below two inches, core cooling can be achieved without accumulators.

4. The staff evaluated the impact on core damage frequency of the containment isolation system (CIS) failures (a lower containment back pressure resulting in degradation in core cooling) following a medium or a large LOCA event. Based on the licensee's response (IE4 of Appendix 3 of the TER) to the staff's RAI on this issue, the staff accepts the licensee's statement that, with the failure of the CIS the peak clad temperature (PCT) will increase to a temperature of not more than 2510 degrees F. With the LHSI and CIS functioning successfully, the PCT will be below the regulatory temperature limit of 2200 degrees F. Because the zirconium phase transition temperature is about 2900 degrees F, beyond which a core damage will be most likely, the staff agrees with the licensee's statement that successful operation of the LHSI system with the failure of the CIS will not result in a severe core damage event.

5. The staff evaluated the impact on core damage frequency of a modeling omission related to the need for switchover to hot leg recirculation in order to avoid boron precipitation following a large LOCA event. The licensee stated that failure to switchover would not lead to core damage, so it was not included in the PSA. However, the licensing basis for the plants assumes that failure to switchover would lead to core damage. The licensee's response (IE5 of Appendix 3 of the TER) to the staff's RAI on this issue, stated that if the event is included, the CDF associated with operator failure to switchover to hot leg recirculation following a large LOCA event is about 2E-8 (0.01 percent of overall CDF). The staff accepts the licensee's calculation of the contribution to the CDF.

6. The staff evaluated the modeling adequacy in characterizing the small break LOCA event frequency as documented in the PSA. For pipe breaks below 0.5 inch diameter, normal charging makeup flow will be sufficient to provide coolant makeup to the reactor. The fluid loss through a single instrument tube break could also be compensated by the normal charging makeup capability (IE6 of Appendix 3 of the TER). Thus the staff judges that a 0.5 inch size will be the lower end of the small LOCA break sizes. Therefore, the licensee's estimate of the small break LOCA frequency based on a 0.5 inch size break is acceptable to the staff.

7. The staff evaluated the success criteria employed for depressurization of the reactor primary system following a SGTR event and has found modeling conservatism. The PSA did not take credit for use of the turbine bypass steam dump system as a means of decay heat removal. Also, the PSA has not taken credit for remaining at hot standby with AFW system makeup to the steam generators (SG) in events where the RHR shutdown cooling mode is not available. The staff accepts the STP findings that the above core cooling criteria are conservative.

8. The staff evaluated the adequacy of the logical minimum number of system trains needed to provide emergency boration, overpressure protection features, coolant makeup to the primary system, and subsequent decay heat removal following a failure-to-scram event. The review indicates that the success criteria basically requires one of two boric acid transfer pumps for emergency boration, two PORVs and two safety valves to open (or all three safety valves to open) for primary system overpressure protection,

one PORV and one safety valve to open (or two safety valves to open) for steam generator overpressure protection, one of three HHSI trains and one of three LHSI trains for primary system coolant makeup, and two of four AFW trains for decay heat removal. The staff accepts the PSA findings in this area.

9. The staff evaluated the minimum number of trains needed for containment cooling following a postulated LOCA event or a transient-induced LOCA event. One LHSI train or two RCFC trains are required to remove the decay heat from the South Texas containment. The staff accepts the PSA findings in this area.

The staff' review also yielded the following safety insights:

1. If it is required to keep the reactor in the hot standby mode for an extended period of time following a transient or a SGTR event, makeup water to the AFW storage tank must be provided. The primary source of makeup water to the AFW storage tank is the hotwell condenser system. While not considered in the PSA, makeup water to the AFW storage tank could also be accomplished through either the demineralized water system or the fire water system. These systems are designed with industrial grade (not seismic category I) structures and components; but in an area of low seismic activity such as STP they could be considered as non-safety backup systems.

2. For a small LOCA event or a transient-induced LOCA event, the HHSI system is required to function in the recirculation mode, taking suction from the containment sump. During this mode of operation, the RHR system, including the RHR heat exchanger, is not needed to remove decay heat. Decay heat removal will be accomplished by means of the RCFC, CCW and ECW systems. If the RCFC system should fail, then the reactor primary system depressurization function, the LHSI system, the RHR heat exchanger, the CCW system, and the ECW system are required to function.

3. One train of the LHSI system is sufficient for both coolant injection and recirculation following a large LOCA event. When the recirculation mode of the LHSI system (with RHR heat exchangers) is not available, two of the three RCFC trains (without RHR heat exchangers) are needed to remove decay heat from the containment. Thus, the importance (with respect to reliability requirements) of the LHSI system to remove decay heat following a LOCA event is significant.

4. The staff considered the potential failure of the STP vessel following a failure-to-scram event. For a failure-to-scram event, the PSA has employed generic thermal-hydraulic analyses to establish core cooling success criteria. For example, the PSA assumes that a moderator temperature coefficient of -8 pcm per degree-F could lead to an event that could result in pressurization of the primary system (including the vessel head) to about 2790 psig (following a turbine trip event). The PSA also assumes that the primary system will fail, resulting in a LOCA event, only if the ASME Level C (a pressure equivalent of about 3200 psig) limit is exceeded. Therefore, a vessel failure event should not occur.

3.1.5 Data Analyses

The staff evaluated the adequacy of the data (random failures, common cause failures, test and maintenance unavailabilities) used to quantify the system unreliability estimates and sequence frequency estimates. Overall, the licensee has not developed plant-specific data (in particular, common cause data) for sequence frequency estimation purposes. The STP facility was under operating license (OL) review, and the facility was not licensed to operate, at that time the PSA was performed. Therefore, with the exception of a few components, the licensee has made use of the collection of generic data documented by its contractor, Pickard, Lowe and Garrick (PL&G). The staff review found that the generic data employed in the PSA (Section 7.7 of the PSA) is a mixture of both nuclear operating experience (until 1987) and data collected by the industry, the DOE National Laboratories, recognized professional societies (such as ASTM, ANS, IEEE, and ASME) and the USNRC. During the review process, some of the PL&G database used for the STP PSA were made available for the staff's review. The review primarily involved a comparison of the PSA data (component failures including common cause failures) with those documented in the NUREG-1150 supporting analyses. The results of a summary comparison are documented in Section 3.4 of NUREG/CR-5606. The following is a summary of the major highlights:

1. The PSA has used a total of about 2.6 trips per calendar-year as opposed to 6.6 trips per reactor-year in the NUREG-1150 report. The staff notes that in future PSA updates, actual STP trip data will be considered for inclusion.

2. The frequency of the loss of feedwater events (considering recoveries) is slightly higher than that estimated for the NUREG-1150 report.

3. The PSA's frequency estimate for the SGTR event (2.8E-2 per reactor-year) is higher than that estimated for the NUREG-1150 report. The PSA estimate is based on single tube failure events.

4. The mean check valve failure probability per demand for the PSA is three times higher than that estimated for the NUREG-1150 analyses. The staff finds that this difference is within the range of the NUREG-1150 uncertainty estimates.

5. The fail-to-reclose probability for the STP PORVs is higher than that estimated in the NUREG-1150 analyses.

6. The fail-to-run probability (per hour basis) for the turbine driven AFW pump of the STP facility is lower (by a factor of five) than that of the NUREG-1150 plants. However, the staff finds that the mission time adopted in many dominant sequences is about one to two hours, and therefore, the use of a lower estimate for the AFW turbine-driven pump will not significantly change the estimated frequency of the station blackout sequences.

The estimates used for the components noted above reflect PLG's method of generic data applicability analyses, and the STP-specific design features. When compared to those used in NUREG-1150, the reasons for these variances are acceptable to the staff.

The staff notes that the PSA has made use of the Multiple Greek Letter (MGL) method to quantify the common cause contributions to the three-train system failures (applicable to the STP facility) to the overall system failure. The licensee's discussions related to common cause methods, data classification and screening, and development of the STP-specific MGL parameter distributions are documented in Section 7 of the PSA. The staff notes that the PL&G generic data for individual components (such as diesel generators, pumps, check valves, motor operated valves, PORVs, Safety Relief Valves (SRVs), fans, circuit breakers, level sensors) have been used as the basis for the common cause parameter quantification. The staff also notes that the data from this generic data base has been screened for its applicability to the STP facility. Moreover, the staff believes that the PSA's documentation related to the method of quantifying common cause failures is outstanding. The staff accepts the licensee's method of common cause analyses.

The staff has also evaluated the modeling adequacy of test and maintenance contributions to the overall system unavailabilities. The contributing components which have been modeled are online maintenance, unscheduled maintenance, preventive maintenance, scheduled testing, online testing, unscheduled testing, and testing after maintenance. Again, the PL&G generic data base has been used in screening and categorizing these contributing elements in developing the probability distributions for maintenance frequencies and durations (Section 7.5 of the PSA). The staff notes that the STP specific design features and maintenance policies and procedures have been used in applying the generic data base to estimate maintenance frequencies and durations. The staff also notes that the STP specific technical specifications with respect to allowed outage time and test intervals have also been considered, as applicable, in estimating frequencies and durations. A detailed summary of these frequencies and durations along with the distributions is documented in Table 7.5-1 and 7.5-2 of the PSA. The staff finds that these estimates are reasonable and comparable with those estimates documented in the NUREG-1150 report. The staff accepts the licensee's method of estimating the test and maintenance contributions to the overall system unavailabilities.

3.1.6 <u>Human Reliability Analysis</u>

The staff has evaluated the adequacy of certain aspects of the HRA such methods, assumptions and probability distributions assigned to estimate the human error probabilities documented in the STP PSA. The licensee provided its response to the staff's RAI in the above HRA areas. Basically, the PSA has used the modified SLIM, SHARP, and THERP methods as part of the overall HRA methods. The principles of the SHARP method have been used to identify the critical dynamic human actions, including recovery actions (to be initiated following a transient or a postulated LOCA event), to be modeled as part of the event trees. The PSA has also performed a transient-specific thermal-hydraulic calculation to estimate the steam generator boil-dry time and the RCP seal failure time and factored them into the STP HRA accordingly. The staff also notes the licensee's room heat-up and thermal fragility calculations were performed as part of the loss of HVAC scenarios. The modified SLIM method has been used to quantify the performance shaping factors (PSFs) used in characterizing the attributes of a particular human action to be modeled in the event sequence and the associated uncertainty distributions. The tabulated human error probability estimates and dependency correlations from NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plants Application," have been used to quantify the human actions modeled as part of the system analyses. The staff accepts these enhanced modeling aspects of the HRA analyses. The following is a summary of our technical review findings:

1. The staff evaluated the basis for the probability used for recovery of the offsite power, the initially-lost diesel generators, the turbine-driven AFW pump following a station blackout event, and the initially-lost chillers (prior to the heat-up of the 4.16 KV switchgear) following a loss of HVAC scenario. These probability estimates are comparable to those estimated for the NUREG-1150 analyses. Thus the staff accepts these probability estimates.

2. The staff also evaluated the appropriateness of the application of the miscalibration probability estimate for the Seabrook facility to the STP facility. The licensee's response in this review area included a design comparison analysis of the facility instrumentation hardware, configuration, and calibration procedures for the two facilities. The staff has reviewed this response and has found it to be appropriate for the South Texas facility.

3.1.7 Sequence Quantification

The licensee's discussions related to quantification methods (including the method of crediting scenario-specific operator recovery actions) of all end states of the developed event trees for transients, LOCAs, ATWS events, and SGTR events are documented in Sections 5.5 and 5.6 of the PSA. During the review process, the staff found several disagreements between the table of dominant accident sequences (provided separately) and the system description split fraction quantification in the PRA. The differences were in the areas of AFW train combinations and the diesel train combinations. In response to a staff RAI, the licensee provided clarification on several items, confirmed the staff's assessment on others, and based on further review, identified one additional error. There was no change to the overall core damage frequency (CDF). The licensee committed to include the corrections in the next PSA update. The details of these findings are documented in Item C of Section 3.6 of NUREG/CR-5606. With the exception of a few sequences previously identified, the staff accepts the PL&G's method of quantification of event trees along with their frequency estimates.

The following is a summary of the staff's portrayal of potential core damage sequences:

1. The PSA has developed a large number (millions) of sequences with frequency estimates ranging as low as 1 E-10 per reactor-year. There are about 225 sequences having frequency estimates greater than 1 E-7 per reactor-year, and about 21 sequences having frequency estimates greater than 1 E-6 per reactor-year. Although the licensee identified the 225 sequences to be "dominant," the staff's review involved a focus on only the first 21 of the dominant sequences.

2. The total mean CDF is about 1.7 E-4 per reactor-year. The staff notes that STP is a three-train plant which has been designed and built based on an "N+2" concept for many accident scenarios. The staff also notes that the STP facility has three motor-driven AFW pumps and one turbine-driven AFW pump to remove decay heat.

3. Of the above 21 sequences, 13 sequences are initiated by a loss of offsite power event. Of these 13 sequences, 8 sequences involve station blackout events (4 events with failure of the turbine-driven AFW train, 3 events with RCP seal failures, and 1 event with a stuck-open PORV event). The remaining five loss of offsite power sequences involve combinations of independent failures of the diesel generators, the motor-driven AFW trains, the turbine-driven AFW train, and the ECW trains.

   The relative contribution of the loss of offsite power events to the overall core damage frequency is about 53 percent. The station blackout core damage frequency is about 3 E-5 per reactor-year.

4. Two sequences are initiated by SGTR events for the STP facility. These sequences involve a failure to depressurize the reactor below the steam generator PORV setpoint, and a failure to isolate the stuck-open PORV on the affected steam generator. The SGTR sequence frequency is 2.5 E-6 per reactor-year.

5. Two sequences are initiated by a loss of the Electrical Auxiliary Building (EAB) HVAC system. The failure of the EAB HVAC system is expected to result in a failure of all three trains of the 4.16 KV buses (due to overheating of the 4.16 KV switchgear) which results in a demand for RCP seal cooling by means of the PDP pump and the TSC diesel generator, and a demand for coolant makeup to the steam generator through the turbine-driven AFW train. The frequency estimate of these sequences is about 9 E-6 per reactor-year.

6. There are two sequences initiated by a reactor trip with a combined sequence frequency estimate of 3.7 E-6 per reactor-year. These sequences involve a failure of the secondary side decay heat removal system (four-train AFW system), and a failure to provide primary side decay heat removal by means of feed and bleed operation (through both pressurizer PORVs) in a timely fashion, or a failure to provide long-term stabilization of the plant. The "long term stabilization of the plant" refers to a stable plant state where core decay heat is being removed

through the steam generators, and the steam generator coolant makeup is being accomplished through the AFW storage tank. It should be noted that makeup to the AFW storage water tank is not required until several hours (when the level in the tank falls below a limit of 138,000 gallons) after the reactor trip.

7. There is one sequence initiated by a turbine trip with a sequence frequency estimate of 2 E-6 per reactor-year. This sequence also involves a failure to provide long-term stabilization of the plant.

8. There is one sequence initiated by a partial loss of MFW event with a sequence frequency estimate of 2.2 E-6 per reactor-year. This sequence also involves a failure to provide long-term stabilization of the plant.

3.1.8 Comparison with Results From NUREG-1150

A summary of relative contributions of various initiating events to the overall CDF is shown in Figure 1.

The staff has compared the results of the STP sequences with those of the NUREG-1150 analyses and has concluded the following:

1. Unlike the NUREG-1150 findings, the frequencies of small LOCA sequences (at the STP facility) involving recirculation failures have been estimated to be insignificant and are lower than 1 E-6 per reactor-year. This is primarily due to the fact that, unlike the NUREG-1150 plants, the ECC pumps of the STP facility are self-cooled, and no forced cooling is needed for the ECC pump rooms, and the switchover from injection to recirculation following a postulated LOCA event is an automatic action at the STP facility.

2. The frequency estimates of sequences initiated by other events, such as loss of offsite power, SGTR, reactor trip, turbine trip, loss of MFW, and interfacing LOCA, are closely comparable (within a factor of 2 to 5) to the corresponding estimates documented in the NUREG-1150 analyses. The staff notes that the differences in methods (use of shorter vs longer event trees; Shoreham Contention 7B) of these two probabilistic safety analyses do not play a major role in reaching the above conclusion.

3. Proper risk application of the PRA will obviously quire a thorough understanding of and attention to the first 21 dominant sequences. However, attention should also be paid to understanding the safety details that could be gained from the remaining sequences (a frequency range of greater than 1E-8 per reactor-year). The latter set could be as important as the first in order to characterize a substantial change in core damage frequency, if any, due to future identification of potential defects in component design and installation procedures. The staff also believes that activities for which the STP PSA could be used include: (1) identification of areas for further design and/or operational improvements with respect to substantial reduction in overall core damage frequency (such as IPE and IPEEE activities), (2) review of the licensee's modifications to the current STP technical specifications, and (3) use of

the STP PSA dominant accident sequences in training the STP operators (e.g., updating STP-specific operator training modules to reflect the required recovery actions in responding to critical multiple failures) regarding the role of certain failures of critical systems and equipment and the required recovery actions for transients. The extensive probabilistic and reliability knowledge from the STP PSA could also be used in updating the training simulator.

4. Maintenance of the STP PSA to reflect "current" plant, accommodate future plant-specific experience (revised component failure data and new events or sequences based on U.S. and foreign experience), update research knowledge (e.g., new sequences, new accident phenomena, and new consequence methods) and hardware and procedural modifications will be a valuable tool for the licensee.

3.2 Fire Analysis

The staff has reviewed the applicable portions of the fire analyses documented in the STP PSA and SNL's technical findings on these fire analyses. The licensee's analyses of fire zone-specific combustibles, postulated fire scenarios, fire data analysis, method of screening analysis and its results, and frequency estimates of the screen scenarios are documented in Sections 8, 9, and Appendix D of the STP PSA. As part of the review of these selected fire scenarios, SNL performed a technical review of the fire analyses. In addition to RAIs the staff, with the help of the SNL staff and plant fire protection engineers, conducted a plant walkdown of critical fire zones to obtain first-hand information on the amount of zone-specific combustibles, fuel sources, location of fire detection and suppression systems, and information on the applicability of generic fire data to the STP-specific fire zones. The additional information gathered during the plant walk-down, along with information provided by the licensee in its response to the RAIs, was included in the TER on fire risk review. These fire risk evaluation findings are documented in Section 6 and Appendix 6 of NUREG/CR-5606.

3.2.1 Screening Criteria

The staff evaluated the adequacy of the screening criteria used by the PSA to exclude the frequency estimate of a single zone fire scenario if it exceeded one tenth of one percent of the overall core damage frequency estimated for transients and LOCAs (2 E-7 per reactor-year). This approach was considered necessary by the licensee for the management of the enormous number of fire sequences that could be expected from a fire event tree. After review of the process the staff concluded that use of the screening criteria would not have eliminated from further consideration any significant fire induced contributors to overall core damage frequency.

3.2.2 Adequacy of Overall Fire Analysis Model

The staff evaluated the modeling adequacy of the fire detection and suppression systems along with the assignment of geometry and severity factors used for postulated fires in the various fire zones. Fire zones have been screened based on two levels of screening criteria. The staff has found that, unlike other fire PRAs evaluated, detection and suppression systems located in various fire zones have been implicitly modeled. The PSA has assigned severity factors in order to model the characteristics of the detection and suppression systems for critical fire zones. As part of this evaluation, the staff requested the licensee to perform a sensitivity analysis to remove such severity factors from the fire scenario frequency estimates and to provide results of the sensitivity analysis including the impact on the zone-specific fire sequence frequency estimate for selected fire zones. These sensitivity results by the licensee which are documented in Appendix 6 of NUREG/CR-5606 indicate that the total fire frequency for these critical zones is about 1.5 E-5 per reactor-year. The staff has reviewed these results and found them acceptable.

The total main control room (MCR) fire frequency estimated by the licensee is about 5 E-3 per reactor-year and is based on a systematic documentation of all reported fire data since 1937. The staff accepts this estimate for fire scenario frequency estimation purposes.

The staff evaluated the modeling adequacy of postulated main control room fires at the STP facility. In particular, the staff and its contractor evaluated the modeling adequacy of the propagation characteristics of postulated panel fires, the completeness of the PLG panel fire data, and the appropriateness of the severity factor assigned for panel fires in the control room. The licensee provided its response to the staff's RAI in these review topics. Review of this response indicates that these severity reduction factors range from 0.0023 to 0.028 depending upon the size of the postulated fire and the location of the postulated fire (such as the fire at a panel interface). These reduction factors have been assigned based on a licensee review of the PL&G panel fire data base established for panels located in the MCR, remote shutdown room, and motor control centers. The licensee concluded that the minimum effective damage radius for a postulated fire to cause significant fire damage is about four feet. Staff review has also found that only one out of 16 panel fires, which have actually occurred at various buildings of current operating plants, has spread more than one foot with respect to significant fire-induced damage. The staff notes that this PL&G fire data has been used in developing a propagation characteristics curve which is used to obtain a severity factor for a given propagation distance to be analyzed for a postulated fire. The staff also notes that, during the fire review, 3 additional panel fires (which occurred at Rancho Seco, Calvert Cliffs, and Beaver Valley) have been added to the original 13 panel fires (analyzed in the STP PSA) and have been used to revise the propagation characteristics curve to obtain a revised severity factor. The licensee has also performed a sensitivity analysis of the revised severity factor on the overall control room fire scenario frequency estimate. The results of the above sensitivity analysis indicate that the incorporation of the additional three panel fire data points does not significantly increase the overall fire frequency estimate. The staff has reviewed these results and found them acceptable.

The staff has also evaluated the appropriateness of recovery actions to be performed by the MCR operators following a postulated fire in the MCR. The licensee stated that a fire in the MCR would disable safety system equipment controls and instrument indications which could be restored by a transfer of control and monitoring actions to the auxiliary shutdown panels (ASP) subsequent to a manual reactor trip from the MCR and immediate abandonment of the MCR. Staff review found that credit for the transfer of equipment control and monitoring functions was not taken in the fire scenario frequency estimation method. However, operator recovery of safety system equipment from the ASP (a failure probability of 0.2 per demand) has been modeled in the fire scenario frequency estimation method. The staff reviewed the modeling aspects of the ASP actions and their failure probability assignment and found them acceptable.

3.2.3 Adequacy of Analytical Steps

The staff evaluated the adequacy of the analytical steps involved in the overall fire probabilistic method based on the responses provided to the staff's RAI for the 4.16 KV switchgear (SWG) zone (Zone 4) fires. A description of the analysis is provided in Section 6.4 and Appendix 6 of the TER.

3.2.3.1 Zone Specific Initiating Fire Frequency

The method of estimating zone-specific initiating fire frequency was evaluated by the staff. The overall SWG Zone 4 fire frequency is about 1.4 E-3 per reactor-year (Table 8.5-2 of the STP PSA) and is based on a systematic analysis of plant-specific information. The Zone 4 frequency was estimated by multiplying the allocated frequency (0.048 per reactor-year) for the Mechanical and Electrical Auxiliary Building (MEAB) by a normalized area modification factor estimated for Zone 4. The area factor is characterized by the fraction of floor area of a particular zone (in percent) of the total area of the building which contains that zone. The area factor for Zone 4 is about 1.4 percent. The modification factor is characterized by the occupancy and the traffic pattern of a particular zone. This modification factor for each zone is assigned by the STP fire protection engineers and is based on the assumption that the frequency of a fire in any given zone is mostly influenced by the zone location in a given building and the combustible contents in that zone. The assigned modification factor for Zone 4 is about 1.9. Thus, the normalized area modification factor was computed by dividing the product of area factor and modification factor for Zone 4 by the sum of all the similar products for all zones in MEAB. The staff accepts the Zone 4 fire frequency along with the fire frequency estimate of 1.4 E-3 per reactor-year.

The Zone 122 fire frequency estimate of 2.17 E-3 per reactor-year is one of the larger (4.5 percent) contributors to the overall fire frequency for the MAB but was screened from further consideration during the analysis. Thus, the staff evaluated the basis for the screening of fires in Zone 122 from the Level 3 evaluation. A fire in this zone results in a small LOCA event with a subsequent failure of the "C" train of the CCW system. Since the additional system failures that are modeled as part of the Level 2 screening analysis fall into the Class 2 scenario (an event causing a transient or a LOCA event and one or more failures of trains of a single safety system), a potential core damage event following a Zone 122 fire will also incorporate failure of the remaining trains of the CCW system and the HHSI system trains. Thus, the resulting core damage frequency was estimated as about 1.5 E-7 per reactor-year. Also, the licensee's sensitivity analyses, in which the geometry and severity factors (that are evaluated as part of the Level 3 evaluation stage) for this Zone 122 were removed altogether, indicate that the fire-induced core damage frequency (end state 43) for Zone 122 is about 2 E-6 per reactor-year which is only about 1.2 percent of the overall CDF. The staff finds these analytical steps used in the CDF estimates from fires to be acceptable.

3.2.3.2 Random Failure Contributions

The modeling adequacy of the random failure contributions for Zone 4 fires, as part of the Level 1 and Level 2 screening stages, and the appropriateness of the licensee's assignment of severity reduction factors as part of the Level 3 evaluation stage of the PSA fire risk analysis, were evaluated by the staff. Since the original PSA did not provide sufficient documentation for these review topics, the licensee provided its response, in detail, to the staff's RAI. The frequencies of fire scenarios (end states 11, 12, 15, 16, 19, and 20) for Zone 4 were estimated and compared with the frequency estimates of the corresponding end states of a transient event. At the Level 1 screening stage, the fire scenario end states with frequencies less than one percent of the frequency estimate of the corresponding end state of the transient event, were screened out from further analysis. At the Level 2 screening stage, credit was taken for these safety systems which are not affected by Zone 4 fires, and the fire frequencies for all resulting sequences (end states 11, 12, 15, 16, 19, and 20) were estimated (Tables 4-9 and 4-11 of Appendix 6 of NUREG/CR-5606). Then, the fire sequences (end states) with frequencies less than one tenth of one percent of the total core damage frequency estimate for the transient events (about 1.7 E-7 per reactor-year) were screened out from further analysis. In summary, for Zone 4, at the Level 1 screening, all sequences passed for Level 2 screening analysis. At the Level 2 screening, only end states 11 and 12 passed for Level 3 evaluation.

The Level 3 evaluation took into account the severity reduction factors in addition to credit for the systems unaffected by the Zone 4 fire. The reduction factors reflected the conditional probability of failures of the fire-induced safety system components such as power cables, control cables and circuit breakers of CCW pump A, ECW pump A and AFW pump A; control cables of the PDP; control cables of pressurizer PORV 655A; the control cables and power cables of the pressurizer PORV 655A block valve; and the ventilation fan motor contactors of AFW pump A and CCW pump A. The staff believes that the Level 3 analysis conducted by the licensee for zone 4 is a realistic fire probabilistic analysis. Therefore, the staff accepts the results of the Level 3 evaluation results provided by the licensee for Zone 4 fires. The staff also notes that the severity reduction factors were assigned based on engineering judgment and knowledge obtained from previous fire PRAs.

3.2.3.3 Fire Sequence Frequencies for Cable Spreading Rooms

The staff evaluated critical modeling aspects involved in estimating the zone specific fire sequence frequencies for cable spreading zones/rooms (CSZ). There are three CSZs (Zone 47, Zone 57, and Zone 60) evaluated in detail for the STP facility. The total fire frequency for all these zones is about 2.4

E-3 per reactor-year. A large portion of this estimate (1.07E-3 per reactor-year) has been allocated to Zone 47. These estimates are based on a systematic documentation of all reported events (cable fires, panel fires, and transient combustible fires) for a typical auxiliary or reactor building of a nuclear power facility since 1987. The total fire frequency estimated by the STP PSA for the auxiliary building is about 4.8 E-2 per reactor-year. This frequency was then partitioned according to the area and the occupancy and traffic characteristics of each of the three CSZs zones indicated above. The staff noted that such a method of estimating fire initiating frequency data is different from past fire risk analysis practices in that past fire PRAs have estimated the fire initiating frequency for a CSZ based on reported data for the auxiliary building alone. As part of the resolution of this data modeling issue, the licensee provided, in response to the staff's RAI, the results of a sensitivity analysis, specifically the impact on the overall core damage frequency of increasing (by a factor of 10) the CS zone fire frequency. The impact was found to be an insignificant increase over the originally estimated overall fire sequence frequency estimate. The staff accepts the results.

The modeling aspects of additional failures of the systems and components (unaffected by the postulated Zone 47 fire) modeled as part of the level 2 screening analysis were evaluated by the staff and were found to be acceptable to the staff.

The adequacy of the licensee's assignment of reduction factors modeled as part of the Level 3 screening analysis was evaluated by the staff. As indicated in previous paragraphs, the licensee also performed a sensitivity analysis by removing altogether the geometry and severity factors as part of estimating the fire-induced core damage frequency for the CSZ fires. The results of this sensitivity analysis found that the Zone 47 fire yielded a total core damage frequency estimate (1.34 E-6 per reactor-year) calculated for four end states (53, 59, 66, and 72). This is considered a bounding estimate of fire-induced core damage frequency. The staff accepts these results.

3.2.3.4 <u>Fire Sequence Frequencies for Turbine Building</u>

The details of the turbine building (TB) fires and their significance on the overall fire sequence frequency estimates were provided in the licensee's response to the staff's RAI in this review area. Based on these responses, the staff evaluated critical modeling aspects involved in estimating the fire sequence frequencies for turbine building fires and fires in the 13.8 KV switchgear room. The total TB fire frequency involving a non-recoverable loss of offsite power is about 2.23 E-3 per reactor-year. The staff notes that this frequency estimate consists of large TB fires and 13.8 KV switchgear room fires.

The TB large fire frequency is about 2 E-3 per reactor-year. This estimate was based on the allocated TB fire frequency (0.047 events per year during plant operation) and one large fire event assigned to the TB of the STP facility out of a total of 23 TB fire events, which have been reported for various nuclear facilities in the United States and Europe. The PSA has characterized 13 fires out of 23 TB fire events that involved a main turbine-generator located in the TB of a typical nuclear power plant. However, only one of the TB fire events has been characterized as a fire event large enough to potentially disable both the 125 V DC control power cables from the TB battery bus and the 125 V DC control cables of the EAB battery bus. The large fire event is one that could result from a rupture of a hydraulic oil line located in a typical TB. Since these control power cables control the switchgears of 13.8 KV buses F, G, and H, given a large fire in the TB, a loss of offsite power event could occur with no timely recovery. The assigned conditional probability is about 0.043 per demand. The staff accepts the licensee's frequency estimate of the large TB fire.

The 13.8 KV switchgear room fire frequency is about 1.9 E-4 per reactor-year. This estimate was based on the allocated mean TB fire frequency (0.047 events per year) and an adjustment factor assigned on a Bayesian estimate (a prior distribution of zero TB switchgear fire events out of 23 TB fire events) for the fraction of the TB fires that occurred in a TB switchgear. The fraction of the TB fires applicable for the switchgear room is estimated to be about 0.04. The adjustment factor takes into account an assumption that about ten percent of the total TB fires could result in damage to all three 13.8 KV buses. The above ten percent assignment is based on the fact that the switchgear cabinets of buses F, G and H are widely separated apart. The staff accepts the licensee's frequency estimate of the 13.8 KV switch gear room fire.

As part of the Level 1 screening analysis, an estimate for the frequency of a non-recoverable loss of offsite power event was obtained (see pages 309 thru 312 of the TER). This estimate is about 0.046 per year. Since the total TB fire frequency (2.2 E-3 per year) involving a non-recoverable loss of offsite power event is greater than one percent of the corresponding transient event frequency (0.01*0.046 = 4.6 E-4 per year), the licensee further evaluated the TB fire scenario analysis as part of the Level 2 screening analysis. Level 2 screening considered the dominant additional system failures that must occur before core damage as well as an independent failure of the 138 KV emergency line (referred to as The Blessing Line). After including the additional failures, the TB fire-induced core damage frequency estimate was about 3 E-7 per reactor-year. Since this estimate is greater than one tenth of one percent of the transient-induced core damage frequency (about 2 E-7 per reactor-year), a Level 3 evaluation would normally have been conducted by the licensee. However, the licensee stated that the Level 2 evaluation included some very conservative assumptions. Therefore, a Level 3 evaluation, which calls for incorporation of reduction factors of the affected systems in the TB fire sequences, was not performed. The staff accepts the licensee's analysis for concluding the TB fires to be insignificant contributors to the CDF and for not considering them further.

3.2.4 Staff Observations of The South Texas PSA

1. The STP facility is a three-train plant and includes physical separations for the safety system components and cable routings. The barriers (walls, ceilings, floors, curtains, doors, and penetrations) separating the critical fire zones have been built to withstand a three-hour fire in all fire zones analyzed in the STP PSA. The staff's review of the STP fire

PSA indicates that the removal of credit for the suppression and detection systems could result in an increase of about 2 E-5 per reactor-year in the overall core damage frequency.

2. The MCR for the STP facility has been designed and built such that a fire in any given control panel would be detected in a timely fashion by the smoke detectors placed near the intake to the HVAC system inside the enclosed control panel housing. Also, to a great extent, separation between controls in a given panel has been provided. For postulated MCR fires, operation from the remote ASP, in addition to its defined functions, was found to be very useful, including the establishment of seal cooling by means of the TSC diesel generator and the PDP system.

3. The staff notes the licensee's assigned estimates of the fire scenario-specific severity reduction factors modeled for various safety systems and components in the STP fire PSA and acknowledges that they are one-of-a-kind analyses. However, these factors are not based on a formal fire engineering type analysis, but based on engineering judgments by thermal experts.

4. A zone-specific walk-down by the staff indicates that the sprinkler and spray nozzles are located above the cable trays and have fusible links. The fire suppression system of the STP facility does not make use of carbon dioxide as a suppression agent.

5. While the estimate of the TB fire-induced core damage frequency is about 3 E-7 per reactor-year, the staff also notes that the STP facility has an additional offsite AC power line (138 KV) that will not be damaged by the postulated TB fires. This additional 138 KV line is also expected to provide independent AC power to the in-plant safety system equipment following a TB fire scenario. The staff believes that the above site feature is unique to the STP facility, and has resulted in a significant reduction in overall fire core damage frequency estimates.

4.0 Conclusions

The questions raised by the staff and SNL have been satisfactorily addressed by the licensee. Staff review has identified both modeling optimism and pessimism in the PSA. Overall, the modeling optimism and pessimism have been found to have negligible impact on the PSA's estimate of the overall core damage frequency. The PSA has estimated the overall mean core damage frequency to be 1.7 E-4 per reactor-year. This frequency estimate is well within the range of core damage frequency estimates provided for similar Westinghouse PWR facilities. However, it should be noted that, unlike other PWR facilities, the STP facility has been built to operate based on an "N+2" concept (except for postulated pipe break events and station blackout events). The mean core damage frequency (1.7 E-4 per reactor-year) estimated for the STP facility is based on significant separation (both electrically and mechanically) of the three safety system trains (N+2 concept) for each unit of the two unit facility. However, the small difference between the core damage frequency estimate for the STP facility and that for other similar Westinghouse plants is primarily due to the fact that the station blackout sequences at the STP facility have been found to have only an "N+1" protection as is the case for other PWR facilities.

4.1 Internal Events

The staff has reviewed the methods, data, and assumptions of the STP PSA, along with the licensee's response to the staff's questions during the PSA review. The review results indicate that there is no unique outlier that contributes significantly (a single sequence exceeding well above 1E-4 per reactor-year) to the overall mean core damage frequency. This is primarily due to the additional redundancy for the safety systems and the provisions for separation of various safety systems which have been built into the facility.

However, one sequence out of many hundreds of sequences (a total of 225 sequences that have a frequency estimate greater than 1E-7 per reactor-year) has been estimated to have a frequency estimate of more than 1E-5 per reactor-year. This sequence is a station blackout sequence involving failure of the turbine-driven train of the AFW system. It should be noted that, because of this sequence, the facility has been found not to have an "N+2" protection. This is the major reason that the STP facility has the same core damage frequency estimate that could be expected for a plant built on a "N+1" concept. However, the staff does not consider a frequency estimate of 1E-5 for any decay heat removal sequence as an outlier for any licensed nuclear facility in the U.S.

There are 21 sequences that have been estimated to have a frequency of more than 1E-6 per reactor-year. These 21 sequences and the remainder of thousands of sequences that have a frequency estimate of greater than 1E-8 per reactor-year collectively yielded a core damage frequency of more than 1E-4 per reactor-year. The remainder of sequences (those with frequency estimates of less than 1E-8 per reactor year) contributed approximately 7E-5 per reactor year to the overall CDF. This is a normally expected estimate for a typical Westinghouse PWR plant that has been built and licensed to operate on an "N+1" concept. This level of CDF can be expected from normal random failures of safety system components.

The staff accepts the PSA's logical minimum number of safety system trains needed to prevent a core damage event following a transient or a LOCA event. It also accepts, for accident management purposes, the steam generator boil-dry time, and the seal failure time as calculated during the PSA review.

4.2 Fire Analysis

Based on its review of the zone-specific details related to fire screening methods, fire data analysis, associated assumptions, and fire protection features built into the STP facility along with the licensee's response to the staff's three sets of RAIs during the fire review process, the staff concludes that the frequency of a single fire sequence at the STP facility is not expected to exceed an estimate of about 2E-7 per reactor-year. The staff also concludes that, on the basis of a conservative analysis, the frequency of a single fire sequence at the STP facility is not expected to exceed an estimate of about 2E-6 per reactor-year. The conservative fire analysis includes the impact of removing the geometry and severity factors used to model the propagation and suppression characteristics, from a probabilistic fire model. It is believed that the conservative fire analysis provides an estimate on the uncertainty range of core damage frequencies resulting from fires. The staff's review of the zone specific fire analysis supports this conclusion and, therefore, the screening-type fire analysis of the STP PSA is acceptable to the staff with the exception of improvements in documentation in certain review topics as indicated in the licensee's response to the staff's RAI.

The application of various screening criteria is useful only to assure that the frequency of a single fire sequence will not exceed an estimate of 2 E-7 per reactor-year at the STP facility and not in determining the total fire contribution to the overall core damage frequency estimate and its uncertainty. However, the staff believes that the fire risk analysis based on a well-thought-out screening criterion (such as the STP fire PSA screening criterion) is useful in identifying and/or exploring significant accident vulnerabilities within the limited review resources.

The staff also concludes that the licensee's overall fire analysis is acceptable to demonstrate that a postulated fire in any given fire zone will not result in a core damage event with a frequency which will exceed, on a realistic basis, one percent of the overall transient-induced frequency estimate. This statement is based on the zone-specific and realistic fire analysis, and a systematic comparison of the above results with the corresponding transient-induced sequence analysis results. This statement is also based on the fact that the licensee's fire analysis reflects the location of the safety system equipment and the cable routing, which are based on a three train (including three separate fire zones for each safety system) design concept. The only exceptions to the above are the fire zones located in the MCR and the TB, for which detailed location-dependent analyses have been performed to demonstrate that the fires are inconsequential contributors to the core damage frequency. There is no unique design feature that contributes a substantial increase in the overall core damage frequency due to the postulated zone-specific fires. However, the unique design feature related to the way of routing of the offsite AC power supply cables to the TB of the STP facility is found to have significantly reduced the overall fire-induced core damage frequency.

Attachments:
1. Technical Evaluation Report
2. Figure

Principal Contributor:
E. Chelliah

Date: January 21, 1992

South Texas Facility

Total Core Damage Frequency: 1.7E-4 /RY

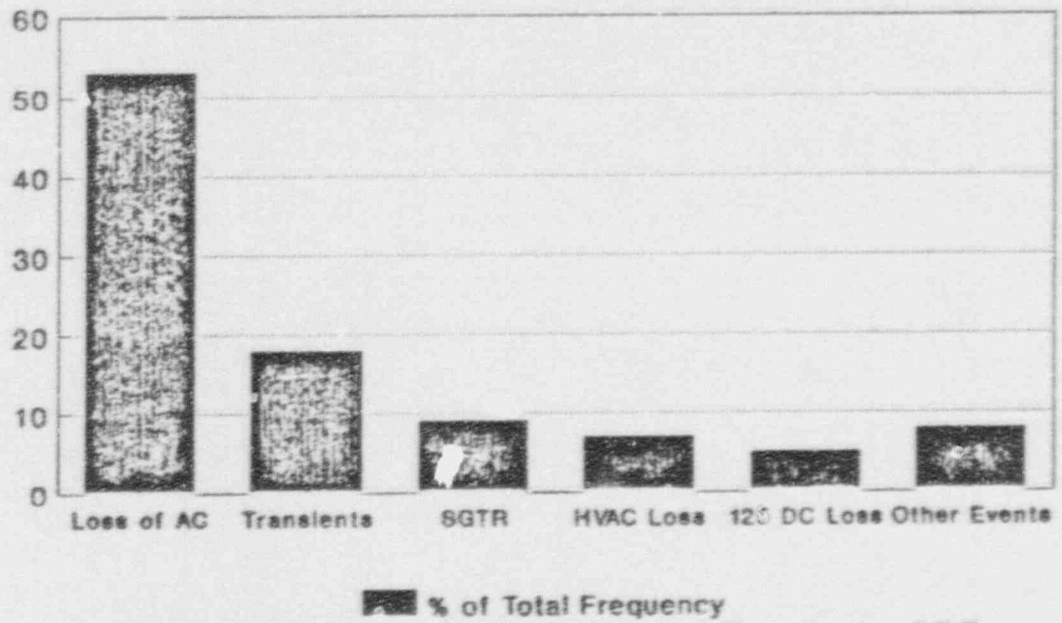

Figure 1: A Relative Contribution of Events to CDF

Seismic Events contribute only 1%.

South Texas Facility

Total Core Damage Frequency: 1.7E-4 /RY

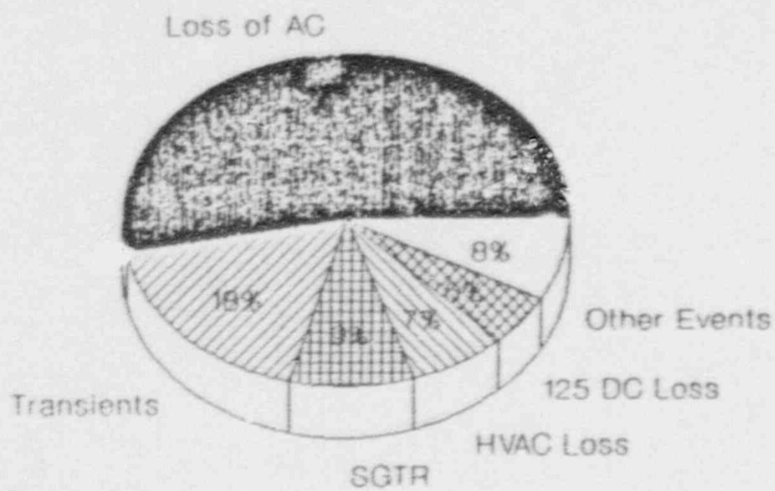

Figure 1: A Relative Contribution of Events to CDF

Seismic Events Contribute only 1%.