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April 11, 1990
ST-HL-AE-3414
File No: G20.1
10CFR50.12

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project Electric Generating Station
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Responses to the
Request for Additional Information from
Sandia National Laboratory

Reference: (1) Letter from M. A. McBurnett to the U. S. Nuclear Regulatory Commission dated March 1, 1990. (ST-HL-AE-3380)

Enclosed are responses to questions raised by Sandia National Laboratory (SNL) regarding the South Texas Project Electric Generating Station (STPEGS) Probabilistic Safety Assessment (PSA) fire analysis. The responses to Questions Q1 and Q2 were transmitted in Reference 1. These responses have been supplemented with additional comments and are therefore resubmitted. The response to Question Q3 related to the dominant fire scenario frequency screening criteria and to Question Q4 related to the internal events frequency screening criteria are attached. This submittal completes HL&P's responses to these questions.

If you should have any questions on this matter, or the attachments, please contact Mr. A. W. Harrison at (512) 972-7298 or myself at (512) 972-8530.

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SDP/hg

Attachment: Response to Questions Q1 through Q4
regarding the STPEGS PSA

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Revised 12/15/89

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Responses to Questions Q1 Through Q4 from Sandia National Laboratory Regarding the STPEGS PSA

Q1: One of the screening criteria employed was that if only one of three safety trains was in a fire area, then this area was screened from further analysis. However, at Peach Bottom the two most dominant fire areas had only one of three safety trains. Each of these areas was two orders of magnitude higher than the dominant fire scenario at STP. In light of the Peach Bottom results, please list which areas were screened by this step and list what safety systems or their associated cabling are present.

Response:

In accordance with Section 8 (Spatial Interactions Analysis) of the South Texas Project Electric Generating Station (STPEGS) Probabilistic Safety Assessment (PSA), Subsection 8.5.3 (Scenario Impact Evaluation) the only areas screened from any quantitative review are areas in which events do not effect any system and do not cause any initiating event in the PSA. The following discussion provides additional clarification of the Spatial Interactions Analysis which was performed.

The STPEGS PSA utilizes a spatial interactions screening analysis as the basis for the fire analysis performed in the PSA. The Spatial Interactions Analysis is described in Section 8 of the PSA. This spatial interactions analysis (SIA) identifies locations in the plant which correspond with the fire zones identified in the STPEGS Fire Hazard Analysis Report (FHAR). Each zone is associated with a fire frequency and a specific inventory including equipment, components, control cable, power cable, other hazard sources, and mitigative features. These areas are then considered as potential fire locations which define scenarios requiring evaluation. These scenarios are summarized in Appendix D, Table D-6, in volumes 6, 7 and 8 of the PSA.

In order to perform the evaluation, each scenario is assigned to one or more of four classes (Class 0, 1, 2 or 3), and then further identified as meeting one or more of ten guidelines which specifies the basis for initial screening. These classes and criteria are defined in Section 8, pp. 8.5-3&4 of the PSA. The class and applicable guidelines for each scenario (Items 10 & 11) are identified in Table D-6. It is also indicated in this table, based on the application of the guidelines, whether further quantitative screening (i.e., beyond the guidelines) is to be performed (Item 9).

Class 1, 2 or 3 scenarios were subjected to initial quantitative screening per the applicable guidelines. Class 2 includes all scenarios which affect one or more trains of a single system only (for those systems which are modelled in the PSA). Only Class 0 scenarios ("scenario does not affect any system and does not cause any initiating event in the plant model") are ruled out from further consideration (per guideline 1, "if a scenario is in Class 0, its further study is not warranted for purposes of risk assessment.")

Supplemental Comments on Q1:

Comparison to the Peach Bottom plant is inappropriate. The Peach Bottom units are BWRs which were constructed in the late sixties and early seventies and went into commercial operation in 1973 and 1974. The South Texas Project units are state-of-the-art PWRs completed in 1988 and 1989 having a combination of redundancy and physical separation which makes direct comparison to other plants inappropriate.

In the case of the STPEGS three-independent-train safety system design, a fire at STPEGS which affects one train and does not cause a plant trip would put the plant into a state which could be compared to recently licensed PWRs which have two trains and which are in normal operation. A fire at STPEGS which disables a single train and which causes a plant trip could be compared to those same PWRs after a turbine trip. Since a fire initiating event frequency is approximately three orders of magnitude lower than a turbine trip, single-train fire scenarios are not an issue for STPEGS. In any case, these events are compared to the frequency of a similar system state from random failures, and if significant (i.e., more than one or two percent) they may be added to the system unavailability frequency. A review of the plant level results (i.e., sequences) provides confidence that this screening is acceptable. To analyze each single train fire scenario in detail would result in a high level of effort without commensurate value being added to the analyses.

Q2: The most dominant scenario was in the control room. However, the methodology employed in the quantification varies substantially from past PL&G fire PRAs and also is at variance with testing results from large scale enclosure tests. In past PL&G fire PRAs, the control room has been assumed to be abandoned and control of the plant is taken from the remote shutdown panel. Sandia sponsored large scale enclosure tests have shown that cabinet fires generate such intense smoke that within 6-8 minutes control of the plant from the control room would be virtually impossible. These tests were conducted with control room ventilation rates of up to ten room changes per hour. Therefore, the most likely scenario would be smoke-forced abandonment of control room and subsequent control of the plant from the remote shutdown panel. If the remote shutdown panel is truly independent of the control room, then it makes no difference whatsoever where the fire originated because all initial potential damage to safety controls would be bypassed. Please explain why STP is either at variance in control room design from past PL&G PRAs or what other factors led the analysts to modify their previous methodology. Using the past methodology for control room analysis would have the effect of increasing core damage frequency estimates by a factor of approximately fifty.

Response:

Several factors have influenced the approach taken in the STPEGS PSA to the control room fire analysis. Factors which influenced this approach include a more detailed focus on the modelling of external events such as fires in the control room, an expanded data base for control room fire events such as that utilized in the fire analysis performed on the Surry plant for NUREG-1150, and the impact of the STPEGS independent three-train design on the consequences of fires.

Past PRAs have focused more on the internally-initiated event analysis due to the greater interdependency of systems design in older plants than the independent three-train design of STPEGS. As a consequence, the approach taken in previous PL&G fire PRAs has been more conservative in assuming abandonment of the control room in the case of a fire while concluding that even in such case, fire-induced core damage is a relatively small contributor (on the order of 10% plus or minus).

The STPEGS PSA fire analysis assumes a mean initiating event frequency of 4.9E-3 for control room fires. This frequency is taken from a paper by M. Kazarians and G. Apostolakis ("Modeling Rare Events: The Frequencies of Fires in Nuclear Power Plants," June 1982). This control room fire frequency is based on a single event which occurred during shutdown at Three Mile Island in 1979. The fire analysis completed for NUREG-1150 for the Surry

Power Station uses an initiating event frequency of 1.8E-3 (NUREG/CR-4550, "NUREG-1150 External Event Risk Analyses: Surry Power Station," September 1989, Table 5.5), a factor of approximately 3 lower than that used in the STPEGS PSA. This control room fire frequency is based on four events between 1978 and early 1983, including the Three Mile Island event (NUREG-4550, Appendix E, p. E-9). None of the four control room fires in the data base lead to the abandonment of the control room. NUREG-4550 assumes that 1 of 10 control room fires leads to abandonment of the control room (see Section 5.10.4 of NUREG-4550).

The STPEGS control room design is such that a fire on a control panel would be quickly detected by smoke detectors placed near the intake to the CR HVAC system inside the enclosed control panel housing. Separation is provided between panels and to a great extent between controls on the same panel. The fire would be extinguished quickly because of the detection and HVAC design and because the control room is continuously manned. NUREG-4550 also takes credit for a factor of 10 reduction in control room fire frequency because of continuous occupation (Section 5.10.4 of NUREG-4550). STPEGS has not taken this credit.

At STP, transfer of control to the auxiliary shutdown panel (ASP) provides control of safe shutdown equipment independent of the control room. A fire in the control room would disable equipment controls which would be restored by transfer to the ASP. The assumption in the STPEGS fire analysis does not take credit for transfer to the ASP since the equipment controls disabled by the control room fire represent the more limiting condition in terms of equipment available for plant shutdown.

Supplemental Comments on Q2:

The point made by the SNL reviewer that smoke from a fire in the control room is an important factor which may limit actions taken by an operator in the control room is a good one. It is also true that the abandonment of the control room is not explicitly modelled in the fire analysis. However, the analysis which is performed allows for operator actions in such a general and conservative way that plant control from the ASP or a local control panel would be an implicit alternative.

For example, Scenarios 2 through 6 (see pp. 9.4-6 through 9.4-10) consider various fires affecting loss of Component Cooling Water (CCW) and/or Essential Cooling Water (ECW). In each case, in order to restore cooling water to the Reactor Coolant Pump (RCP) seals, the use of the ASP was considered to restore the CCW/ECW function (in these sections, the term "hot shutdown panels" was used to refer to the ASP). The unlikelihood of restoration of CCW/ECW in these cases was 1.4E-2. This function could also be restored from a local control panel.

The SNL reviewer observes that "if the remote shutdown panel is truly independent of the control room, then it makes no difference whatsoever where the fire originated because all initial potential damage to safety controls would be bypassed". This cannot be the case even if the remote shutdown panel is independent, as is STPEGS's, since the location of the fire would influence the precise impact on the plant, timing of the scenario and time dependent indications to the operator.

For STPEGS, the ASP is located within the same building on a lower level which could be reached in a timely manner. Procedures provide for shift of control to the ASP in the event the control room becomes uninhabitable. Operator training and demonstrations provide confidence that the operators will effectively and efficiently take control from the ASP in order to shut the plant down. Cold-shutdown can be achieved from the ASP.

Of the 23 fire scenarios considered for the control room, other than the 5 referred to above, all assume failure of unspecified recovery actions by the operators with a likelihood of 0.2. This value is considered very conservative (i.e., high) as evidenced by the value of 1.4E-2 for the 5 discussed above for action taken from the ASP. The unspecified actions could include failure to take additional action in the control room and failure to take control of the plant from the ASP. If this were the case, and no additional recovery actions were taken from either the control room or the ASP, which is highly unlikely, then all of the fire results listed in Table 9.4-3 would be considered as the final fire results. In this case, the total fire induced core damage frequency would be approximately 2.5E-6, or about 1.5% of the CDF.

Q3: The dominant fire scenario frequency was approximately 1.0E-7 per year. One screening criteria to eliminate fire areas was at a frequency of 2.0E-7 per year. I feel it is inappropriate to set screening levels above the ultimate total fire-induced core damage frequency. Please list which fire areas were eliminated by this consideration and what safety equipment they contain.

Response: Fire areas are not screened by application of this criteria.

Supplemental Comments on Q3:

The comparison of the screening value of 2.0E-7 to 1.0E-7 as the "ultimate total fire-induced core damage frequency" is incorrect. The total core damage frequency resulting from fire-initiated events is approximately 5.06E-7, which is 0.3% of the total STPEGS estimated core damage frequency (CDF) of 1.67E-4. Thus, the screening criteria of 2.0E-7 is below the total core damage frequency due to fires. This total for fires is due to two fire scenarios, including 4 sequences, all of which occur in the control room. Fires in other locations were determined to be insignificant contributors to CDF.

The value of 0.3% as the percentage contribution of fires to STPEGS CDF was previously provided to SNL at the meetings held in STPEGS offices on November 28-30, 1989. This was in response to a question regarding the core damage frequency resulting from fires at the meeting with NRC and SNL personnel in Albuquerque on August 8, 1989. In addition, HL&P provided information regarding the dominant sequence at STPEGS due to a fire.

One correction should be noted to the information provided to the NRC and SNL at the November meeting (these meeting minutes have not been issued by the NRC at this time, so no reference is provided). The dominant fire sequence due to fires is approximately 1.9E-7 per year or approximately 0.1% of total CDF as previously indicated. However, the dominant sequence is as shown in Table 1. Table 1 also includes the sequence previously provided which is actually the third fire sequence in magnitude.

For additional discussion related to this question, see the section "Additional Comments" below.

Q4: Another screening criteria was to eliminate fire areas of 10% of internal events frequency for a similar end state. Once again, this has the potential for elimination of fire areas with contributions to core damage greater than the ultimate dominant scenario. Please list what fire areas were eliminated in this step and what safety equipment they contain.

<u>Response:</u> Fire areas are not screened by application of this criteria.

Additional Comments:

In regards to the methodology and reporting employed in the fire analysis:

- o Insufficient documentation exists in the report to do an adequate review of results of methodology employed.

- o Screening criteria are non-conservative and have the potential to dismiss relatively dominant (when compared to total fire-induced core damage frequency) fire areas.

- o The control room analysis does not appear to have used past PRA and fire testing insights and, therefore, may have substantially underestimated core damage frequency.

Response:

o <u>Insufficient Documentation</u>

HL&P has submitted documentation to support the review of the PSA in accordance with the guidance given in GL 88-20. It is true that most of the actual calculations performed to establish the contribution to core damage are not reproduced in the South Texas Project Electric Generating Station (STPEGS) Probabilistic Safety Assessment (PSA). The PSA as it currently stands is very voluminous (27 volumes), and it was never the intent to include the calculation details. The methodology is described in the Sections 8 and 9 as discussed below. The actual calculations, consisting of numerous volumes and computer runs, were shown and identified to SNL personnel and were available for review by SNL personnel during the plant visit on November 28-30. At that time SNL personnel indicated that it was not necessary to review this documentation. HL&P believes that the documentation in the PSA provides the information required to answer the questions regarding methodology and to provide the details which have been addressed to the HL&P to date.

The documentation of the fire analysis and the results of the methodology employed is extensively documented in the STPEGS PSA. Table D-6 in volumes 6, 7 & 8 of the PSA catalogs and summarizes, among other events considered, all fire scenarios considered in the fire analysis. Each scenario lists the location, initiating event frequency, potentially affected equipment and components, additional factors affecting propagation, classes and categories which specifies the basis for screening, and the result of the initial quantitative screening process. The methodology utilized for the fire screening analysis is completely stated in Sections 8 (Spatial Interactions Analysis) and 9 (Internal Fires Analysis) with detailed examples of each which are in fact the dominant scenarios.

If after review of this information you determine that the actual calculations must be reviewed, HL&P requests that you return to the STP site to review the material.

o <u>Screening Criteria are Non-conservative</u>

HL&P considers that the screening criteria used are conservative and that the use of these criteria will identify any significant fire sequences which are similar in magnitude to the (already small) total fire-induced core damage frequency. Based on the four questions provided in the letter which conveyed these general comments (i.e., Sandia National Laboratory to Nuclear Regulatory Commission dated January 3, 1990), the following discussion assumes that this concern relates directly to those questions.

With regard to Q1, the only fire <u>areas</u> screened <u>without any quantitative evaluation</u> are addressed in Section 8 and are areas which do not effect any system and do not cause any initiating event in the PSA (i.e., Class 0 scenarios. See p. 8.5-3). <u>All other fire areas are quantitatively evaluated.</u> Tables 8.6-1 through 8.6-6 (pp. 8.6-3 through 8.6-24 inclusively) summarize the scenarios cataloged in Table D-6 for all types of events evaluated using the spatial interaction approach, including fires. Table 8.6-7 summarizes the results of the initial quantitative evaluation using the quantitative criteria stated in Section 8 (p.8.5-4 and p.8.5.-5). Application of the criteria to Tables 8.6-1 through 8.6-6 to produce Table 8.6-7 is straight forward. (Note: There are a few omissions from Table 8.6-7 not involving fire scenarios which were evaluated separately and found to be unimportant.) Each of these tables state the general impact of the event on the plant. Reference to Table D-6 in volumes 6, 7 & 8 provide the specific equipment effected. The fire scenarios included in Table 8.6-7 are then evaluated in Section 9.

The analyses in Section 9 of the PSA apply to specific equipment states which result from event trees developed and quantified as described in this section for each fire scenario. The use of screening criteria in this section apply to specific individual sequences resulting from the event tree quantification. <u>Fire areas are not screened in this section</u>; individual sequences representing specific equipment failure states are developed, quantified and evaluated. With regard to Question 3, the screening criteria referred to is used in Section 9.3.3 ("Step 3 - Second Level of Screening", pp. 9.3-4 and 9.3-5) and applies to sequences, not to fire areas. With regard to Question 4, the screening criteria referred to is used in Section 9.3.2 ("Step 2 - Event Tree Quantification and First Level of Screening", pp. 9.3-2 through 9.3-4) and applies to sequences, not to fire areas. The object of the screening by the application of these criteria is a specific sequence or equipment state, not a fire area. The application of the screening criteria is considered acceptable since, in each case, they are applied to specific sequences, not fire areas, and additional equipment failures and/or failure of operator actions must occur before core damage results. For example, with regard to Question 3, a screening criteria of 2.0E-7 as applied to a specific sequence which by itself does not lead to core damage is reasonable, even though the dominant fire sequence frequency is approximately 1.9E-7. Fires only contribute approximately 0.3% of core damage frequency and the sequences being screened by this criteria are less than approximately 0.1% of CDF and do not lead by themselves to core damage.

o Control Room Analysis Does Not Use Past PRA and Fire Testing Insights

The STPEGS fire analysis does use past PRA and fire testing insights. The response to Question 2 addresses this concern as it applies to the STPEGS PSA.

A PRA utilizes plant experience to the extent it is available to estimate the likelihood of events. The data for fires in control rooms, although sparse, does not support the contention that any fire in the control room leads to abandonment. To the contrary, of the four minor fire events in the data base for control rooms, including one in 1979 at Three Mile Island when the plant was in a shut-down condition, all occurred in 1983 or earlier and no fire led to abandonment of the control room. The requirements of 10 CFR 50 Appendix R, "Fire Protection", went into effect in February 1981 and the implementation of its requirements since that time would be expected to favorably influence the unlikelihood of fires in nuclear power facilities. The STPEGS control room fire analysis assumes an initiating event frequency of 4.9E-3 based on this early experience rather than the less-conse_vative frequency of 1.8E-3 used in the NUREG-1150 control room fire analysis for the Surry plant.

SNL has conducted fire experiments which indicate that cabinet fires generate such intense smoke that within 6-8 minutes control of the plant from the control room would be virtually impossible. While this may occur, experience indicates that such fires are rare, and in fact have not happened. Even in the recent fire at the Vandellos plant in Spain where intense smoke entered the control room from a fire outside the control room (an oil fire lasting several hours in the turbine building in which the control room is located), operators were not forced to abandon the control room. In fact, the NUREG-1150 external events analysis for the Surry plant which was performed in part by SNL personnel (NUREG-4550, "NUREG-1150 External Event Risk Analyses: Surry Power Station", Section 10.5.4), assumed that only 1 of 10 control room fires lead to the abandonment of the control room.

Concern has been expressed that the STPEGS control room fire analysis did not assume the control room was abandoned in the event of any fire. Abandonment of the STPEGS control room would result in transfer of control to the auxiliary shutdown panel (ASP). A fire in the control room would disable equipment controls which would be restored by transfer to the ASP. All three trains of safety systems at STPEGS are controlled from the ASP, not just a single pathway as specified in Appendix R. The assumption in the STPEGS fire analysis does not take credit for transfer to the ASP since the equipment controls disabled by the control room fire represent the more limiting condition in terms of equipment available for plant shutdown and therefore is conservative.

Table 1
Summary of Sequences Initiated By Fire

Sequence	Frequency	Description
1	1.913E-7	FR18*AFR*(Success Terms) FR18 = 2.100E-6 (see PSA p.9.4-18). Control Room fire, Scenario 18, initiating event. Fire disables EAB/CR HVAC controls. AFR = 1.096E-1 (see PSA p.5.5-77). AFW train D fails. Success Terms = 8.312E-1 (see Note 1).
2	1.445E-7	FR18*PDH*(Success Terms) FR18 = 2.100E-6 (see PSA p.9.4-18). Control Room fire, Scenario 18, initiating event. Fire disables EAB/CR HVAC controls. PDH = 9.297E-2 (see PSA p.5.5-78). Failure of positive displacement pump given no charging and all support available. Success Terms = 7.401E-1 (see Note 1).
3 (Note 2)	9.949E-8	FR18*ORM*(1-CPC) FR18 = 2.100E-6 (see PSA p.9.4-18). Control Room fire, Scenario 18, initiating event. Fire disables EAB/CR HVAC controls. ORM = 6.161E-2 (see PSA p.5.5-8). Operator fails to start a train of HVAC having no automatic start signal. CPC = 2.31E-1 (see PSA p.5.5-8). No support available (1-CPC means support is available).
4	5.058E-8	FR23*OBA*(Success Terms) FR23 = 1.600E-6 (see PSA pp.9.4-17,18). Control Room fire, Scenario 23, initiating event. Fire disables all four trains of AFW. OBA = 4.802E-2 (see PSA p. 5.5-79). Operators open 2/2 PORVs for bleed and feed. Success Terms = 6.583E-1 (see Note 1).

Note 1: The frequency for successful operation of the remaining systems is not shown, but is included in the total sequence frequency.

Note 2: Previously provided to NRC and SNL personnel as the "Top Ranking Fire Event".