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

May 9, 2007
NOC-AE-07002163
10CFR50.36
10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information on
Risk-Informed Technical Specification Amendment Request

References: 1. Letter from David W. Rencurrel to NRC Document Control Desk dated December 28, 2006, "Revised Broad Scope Risk-Informed Technical Specification Amendment Request" (ML070040247, NOC-AE-06002036)
2. Letter from Charles T. Bowman to NRC Document Control Desk dated February 28, 2007, "Response to NRC Requests for Additional Information on STPNOC Proposed Risk Managed Technical Specifications" (ML070670369, NOC-AE-07002112)
3. TAC Nos. MD 2341 & MD 2342

In References 1 and 2, the STP Nuclear Operating Company (STPNOC) submitted a revised license amendment request for a broad scope risk-informed set of Technical Specification (TS) changes and a response to a NRC request for additional information (RAI). This submittal responds to additional NRC questions.

Attachment 1 responds to NRC questions on the STP application.

Attachment 2 is the insert for the TS Bases.

There are no commitments in this submittal.

STI: 32157373

A001

If you have any questions, please call Wayne Harrison at 361-972-7298 or me at 361-972-7454.

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

Executed on May 9, 2007.
Date

Charles T. Bowman

C. T. Bowman
General Manager, Oversight

Attachments:
1. Response to NRC Request for Additional Information
2. Bases Inserts

cc:
(paper copy)

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

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

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

Richard A. Ratliff
Bureau of Radiation Control
Texas Department of State Health Services
1100 West 49th Street
Austin, TX 78756-3189

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

(electronic copy)

A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP

Mohan C. Thadani
U. S. Nuclear Regulatory Commission

Thad Hill
Eddy Daniels
Marty Ryan
Harry Holloway
Steve Winn
NRG South Texas LP

Ed Alarcon
J. J. Nesrsta
R. K. Temple
Kevin Pollo
City Public Service

Jon C. Wood
Cox Smith Matthews

C. Kirksey
City of Austin

# Response to NRC
# Request for Additional Information

# Attachment 1

## Response to Request for Additional Information

**1. The revised 6.8.3.k for the Configuration Risk Management Program (CRMP) identifies changes in core damage frequency (CDF), large early release frequency (LERF), or cumulative core damage probability. In fact, NEI 06-09 assesses risk-informed completion times (RICTs) based on cumulative core damage probability (CDP) and cumulative large early release probability (LERP). Similarly, the TS bases identifies only the incremental core damage probability (ICDP) limit explicitly. The TS and bases need to correctly refer to the appropriate risk metrics used in NEI 06-09.**

**In addition, the staff believes the 30-day backstop should be referenced in TS 6.8.3.k.**

Response: See the revised TS 6.8.3.k below and the attached Bases inserts. The changes to TS 6.8.3.k will be also be submitted separately as a supplement to the license amendment request.

k. Configuration Risk Management Program (CRMP)

~~A program to assess changes in core damage frequency and cumulative core damage probability resulting from applicable plant configurations. The program should include the following:~~ **A program to calculate risk-informed completion time in accordance with NEI 06-09, "Risk-Managed Technical Specifications (RMTS) Guidelines, Rev. 0". The CRMP may be used for calculating a risk-informed completion time only in Mode 1 and Mode 2. In accordance with NEI 06-09, the completion time determined using the CRMP shall not be more than 30 days.**

~~1) Training of personnel~~

~~2) procedures for identifying the generation of risk profiles and the evaluation of risk against established thresholds, and~~

~~3) provisions for evaluating changes in risk resulting from unplanned maintenance activities;~~

**2. For conditions which previously would have been 3.0.3 entries, the bases of each actions should be modified to reflect that the action is not to be voluntarily applied for operational convenience, consistent with the current 3.0.3 bases.**

Response: In accordance with a telephone discussion with the staff, STPNOC understands that the intended limitation on voluntary entry is for conditions where there is a loss of function. See the attached inserts for the Bases for Application of Risk Informed Completion Times.

**3. During our site visit in June 2006, the staff identified some suggestions with regards to the conduct of the uncertainty evaluation, specifically:**

- **Generate functional event trees to identify the scenarios that involve cross system or cross train combinations of equipment out of service as a means of identifying the key uncertainties that need to be addressed.**

  Response: The review of core damage and large early release sequences and Risk Assessment Calculator (RAsCal) results for asymmetries and irregularities is part of the process for evaluating uncertainty in the PRA (see the response to question 5). Functional event trees that examine cross-system or cross-train combinations of equipment out of service were not used.

- **Consider a sensitivity of LERF to the assumption that steam generator tube rupture (SGTR) scenarios do not bin to LERF.**

  Response: The sensitivity of Large Early Release to an Induced SGTR is included as a standard sensitivity case in the graded quality assurance (GQA) risk-ranking process. Spontaneous SGTR sequences contribute approximately 3% to LERF. The question relates to a peer review finding associated with an assumption in the Level 2 model concerning late core damage sequences in the SGTR with an unisolated release path (e.g., a steam generator safety valve). This peer review finding was specifically addressed in the Level 2 analysis update. MAAP analyses support the late timing of core damage with an unisolated steam generator assuming no credit for refueling water storage tank (RWST) refill. Because the time of core damage is greater than 24 hours, and Emergency Response procedures will already have recommended evacuation/sheltering, no sensitivity analysis of this issue was performed.

- **Consider whether new data from EPRI fire tests for hot shorts would change fire frequencies enough to affect the RICT results.**

  Response: Evaluation of new data from the EPRI tests for hot shorts is planned for a future reanalysis of fire hazards at STP and was not considered in the uncertainty evaluation.

- **Consider whether adopting the latest fire frequency data (NUREG/CR-6580) would affect RICT results.**

  Response: Review of the latest fire frequency data presented in NUREG/CR-6850 is planned as part of a future reanalysis of fire hazards at STP. A brief review of the data indicates that control room fire scenarios may decrease in frequency (4.9E-03 to 2.5E-03). This slight decrease does not significantly affect RICT results. Because of the detail provided in NUREG/CR-6850, fire frequency in other areas of the Electrical Auxiliary Building (EAB) may be unaffected, or affected slightly (i.e., small time increment when compared to a 24 hour day). The fire re-screening analysis performed to support the RMTS submittal, and the limited data review of NUREG/CR-6850 indicate that

incorporation of this data may have a slight effect on RICT results, but without performing compartment level analysis as part of a detailed fire reanalysis, the exact effect is unknown.

**How were these suggestions dealt with in developing the uncertainty evaluation?**

Response: The responses to the items above include statements as to whether or not such information is used. Some of the discussion items above offer additional "completeness" in an uncertainty evaluation; however, their impacts are not considered to be large to the point that decision making resulting from a RICT calculation would be altered from a compensatory measures perspective. Also, the impact to a RICT from a quantitative perspective is also considered to be small with respect to the overall time increment usually used for Technical Specification allowed outage times.

**4. Attachment 3 refers to a Table 3 which was not provided. What are the most significant human error probabilities (HEPs) and which RICTs do they impact and how? The point here would be to demonstrate that the RICTs that are affected by these HEPs are either well in excess of 30 days, or that the HEPs were somewhat conservative.**

Response:

The most significant operator actions in the PRA are provided below. Sensitivity analyses on the dominant operator actions were performed as part of the uncertainty evaluation.

Risk Important Operator Actions

| Description of Failed Operator Action | Operator Action Basic Event ID | Basic Event CDF RAW | Impacted PRA Equipment |
|---|---|---|---|
| Operator initiates smoke purge for loss of EAB HVAC cooling event | VE_OPER_OSA | 10.0 | • Essential Chill Water<br>• Essential Cooling Water |
| Operator places residual heat removal (RHR) in service for SGTR event | RH_HUMA_ERR_0001 | 7.1 | • Steam Generator Isolation (main steam isolation valves, steam generator power-operated relief valves, steam generator safety relief valves)<br>• RHR<br>• Component Cooling Water to RHR Hx |
| Operator manually trips reactor for ATWS event | RT_OPER_OTA | 4.1 | • 1E 120 VAC Inverters<br>• Rx trip breakers<br>• Solid State Protection Logic |
| Operator starts positive displacement pump (PDP) for loss of reactor coolant pump (RCP) seal cooling event | CV_HUMA_ERR_0003 | 2.7 | • EAB HVAC<br>• SI Recirculation<br>• Essential Chill Water<br>• SI Room Coolers<br>• HHSI pumps |

| Description of Failed Operator Action | Operator Action Basic Event ID | Basic Event CDF RAW | Impacted PRA Equipment |
|---|---|---|---|
| Operator manually starts auxiliary feedwater (AFW) when ESFAS fails for control room fire event | SS_OPER_ORM | 2.7 | • CRE HVAC<br>• SG SRVs |
| Operator trips RCP on loss of RCP motor cooling event | CV_HUMA_ERR_001B | 2.7 | • Seal Injection valves<br>• EAB HVAC<br>• SI Recirculation<br>• Essential Chill Water<br>• Essential Cooling Water |

The PRA equipment identified above have notable increases in basic event importance when sensitivity models are quantified with the associated operator action HEP increased by a factor of 10. Variation in the operator action HEP values can potentially affect the equipment RICTs. Assuming single train unavailability for the above equipment, the associated RICTs are substantially greater than 30 days. When two or more trains of the equipment are unavailable, the RICT can be shorter than 30 days and thus variation in HEP values could impact the RICT. The important modeled operator actions described above are proceduralized and the associated HEPs have reasonable values such that they are not considered a key source of uncertainty.

**5. The process described by STP for conduct of the uncertainty evaluation during the site visit included a number of good features. Was this the process which was used?**

Response:

Yes, in general the process that we used was what was discussed in the NRC's site visit in June 2006. The general process is again described below.

Basically, we perform GQA sensitivity calculations which investigate several of the traditional sources of uncertainty. Prior to signing off the model, we investigate initiating event contribution, review sequence reasonableness to a fairly deep level, and confirm consistent results (no unidentified asymmetries). Cross-comparisons within RAsCal results are also used to identify asymmetries. Dominant initiators are investigated in more detail to determine reasonableness. Traditional PWR core damage sequences and large early release sequences are reviewed against comparable Westinghouse PWR results and against the cross-comparison performed by the Pressurized Water Reactor Owners Group (PWROG) in support of Mitigating Systems Performance Index (MSPI). The STP PRA sequences for internal events are consistent with other similar Westinghouse four loop PWRs given our system differences, four train AFW, three train SI, in general no cross-ties, etc. Our results are significantly different in the inclusion of site-specific external events and in the site-specific support system initiating events. These issues are included as part of the uncertainty calculation. One of the proposed sensitivity cases for the support system initiating events, investigation of the common cause failure fractions (Multiple-Greek-Letter methodology) was not performed as other sensitivity calculations (e.g., Loss of EAB HVAC) indicated the importance of the assumptions associated with the modeling.

**Follow-on from May 3, 2007 telecon: Please confirm that in the uncertainty analysis, the focus of the search for uncertainties was on those RICTs that were 30 days or less, and that none of these was significantly affected by a key uncertainty.**

The STP PRA uncertainty analysis focused on sources of uncertainty in the PRA without consideration of a specific application. After the significant sources of uncertainty were identified, the effects of the uncertainties were reviewed to ensure the effects are adequately addressed in the various risk-informed applications.

For RMTS, the key sources of uncertainty were reviewed against the RICTs to ensure no RICT less than 30 days was significantly affected by the key sources of uncertainty. Single train configurations are unaffected by uncertainty considerations. Cross-train and multi-train configurations are affected by the key sources of uncertainty identified in the uncertainty analysis; however, the dominant causes of uncertainty in the PRA are considered conservatively modeled which results in a lower calculated (i.e., more restrictive) RICT for a given configuration.

# Bases Inserts
# (For Information Only)

# Attachment 2

Bases for Application of Risk-Informed Completion Times

The Configuration Risk Management Program (CRMP) as described in TS 6.8.3.k and associated reference to NEI 06-09, "Risk Managed Technical Specifications Guidelines," establishes provisions for performing a risk assessment to determine required actions and allowed outage times for specifically identified specifications for structures, systems, and components. Application of the risk assessment is consistent with the requirements of the Maintenance Rule, 10CFR50.65(a)(4), to assess and manage the increase in risk that may result from maintenance activities. The process to manage the risk assesses the rate of accumulation of risk in plant configurations and determines the allowed outage time (AOT) by calculating the time required to cross the incremental core damage probability threshold of 1.0E-05 or the incremental large early release probability threshold of 1.0E-06.

The CRMP and TS 6.8.3.k establish a backstop AOT of 30 days. This backstop AOT prevents allowing a component with little or no risk significance from being inoperable indefinitely and resulting in a de facto change to the design or licensing basis of the plant.

Application of the risk assessment to manage allowed outage time in different plant configurations is complemented by the station's programs to monitor performance indicators for long-term availability of risk-significant components. The requirement to achieve acceptable long-term performance indicators provides a significant disincentive against extending baseline AOTs to the detriment of component or system availability.

The CRMP as described in TS 6.8.3.k and associated reference to NEI 06-09 establishes the conditions for performance of the risk assessment. The LCOs subject to the CRMP specifically reference the CRMP. The baseline AOT or required completion time specified in the LCO may be used to apply the CRMP to determine an alternate AOT and risk management actions.

The CRMP may be used for calculating a risk-informed completion time (RICT) only in Mode 1 and Mode 2. If a MODE is entered where the PRA cannot be applied, the risk-informed provisions of the CRMP may not be applied and only the remaining frontstop completion time for the subject TS shall apply from the time of the MODE change. If the affected TS frontstop has been exceeded, then the applicable MODE 3 – MODE 5 transition times shall apply from the time MODE 3 was entered (e.g., "..be in COLD SHUTDOWN in the next 30 hours").

Although the CRMP may be applied to extend the allowed outage time for a referencing TS, except for the extension in the allowed outage time, the other requirements of the referencing TS continue to apply. For instance, if the CRMP is applied to extend the allowed outage time for Train A ECW (TS 3.7.4.a), the provisions of TS 3.7.4.b. will apply if another ECW train becomes inoperable.

TS 3.0.2 applies to the RMTS ACTION statement allowance to calculate a risk-informed completion time (RICT). If the component is restored or if a RICT is calculated in accordance with the ACTION before a required shutdown is completed, the shutdown need not be completed.

The requirement to continually determine the acceptability of the plant configuration means that once the subject LCO has exceeded the frontstop AOT, the risk assessment must be reperformed to determine the need to adjust the required action and time limits for any affected TS component based on the risk associated with any CRMP component that subsequently

becomes inoperable or non-functional. This requirement provides assurance that the configuration risk is adequately assessed.

The CRMP is applied with the referencing specification and the ACTION required by the referencing specification must be taken if the configuration risk exceeds the 1E-05 incremental core damage probability risk threshold or the 1E-06 incremental large early release probability risk threshold. The CRMP recognizes that the plant is in an extended AOT that has a specified required action if the required action time is exceeded. In a configuration where the risk exceeds one of these thresholds, the calculated RICT has been exceeded and the action required at the expiration of the LCO AOT must be taken.

Application of the CRMP will provide action for conditions where more than one train or channel of a function is inoperable. In accordance with NEI 06-09, a RICT may not be applied for configurations where there is a complete loss of function or for pre-planned activities when all trains of equipment required by the TS LCO would be inoperable. It is permissible to apply a RICT for emergent conditions where all trains of equipment required by the LCO are inoperable provided one or more of the trains are functional as described in the guidance.

If a component is determined to be inoperable, it may still be considered to have PRA Functionality for calculation of a RICT if there is reasonable assurance that it can perform its required functions for events not affected by the degraded or non-conforming condition and if the condition can be quantified in the PRA. If these conditions are not met, the component will be assumed to be non-functional for calculating the RICT; i.e., it will have no PRA Functionality.

Examples of where a component has PRA Functionality such that the condition could be quantified in the determination of an allowed outage time are listed below:

- SSCs that don't meet seismic requirements but are otherwise capable of performing their design function.
- SSCs that are inoperable but secured in their safe position (e.g., a closed containment isolation valve).
- SSCs powered from a source other than their normal power source, provided the alternate power source is modeled in the PRA.
- An SSC with an inoperable automatic function if the manual actuation of the SSC is modeled in the PRA (e.g., a diesel generator with an inoperable sequencer). Actuation channels are associated with their actuated components or trains. Loss of actuation channels is not considered a Loss of Function unless no train of the actuated SSC function has PRA Functionality.
- An SSC that is functional for mitigation of a set of events (e.g. steam generator tube rupture, small break LOCA) but is not functional for other events for which it is credited (e.g. large break LOCA or steam line break), providing the PRA model can quantify the risk for the calculation of a RICT. An example of this type of condition is degradation of environmental qualification.

Reference 1 specifies the criteria for determining functionality.

Reference:

1. NEI 06-09 "Risk Managed Technical Specifications Guidelines"