PMSTPCOL PEmails

From: Elton, Loree [leelton@STPEGS.COM]
Sent: Thursday, December 03, 2009 4:48 PM
To: Muniz, Adrian; Dyer, Linda; Wunder, George; Tonacci, Mark; Eudy, Michael; Plisco, Loren; Anand, Raj; Foster, Rocky; Joseph, Stacy; Govan, Tekia; Tai, Tom
Subject: Transmittal of U7-C-STP-NRC-090216
Attachments: U7-C-STP-NRC-090216.pdf

Please find attached a courtesy copy of letter number U7-C-STP-NRC-090216, which contains supplemental and revised responses to previously submitted NRC staff questions included in Request for Additional Information (RAI) letters related to Combined License Application (COLA) Part 2, Tier 2 Chapter 19.

The official version of this correspondence was signed today and will be placed in today's mail. Please call me if you have any questions concerning this letter.

Thank you,

Loree Elton

Licensing, STP 3 & 4
leelton@stpegs.com
361-972-4644

Hearing Identifier: SouthTexas34Public_EX
Email Number: 1942

Mail Envelope Properties (C7F098E3C31A0141A02043F0B8E656EE25648BDD6A)

Subject: Transmittal of U7-C-STP-NRC-090216
Sent Date: 12/3/2009 4:47:48 PM
Received Date: 12/3/2009 4:47:59 PM
From: Elton, Loree

Created By: leelton@STPEGS.COM

Recipients:
"Muniz, Adrian" <Adrian.Muniz@nrc.gov>
Tracking Status: None
"Dyer, Linda" <lcdyer@STPEGS.COM>
Tracking Status: None
"Wunder, George" <George.Wunder@nrc.gov>
Tracking Status: None
"Tonacci, Mark" <Mark.Tonacci@nrc.gov>
Tracking Status: None
"Eudy, Michael" <Michael.Eudy@nrc.gov>
Tracking Status: None
"Plisco, Loren" <Loren.Plisco@nrc.gov>
Tracking Status: None
"Anand, Raj" <Raj.Anand@nrc.gov>
Tracking Status: None
"Foster, Rocky" <Rocky.Foster@nrc.gov>
Tracking Status: None
"Joseph, Stacy" <Stacy.Joseph@nrc.gov>
Tracking Status: None
"Govan, Tekia" <Tekia.Govan@nrc.gov>
Tracking Status: None
"Tai, Tom" <Tom.Tai@nrc.gov>
Tracking Status: None

Post Office: exgmb1.CORP.STPEGS.NET

Files	**Size**	**Date & Time**
MESSAGE	618	12/3/2009 4:47:59 PM
U7-C-STP-NRC-090216.pdf	182009	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

STP Nuclear Operating Company

South Texas Project Electric Generating Station 4000 Avenue F – Suite A Bay City, Texas 77414

December 3, 2009
U7-C-STP-NRC-090216

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

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Response to Request for Additional Information

Attached are supplemental responses and revised responses to previously submitted NRC staff questions included in Request for Additional Information (RAI) letters related to Combined License Application (COLA) Part 2, Tier 2 Chapter 19. The attachments contain the supplemental or revised responses for the following RAI questions.

19-7, Supplement	19.01-29, Supplement
19-16, Supplement	19-11, Revised Response
19-23, Supplement	19-13, Revised Response
19.01-13, Supplement	19.01-1, Revised Response
19.01-23, Supplement	

When a change to the COLA is indicated, the change will be incorporated into the next routine revision of the COLA following NRC acceptance of the RAI response.

There are no new commitments in this letter.

If you have any questions regarding these RAI responses, please contact Scott Head at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

STI 32584943

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

Executed on 12/3/09

Mark McBurnett
Vice President,
Oversight & Regulatory Affairs
South Texas Project Units 3 & 4

dws

Attachments:

1. Question 19-7, Supplement
2. Question 19-11, Revised Response
3. Question 19-13, Revised Response
4. Question 19-16, Supplement
5. Question 19-23, Supplement
6. Question 19.01-1, Revised Response
7. Question 19.01-13, Supplement
8. Question 19.01-23, Supplement
9. Question 19.01-29, Supplement

cc: w/o attachment except*
(paper copy)
Director, Office of New Reactors
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

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

Kathy C. Perkins, RN, MBA
Assistant Commissioner
Division for Regulatory Services
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

Alice Hamilton Rogers, P.E.
Inspection Unit Manager
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

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

*Steven P. Frantz, Esquire
A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Ave. NW
Washington D.C. 20004

*George F. Wunder
*Michael Eudy
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

(electronic copy)
*George Wunder
*Michael Eudy
Loren R. Plisco
U. S. Nuclear Regulatory Commission

Steve Winn
Joseph Kiwak
Eli Smith
Nuclear Innovation North America

Jon C. Wood, Esquire
Cox Smith Matthews

J. J. Nesrsta
R. K. Temple
Kevin Pollo
L. D. Blaylock
CPS Energy

RAI 19-7

QUESTION

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

SUPPLEMENT TO RESPONSE

The FSAR Table 19.2-2 will be modified in a future update to the COLA to reflect the impact on Core Damage Frequency (CDF) as a result of Departure STP DEP 1.1-2.

Departure Number	Certified Design Basis(DCD)	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP 1.1-2 Dual Units at STP 3 and 4	Single Unit site..	Dual Unit site with common fire protection system.	No significant effect on CDF, no change to the PRA, editorial changes for fire protection system [See Chapter 19I.3.1, 19L.8, 19M.6.3, 19Q.4.4]

RAI 19-11

QUESTION:

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

RESPONSE:

The departure clarifies the STP 3&4 containment spray logic design by 1) emphasizing that the LPFL mode has precedence over the containment spray below reactor vessel water Level 1, 2) clarifying initiation of drywell and wetwell sprays, and 3) clarifying interlocks associated with RHR operation. The logic changes for the wetwell spray valves and suppression pool return valves are made with no change in the DCD-required functional or safety requirements.

The containment spray function was not specifically modeled in the internal events PRA prepared to support the DCD. No credit was taken for the containment spray function in evaluating the core damage frequency. The Level 2 portion of the PRA takes credit for the spray function in evaluating the radioactive release consequences (categories and their frequencies). In this evaluation, the spray function is modeled with an operator action, but the control and logic associated with the spray function is not modeled. Therefore, this departure will not change the results of the PRA.

No COLA revision is required as a result of this RAI response.

REVISED RESPONSE

The second paragraph in the response to RAI 19-11 is revised and replaced as follows:

> ~~The containment spray function was not specifically modeled in the internal events PRA prepared to support the DCD. No credit was taken for the containment spray function in evaluating the core damage frequency. The Level 2 portion of the PRA takes credit for the spray function in evaluating the radioactive release consequences (categories and their frequencies). In this evaluation, the spray function is modeled with an operator action, but the control and logic associated with the spray function is not modeled. Therefore, this departure will not change the results of the PRA.~~
>
> The containment spray function was modeled in the Level 2 portion of the internal events PRA prepared to support the Design Control Document (DCD). The Level 2 portion of the PRA takes credit for the spray function in evaluating the radioactive release consequences (categories and their frequencies). In the Level 2 PRA, the spray function is modeled with an operator action, but the control and logic associated with the spray function is not modeled. The Standard Safety Analysis Report, Appendix 19D.6.3.3 and Figure 19D.6-8 provides details of the system model. Appendix 19E.2 of the DCD provides additional details concerning the use of the containment spray function in the ABWR PRA. Because this departure does not affect the containment spray function modeled in the ABWR PRA, this departure does not change the containment spray model or the results of the PRA.

No COLA revision is required as a result of this RAI response.

RAI 19-13

QUESTION

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

RESPONSE

The safety/relief solenoid valves are included in the PRA described by the DCD however the testing of the safety/relief valves described in Section 7.3.1.1.1.2 (g) and modified by STD DEP 7.3-16 is not included in the PRA described by the DCD. The DCD testing restriction stated:

> *The pilot solenoid valves can be tested when the reactor is not pressurized.*

The reactor pressure restriction is removed with STD DEP 7.3-16.

The ability to perform testing during plant operation enhances the ability to schedule and perform planned and preventive maintenance which leads to improved equipment reliability and reduced on-line unavailability. This is the potential benefit identified in Table 19.2-2.

No COLA revision is required as a result of this RAI response.

REVISED RESPONSE

The third paragraph of the response to RAI 19-13 is revised and replaced as shown below, and a new paragraph added:

> ~~The ability to perform testing during plant operation enhances the ability to schedule and perform planned and preventive maintenance which leads to improved equipment reliability and reduced on-line unavailability. This is the potential benefit identified in Table 19.2-2.~~
>
> The ability to perform testing during plant operation enhances the ability to schedule and perform planned and preventive maintenance which leads to improved equipment reliability and reduced on-line unavailability. This is the potential benefit identified in Table 19.2-2, for the plant-specific PRA required to support plant operation in accordance with 10CFR50.71(h).
>
> There is no change to the PRA described in the ABWR Design Control Document (DCD) as a result of this Departure.

No COLA revision is required as a result of this revised RAI response.

RAI 19-16

QUESTION

STD DEP 7.7-1, RPV Water Level Instrumentation

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

SUPPLEMENT TO RESPONSE

This response supplements the previous response to this Request for Additional Information.

The Design Control Document (DCD), Section 1A.2.16 discusses the ABWR design in response to NRC Staff concerns about the potential for reactor pressure vessel water level measurement errors resulting from dissolved non-condensable gases in the water column in the reactor pressure vessel reference leg water level instrument lines (NRC Generic Letter 92-04 and NRC Information Notice 93-27). Continuous purging of the reactor pressure vessel reference leg water level instrument lines using Control Rod Drive (CRD) System is implemented in the ABWR design.

Standard Departure STD DEP 7.7-1 consists of editorial changes to the description of this function in Section 7.7.1.1 as follows:

(6) Reactor Vessel Water Level

(e) Reactor Well Water Level Range

The concern that non-condensable gasses may build-up in the water column in the reactor vessel reference leg water level instrument lines, i.e., the reactor vessel instrument lines at the elevation near the main steam line nozzles, has been addressed by continually flushing these instrument lines with water supplied by the Control Rod Drive (CRD) System <u>for those instrument lines with a condensing chamber. This applies to (a) through (d) above</u>.

Section 4.6.1.2 and 15B.2.3 of the reference ABWR DCD were modified as part of this Departure to clarify that the Control Rod Drive Hydraulic System (CRDHS) supplies purge flow for the Nuclear Boiler System instrument lines.

Generic Letter 92-04, Information Notice 93-27, and NRC Bulletin 93-03 describe the concern as a result of observed degassing in the BWR reference legs during cooldown and depressurization of operating BWRs. Once the plant enters MODE 4, and continuing into MODE 5, the reactor is sub-cooled (≤93ºC) and fully depressurized. As a result, during MODE 4 and MODE 5, degassing in the reactor vessel reference legs is not of concern, and there is no need for continued supply of reference leg purge from the CRDHS. The Shutdown Level and Reactor Well indications (LI 605 and LI 604 respectively) do not require purge flow from the CRDHS.

A review of operating BWR requirements for similar systems and functions indicates that the Reference Leg Backfill System is required in MODES 1 through 3. Once the reactor is depressurized, the backfill system is no longer required to operate. During startup, upon entry into MODE 3, the reference leg purge is reestablished. If CRD pumps stop running during operation in MODES 1 or 2, the reactor is tripped and the plant is in MODE 3. If the CRD pumps stop operating in MODE 3, the restoration time for reference leg flow unavailable to more than one instrument line are typically 7 days based on a review of operating BWR documents.

The Shutdown PRA described in the DCD is unaffected by this departure, which provides only editorial changes and clarifications.

Table 19.2-2 of the FSAR will be modified as described below to clarify this Departure and the effect on the Shutdown PRA.

Departure Number	Certified Design Basis(DCD)	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.7-1 RPV Water Level Instrumentation	All instrument lines are flushed, ~~even when they do not need to be~~ including those without a condensing chamber.	~~Condensable gas build-up in reactor vessel reference leg water level instrument lines is addressed by using CRD water to continually flush~~Clarified that instrument lines having condensing chambers are the only ones with continuous flushing from the CRD System.	~~Not explicitly modeled in the PRA. Potential beneficial effect for plant-specific PRA.~~ Clarification to text only. No effect on the Operating or Shutdown PRAs.

RAI 19-23

QUESTION

The statements in Section 19N of the STP FSAR, Revision 2, do not appear to be consistent with the statements in Section 19N of the ABWR DCD ("Analysis of Common-Cause Failure of Multiplex Equipment") in accordance with departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture). The inconsistencies are as follows:

1) Section 19N.5 of the ABWR DCD states "The effects of EMUX CCF are included in the quantification of core damage frequency in the internal events analysis of Appendix 19D. Additional discussion is given herein to provide further information and insight into the nature of EMUX CCF contribution to core damage frequency." In section 19N.5 of the STP FSAR, Revision 2, it appears that the term CCF is missing in this statement under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

2) Figure 19N-4 of the ABWR DCD states: "AUTOMATIC INITIATION THRU EMUX". Figure 19N-4 of the STP FSAR, Revision 2, does not appear to address this statement for departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

3) Section 19N.3 of the ABWR DCD states: "(10) To reduce the probability of spurious initiation of ECCS, two SLUs are used in parallel within a division, with 2/2 voting at the final channel output to initiate equipment actuation." In section 19N.3 of the STP FSAR, Revision 2, the term SLU appears to be used incorrectly in this statement under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

4) Section 19N.3 of the ABWR DCD states: "(12) Control room indications, annunications, and alarms associated with EMUX transmitted control signals are dependent on correct operation of EMUXs." Section 19N.3 of the STP FSAR, Revision 2, does not include this statement under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

5) Section 19N.4.5 of the ABWR DCD states: "Only the analog-to-digital converters of the RMUs require calibration." In Section 19N.4.5 of the STP FSAR, Revision 2, the term ECFs appears to be used incorrectly in this statement under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

6) Section 19N.5.1 of the ABWR DCD states: "If there were sufficient experience data for multiple failures of solid-state multiplexing equipment, the experience data would be used directly and there would be no need for use of the beta-factor model. However, there is a dearth of multiple-failure data pertaining to solid-state multiplexer equipment, particularly equipment with a self-test feature. The alternative is to evaluate or estimate the relative susceptibility of the EMUX to multi-divisional failures through use of the beta-factor. A recent report by the Electric Power Research Institute (EPRI) (Reference 19N-1) discusses the beta-factor model and lists representative values for beta. The values listed generally range from 0.1 down to about 0.01, but there is no value given specifically for solid-state multiplexing equipment." Section 19N.5.1 of the STP FSAR,

Revision 2, does not appear to address this statement for departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

7) Section 19N.5.1 of the ABWR DCD states: "The random unavailability of the RMUs and TLUs is derived from an expected mean time between failures (MTBF) and a mean time to detect and repair a failure (MTTR)." In section 19N.5.1 of the STP FSAR, Revision 2, the term TLU appears to be used incorrectly under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture) and Figure 7.9S-1 (Data Communication Interfaces).

8) In section 19N.3, Items (1), (3), (6), (8), and (9) of the STP FSAR, Revision 2, the terms DTU and TLU appear to be used incorrectly under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture) and Figure 7.9S-1 (Data Communication Interfaces).

The staff requests that the applicant address the above inconsistencies and revise Section 19N of the STP FSAR, as necessary.

SUPPLEMENT TO RESPONSE

The response provided to this RAI are supplemented with the following information as a result of changes associated with Combined Operating License Application (COLA) Revision 3.

QUESTION

6. Section 19N.5.1 of the ABWR DCD states: "If there were sufficient experience data for multiple failures of solid-state multiplexing equipment, the experience data would be used directly and there would be no need for use of the beta-factor model. However, there is a dearth of multiple-failure data pertaining to solid-state multiplexer equipment, particularly equipment with a self-test feature. The alternative is to evaluate or estimate the relative susceptibility of the EMUX to multi-divisional failures through use of the beta-factor. A recent report by the Electric Power Research Institute (EPRI) (Reference 19N-1) discusses the beta-factor model and lists representative values for beta. The values listed generally range from 0.1 down to about 0.01, but there is no value given specifically for solid-state multiplexing equipment." Section 19N.5.1 of the STP FSAR, Revision 2, does not appear to address this statement for departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture).

RESPONSE

Revision 3 to the STP 3&4 COLA corrected the language in Section 19N.5-1 to read:

"If there were sufficient experience data for multiple failures of solid-state ~~*multiplexing*~~ <u>digital communications</u> *equipment, the experience data would be used directly and there would be no need for use of the beta-factor model. However, there is a dearth of multiple-failure data pertaining to* ~~*solid-state multiplexer*~~ <u>such</u> *equipment, particularly equipment*

with a self-test feature. The alternative is to evaluate or estimate the relative susceptibility of the ~~EMUX~~ ECF *to multi-divisional failures through use of the beta-factor.*

A recent report by the Electric Power Research Institute (EPRI) (Reference 19N-1) discusses the beta-factor model and lists representative values for beta. The values listed generally range from 0.1 down to about 0.01, but there is no value given specifically for solid-state ~~multiplexing~~ digital communications *equipment."*

QUESTION

8. In section 19N.3, Items (1), (3), (6), (8), and (9) of the STP FSAR, Revision 2, the terms DTU and TLU appear to be used incorrectly under departure STD DEP T1 3.4-1 (Safety-Related I&C Architecture) and Figure 7.9S-1 (Data Communication Interfaces).

RESPONSE

Revision 3 to the STP 3&4 COLA corrected the language in the identified Sections of 19N.3 to read:

"(1) There is complete separation of ~~RMUs, DTMs~~ RDLCs, *Digital* Trip Units Function (DTF) components, DLCs (performing the Safety Logic Function (SLF)), *Trip Logic ~~Units (TLU)~~* Function (TLF) components*, sensors and ECCS actuators, etc., between the four safety divisions of control and instrumentation."*

"(3) There is separation of ~~DTM~~ DTF *and ~~TLU modules~~* TLF components *within a division along the lines of "~~de-energize~~* deenergize *to operate" and "energize to operate" functions, i.e., RPS, and MSIV signals are processed by different ~~DTM~~* DTF *and ~~TLU~~* TLF *modules than the ~~DTM and SLU~~* DTU and DLC *modules used for ECCS control and PCV isolation (PCV isolation is also ~~deenergize~~* deenergize*-to-operate). The RPS/MSIV process channel is "deenergize to operate", while the ESF process channel is predominantly "energize to operate."*

"(6) Comparison of a sensed input to a setpoint for generating a trip is done by a ~~DTM~~ DTF. *Coincident 2/4 trip logic processing for generating a divisional output trip is done by a ~~TLU~~* TLF *or* DLC performing the SLF*."*

"(8) Manual scram is implemented by hard wire to the scram pilot valve solenoids and does not depend on the correct operation of the ~~DTM~~ DTF *or ~~TLU~~* TLF*."*

"(9) A bypass of the RPS output logic unit is a manual, division out-of-service bypass, which allows repair of the ~~DTM~~ DTF *or ~~TLU~~* TLF *of that division without a half scram condition or half MSIV isolation condition. Only one division can be bypassed at a time."*

No COLA revision is required as a result of this supplemental RAI response.

RAI 19.01-1

QUESTION

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

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

REVISED RESPONSE

The Loss of Offsite Power Initiating Event Frequency data table included with the original response to RAI 19.01-1 is revised as shown below to reflect the Electric Reliability Council of Texas (ERCOT) data from NUREG/CR-6890 used in the loss of offsite initiating event frequency sensitivity study.

Loss of Offsite Power Initiating Event Frequencies

Basic Event Name	Duration of the Loss of Power	Original Frequency (/yr) Table 19D.3-1 (SSAR)	Updated Frequency (/yr)
Included as part of transients – not modeled separately	Less than 30 minutes	0.0579	0.01517
TE2	30 minutes to 2 hours	0.0246	0.00645
TE8	Two hours to 8 hours	0.0158	0.00414
TEO	Greater than 8 hours	0.0017	0.00045
TE	Total Frequency for Loss of Offsite Power	0.100(1)	0.02620

Notes

(1) The frequency, 0.1, represents a upper 90% confidence bound for loss of offsite power frequency, which was used in the ABWR SSAR Loss of Offsite Power evaluations.

Historical Line Outage Data from 1980 through 2006 is included in the STP 3&4 FSAR in Table 8.2-3. The response to RAI 08.02-7, transmitted under letter number U7-C-STP-NRC-090039, dated May 18, 2009, and the response to RAI 08.02-22, transmitted under letter number U7-C-

STP-NRC-090204, dated November 18, 2009, provide additional information on the line outage data included in FSAR Table 8.2-3.

The proposed change to Subsection 19.9.6 will be revised, as shown below:

19.9.6 Confirmation of Loss of AC Power Event

The following site-specific supplement addresses COL License Information Item 19.6.
The site-specific frequency estimate for the loss of AC power event (Subsection 19D.3.1.2.4) is complete. The assessment addressed site-specific parameters such as specific causes (e.g., a severe storm) of the loss of power, and their impact on a timely recovery of AC power using data from NUREG/CR-6890 for the Electric Reliability Council of Texas (ERCOT). This evaluation verified that the overall risk impact of grid events at STP is bounded by the original Subsection 19D analysis.

RAI 19.01-13

QUESTION

In Section 19.2, Table 19.2-2 of the STP FSAR, Rev. 2, “PRA Assessment of STP COLA Departures from ABWR DCD”, Departure STD DEP T1 2.4-2 (Feedwater Line Break Mitigation), it is stated that this departure is not explicitly modeled in the ABWR DCD PRA. In the ABWR DCD original design, the feedwater was assumed to be unavailable when hotwell inventory was depleted. No automatic isolation of feedwater flow was assumed. In ABWR Standard R-COL design modification, the condensate pumps are tripped in the event of high containment pressure from Feedwater line break.

Please explain whether this design change was included in the STP plant-specific PRA model. If so, explain its impact on the PRA results.

SUPPLEMENT TO RESPONSE

The departure described in STD DEP T1 2.4-2 (Feedwater Line Break Mitigation), was required to correct an assumption in the FWLB analyses concerning the amount of water continuing to be supplied to the containment in the event of a feedwater line break inside containment.

From the departure description in Part 7, Chapter 2.1:

> “The FWLB is the limiting design basis accident for ABWR primary containment vessel (PCV) peak pressure response. This is because blowdown flows from both the reactor pressure vessel (RPV) side and the balance of plant (BOP) feedwater side contribute to the peak pressure response.
>
> The licensing basis for ABWR is no operator actions for 30 minutes for design basis accidents, as discussed in DCD Tier 2, Subsections 6.2.1.1.3.3.1.2 and 6.2.1.1.5.6.1. With the current ABWR design, the only mitigation option available, for limiting the containment pressure, would be operator action using the non-safety trip of the condensate and/or feedwater pumps.
>
> Therefore, high drywell pressure signals that would already be existing in the Leak Detection & Isolation (LDS) logic of the Safety System Logic & Control (SSLC) are used, in conjunction with the added differential pressure signals between the two feedwater lines, to identify a FWLB in containment and to then trip the condensate pumps.”

The condensate system, as modeled in the ABWR DCD PRA, provides a source of low pressure injection to the reactor and is modeled in Top Event Q in the PRA event trees. This top event is unaffected by the changes described in departure STD DEP T1 2.4-2.

Because the departure is specifically directed at a design basis initiating event that is not included in the ABWR PRA described in the Design Control Document, and because the initiating signal requires coincident high drywell pressure and differential pressure signals between the two feedwater lines to identify a feedwater line break in containment, there is no effect on the condensate system as modeled in the ABWR DCD PRA.

No COLA revision is required as a result of this supplemental RAI response.

RAI 19.01-23

QUESTION

A list of new components and their locations in the Turbine building for STP units 3&4 is provided in Table 9A.6-4 in STP FSAR Section 9A.6, Fire Hazard Analysis Database. However, the impact of these additional components on the FIVE (Fire-induced Vulnerability Evaluation Methodology) results associated with Turbine building was not discussed in Section 19M.

Please explain whether these additional components are included in the fire risk assessment and, if so, please discuss their impact on the FIVE (Fire-induced Vulnerability Evaluation Methodology) results.

Please explain whether these additional components are included in the fire risk assessment and, if so, please discuss their impact on the fire PRA results.

SUPPLEMENT TO RESPONSE

The following response supplements the previously submitted response to RAI 19.01-23.

Appendix 19M of the Design Control Document (DCD), Section 19M.1, summarized the five bounding fire scenarios from the FIVE fire screening assessment. One of the five scenarios was the turbine building fire: From 19M.1:

> "The fifth and final scenario examines the consequences of a fire in the turbine building based upon the assumption that resulting loss of off-site power bounds the possible outcomes of this initiator."

Section 19M.2, Basis of the Analysis, describes turbine building fires as follows:

> (3) Turbine building—As documented in Subsection 9A.5.5.1, fire induced failure of the small amount of safety-related sensors located in this building cannot prevent safe shutdown of the plant. The turbine building is included in the analysis because a turbine building fire could result in a plant shutdown concurrent with a loss of off site power.

The turbine building itself does not contain any equipment necessary to achieve safe-shutdown for the ABWR design. In order to affect the FIVE screening assessment, potential departures to the ABWR turbine building design described in the DCD must either:

> (1) Cause a new scenario that was not assumed in the original screening, or,
> (2) Result in a significant increase in fire frequency, or
> (3) Significantly modify the consequences of the scenarios considered.

From Table 19M-1 of the ABWR Standard Safety Analysis Report (SSAR), Turbine Building core damage frequency results from the FIVE screening are:

Initiators and Conditions	**Fire Ignition Frequency**	**Core Damage Frequency Per Year**
Turbine Building Fire	1.85E-01	2.19E-07

The major components contributing to the Fire Ignition Frequency listed above are presented in Table 19M-7 for the Turbine Building. None of the Turbine Building departures described in the STP 3&4 FSAR affect the entries in Table 19M-7. None of the Turbine Building departures in the STP 3&4 FSAR create a new fire scenario for the Turbine Building. None of the Turbine Building departures in the STP 3&4 FSAR modify the consequences of the scenarios considered in Appendix 19M of the SSAR and DCD. Therefore, there is no affect on the fire screening assessment described in Appendix 19M of the DCD for the Turbine Building departures described in the STP 3&4 FSAR.

No COLA revision is required as a result of this supplemental RAI response.

RAI 19.01-29

QUESTION:

In STP design departures (for example, new fans in the RSW/UHS system, 4KV switchgear related components in the MVES), there are some new basic event failure data which were not included in the original ABWR PRA Database (SSAR Section 19D.6). The staff requests that the applicant provide a list of these new basic events and the references where this data was obtained and how the data was calculated.

SUPPLEMENT TO RESPONSE

To support the review of STD DEP 8.3-1, Plant Medium Voltage Electrical System Design, and potential incorporation into the STP 3&4 PRA, a review of various data sources was performed for breaker operating history. The electric power system described in the Design Control Document (DCD) consisted of 6.9kV non-Class 1E and 6.9kV Class 1E distribution systems. The STP 3&4 medium voltage distribution system consists of a 13.8kV/ 4.16kV non-Class 1E system and a 4.16kV Class 1E system. There is potentially a difference in operating data between the DCD 6.9kV power distribution breakers and the 13.8kV/4.16kV power distribution breakers in the STP 3&4 design. The breakers of interest are those that are required to open or close as a result of a plant initiating event such as loss of offsite power, specifically, those breakers associated with isolation of the offsite sources, and the stripping and repowering of essential plant loads on the diesel generators. For the other typical breaker failure mode,"Transfer Open During Operation" the failure rate is very low when compared to the rate for the active open and close failure modes. Power distribution designs in the U.S. include 13.8kV, 6.9kV and 4.16kV distribution voltages.

The EPRI Advanced Light Water Utility Requirements Document (URD), Volume II, ALWR Evolutionary Plant (Reference 1) identifies breakers (4kV) with a failure rate of 3.0E-04 per demand. One of the plant sources listed in the URD has a 6.9kV/480V non-Class 1E and 480V Class 1E distribution system. The data from this design was included with the 4.16kV plant designs. NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants (Reference 2), includes circuit breaker failure data for breakers used in power distribution, with no voltage specified. The failure rate in NUREG/CR-6928 is 2.5E-03 per demand.

As a further check, the operating experience at STP Units 1 and 2 for the 13.8kV/4.16kV distribution design was reviewed. All power distribution breakers ≥ 4.16kV are assembled into two data variables for fail to open on demand, 4.9E-04, and fail to close on demand, 1.1E-03. Demand failure information from 13.8kV and 4.16kV breakers is included in the periodic updates of these data variables.

This review indicated that there was not a reported difference in failure data between different distribution voltage designs, therefore the data supporting the ABWR PRA was chosen to represent the revised 4.16kV distribution system for STP Units 3&4.

No COLA revision is required as a result of this supplemental RAI response.

References

1. Advanced Light Water Reactor Utility Requirements Document, Volume II, ALWR Evolutionary Plant, EPRI, TR-106780 Revision 8, March, 1999.
2. Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928, USNRC, February 2007.