Nuclear Operating Company

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

July 23, 2009
U7-C-STP-NRC-090079

U. S. Nuclear Regulatory Commission

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

Attached are responses to NRC staff questions included in Request for Additional Information (RAI) letter number 124 related to Combined License Application (COLA) Part 2, Tier 2 Chapter 19. The attachments contain the responses to the RAI questions listed below:

19.01-1	19.01-5	19.01-9
19.01-2	19.01-6	19.01-10
19.01-3	19.01-7	19.01-11
19.01-4	19.01-8	19.01-12

The response to RAI 19.01-11 modifies the response to RAI 02.04.04-9, supplement 1, contained in letter U7-C-STP-NRC-090012, dated 2/23/2009.

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 commitments in this letter.

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

STI 32506809

D091
NRO

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

Executed on 7/23/09

Scott Head
Manager, Regulatory Affairs
South Texas Project Units 3 & 4

dws

Attachments:

1. Question 19.01-1
2. Question 19.01-2
3. Question 19.01-3
4. Question 19.01-4
5. Question 19.01-5
6. Question 19.01-6
7. Question 19.01-7
8. Question 19.01-8
9. Question 19.01-9
10. Question 19.01-10
11. Question 19.01-11
12. Question 19.01-12

cc: w/o attachment except*
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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.

RESPONSE

A sensitivity analysis comparing the ABWR loss of offsite power results, including initiating event frequency and recovery data, to similar area specific data in NUREG/CR-6890 was performed for COLA Rev. 0 and reperformed using the reconstituted PRA model of the ABWR. Using the data from NUREG/CR-6890 for ERCOT, there is a decrease in core damage frequency from loss of offsite power initiating events, which confirms the frequency estimates for the loss of offsite power events, including specific causes such as a severe storm, used in SSAR Subsection 19D.3.1.2.4 are bounding for the STP 3 and 4 site. There is no change to the PRA results or insights described in the DCD as a result of this sensitivity analysis. The detailed sensitivity analysis is available on site for staff review. The table below presents a comparison of the input data for the loss of offsite power sensitivity analysis against the original data in SSAR Appendix 19D.3.

Table-1, 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	0.0208
TE2	30 minutes to 2 hours	0.0246	0.0088
TE8	Two hours to 8 hours	0.0158	0.0057
TEO	Greater than 8 hours	0.0017	0.00061
TE	Total Frequency for Loss of Offsite Power	0.1[(1)]	0.0359

Note

(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.

The COLA will be revised to clarify the use of the NUREG/CR-6890 loss of offsite power data and the results of the sensitivity analysis.

Tier 2 Subsections 19.2.3.2 and 19.2.3.3 will be incorporated by reference with no departures or supplements, as shown below:

~~19.2.3.2 Failure Probability and Field Experience~~

~~The information in this subsection of the reference ABWR DCD is incorporated by reference with the following site-specific supplemental information.~~

~~The expected loss of offsite power frequency has been supplemented to reflect updated information and site-specific data, and utilized in calculating PRA outputs to use in assessing changes in results and insights (Delta-PRA) to confirm continued compliance with requirements.~~

~~19.2.3.3 Initiating Accident Events~~

~~The information in this subsection of the reference ABWR DCD is incorporated by reference with the following site-specific supplemental information.~~
~~The expected loss of offsite power frequency has been supplemented to reflect updated information and site-specific data, and utilized in calculating PRA outputs to use in assessing changes in results and insights (Delta-PRA) to confirm continued compliance with requirements.~~

Subsection 19.3.1.1 will be revised as shown below:

19.3.1.1 Accident Initiators

The following site-specific supplement addresses frequency of initiating events.

The total frequency of transient initiators used in these evaluations is based upon a 1985 analysis of operating plant data (Reference 19.3-1). The frequency of transients is a design requirement prescribed in the Advanced Light Water Reactor (ALWR) Requirements Document (Reference 19.3-2). Apportioning of the expected transient frequency by initiating event was done on the basis of historical electrical grid and BWR performance data as described in Subsection 19D.3.

In order to verify that the Subsection 19D.3 remains bounding for the STP 3 & 4, loss of offsite power and power recovery data from NUREG/CR-6890 (Reference 19.3-8) was also evaluated in a sensitivity study. Industry composite data in NUREG/CR 6890 for the Energy Reliability Council of Texas (ERCOT) 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.

And finally, 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 Energy 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 19.01-2

QUESTION

In Section 19R.4.4 ("Control Building") of the STP FSAR, Revision 2, the set of assumptions for the "worst case" control building flood is presented twice with each having a different assumption for pipe length between the ultimate heat sink and the RCW/RSW room. The staff requests that the applicant clarify in the STP FSAR the assumptions for the "worst case" control building flood.

RESPONSE

These errors in Revision 2 will be corrected in the next routine revision of the COLA. The corrected text is included below.

19R.4.4 Control Building

STP DEP 19R-1

~~The following site-specific supplement presents the analysis performed for the RSW pump house internal flood.~~

The RCW/RSW rooms contain two sets of diverse safety grade level sensors in a two out of four logic. The first set is located at 0.4 meters from the floor and is intended to alert the control room operator to investigate for the presence of water in the RCW/RSW rooms. The second set of sensors are located at 1.5 meters and informs the control room operators that a serious condition exists that needs immediate attention. In addition, the upper level sensors trip the RSW pumps and close redundant supply side *motor operated isolation valves in the RSW system of the affected division.* Redundant motor-operated valving is provided to ensure that the UHS basin water does not gravity drain to the control building.

~~*Anti-siphon capability (e.g., vacuum breakers, air breaks) is included to prevent continued flooding in the event that the RSW pump is tripped but the isolation valves do not close.*~~ *Figure 19R-2 depicts the RSW system. Given that the pumps have tripped, actuation of the* ~~*anti-siphon*~~ redundant automatic isolation *capability will terminate the flood. The ABWR UHS cannot gravity drain into the control building.*

From the above, it is concluded that the only flooding concern in the control building is a leak in the RSW system that threatens the RCW system motors in the RCW/RSW rooms. If the upper level sensor alarms, it is a clear indication of a major RSW system leak in the RCW/RSW room.

The following assumptions are used in this "worst case" control building flood:

~~*(1) The ultimate heat sink (UHS) is at an elevation higher than the control building RCW/RSW rooms such that siphoning*~~ draining ~~*of UHS water through the RSW system to the RCW/RSW rooms is possible.*~~

~~*(2) There is a maximum of 4000 meters of pipe (2000 each for supply and return) between the UHS and the RCW/RSW room which can be discharged to the RCW/RSW room following RSW pump trip.*~~

~~*(2) (3) The size of the RSW crack is about 103 cm² (16 in²) per ANSI/ANS-58.2 and BTP MEB 3-1.*~~

~~*(3) (4) The leak occurs in the RCW/RSW room.*~~

~~*(4) (5) No operator action was assumed.*~~

~~*The results of this "worst case" control building flood are:*~~

~~*(1) A leak occurs in the RCW/RSW room with the RSW pump running and the lower level sensor alarms at 0.4 meters.*~~

~~*(2) The water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped at 1.5 meters.*~~

~~*(3) Water flows into the RCW/RSW room from the 4000 meters of RSW pipe*~~ piping ~~*outside the control building.*~~

~~*(4) No water leaves the flooded room and only one division of RCW is affected.*~~

(~~5~~1) The ultimate heat sink (UHS) is at an elevation higher than the control building RCW/RSW rooms such that draining ~~*siphoning*~~ *of UHS water through the RSW system to the RCW/RSW rooms is possible.*

(~~6~~2) There is ~~*a maximum*~~ *approximately* ~~*of*~~ 580~~*4000*~~ *meters of pipe (*270 m~~*2000 each*~~ *for supply and* 310 m *return) between the UHS and RCW/RSW room which can be discharged to RCW/RSW room following RSW pump trip.*

(~~7~~3) The size of the RSW crack is about 103 cm^2 (16 in^2) per ANSI/ANS-58.2 and BTP MEB 3-1.

(~~8~~4) The leak occurs in the RCW/RSW room.

(~~9~~5) No operator action was assumed.

The results of this "worst case" control building flood are:

(1) A leak occurs in the RCW/RSW room with the RSW pump running and the lower level sensor alarms at 0.4 meters.

(2) The water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped and redundant supply isolation valves are automatically isolated at 1.5 meters.

*(3) Water flows into the RCW/RSW room from the ~~4000~~*580 *meters of RSW pipe outside the control building.*

(4) No water leaves the flooded room and only one division of RCW is affected.

RAI 19.01-3

QUESTION

Section 19R.4.6 ("RSW Pump House") of the STP FSAR, Revision 2, states "The results of this 'worst case' RSW pump house flood are: (1) A leak occurs in the RSW pump room and the lower level sensor alarms at 0.4 meters." The results for this "worst case" RSW pump house flood appears to be incomplete. The staff requests that the applicant provide additional information on the results for this "worst case" RSW pump house flood (e.g., With no operator action, the water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped and redundant supply isolation valves are automatically isolated at 1.5 meters.).

RESPONSE

The last paragraph in supplemental Subsection 19R.4.6 "RSW Pump House" of Appendix 19R will be corrected in the next routine revision of the COLA to provide complete information as shown below:

The results of this "worst case" RSW pump house flood are:

(1) A leak occurs in the RSW pump room and the lower level sensor alarms at 0.4 meters.

(2) The water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped at 1.5 meters.

(3) Water flows into the RSW room from the UHS.

(4) No water leaves the flooded division until it exits the HVAC supply and return at the roof of the RSW pump house. Only a single division of RSW and ECCS is affected.

From the above, it is concluded that there are no flooding concerns in the RSW pump house because most sources of water are either not large enough or leak at small enough rates that no equipment damage could reasonably occur. The only potential water source of concern is the RSW system and only one division of RSW would be affected. The reactor could be brought to safe shutdown using equipment from the other two divisions.

RAI 19.01-4

QUESTION

The departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the plant-specific probabilistic risk assessment (PRA) for reactor building flooding in Section 19R.5 ("Probabilistic Flood Assessment") of the STP FSAR, Revision 2. For example, these departures can impact the failure probabilities associated with the top events for bringing reactor to safe shutdown condition in the reactor building flooding event trees (refer to Figures 19R-11 to 19R-13 in the ABWR Standard Safety Analysis Report). Also, Section 19R.5 did not provide the plant-specific core damage frequency (CDF) for reactor building flooding. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.)

Therefore, the staff requests the applicant provide the plant-specific CDF value for internal flooding of reactor building, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

RESPONSE

RG 1.206 Section C.I.19 applies to COL applications that do not reference a design certification. RG 1.206 Section C.III.I.19 applies to COL applications, such as the application for STP Units 3 and 4, that reference a design certification. RG 1.206 Section C.III.I.19 states "In cases where it can be shown that assumptions in the certified design PRA (1) bound certain site-specific and plant-specific parameters, and (2) do not have a significant impact on the PRA results and insights, no change to the design certification PRA is necessary." As a result of the departures discussed above, no significant changes to the ABWR PRA results have been identified by STP using the guidance provided by RG 1.206, Section C.III.I.19.

Since the departures listed do not significantly affect the PRA results referenced in the ABWR DCD as described in COLA Chapters 19.3 and 19.4, no update to the PRA for reactor building flooding is required under RG 1.206.C.III.I.19.

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

RAI 19.01-5

QUESTION

Departure STD DEP 10.4-2 increased the number of circulating water pumps to four. This can impact the plant-specific probabilistic risk assessment (PRA) for turbine building flooding in Section 19R.5.3 ("Turbine Building") of the STP FSAR, Revision 2. For example, this departure can impact the failure probabilities associated with top events "PTRIP" and "VCLOSE" in the turbine building flooding event tree (refer to Figure 19R-8 "Turbine Building Flooding, High PCHS" in the ABWR Standard Safety Analysis Report). In addition, the departures that were considered in the internal events PRA (e.g. TD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the failure probabilities associated with the top event for bringing reactor to safe shutdown condition in the turbine building flooding event tree. Also, Section 19R.5.3 did not provide the plant-specific core damage frequency (CDF) for turbine building flooding. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the plant-specific CDF value for internal flooding of turbine building, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

RESPONSE

The response of the plant to failure in a main circulating water piping assumed that even if automatic protection did not work, the water would exit the turbine building through the truck doors. From DCD Appendix 19R.1:

> "In the unlikely event this automatic protection fails and the operator fails to take any action, potential flood waters would still be prevented from reaching the service building. Potential flood waters would be expected to exit the turbine building through the non-watertight truck entrance door."

Increasing the number of circulating water pumps does not affect the level setpoints at which the circulating water pumps trip and the pump isolation and condenser isolation valves close, or the plant response to circulating water flooding event, therefore, as described in the COLA, there is no change to the PRA results presented in DCD.

Top event PTRIP in the Turbine Building Flooding (High PCHS) event tree, SSAR Figure 19R-8 has no branch in the event tree for the High PCHS design because trip of the circulating water pumps does not stop circulating water flow and is therefore unaffected by the number of circulating water pumps in the circulating water system. Top event VCLOSE is also unaffected by the changes associated with STP DEP 10.4-2. The function modeled by VLCOSE includes the condenser isolation valves, one for each condenser element, and the circulating water pump isolation valves. The value presented in Figure 19R-8, 1.2E-03, as derived from the data presented in SSAR Table 19R-4, represents failure of one of three isolation valves (condenser isolation valves) and common cause failure with any of the pump isolation valves, represented by the beta factor of Table 19R-4. There is no change to the modeling of the Turbine Building Flooding event tree presented in Figure 19R-8 of the SSAR.

The other departures identified above do not significantly affect the PRA results referenced in the ABWR DCD as described in COLA Chapters 19.3 and 19.4, so no update to the PRA for turbine building flooding is required under Regulatory Guide 1.206.C.III.I.19.

RG 1.206 Section C.I.19 Appendix A applies to COL applications that do not reference a design certification. COL applications, such as the application for STP Units 3 and 4, that reference a design certification are subject to RG 1.206 Section C.III.I.19. That section states: "In cases where it can be shown that assumptions in the certified design PRA (1) bound certain site-specific and plant-specific parameters, and (2) do not have a significant impact on the PRA results and insights, no change to the design certification PRA is necessary." No significant changes to the PRA results for the ABWR have been identified by STP using the guidance provided by RG 1.206, Section C.III.I.19.

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

RAI 19.01-6

QUESTION

Departure STD DEP 9.2-5 increased the RSW flow rate per pump from 1800 m^3/h to 3290 m^3/h and increased RSW pipe sizes. This can impact the plant-specific probabilistic risk assessment (PRA) for control building flooding in Section 19R.5.4 ("Control Building") of the STP FSAR, Revision 2. For example, this departure can impact the timing associated with operator actions in top events "OPACT1", "OPACT2" and "OPACT3" in the event tree for control building flooding due to RSW line break (refer to Figure 19R-9 "RSW Control Building Flood" in the ABWR Standard Safety Analysis Report). In addition, the departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the failure probabilities associated with the top events for bringing reactor to safe shutdown condition in the control building flooding event trees. Also, Section 19R.5.4 did not provide the plant-specific core damage frequency (CDF) for control building flooding due to RSW line and fire water system breaks. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the plant-specific CDF values for internal flooding of control building due to RSW line and fire water system breaks, and describe the risk impact that the departures have on these PRA results (also provide necessary quantitative information that supports this description).

RESPONSE

RG 1.206, Section C.I.19 applies to the STP PRA for any significant differences as a result of changes to the certified design. No significant changes to the PRA results have been identified using the guidance provided by RG 1.206, Section C.III.I.19 therefore RG 1.206, Section C.I.19 does not apply to the STP COLA , and STP is required to use the design certification PRA in accordance with 10 CFR 52.79(d)(1).

The departures listed do not significantly affect the PRA results described in the DCD, as described in Chapter 19.3, so no change to control building flooding from these departures is required under Regulatory Guide 1.206.C.III.I.19.

RSW pump flow rates do not directly affect the computed leakage from the postulated RSW pipe failure, as this leakage is based only on the operating pressure within the pipe, the pipe crack size, and the volume of the RSW piping which contributes to the flood source. Larger pipe diameters are offset by the reduced amount of piping associated with the redesigned RSW system. Because the break size associated with the increased pipe diameter is bound by the size assumed in the DCD, and the increased flow rate of the RSW pumps does not affect the flow rate out of the break, there is no significant effect on operator timing, and no change to PRA described in the DCD. The revised water volume in the Control Building basement from the RSW pipe failure described in Appendix 19R is approximately 6,500 ft^3 (~184 m^3), with automatic isolation. This results in a water level of 7.6 ft (~2.3 m), which is well below the 5 m maximum of the RSW Design Description in Tier 1, Section 2.11.9. The lower result is due to the significantly shorter length of RSW pipe that drains into the Reactor Cooling Water (RCW) pump room from the RSW system following RSW train isolation and drain down.

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

RAI 19.01-7

QUESTION

Section 19R.5.6 ("RSW Pump House") of the STP FSAR, Revision 2, states "Unisolated breaks in the fire water system could cause inter-divisional flooding since the RSW divisional separation splits the RSW pump house into three, watertight compartments." However, Section 19R of the STP FSAR does not provide or describe a probabilistic risk assessment (PRA) for internal flooding due to unisolated breaks in the fire water system in the RSW pump house. The staff requests the applicant describe in the STP FSAR the PRA internal flooding analysis for this scenario, or justify why it should not be included.

RESPONSE

Floods associated with fire water system leaks, piping failures and usage in the RSW pump house are less significant than flood from the RSW piping as described in Appendix 19R.1 of the COLA because of lower water flows and external water isolation capability. If analyzed, fire water floods would be bound by the results of the RSW piping floods which are included in Appendix 19R.

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

RAI 19.01-8

QUESTION

Section 19R.5.6.1 ("RSW Line Breaks") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for internal flooding due to reactor service water (RSW) line breaks in the RSW pump house. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.).

Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for internal flooding due to RSW line breaks in the RSW pump house:

- Total CDF for this internal flooding event,
- PRA significant accident sequences and their mean CDFs,
- Initiating event frequency estimation and its basis, and
- Top event failure probabilities and their basis.

Also, for this internal flooding event, explain where the PRA assumes the worst case pipe break to occur (e.g., pipe break occurs downstream of the RSW pump discharge motor-operated valve, pipe break occurs upstream of the RSW pump discharge motor-operated valve).

RESPONSE

COL applications, such as the application for STP Units 3 and 4, that reference a design certification are subject to RG 1.206 Section C.III.I.19. That section states: "In cases where it can be shown that assumptions in the certified design PRA (1) bound certain site-specific and plant-specific parameters, and (2) do not have a significant impact on the PRA results and insights, no change to the design certification PRA is necessary." No significant changes to the PRA results for the ABWR have been identified by STP using the guidance provided by RG 1.206, Section C.III.I.19.

Consistent with the DCD and Standard Safety Analysis Report (SSAR), a screening evaluation was performed using the PRA information in Appendix 19R of the SSAR. This evaluation resulted in a "very small change" in total CDF when compared to the SSAR internal events results.

The results of the analysis will not be included in the STP 3&4 FSAR in order to maintain consistency with the other internal flooding results discussed in the DCD, but are presented for review by the staff:

Total CDF for this event from the screening assessment is 3.8E-08/yr

PRA significant accident sequences: See the assessment included with this response on the following pages.

Initiating event frequency estimation and its basis. The frequency, 1E-02/yr, was obtained from the ABWR SSAR Table 19R-3 as required by 10CFR52.79(d)(1) and the basis is as described in Appendix 19R of the SSAR.

Reliability data and conditional failure probabilities were obtained from the ABWR SSAR, Table 19R-4 and Table 19R-5, as required by 10CFR52.79(d)(1) and the basis is as described in Appendix 19R of the SSAR.

The RSW pump house flood was analyzed assuming an unisolable break upstream of the pump discharge MOVs.

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

III. Screening Analysis

Assumptions

The following assumptions are used in the screening evaluation for the RSW pump house internal flood.

1. Sump level sensors. Two sets of sensors are provided, sump sensors and floor sensors. Sensors are arranged such that inadvertent isolation signals are minimized, but valid high-high level signals will be received.
2. Automatic isolation of MOVs on receipt of a high-high RSW pump room level signal.
3. Watertight doors seal both ways. There are four watertight doors between the three divisions, two at the pump room level and two at the electrical equipment room level.
4. Watertight doors are alarmed if not closed at a security alarm station. The doors are alarmed in the Control Room if not dogged closed.
5. Control building flood frequency is similar to RSW pump house flood frequency. The frequency of Control Building floods used in the ABWR is based on the piping systems in the building. The only flood presented for the Control Building that was analyzed is due to floods from RSW piping in the RCW heat exchanger rooms.
6. Flooding originating in multiple rooms is not analyzed. Multiple simultaneous floods in piping from natural causes is a random process. The systems are designed for the SSE, and the suction piping is designated in accordance with pipe break exclusion criteria. The likelihood of multiple flood events simultaneously is extremely small.
7. No credit is taken for design of the suction piping (which is in accordance with break exclusion criteria).
8. Isolable floods are not quantified, the CDF is very much less than unisolable floods due to signal reliability and multiple MOVs capable of isolating the flood.
9. A through wall crack is assumed with flow in excess of the room sump pump capability(ies). This is the initiating event for the RSW pump house internal flood scenarios.
10. Operator action to unlock and close the manually operated suction isolation valve is not credited.
11. Common cause failure of multiple watertight doors is quantified as a single basic event for each pump room flood scenario.

Data

Data for the screening analysis is taken from data presented in the SSAR Section 19R for the internal flooding analysis of the Control Building Flood. This data is summarized below.

Table 19R-3, ABWR Flood Frequency (per reactor year)	
Flooding Location (Equivalent ABWR)	**ABWR**
Control Building	1.00E-02

Table 19R-4, Reliability Data for ABWR Probabilistic Flood Analysis		
Component/Element	**Failure Mode**	**Failure Rate (per demand except as noted)**
Level Sensors	Fail to Operate	1.00E-03
Isolation Valve	Fail to Close	4.00E-03
Operator Fails to Act	Available Time < 12 min	5.00E-01
	Available Time < 30 min	1.00E-01
	Available Time < 1 h	5.00E-02
	Available Time > 1h	1.00E-02
Watertight doors	Fail to Stay Closed	1.00E-03
	Common Cause	2.50E-05

Table 19R-5, Conditional Failure Probability of Safe Shutdown	
Conditional Event	**Failure Probability of Safe Shutdown (per demand)**
All ECCS Divisions Available	1.05E-08
One ECCS Division Unavailable	8.70E-07
Two ECCS Divisions Unavailable	2.70E-06
Three ECCS Divisions and Power Conversion System Unavailable	1.00E-01

Analysis

DCD Section 19.9.10(3)(a) identifies ANSI/ANS 58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture", as the basis document for determining the maximum volume and flow rate of potential flood sources. For "moderate energy" piping, such as the RSW piping in the pump house, Section 4.3.5 specifies through-wall cracks and leakage cracks be evaluated. For the purposes of this analysis, a through-wall crack delivering water in excess of the sump pump capabilities is assumed as the initiating event.

Isolable floods are not quantified in this screening analysis. The flood frequency times the likelihood of isolation failure with no operator recovery is less likely than the flood scenarios considered.

Three general flood scenarios are considered. All flood scenarios initiate with a through wall crack in the RSW supply piping in the RSW pump house between the RSW pump house wall and the first automatic isolation valve.

Scenario One. If neither watertight door within a division fails, the flood will proceed to encompass the electrical equipment room and flow out the HVAC vents in the roof of the RSW pump house. Because the flood is unisolated, the UHS will eventually drain to approximately 50 ft MSL. Prior to reaching the final level, the plant technical specifications will require a plant shutdown due to the loss of the UHS level. In general terms, the failure expression is:

> (Flood initiates) X (Conditional Probability of Core Damage given one ECCS Division is unavailable)

Scenario Two. If any one of the watertight doors that isolate the affected RSW division from the unaffected divisions fails, then the flood will disable two RSW divisions, following the same general progression described above. The general failure expressions are:

Division A or C

> (Flood initiates) X (failure of pump level watertight door or electrical equipment room watertight door) X (Conditional Probability of Core Damage given two ECCS Divisions are unavailable)

Division B

> (Flood initiates) X (failure of either pump level watertight door or either electrical equipment room watertight door) X (Conditional Probability of Core Damage given two ECCS Divisions are unavailable)

Scenario Three. If a second watertight door between the affected rooms and the unaffected room fails, then a complete loss of RSW is assumed. This scenario includes the random failure of two doors, and the common cause failure of multiple doors.

> (Flood initiates) X [(failure of pump level watertight door or electrical equipment room watertight door) X (failure of a second pump level watertight door or electrical equipment room watertight door) or common cause failure of multiple doors] X (Conditional Probability of Core Damage (CCDP) given all ECCS Divisions are unavailable)

The following table presents the results of the three scenarios for the three pump room locations and presents the core damage frequency (CDF) calculated for internal flooding in the RSW pump house.

			IE Frequency/yr			
			RSW/ECCS Division Affected			
			One	Two	Three	Three (CCF)
Flood Pump Room A			3.33E-03			
	(WT Door A1 + WT Door A2) * Flood			6.67E-06		
	(WT Door A1 + WT Door A2) * (WT Door C1 + WT Door C2) * Flood				1.33E-08	
	CCF Single Term * Flood					8.33E-08
Flood Pump Room C			3.33E-03			
	(WT Door C1 + WT Door C2) * Flood			6.67E-06		
	(WT Door C1 + WT Door C2) * (WT Door A1 + WT Door A2) * Flood				1.33E-08	
	CCF Single Term * Flood					8.33E-08
Flood Pump Room B			3.33E-03			
	(WT Door A1 + WT Door A2 + WT Door B1 + WT Door B2) * Flood			1.33E-05		
	(WT Door C1 + WT Door C2) * (WT Door A1 + WT Door A2) * Flood				1.33E-08	
	CCF Single Term * Flood					8.33E-08
		IE Sum/yr	1.00E-02	2.67E-05	4.00E-08	2.50E-07
		CCDP	8.70E-07	2.70E-06	0.1	0.1
	CDF	3.78E-08	8.70E-09	7.20E-11	4.00E-09	2.50E-08

The total core damage frequency calculated is less than the total core damage frequency calculated for the DCD ABWR PRA, but slightly higher than the result for the Control Building flooding scenarios, and is dominated by common cause failure of watertight doors. Several conservative assumptions in the calculation, if adjusted, would act to reduce the core damage frequency presented above.

- Credit for operator action to close the normally locked-open manual isolation valve on the pump suction. With an available response time of 30 minutes, the core damage frequency calculated above would be reduced to 3.8E-09, which is approximately the same as the value calculated for the Control Building floods.

- Credit for enhanced capability piping design, e.g., in accordance with pipe break exclusion criteria. The data for flooding rates was taken directly from the ABWR DCD and SSAR. A significant amount of work has been accomplished recently to determine and validate piping failure rates for water systems in nuclear power plants. EPRI Technical Report 1013141, Pipe Rupture Frequencies for Internal Flooding PRAs, Revision 1, March 2006, indicates in Table ES-1 that mean flood failure rates per operating year-linear foot for flood (rates between 100 and 2000 gpm and > 24") are approximately 3.4E-08 to 8.2 E-08. With the enhanced piping design for the RSW suction, the failure rates should be even lower. An explicit calculation using the actual piping configuration and the revised EPRI failure data would also serve to reduce the likelihood of RSW flooding in the pump house.

- No credit for additional monitoring of the watertight doors. The ABWR DCD did not credit alarms if a watertight door is not dogged. Limit switches are provided for the watertight doors in the RSW pump house and alarmed at the security alarm station if not closed and in the alarmed in the Control Room if the doors are not dogged.

With any of these additional considerations included in the analysis, the expected core damage frequency from internal flooding in the RSW pump house would be less than 3.8E-08 per year and would be bound by the calculations presented in the original SSAR and DCD.

RAI 19.01-9

QUESTION

The reference numbers provided in Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, do not correspond to the references in Section 19R.8 ("References"). The staff requests the applicant correct this inconsistency in the STP FSAR.

RESPONSE

These editorial errors in Appendix 19R, supplemental Subsection 19R.7, will be corrected in the next routine revision of the COLA as described below:

References in Subsection 19R.7.2, Identify and Screen Initiating Events, in the third, fourth, and ninth paragraph will be corrected as follows:

> Based on analysis performed for STP 1&2 (Reference ~~19R.7-1~~ 19R-1), landslides are not considered a threat to the STP site. Therefore, landslides are screened as potential external flooding initiating events.
>
> Analysis for STP 1 and 2 (Reference ~~19R.7-1~~ 19R-1) also concluded that tsunamis cannot affect the site. Therefore, tsunamis are screened from consideration as initiating events.
>
> The STP site is located on the Colorado River at river mile 16.4, upstream from the Gulf of Mexico. The potential for dams upstream of the site to cause plant flooding was evaluated as part of the original licensing for Units 1&2. The analyses for Units 1&2 (Reference ~~19R.7-1~~ 19R-1) show that a maximum flood level of 32.0 ft MSL is expected at the STP site from a single upstream dam break. Since this level is below the elevation of Unit 3&4 plant buildings, single upstream dam breaks can be screened from further consideration as external flooding initiating events.

References in Subsection 19R.7.4.1, Main Cooling Reservoir Breach, second paragraph, wil be corrected as follows:

> A breach of the main cooling reservoir could occur suddenly or progress over many minutes. A discussion of previous dam breaches notes that the failure time of most breaches is 15 minutes to one hour from the time of inception to completion of the breach. However, some breaches became fully developed in as little as 6 minutes while others took more than 7 hours. It was also noted that half the breaches identified occurred in less than 1.5 hours. Therefore, it is concluded that, while there is a good deal of uncertainty and variability associated with the breach time, 15 minutes to one hour would likely be conservative. Breach width was also noted to be typically 2 to 5 times dam height (Reference ~~19R.7-2~~ 19R-2). The timing of the breach along with the width of the breach affects the height of water that reaches plant buildings. Smaller breaches or

breaches that take longer to develop would result in a lower level of water on plant buildings. For smaller and slower-developing breaches, it can be expected that water would not rise above grade elevation on plant buildings. For larger and faster-developing breaches, water level on plant buildings would be higher. The analysis, originally documented in the IPEEE of Units 1&2 (Reference ~~19R.7-3~~ 19R-3), considered that failures of the MCR are equally likely to occur anywhere along the perimeter and excluded from consideration that portion of MCR failures that would direct water away from plant buildings. MCR failures that would result in water flowing away from the site would not be considered as external flooding initiating events, consistent with the analysis presented in Reference ~~19R.7.3~~ 19R-3. This assumption is considered reasonable since the land around the MRC generally slopes southward towards the Colorado River. This analysis assumed that any breach of the main cooling reservoir that is included in the initiating event definition is sufficiently large that water level will rise above the entrances to plant buildings. This analysis also assumed that the main cooling reservoir breach would cause a loss of offsite power either because of failure of the switchyard equipment or the plant auxiliary transformers that are impacted by the floodwaters. Furthermore, this analysis assumed that the loss of offsite power is not recoverable for several days.

RAI 19.01-10

QUESTION

Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for external flooding due to Main Cooling Reservoir (MCR) breach. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.). Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for external flooding due to MCR breach:

- Total CDF for this external flooding event
- PRA significant accident sequences and their mean CDFs
- Initiating event frequency for MCR failures that could impact STP Units 3 and 4 and the basis for this frequency
- Top event failure probabilities and their basis, and
- Failure probability for operator action to close control room watertight access door and the basis for this failure probability.

Also, confirm the staffs interpretation that the watertight control room access door is normally closed (except for intermittent ingress and egress), but the MCR external flooding PRA conservatively assumes this door to be open prior to MCR breach.

RESPONSE

The main cooling reservoir breach evaluation results described in the COLA would not significantly affect the Level 1 results presented in the DCD/SSAR if they were summed with the internal events results. In order to remain consistent with the evaluations performed for other traditional external events (fire and seismic), the external flooding analyses were treated as screening evaluations and not considered for inclusion with the Level 1 results discussed in the DCD. The important risk-insights are incorporated into COLA Chapter 19 where appropriate (e.g., watertight doors, operator training, etc.). The status of the watertight door described in Appendix 19R, e.g., always open or intermittently open, has not been firmly decided at this time. As noted in this RAI question, the PRA assessment assumes the door is open.

The detailed screening evaluation is available at the site for review by the NRC staff.

Using failure and core damage sequence information contained in Chapter 19R of the SSAR and the reservoir breach initiating event frequency from the STP 1 and 2 PRA, the following screening results were obtained:

Total CDF – 1.1E-07 per year (/yr)

Significant Sequences

Breach and operator failure to close Control Building access door – 1.0E-07/yr
Breach and watertight doors fail – 9.0E-09/yr
Breach and failure to bring reactor to safe shutdown – 1.2E-11/yr

Initiating Event Frequency -1.0E-06/yr

Basis for Initiating Event frequency

The STP 1&2 Individual Plant Examination for External Events (IPEEE), Reference 1, describes the development of the Main Cooling Reservoir (MCR) Breach initiating event frequency. The generic dam breach initiating event frequency, is which is consistent with SDP Phase 3 Risk Assessment of Operational Events Volume 2, External Events, Reference 2, was modified in the IPEEE for failure modes that are not applicable to the MCR design, such as overtopping, slope protection erosion, and sliding. Additional adjustments in the breach failure rate for the large breach width necessary to flood safety related structures (3000 ft in the IPEEE), and the geometry (specific location) of the breach in the MCR circumference are also developed in Reference 1. The IPEEE MCR breach initiating event frequency has been modified in the current STP 1 and 2 PRA based on successful operation of the MCR for at least five years, which is also consistent with the information developed in Reference 2. The STP 1 and 2 PRA has been reviewed by the NRC staff using the guidance provided by Regulatory Guide 1.200, Revision 1, in support of the Risk Managed Technical Specifications in effect at STP Units 1 &2, reference 3.

The STP 3 and 4 initiating event frequency is based on the STP 1 & 2 PRA initiating event frequency with a correction for the breach width reduction from 3000 ft in the IPEEE to approximately 1000 ft. for Units 3 and 4 This reduction is consistent with the current design basis reservoir breach model described in the Request for Additional Information Response 02.02.04-9, supplement 1, reference 4, which developed a bottom breach width of 380 ft and an average breach width of 417 ft. One thousand feet was assumed in the sensitivity evaluation to encompass the results for either an East breach which affects STP Unit 3 or a West breach which affects STP Unit 4.

Top Event Failure Probabilities - data for watertight door failure and the conditional failure to safely shutdown the plant given the reservoir breach were obtained from the ABWR SSAR as required by 10CFR52.79(d)(1).

Failure probability for operator action to close control room watertight access door and basis – data obtained from the ABWR SSAR as required by 10CFR52.79(d)(1).

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

REFERENCES

1. South Texas Project Electric Generating Station, Level 2 Probabilistic Safety Assessment and Individual Plant Examination, August 1992.
2. SDP Phase 3 Risk Assessment of Operational Events Volume 2, External Events, Revision 1.01, January, 2008
3. Amendments 179/166 to the STP Unit 1 and Unit 2 Operating Licenses, July 13, 2007.
4. Letter, S. Head to Document Control Desk, "Supplemental Responses to Requests for Additional Information," dated 2/23/2009, U7-C-STP-NRC-090012, ML090710301 and ML 090710302.

RAI 19.01-11

QUESTION

Section 19R.7 ("External Flooding Evaluation") of the STP FSAR, Revision 2, qualitatively describes the plant-specific probabilistic risk assessment (PRA) for external flooding due to multiple, concurrent upstream dam (MCUD) failures. 10 CFR 52.79(a)(46) states that a combined license (COL) application must contain an FSAR that includes a description of the plant-specific PRA and its results. In addition, Regulatory Guide 1.206, Section C.I.19, Appendix A explains what the results should include (e.g., core damage frequency, CDF, significant core damage sequences, importance measures, and etc.). Therefore, the staff requests the applicant provide the following information associated with the plant-specific risk for external flooding due to MCUD failures:

- Total CDF for this external flooding event,
- PRA significant accident sequences and their mean CDFs,
- Initiating event frequency for MCUD failures and the basis for this frequency, and
- Top event failure probabilities and their basis.

Also, Section 19R.7.3 ("Quantification of External Flooding Initiating Event Frequency") of the STP FSAR, Revision 2, states "The frequency of multiple, concurrent upstream dam breaks considers the failure of three dams, the S. W. Freese, Buchanan, and Mansfield Dams. ... Downstream of the S. W. Freese Dam is the Buchanan Dam. It is assumed that failure of the Buchanan Dam is dependent on the failure of the S. W. Freese Dam. Table 19R-4 gives values for common cause factors. Although not considered a common cause failure in the traditional sense, the second and third dam failures are analyzed using the common cause factors from Table 19R-4. Using the Beta factor from Table 19-4, failure of the Buchanan Dam, given failure of the S. W. Freese Dam is calculated. Failure of the third dam, the Mansfield Dam, given failure of the first two dams, is calculated using the Gamma factor given in Table 19R-4. The frequency of multiple concurrent dam failures considered as external flooding initiating events is calculated to be very low." The uncertainty associated with the MCUD initiating event frequency could be large. It may be more appropriate to conservatively assume that the Beta or Gamma factors are 1.0. The staff requests the applicant provide the basis for the assumption of analyzing the second and third dam failures using the common cause factors from Table 19R-4.

RESPONSE

This potential design basis external flood has been reanalyzed in response to NRC Request for Additional Information 02.04.04-9 on Chapter 2.4S of the COLA, Reference 1. The new flood height associated with the non-mechanistic, multiple-cascading upstream dam failure scenario described in Chapter 2.4S is 32.5 ft MSL. With wave run-up, the maximum water level from the multiple cascading dam failure is 34.4 ft MSL which is below the openings to safety-related buildings at the STP 3 & 4 site. For this reason, this flood scenario is no longer considered as a potential source of external flooding to be included in the site specific PRA described in

Appendix 19R. Appendix 19R, Appendix 19Q, and Chapter 19.4 will be modified as described below to better describe the multiple cascading dam flooding scenario.

Subsection 19R.7.2, Identify and Screen Initiating Events, second to last paragraph, will be changed as shown below:

> In addition the potential flooding effects from multiple, cascading failures of Colorado River dams upstream of the STP site has the potential to affect safety-related structures. ~~That~~ The analysis described in Chapter 2.4S.4.3.1 shows ~~that~~ a peak still water elevation of 32.5 ft MSL ~~34.1 feet with~~ and a wave runup of 1.9 ft for a resulting flood elevation of 34.4 ft MSL ~~to the 43.7 foot elevation~~. As the building openings are above this water level, ~~Therefore,~~ multiple, ~~concurrent~~ cascading dam failures are not considered as an external flooding initiating event.

Subsection 19R.7.3, Quantification of External Flooding Initiating Event Frequency, second , third, and fourth paragraphs will be deleted:

> ~~The frequency of multiple, concurrent upstream dam breaks considers the failure of three dams, the S. W. Freese, Buchanan, and Mansfield Dams. The analysis assumes that the first dam failure can occur randomly and that the second and third failures are dependent on the previous dam failures. The sequence of events analyzed begins with failure of the S. W. Freese Dam which began operation in 1990.~~
>
> ~~Downstream of the S. W. Freese Dam is the Buchanan Dam. It is assumed that failure of the Buchanan Dam is dependent on the failure of the S. W. Freese Dam. Table 19R-4 gives values for common cause factors. Although not considered a common cause failure in the traditional sense, the second and third dam failures are analyzed using the common cause factors from Table 19R-4. Using the Beta factor from Table 19-4, failure of the Buchanan Dam, given failure of the S. W. Freese Dam is calculated.~~
>
> ~~Failure of the third dam, the Mansfield Dam, given failure of the first two dams, is calculated using the Gamma factor given in Table 19R-4. The frequency of multiple concurrent dam failures considered as external flooding initiating events is calculated to be very low.~~

Subsection 19R.7.4, Accident Sequence Analysis, will be modified as follows:

> ~~STP DEP T1 5.0-1~~
>
> The subsections that follow summarize the accident sequence analysis for the ~~two~~ one events considered as external flooding initiating event~~s~~.

Subsection 19R.7.4.2, Multiple, Concurrent Upstream Dam Failures, will be deleted n its entirety as shown below:

~~19R.7.4.2 Multiple, Concurrent Upstream Dam Failures~~

~~STP DEP T1 5.0-1~~

~~Note that this analysis is developed assuming that the watertight door providing normal access to the main control room is open. This assumption provides a conservative and bounding assessment of risk from external flooding because the watertight door to the main control room would be closed except for intermittent ingress and egress (Refer to FSAR Section 2.4S.10).~~

~~The accident progression for multiple, concurrent upstream dam failures is similar to that of the main cooling reservoir breach except for timing. Since the last dam that would fail, the Mansfield Dam, is nearly 300 miles upstream of the STP site, flood waters from that dam failure would not reach the STP site for many hours. In that time, closure of the normally-open main control room access door would be assured. In addition, compensatory actions such as sandbagging or installation of other temporary flood barriers can be installed around access doors. These additional compensatory actions, however, are not quantified as part of this analysis. This analysis also assumes that the flooding that results from multiple, concurrent upstream dam failures will cause a loss of offsite power either because of failure of the switchyard equipment or the plant auxiliary transformers that are impacted by the floodwaters. Furthermore, this analysis assumed that the loss of offsite power is not recoverable for several days.~~

~~External access points to the control and reactor buildings are provided with normally-closed, watertight barriers or doors designed to withstand the maximum loadings of any potential main cooling reservoir breach, a more severe event than multiple, concurrent upstream dam failures. All these doors are alarmed at the central alarm station so it is unlikely that one would be left open. Failure of any one of these doors would allow water to enter the building and flow through drains, stairways, and non-watertight doors to the essential electrical switchgear rooms below grade. Since there are no internal watertight barriers to protect the rooms on the lower elevations from water that entered the upper elevations, it is conservatively assumed that failure of one of the watertight doors on the reactor building will result in core damage.~~

~~The normal access to the main control building is via the service building through a watertight door on the 2950 mm elevation. In addition, there are other normally-closed watertight doors that provide access to the control building from the service building and that are located either at or below grade. Since the service building is not designed to withstand flooding, it is conservatively assumed that the flooding that results from multiple, concurrent upstream dam failures would result in water entering the service building. If any one of the doors from the service building to the control building fails, then water could enter the control building and cause failure of all three divisions of~~

~~reactor cooling water (RCW) or DC power since these are located below grade. Since there are no internal watertight barriers to protect the rooms below grade in the control building, it is conservatively assumed that failure of one of the watertight doors on the control building will result in core damage.~~

~~The turbine building and service building are not designed to withstand flooding. Therefore, it is conservatively assumed that any equipment in the turbine building or service building is failed by the flooding caused by multiple, concurrent upstream dam failures. PRA-related equipment housed in the turbine building includes the condensate and feedwater systems and the combustion turbine generator (CTG).~~

~~When notified of an upstream dam failure, steps will be taken (Refer to Section 19.9.3) to ensure that the watertight main control room access door will be closed prior to flood waters reaching the STP site. Since many hours are available to effect this action and the action is simple and visually verifiable, the probability of failing to ensure closure of the door is considered sufficiently small as to be neglected. Closing this door prevents water from entering the control building.~~

~~Since the flooding is assumed to cause a loss of offsite power, all equipment powered from non-essential electrical buses would be lost. The loss of offsite power will result in the EDGs starting and loading to their respective essential electrical buses. The CTG is conservatively assumed failed by the flood so failure of all three EDGs would result in a station blackout (SBO). For this analysis, a SBO is assumed to be non-recoverable and results in core damage.~~

~~If one or more EDG starts and loads its respective buses, then the reactor can be brought to safe shutdown using equipment powered from the essential AC buses.~~

~~The accident progression for this event tree is similar to that of a loss of offsite power. However, for multiple, concurrent upstream dam failures, it is assumed that offsite power is not recovered and that failure to insert control rods or a subsequent station blackout result in core damage.~~

Subsection 19R.7.5, Summary of Accident Sequences, will be modified as follows:

~~STP DEP T1 5.0-1~~

The subsections that follows summarizes the determination of the accident sequences developed for the ~~two~~ one events considered as external flooding initiating events. Determination of CDF made use of the existing ABWR PRA logic models and used a process similar to that used to quantify the internal flooding events.

Subsection 19R.7.5.2, Multiple, Concurrent Upstream Dam Failures Accident, will be deleted in it entirety as shown below:

~~**19R.7.5.2 Multiple, Concurrent Upstream Dam Failures Accident**~~

~~STP DEP T1 5.0-1~~

~~Four accident sequences lead to core damage. Core damage results if any one of the top events fails. Development of each of the top events is discussed below.~~

~~***IEDAM - Multiple Concurrent Upstream Dam Failures***~~
~~This top event represents the failure of the three dams upstream of the STP site on the Colorado River. This event is described above.~~

~~***WTDOOR - Reactor Building and Control Building External Watertight Doors Fail***~~
~~This top event represents failure of the watertight doors to prevent flood waters from entering either the control building or the reactor building. Because of the long time available for notification and action following failure of the last dam on the Colorado River, it is assumed that the failure probability of operator action to close the normally-open watertight door to the main control room can be neglected.~~

~~Any one of the nine watertight doors that allow access to the reactor building or control building could randomly fail. Using the values in Table 19R-4, the probability of random door failures that allow water to enter either the control building or the reactor building is calculated.~~

~~***C - Failure To Insert Control Rods***~~
~~This top event represents failure to insert the control rods on the loss of offsite power caused by the external flooding event. The probability of this event is taken from the internal events PRA models.~~

~~***PO1 - SRVs Fail To Open (After Scram)***~~
~~This top event represents failure of the safety relief valves (SRVs) to open after a reactor trip. The probability of this event is taken from the internal events PRA models.~~

~~***SSD - Reactor Brought To Safe Shutdown Condition***~~
~~This top event represents failure to bring the reactor to a safe shutdown condition. This top event is described in Section 19R.7S.5.1.~~

~~Since failure of each of the top nodes on the IEDAM event tree results in core damage and since each of the top nodes is independent of the others, the total CDF for an external flooding event caused by multiple, concurrent upstream dam failures is the product of the initiating event frequency, the success probability of any previous nodes, and the top node failure probability. The total CDF for a breach of the main cooling reservoir is determined to be very low.~~

Subsection 19R.7.5.3, Total External Flooding Event CDF, will be modified as shown below:

~~STP DEP T1 5.0-1~~

The total CDF from the external flooding event~~s is obtained by summing the CDF from each of the events above and~~ is determined to be very low.

Subsection 19R.7.7, Operator Actions Related to External Flooding, will be modified as follows:

~~STP DEP T1 5.0-1~~

One operator action is important to external flooding risk. This action, timely closure of the watertight door at the entrance to the main control room is similar to the event included in section 19R.6.4. However, the cues to initiate the action for the external flooding event~~s are~~ is different than for internal flooding.

Subsection 19R.7.8, External Flooding Reliability Goals (Input to RAP), will be modified as follows:

~~STP DEP T1 5.0-1~~

The results of the external flooding analysis show that watertight doors are important to reducing external flood-related risk. Watertight doors are included as input to the RAP because of internal flooding events. The information from Section 19R.6.5 related to watertight doors is also applicable to the external flooding events and is applied to all external watertight doors on the reactor and control buildings.

In addition, changes to FSAR Appendix 19Q, second to the last paragraph, will be made as indicated below:

19Q.6 Flooding and Fire Protection

External Flooding Risk

STP DEP T1 5.0-1

Appendix 19R presents the analysis performed for external flooding at STP Units 3 & 4 for power operation. The events considered include: The cascading failure ~~Failure~~ of upstream dams on the Colorado River; probable maximum precipitation (PMP) events; main cooling reservoir breach; tsunamis, etc. The breach of the main cooling reservoir is the design basis flood for STP Units 3 & 4. The cascading failure of upstream dams on the Colorado River scenario and the PMP scenario result in water level slightly above grade, but less than the flood level due to the main cooling reservoir breach, and are much more slowly developing floods. If external flood barriers are open or removed and cannot be restored prior to high water levels reaching the site, then core damage is assumed. An operating procedure for severe external flooding will be developed and implemented prior to fuel loading. (COM 19.9-3).

And finally, changes to COLA Chapter 19.4 will be made as indicated below to supplemental paragraph in Subsection 19.4.5, ABWR Probabilistic Flooding Analysis

The ABWR Probabilistic External Flooding Analysis screened all ~~except two~~ but one external flooding events from consideration because flood waters would not rise to an elevation above the entrances to plant buildings. The ~~two~~ external flooding events with the potential to result in core damage ~~are a)~~ is a breach of the main cooling reservoir ~~and b) multiple, concurrent failures of upstream dams~~.

The ~~Both~~ external flooding event~~s are assumed~~ is assumed to cause a non-recoverable loss of offsite power as well as fail all equipment in the turbine building and the fire protection pump house.

Failure of any watertight door to prevent water from entering the control building was assumed to result in core damage because all three essential DC divisions and the main control room are located below grade and there are no internal watertight barriers that would prevent water that enters the control building from failing all three DC divisions or the main control room. For a breach of the main cooling reservoir, timely operator action is required to close the normally-open main control room access door. ~~For multiple, concurrent upstream dam failures, many hours are available from failure of the last dam until flood waters reach the site. Therefore, the operator action to close the main control room access door for the multiple, upstream dam failures is considered assured.~~

REFERENCES:

1. Letter, S. Head to Document Control Desk, "Supplemental Responses to Requests for Additional Information," dated 2/23/2009, U7-C-STP-NRC-090012, ML090710301 and ML 090710302.

RAI RAI 19.01-12

QUESTION

The third paragraph of Section 19K.11.1 of the ABWR DCD ("Component Inspections and Maintenance") states "Multiplexers which provide multiple signals to several systems are identified by the Level 1 analysis as high importance components. Safety system multiplexers have a built-in self test that checks circuits frequently. In addition, one of four multiplexers can be bypassed and tested during plant operation without loss of system function. ..." Section 19K.11.1 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"). The staff requests that the applicant address the above comment and revise Section 19K.11.1 of the STP FSAR as necessary.

RESPONSE

STD DEP T1 3.4-1 changes the ESF Logic and Control System (ELCS) and Neutron Monitoring System (NMS)/Reactor Trip and Isolation System (RTIS) equipment from multiplexer-based system to the use of Remote Digital Logic Controllers (RDLCs). The function of the ELCS and NMS/RTIS are unchanged. The incorporation of STD DEP T1 3.4-1 included changes to two other paragraphs in COLA Section 19K.11.1, but inadvertently did not address the change in hardware for the third paragraph of the DCD as noted in the RAI. The suggested changes to the third paragraph of this section of the DCD, will be made as provided below. Changes for the COLA are highlighted in gray shading. (The prior paragraph of the COLA is shown for reference in locating the new COLA paragraph).

> *The system of greatest FV importance* with respect to outage time *is the RCIC System, which has been assigned a small unavailability for test and maintenance. The amount of time the RCIC System is unavailable because of test and maintenance should be monitored to assure that it remains within the specified assumption annually. Sensitivity studies of increased SSC unavailabilities showed that an increase in RCIC unavailability would cause the greatest increase in estimated core damage frequency of any SSC. The RCIC System was also found to be the most sensitive system to increased outage time assumptions. The highest contributor to uncertainties in the CDF as well as the CDF estimate was RCIC test and maintenance.*
>
> STD DEP T1 3.4-1
>
> The Remote Digital Logic Controller (RDLC) performs the Remote Input/output Function (RIF). Components that provide this function ~~*Multiplexers which provide multiple signals to several systems*~~ *are identified by the Level 1 analysis as high importance components. Safety system* RDLCs ~~*multiplexers*~~ *have* self-diagnostics ~~*a builtin self test*~~ *that* detect failures during on-line operation ~~*checks circuits frequently*~~. *In addition, one* division *of* ~~*four*~~ RDLCs ~~*multiplexers*~~ *can be bypassed and tested during plant operation without loss of system function. Such tests provide a* periodic verification ~~*complete simulation*~~ *of the* RDLC operability ~~*multiplexer signals, more than included in the self test*~~. *During plant outages more detailed* RDLC ~~*multiplexer*~~ *tests are possible,*

including a complete system test and identification of signal errors. These tests will include verification that the remote RDLCs ~~*multiplexing units*~~ *function properly.* RDLC RIF ~~*Multiplexer*~~ *tests that are suggested as part of the RAP are given in Table 19K-4.*