PMSTPCOL PEmails

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Sent: Wednesday, January 20, 2010 6:07 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 Letter U7-C-STP-NRC-100023
Attachments: U7-C-STP-NRC-100023.pdf

Please find attached a courtesy copy of letter number U7-C-STP-NRC-100023, which contains responses to the NRC staff question included in Request for Additional Information (RAI) letter number 296 related to Combined License Application (COLA) Part 2, Tier 2 Chapter 19, a revision to the response to RAI 19-3, and a supplemental response to RAI 19-14.

The official version of this correspondence will be placed in today's mail. Please call Bill Stillwell at 361-972-7581 if you have any questions concerning this letter.

Thank you,

Loree Elton
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Hearing Identifier: SouthTexas34Public_EX
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January 20, 2010
U7-C-STP-NRC-100023

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

References:
1. Letter, Scott Head to Document Control Desk, "Response to Request for Additional Information" dated September 15, 2009, U7-C-STP-NRC-090144 (ML092600801)
2. Letter, Mark McBurnett to Document Control Desk, "Response to Request for Additional Information" dated July 13, 2009, U7-C-STP-NRC-090064 (ML092740559)

Attached is the response to the NRC staff question included in Request for Additional Information (RAI) letter number 296 related to Combined License Application (COLA) Part 2, Tier 2 Chapter 19. Attachment 2 to this letter revises the response to RAI 19-3 that was provided in Reference 1. Additionally, Attachment 3 to this letter supplements the response to RAI 19-14 that was provided in Reference 2.

The attachments provide the responses to the following NRC staff questions as described above:

19-30
19-3 Revised Response
19-14 Supplemental Response

There are no new commitments in this letter.

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.

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

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

Executed on 1/20/2010 MAMcBurnett

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

dws

Attachments:

1. Question 19-30
2. Question 19-3, Revision 2
3. Question 19-14, Supplemental Response

cc: w/o attachment except*

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RAI 19-30

QUESTION

At the staff audit of the South Texas Projects Unit 3 and Unit 4 PRA on September, 23, 2009, the staff reviewed the calculation, "External Flooding Event Breach of the Main Cooling Reservoir (MCR)". The calculation was dated April 20, 2009 and was referenced in the applicant's RAI response to 19.01-10 which discussed the PRA for external flooding due to MCR breach. The staff then reviewed Section 2.4S.4.1.2 of the FSAR which evaluates postulated failure of the MCR. Based on staff review of these two documents, the staff requests that the applicant address the following questions:

1. Section 2.4S.10 of the FSAR states: "All safety-related facilities in the power block are designed to be water tight at or below elevation 40.0 ft MSL. All water tight doors and hatches are normally closed under administrative controls and open outward. … <u>An MCR embankment breach near the STP 3 & 4 power block area would not provide sufficient time for implementation of emergency operating procedures or flood warning systems.</u> As all water-tight doors and hatches are to remain in a closed position, no emergency operating procedures or plant Technical Specifications (plant shutdown), which are discussed in Subsection 2.4S.14, are required for implementation of flood protection measures." The MCR external flooding PRA analysis described in Section 19R of the FSAR is not consistent with the above statement in that under Section 19R the water tight door between the service building and the control building is normally open and takes credit for emergency operating procedures and operator action to close this water tight door during MCR breach. Please clarify this inconsistency and revise the FSAR as appropriate.

2. In STP's response to RAI 19.01-10, STP stated that the overtopping, slope protection erosion, and sliding failure modes are not applicable to the MCR design. Please justify why these failure modes are not applicable to the MCR design, and provide the basis for the reductions in dam failure frequency as a result of excluding these failure modes. In your discussion on why the MCR cannot overtop, please include the following information:

    a. The maximum pumping capacity to the MCR from the Colorado River and the maximum discharge capacity to the Colorado River.
    b. The frequency at which the MCR levels are monitored and how this information is alarmed/displayed in the control room.
    c. The procedures used to control MCR level, and the response procedures if MCR level becomes too high.

3. Section 19R.7.4.1 of the FSAR states: "A breach of the main cooling reservoir could occur suddenly or progress over many minutes." This section of the FSAR also discusses other dam breaches noting that the failure time of most breaches is 15 minutes to 1 hour, and some breaches become fully developed in as little as 6 minutes. A sudden breach of the MCR (e.g., seismic liquidfication) may not provide sufficient time for the operator to close the water tight door between the service building and the control building (i.e., basic event OCD = 1.0). Please address the external flooding analysis due to sudden MCR breaches.

4. Please assess the impact of Category 4 and 5 hurricanes on the frequency of MCR breach. Address how a storm surge from such a hurricane would affect the MCR levee system and the exterior side of the reservoir that has no liner.

5. Please provide your data sources for dam failures that include infantile dam's failures that were used to support your reduction factor for satisfactory operation of the MCR for five years. Based on staff review of dam failures from the National Performance of Dams Program (NPDP), developed by the Department of Civil and Environmental Engineering at Stanford University, including the Taum Sauk dam failure in 2005, the inclusion of infantile dam failures would result in generic dams break frequencies greater than 1E-4 per year. In addition, it appears that the reduction you credited for satisfactory operation of the MCR seems to be double-counting. Please address these issues in your response.

6. Please justify the factor of three reduction you used, based on the assumption that the location of a breach is limited to a thousand foot section. Please explain why any thousand foot section in the 16,250 foot perimeter facing the safety related buildings can not cause a flood.

7. Please assess the impact of a MCR breach during cold shutdown and refueling if secondary and primary containment has open penetrations to facilitate maintenance. Please consider the elevations of these penetrations in your assessment.

8. Please document if the assumptions, insights, or conclusions in the referenced calculation change given the revised MCR breach evaluation in Section 2.4.4.1.2 of the FSAR.

9. The staff needs more information on the probability (basic event- OCD) of the operator failing to close the single normally open flood door between the service building and the control building. To justify the human error probability 0.1, please provide the following information:

    a. The criterion that you will supply to the guard at security house to determine if the MCR has breached.

    b. The process by which these procedures will be controlled.

    c. The potential for ambiguous visual indication on the occurrence of a MCR breach including: the occurrence of local ponding due to heavy rains and the ability of the guard to identify increased flood levels due to reduced visibility during heavy rain storms, fog, etc., particularly at night time.

    d. Section 19R.7.5.1 of the FSAR states: "…a minimum available warning time from water at the South Security Gate House, approximately El. 32.0' MSL, to water at the entrances to safety-related buildings, El. 35.0' MSL. At least 30 minutes is available for operator action to close the normally open access door between the Service Building and the Control Building once water reaches the South Security Gate House." Please sufficiently justify the operator action time of at least 30 minutes.

RESPONSE

The "Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009," approved February 2, 2009 (reference 1), contains screening criteria for external events other than fire and seismic events in Subsection 6-2.3. This Standard applies to an At-Power Level 1 PRA for operating nuclear power plants. An equivalent Low Power/Shutdown Standard is not yet approved;

however, for the purposes of responding to this Request for Additional Information, the external event screening criteria in the published national standard are selected to provide an additional basis for the response provided below, regardless of the plant operating mode. In NUREG-1407 (reference 2), the NRC recommended a similar set of screening criteria for the Individual Plant Examination of External Events (IPEEE) required of all operating nuclear power plants.

In ASME/ANS RA-Sa-2009, Subsection 6-2.3, the fundamental criteria for screening external events other than fire and seismic events are as below:

> "There are three fundamental screening criteria embedded in the requirements here, as follows. An event can be screened out either
> *(a)* if it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) or a later revision; or
> *(b)* if it can be shown using a demonstrably conservative analysis that the mean value of the frequency of the design-basis hazard used in the plant design is less than $\sim 10^{-5}$/yr and that the conditional core damage probability is $<10^{-1}$, given the occurrence of the design-basis hazard event; or
> *(c)* if it can be shown using a demonstrably conservative analysis that the CDF is $<10^{-6}$/yr."

The STP design for safety-related systems, structures and components satisfies the requirements of Standard Review Plan 3.4.2, Revision 3 and Standard Review Plan 2.4.4, Revision 3, which were in effect at the time of the Combined Operating License Application. Criterion (a) of ASME/ANS RA-Sa-2009 Subsection 6-2.3 is satisfied for the external flood scenarios and these events are screened from detailed quantitative evaluation in the STP 3&4 PRA described in Section 19 of the Final Safety Analysis Report (FSAR). A quantitative screening assessment for breach of the Main Cooling Reservoir (MCR) is described in FSAR Appendix 19R and documented in other parts of Section 19 of the FSAR. The RAI responses provided below are provided for this screening assessment.

1. The response to RAI 02.04.14-1 revised the STP FSAR position on the status of watertight door that provides normal access to the Control Room from the Service Building and stated that the watertight door will be normally open. This facilitates normal access to the Control Room and will reduce the likelihood of door malfunction due to frequent usage. Section 3.8.1 of the FSAR indicated that the normal access watertight doors between the Service Building and Control Room and the two watertight doors between the Service Building and Radwaste Building, i.e. the Reactor Building Access Corridor, are also normally open for the same reason. FSAR Appendix 19R presents the results of a screening analysis for the Main Cooling Reservoir Breach assuming only the normal watertight access door between the Service Building and the Control Room was normally open. As a result of these inconsistencies, the following FSAR Sections will be modified to reflect the expected status of the three watertight doors that will remain normally open to facilitate personnel entrance and exit during normal plant operation. All other watertight doors and openings below the Design Basis Flood Level will remain closed under administrative control during plant operation.

The FSAR will be revised as shown below for the following:

FSAR Section 2.4S.10
FSAR Section 2.4S.14
FSAR Section 19.4
FSAR Section 19.8
FSAR Section 19.9
FSAR Section 19.11
FSAR Appendix 19K
FSAR Appendix 19R

2. The Main Cooling Reservoir (MCR) design precludes the failure modes seepage, slope protection erosion, overtopping, and liquefaction. The MCR design is described in the Updated Final Safety Analysis Report for STP Units 1 &2 Chapters 2.4 and 2.5.

   (a) Pumping rates for the Reservoir Makeup Pumping Facility (RMPF) are contained in the operating procedure for the facility, 0POP02-LM-0001:

      60 CFS – 2 pumps each
      240 CFS – 2 pumps each

   During river pumping operations, the River Flow Rate vs. Maximum Allowed Pumping Rate/Pump Combination is verified at least twice per shift (operating procedure 0POP02-LM-0001, Reservoir Makeup Pumping Facility).

   The maximum discharge capacity is determined from the Spillway Rating Curve, Figure 2.4.8-5 of the UFSAR for Units 1 and 2. At 52.1 feet, the maximum discharge is 4200 CFS. The spillway gates are only opened if water level reaches 49.5 feet with the potential to go higher (operating procedure 0POP02-MC-0001, Cooling Water Reservoir Spillway Gates and Blowdown Operation). Blowdown is the preferred method of level control. The maximum blowdown rate with 7 blowdown valves is 260-308 CFS.

   (b) The MCR level is recorded every 12 hours and reported daily when no pumping or blowdown operations are on-going. During pumping evolutions, water level is monitored at least twice per shift. During discharge evolutions, permit conditions are validated and recorded every shift.

   (c) MCR water level is normally controlled by operating the RMPF when adequate water flow is available in the Colorado River. Make-up flow is controlled by a contract between the Lower Colorado River Authority and STPNOC. Make-up from the Colorado River is stopped when the MCR water level reaches 49.0 ft. If water level reaches 49.5 feet with the potential to go higher, operating procedure 0POP02-MC-0001 is used to reduce the MCR water level using the spillway gates.

3. Sudden catastrophic failure of the MCR is the basis for the design flood levels described in FSAR Section 2.4S.4. Catastrophic failure does not imply a fully developed breach. The flood calculations supporting the design basis flood evaluation in 2.4S.4 postulate a rapidly progressing breach that quickly proceeds to a conservatively determined average breach width of 417 feet. The timing for operator action is based on the time for the water level on site to go from El. 32.0' to El. 34.5 ft' given the design breach width. Liquifaction of the reservoir under Safe Shutdown Earthquake accelerations has been analyzed as part of the licensing of STP Units 1 and 2 and is not a credible failure mode for the Main Cooling Reservoir embankment (UFSAR Section 2.4.4.1.1.3).

4. The impact of hurricanes on the MCR is evaluated in FSAR Chapter 2.4S.5. As shown in Figure 2.4S.5-7, the STP plant site and north side of the MCR is dry for Category 4 and 5 hurricanes. There could be minor flooding on the south embankment of the MCR; however any damage to the south embankment will not produce a flood that affects Units 3&4.

5. The dam failure information was developed to support the Individual Plant Examination for External Events (IPEEE) performed for STP Units 1 and 2 and transmitted to the NRC under STP letter ST-AE-HL-93526, August 31, 1993. As described in Section 3.4.6.5 of the IPEEE, the primary data sources are "Baecher, G. B., M. E. Pate, and R. de Neufuille, "Risk of Data Failure in Benefit-Cost Analysis, Water Resources Research," Vol. 16, No. 3, Pg. 449-456, June 1980," and "Von Thun, J. L., Bureau of Reclamation, Engineering and Research Center, "Application of Statistical Data from Dam Failures and Accidents to Risk-Based Decision Analysis on Existing Dams," October 1985." The base failure rate developed for the IPEEE included all dam failures and noted that approximately one-half of dam failures occur during the first five years after initial fill. A 50% reduction in failure rate, is appropriate based upon this information and the successful operation of the MCR for 25+ years.

6. The reduction in the likelihood of a reservoir breach based on length is based on the physical characteristics of the site. As described in FSAR Section 2.4S.4.1.2, "The northern MCR embankment, near the proposed circulating water intake and discharge pipeline, is the most critical location for piping failure because it is closest to, and inline with, Units 3 and 4. Two breach locations were considered for the analysis, one immediately east and one immediately west of the circulating water pipeline. Further discussion of breach parameter selection is presented in Subsection 2.4S.4.2.2.2.2." The average breach width in FSAR Section 2.4S.4.2.2.2.2 is 417 feet. This breach width is assumed to occur in a 1000 feet section directly south of Units 3 and 4 for the quantitative screening evaluation of the MCR described in FSAR Section 19R. Too far east or west of Units 3 and 4, and the flood water from the assumed breach will be directed away from the units to those areas of the site with a lower elevation, resulting in a reduction in the water level at Units 3 or 4.

   In the analysis described in the Unit 1 and 2 IPEEE referenced above, the UFSAR design basis reservoir breach was an instantaneous removal of 2000 feet of the reservoir levee facing the units. Given this, the IPEEE assumed the breach width that potentially affected either unit was 3000 feet out of the 16,250 foot section of the MCR that faced the units. This resulted in a reduction in the initiating event frequency for the MCR breach of 3000/16250. For Units 3 and

4, the assumed breach width for the PRA screening assessment is 1000 feet because of the smaller design basis breach width and the postulated failure locations. The reduction in the initiating event frequency is 1000/16250. This is reason for the factor of three reduction in the postulated failure frequency of the MCR.

7. The MCR breach failure rate calculated is an annual frequency that is independent of the plant operating status. If the refueling interval is assumed to be 30 days every 18 months, the likelihood of a MCR breach during refueling shutdown is then:

   1E-06 per year x 30/(1.5 * 365) = 5.5E-08 per refueling.

   During refueling, various maintenance openings to the Reactor Building will be open periodically to facilitate maintenance on components. If a MCR breach were to occur while one of the openings was in use, the Reactor Building would flood. This would not necessarily result in core damage or release as the AC Independent Water Addition function provided by the portable diesel-driven firewater pump would still be available.

   The "Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009," approved February 2, 2009 (reference 1), contains screening criteria for external events other than fire and seismic events in Subsection 6-2.3. This Standard applies to an At-Power Level 1 PRA for operating nuclear power plants. An equivalent Low Power/Shutdown Standard is not yet approved; however, for the purposes of responding to this Request for Additional Information, the external event screening criteria in the published national standard are selected to provide a basis for the response provided below. In NUREG-1407 (reference 2), the NRC recommended a similar set of screening criteria for the Individual Plant Examination of External Events (IPEEE) required of all operating nuclear power plants.

   In ASME/ANS RA-Sa-2009, Subsection 6-2.3, the fundamental criteria for screening external events other than fire and seismic events are as below:

   "There are three fundamental screening criteria embedded in the requirements here, as follows. An event can be screened out either
   (a) if it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) or a later revision; or
   (b) if it can be shown using a demonstrably conservative analysis that the mean value of the frequency of the design-basis hazard used in the plant design is less than ~10^{-5}/yr and that the conditional core damage probability is <10^{-1}, given the occurrence of the design-basis hazard event; or
   (c) if it can be shown using a demonstrably conservative analysis that the CDF is <10^{-6}/yr."

   The MCR design basis flood event during shutdown would screen from further PRA evaluation using criterion (c) above.

8. The screening calculation of MCR breach referenced in this RAI was performed at the same time that the revised MCR breach analysis described in Section 2.4S.4.4.1.2 was being developed. The assumptions, insights, and conclusions reflect the design basis flood calculation described in FSAR Section 2.4S.4.

9. a. The timing evaluation for the operator action to close the normally open watertight door from the Service Building to the Control Room and the two watertight doors in the Reactor Building Access Corridor is based on notification from the normal access control point South of Units 3 and 4. The indication of severe flooding from a MCR breach which is time zero, is water entering the normal security access point doors which are at El. 32.0 ft. A more likely indication of issues with the MCR water retaining structure would be a result of the daily inspections or the monthly piezometric measurements of the relief wells. In this case, response measures would be implemented based on engineering recommendations.

   b. Procedures are controlled and maintained in accordance with the established site administrative processes as described in Section 13.5 of the FSAR.

   c. As described in response 9.a. , there is very little likelihood of ambiguous visual indication for the design basis MCR breach scenario. No other design basis external flooding event produces water above the site grade as quickly, and the indication is water coming through the door.

   d. As part of the MCR design basis flood reevaluation, a timing study was performed by the contractor to determine the minimum time for operator action assuming a water level change from elevation 32.0 feet, the elevation of the South access point, to elevation 35.0 feet, the entrance to safety-related structures. As determined in the MCR embankment breach calculation, Warning Time to Close Flood Doors at STP 3 & 4 Power Block Buildings, approximately 30 minutes are available from water entering the Security Access Building to the South edge of the safety-related structures for Units 3 and 4.

REFERENCES

1. Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009, February 2, 2009, American Society for Mechanical Engineers and American Nuclear Society.
2. “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” Report NUREG-1407, U.S. Nuclear Regulatory Commission (1991).

FSAR Changes

FSAR Section 2.4S.10 will be revised as shown below.

2.4S.10 Flooding Protection Requirements

In addition to structural protection against static, dynamic, and erosion flood forces, the safety-related facilities must remain free from flooding and intrusion of water into areas that contain safety-related equipment. All safety-related facilities in the power block are designed to be water tight at or below elevation 40.0 ft MSL. With the exception of the normal access watertight door between the Service Building and the Control Room and the two watertight doors on the Reactor Building Access Corridor, all ~~All~~ water tight doors and hatches are normally closed under administrative controls and all open outward. All ventilation openings are located above elevation 40.0 ft MSL. The UHS and Pump House is designed to be watertight below 50 ft MSL.

~~An MCR embankment breach near the STP 3 & 4 power block area would not provide sufficient time for implementation of emergency operating procedures or flood warning systems. As all water-tight doors and hatches are to remain in a closed position, no emergency operating procedures or plant Technical Specifications (plant shutdown), which are discussed in Subsection 2.4S.14, are required for implementation of flood protection measures.~~

FSAR Section 2.4S.14 will be revised as shown below.

2.4S.14 Technical Specifications and Emergency Operation Requirements

Specific flood protection measures are described in Subsection 2.4S.10. To withstand the static and dynamic forces as a result of the MCR embankment breach, watertight flood protection measures and structural measures are applied to any STP 3 & 4 facilities that have an open passageway to any safety-related facility. With the exception of the normal access watertight door between the Service Building and the Control Room and the two watertight doors on the Reactor Building Access Corridor, ~~Since~~ all watertight doors and hatches for these facilities, at or below 40.0 ft. MSL are to remain in a closed position under administrative control.~~, no emergency operating procedures or plant technical specifications (plant shutdown) are required for implementation of flood protection measures.~~ The emergency procedures for MCR breach described in Section 19.9.3 will be developed consistent with the plant operating procedure development plan in Section 13.5.

FSAR Appendix 19R will be revised as shown below.

19R.6.4 Operator Actions

(4) Ensure that the ~~*Close*~~ *watertight door at* the *entrance to the control room area* and the two watertight doors at the entrances to the Reactor Building Access Corridor ~~is~~ are closed *if floods in the turbine building result in service building flooding.*

19R.7 External Flooding Evaluation

Summarized in the sections below is the external flooding PRA analyses for the STP 3 & 4 plants. External flooding is defined as intrusion of water from sources outside of plant buildings such that the ability of the plant to achieve safe shutdown is affected. The analysis determined the potential core damage frequency (CDF) that could result from external flooding events for each of the new units and was developed assuming that the watertight doors providing normal access to the main control room and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ open. This assumption provides a conservative and bounding assessment of risk from external flooding.

19R.7.4.1 Main Cooling Reservoir Breach

Note that this analysis is developed assuming that the watertight doors providing normal access to the main control room and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ open. This assumption provides a conservative and bounding assessment of risk from external flooding.

With the exception of the normally open access door to the control ~~building~~ room from the service building and the two watertight doors in the Reactor Building Access Corridor, 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.

The normal access to the main control ~~building~~ room is via the service building through a watertight door on the 2950 mm elevation (elevation 35.0). In addition, there are two access doors to the Reactor Building Access Corridor in the Control Building, one from the Service Building and one from the Radwaste Building (elevation 18' 6 ½"). As discussed above, this analysis assumes that ~~this~~ these doors are ~~door is~~ open. The doors are ~~door is~~ oriented such that water external to the control building will seal the door. 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 assumed that a main cooling reservoir breach would result in water entering the service building. If any one of the doors from the service building to the control building is not closed or 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 would result in core damage.

When notified of a main cooling reservoir breach by security personnel, the operators in the main control room staff would ensure that the normally-open, watertight control room access door and the two watertight doors in the Reactor Building Access Corridor are ~~is~~ closed. Closing these doors ~~this door~~ prevents water from entering the control building.

As discussed above, failure to close these doors ~~this door~~ would result in submerging the control building and is conservatively assumed to result in core damage.

If the door to the main control room and the doors to the Reactor Building Access Corridor are ~~is~~ closed, then the event progresses as a loss of offsite power since it is assumed that the MCR breach causes a loss of offsite power.

19R.7.5.1 Main Cooling Reservoir Breach Accident

OCD - Operator Action To Close Control Room Watertight Access Door or RB/CB External 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. Failure of this top event can occur from two causes. First, the operators can fail to close the normally open, watertight doors that provides main control room access from the service building.and the two watertight doors that provide access to the Reactor Building Access Corridor. As described in section above, security personnel are stationed such that they will have a clear view of the area between the main cooling reservoir and plant buildings. This analysis assumes that the security staff is trained and that procedures are in place for them to alert the control room if there are indications of a breach of the main cooling reservoir. Procedures are also assumed to be in place to direct that the main control room access door and the Reactor Building Access Corridor doors be closed immediately on notification of a potential external flooding event (Refer to Section 19.9.3). Furthermore, the analysis assumes that the area between the main cooling reservoir and plant buildings is lighted to an extent that any flow of water from a breach of the main cooling reservoir would be clearly visible to the security personnel at night.

The main cooling reservoir breach analysis described in Section 2.4S.4 was used to develop a minimum available warning time from water at the South Security Gate House, approximately El. 32.0' MSL, to water at the entrances to safety-related buildings, El. 35.0' MSL. At least 30 minutes is available for operator action to close the normally open access doors between the Service Building and the Control Building and the Reactor Building Access Corridor doors once water reaches the South Security Gate House. Once the security staff notifies the control room of the breach, closing and securing the watertight doors takes less than five minutes ~~one minute~~. Therefore, it is assumed that a moderate and adequate amount of time is available to effect the actions to close the ~~control room~~ access doors. Then the failure probability for this event was assigned using the values in the Standard Safety Analysis Report (SSAR) Table 19R-4.

Even if operator action to close the normally-open doors is successful, failure of any one of the watertight doors that allow access to the reactor building or control building could randomly fail. Using the values in the SSAR Table 19R-4, the probability of random door failures that allow water to enter either the control building or the reactor building was calculated.

19R.7.7 Operator Actions Related to External Flooding

One operator action is important to external flooding risk. This action, timely closure of the watertight doors at the entrance to the main control room and the two doors in the Reactor Building Access Corridor is similar to the event included in section 19R.6.4. However, the cues to initiate the action for the external flooding event is different than for internal flooding.

19R.7.9 Conclusions

The conclusions from the ABWR probabilistic external flooding analysis are that the risk from external flooding is acceptably low, even with the assumption that the watertight normal access door to the control room and the two watertight access doors to the Reactor Building Access Corridor are ~~is~~ open. It is also concluded that the incremental risk from external flooding events is within the goals for an increase in CDF or LERF.

FSAR Appendix 19K will be revised as shown below.

19K.10 Identification of Important Capabilities Outside the Control Room

The identified activities outside the control room are:

(8) Closing the normally-open watertight door to the control room and the two watertight doors on the Reactor Building Access Corridor on notification of a main cooling reservoir breach.

FSAR Section 19.4 will be revised as shown below.

19.4.5 ABWR Probabilistic Flooding Analysis

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 and the two access doors to the Reactor Building Access Corridor.

FSAR Section 19.8 will be revised as shown below.

19.8.5.3 Features Selected

Operator Check Watertight Doors are Dogged

The flooding analysis assumes that all watertight doors except the normally-open main control room access door and the two access doors to the Reactor Building Access Corridor, *are closed and dogged to prevent floods from propagating from one area to another* or from outside to the inside.

View of the Main Cooling Reservoir

Plant buildings are located such that security personnel will have a clear and unobstructed view of the main cooling reservoir. Having such a view allows for prompt notification of the main control room so that the normally-open watertight door to the main control room and the two access doors to the Reactor Building Access Corridor can be closed before failure of the main cooling reservoir could be expected to threaten the plant. The area between the plant and the main cooling reservoir is lighted so that clear views are provided at night.

Operator Actions to Ensure Integrity Against External Floods

In addition to having unobstructed views of the main cooling reservoir, security personnel will be trained to alert the main control room immediately to any indication of main cooling reservoir failure. On such notification, personnel in the main control room will ensure that the access door to the main control room and the two access doors to the Reactor Building Access Corridor are ~~is~~ closed immediately. Also, all external doors located below the maximum flood level will be closed and verified on notification of any upstream dam failures. The emergency procedures for Severe External Flooding ensure that watertight barriers are in place and external opening sandbagged prior to the arrival on site of high water levels from external flooding (COM 19.9-3).

FSAR Section 19.9 will be revised as shown below.

19.9.3 Event Specific Procedures for Severe External Flooding

(1) Procedures and training will be developed to ensure that observation of the main cooling reservoir is conducted such that main control room personnel will be alerted on indications of a main cooling reservoir breach. These procedures will also direct that the main control room access door and the two access doors on the Reactor Building Access Corridor will be closed immediately on receipt of such notification.

FSAR Section 19.11 will be revised as shown below.

19.11 Human Action Overview

A new human action is modeled by the STP 3 & 4 external flooding analysis (Appendix 19R) to close the control room watertight access door and the two access doors to the Reactor Building Access Corridor in the event of an external flood. This action has been found to be important and meets the provisions identified in Subsection 19D.7 for important human actions and critical tasks. In addition, Subsection 19.9.3 documents the actions to be completed to ensure the human action's reliability.

QUESTION 19-3

Contributions to LRF and CCFP from severe accidents during low power or shutdown operations were not included in the ABWR SSAR or in the STP 3 & 4 FSAR. More recent design certification PRAs have shown that such scenarios are significant and sometimes dominant contributors to LRF and CCFP. Please discuss the impacts on LRF and the overall CCFP from low power and shutdown scenarios for STP 3 & 4. In addition, please explain whether or not the deletion of the Flammability Control System, including the recombiners, from the STP 3 & 4 design, affects the consideration of hydrogen combustion during the startup/shutdown periods when the containment may not be inerted.

REVISED RESPONSE

The following information revises in its entirety and supersedes the response submitted as Attachment 2 to the letter from Scott Head to Document Control Desk, "Response to Request for Additional Information," dated September 15, 2009, U7-C-STP-NRC-090144 (ML092600801).

DCD Appendix 19QB, which is incorporated by reference in the STP 3 & 4 COLA, discusses potential offsite releases during shutdown. The DCD also considered containment integrity in evaluating the risk during low power and shutdown conditions in DCD Appendix 19Q.4.3, which is incorporated by reference in the STP 3 & 4 COLA. There are no departures that affect the referenced discussions in Appendix 19Q on containment integrity during shutdown. Further discussion of the ABWR containment and offsite releases during shutdown accident scenarios is provided in DCD Appendix 19L.8 which is incorporated by reference in the STP 3 & 4 COLA. There are no departures that affect Section 19L.8. As indicated, severe accidents and offsite releases from Low Power/Shutdown scenarios were evaluated as part of the Design Certification process for the ABWR and there are no departures to the Certified Design that affect these evaluations.

The hydrogen recombiners are removed from the ABWR design as described in Departure STD DEP T1 2.14-1, which incorporates amendments made to 10 CFR 50.44, "Combustible gas control for nuclear power reactors," The amended 10 CFR 50.44 eliminates the requirements for hydrogen control systems to mitigate a design basis LOCA hydrogen release.

The white paper included with this revised response discusses hydrogen control issues and the effect of the removal of the hydrogen recombiners for all operating modes, including low power and shutdown, with an inerted or deinerted containment. As this white paper shows, removal of the hydrogen recombiners is expected to have little or no impact on the LRF during low power and shutdown conditions when the containment is not inerted. Existing ABWR emergency response guidance described in Section 18 of the COLA provides the necessary considerations for the control of hydrogen released during severe accident conditions with the containment deinerted.

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

HYDROGEN RECOMBINERS

HYDROGEN RECOMBINERS

Purpose

The purpose of this summary is to identify the basis for the removal of the requirements for hydrogen recombiners in operating and future plants with inerted containments.

Background

The Three Mile Island (TMI) Unit 2 accident resulted in the generation of hydrogen and release to the containment plus evidence of an ensuing hydrogen deflagration as a result of the core damage that occurred. Lessons learned from the accident led to the establishment of new and revised requirements, including a significant revision to 10CFR50.44 "Standards for combustible gas control systems in light-water-cooled power reactors". This rule was first established in October 1978 and was amended several times prior to the latest amendment dated October 2003. (Reference [1])

All combustible gas control amendments prior to the October 2003 amendment were based on the need to design for combustible hydrogen gases being produced as a result of a postulated design basis LOCA. Various requirements were imposed on all BWRs depending on the type of containment design including the need for recombiners.

The October 2003 amendment is based on risk insights for accidents and studies performed on hydrogen production from accidents. Conclusions of the amended rule are that the hydrogen release from a design-basis LOCA is not risk significant and the risk associated with hydrogen combustion is from beyond design basis severe accidents. This resulted in the determination that hydrogen recombiners are no longer needed and changes can be made to hydrogen and oxygen monitors, which no longer need to meet the requirements of safety related instrumentation. (References [2,3,4])

The requirement for a hydrogen control system to deal with the slow evolution of hydrogen following a Design Basis Accident (DBA) LOCA was a requirement of the original rule. The installation of recombiners and/or vent and purge systems addressed the limited quantity and rate of hydrogen generation that was postulated in the original rule. Recombiners can only deal with a very limited amount of hydrogen and would be completely overwhelmed by the quantity and rate of hydrogen expected to be evolved during the early stages of a core melt accident. Therefore, recombiners are not useful during the three severe accident time regimes (before vessel breach, at vessel breach, after vessel breach) and do not contribute to reducing the risk estimates. (Reference [2])

10CFR50.44 has been modified by the NRC in light of the analysis of hydrogen production during DBA and severe accidents. The latest 10CFR50.44 (Reference [1]) does not require hydrogen recombiners in those plants with an inerted containment.

Discussion of Rule

The installation of hydrogen recombiners and/or vent and purge systems required by 10CFR50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. (Reference [4]) The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity. As stated in the rule change, since hydrogen recombiners are no longer required to respond to a LOCA, the hydrogen recombiners no longer meet Criterion 3 or any of the other criteria for retention in the Technical Specifications. Therefore, the revised rule (Reference [1]) states that the requirements related to hydrogen recombiners currently in the Improved Standard Technical Specifications (ISTS) no longer meet the criteria of 10CFR 50.36(c)(2)(ii) for retention in the Technical Specifications and may be eliminated.

Risk Significance

The risk significance of the systems used to meet the post-LOCA combustible gas requirements of 10CFR50.44 is low. The risk of the design basis LOCA accident itself is low. The recombiners can only process a very limited amount of hydrogen and would be completely overwhelmed by the quantity and rate of hydrogen expected to be evolved in a more risk significant severe accident.

This decision to allow removal of the hydrogen recombiners was based on the following:

- For DBA, NRC noted in Reference [3]:

  "… experience and experiment have shown that during accidents involving core damage sufficient quantities of combustible gases can be evolved to pose a threat for some containments. The most significant risk appears to come from full core melt accidents, which include in-vessel clad metal/water reaction, potentially large quantities of hydrogen entering the containment at vessel failure, and the possibility of core-concrete interaction as the accident continues. On the other hand, design basis LOCA accidents, which involve only minor clad oxidation and in which the reactor vessel and containment does not fail, are not contributors to risk."

- For severe accidents from full power, the hydrogen generation rate is sufficiently large that the H_2 recombiner is not effective in removing H_2 fast enough to prevent a deflagration in a deinerted containment. Inerting is an effective hydrogen control system for all risk-significant degraded core and full core melt accidents in these containments. (Reference [3])

In addition, it is noted that:

- For severe accidents from low power or shutdown with the containment intact, the hydrogen production would be similar to that from full power, although the time to reach the critical hydrogen production rate may be longer (i.e., longer time available for recovery).

  Because small differences in long duration recovery actions are not well characterized using available data or HRA methods, it is judged that the differences between effects of an at-power accident response or a shutdown accident response are not distinguishable.

- For severe accidents from low power without the containment intact, the H_2 recombiners are not effective and their presence is moot.

  During typical refuel operations, the containment would be deinerted and opened within the first 24 hours.

Following issuance of the revised 10CFR50.44 (Reference [1]) recombiners are being removed from their Technical Specifications according to TSTF-447, Reference [4].

Defense-in-Depth

An inerted containment atmosphere ensures there is insufficient oxygen to burn with any hydrogen in the containment.

When inerting systems are unavailable or incapable of controlling combustible gas concentrations, the decisions and actions governing operation of drywell and suppression pool sprays provide a strategy to mitigate the consequences of a hydrogen generation event. Spray operation:

- Reduces containment pressure
- Reduces the flammability of combustible gases through the addition of water vapor to the gas mixture

- Suppresses the temperature and pressure increase following combustion if a deflagration does occur
- Scrubs the containment atmosphere in anticipation of radioactivity release
- Mixes the containment atmosphere to reduce localized buildup of combustible gases

Successful spray operation may also prevent containment venting at rates beyond allowable offsite radioactivity release rate limits for combustible gas control or delay its requirement until systems designed to control combustible gas concentrations can be restored to service.

Therefore, the BWROG EPGs provide the defense-in-depth procedures to cope with combustible gas mixtures when the containment is deinerted. The actions to be taken by the crew include use of drywell sprays and judicious containment purge and vent operation.

Removal of the hydrogen recombiners has no adverse effect on the availability of other systems included in plant-specific SAMGs for combustible gas control.

<u>STP Configuration Impacts</u>

The summary of STP 3&4 plant configurations that may influence combustible gas control strategies using recombiners is provided as follows:

<u>At-Power and Inerted</u>:	STP 3&4 is inerted and this provides the adequate protection of the public from combustible gas mixtures for a DBA or severe accident as indicated in the analysis supporting 10CFR50.44. (References [2,3])
<u>At-Power and Deinerted</u>:	STP 3&4 will have very short durations (governed by Technical Specifications)

during which the containment is allowed to be deinerted with the reactor at power. Risk analyses of operating BWRs have shown that the risk associated with these deinerted times is very low (ref. TS 3.6.3.2).

The hydrogen recombiners are ineffective in treating the risk dominant severe accident challenges of hydrogen production during these deinerted times and therefore would not alter the risk profile.

<u>Shutdown and Inerted</u>:	See the At-Power discussion.
<u>Shutdown and Deinerted and Containment Isolated</u>:	This condition is found to be one of extremely short duration and of lower risk because of the longer allowed times to take mitigative actions. (See the discussion of At-Power and Deinerted.)
<u>Shutdown and Deinerted and Containment Not Isolated:</u>	For these cases, the recombiners would be ineffective because the containment is open to the environment (Reactor Building). The primary method of combustible gas control under these conditions is the purging and venting of any gases from the containment to the environment via the Secondary Containment using normal ventilation systems.

Summary

As part of the revised combustible gas control rule (10CFR50.44, Reference [1]) and the Improved Standard Technical Specifications (ISTS), the hydrogen recombiners are no longer needed to respond to a DBA because of the minimal hydrogen and oxygen generation. In addition, hydrogen recombiners are ineffective in mitigating the risk significant severe accident combustible gas mixtures.

Therefore, the STP 3&4 ABWR has eliminated the recombiners from the design as not required to meet the design requirements and are not needed to reduce risk associated with severe accidents either at-power or during shutdown.

REFERENCES

[1] Code of Federal Regulations 10CFR50.44, 2003.

[2] Mary Drouin, et al., "Feasibility Study for a Risk-Informed Alternative to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-water-cooled Power Reactors", Office of Nuclear Regulatory Research, August 2000.

[3] Memorandum To Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, From Ashok C. Thadani, Director, Office of nuclear Regulatory Research, Subject: RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

[4] Letter from Anthony R. Pietrangelo (NEI) to Dr. William D. Beckner, Program Director, Operating Reactor Improvements Program, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, regarding TSTF-447, Revision 1 – Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors, dated May 12, 2003.

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QUESTION 19-14

Table 19.2-2 of the STP COLA, Revision 2, describes updated Reactor Building Cooling Water System (STD DEP 9.2-1). Table 19.2-2 states that this change is a clarification to text but the Departures Report states that this is a design capacity change. Please clarify and explain how the PRA results are affected due to the design capacity change of the Reactor Building Cooling Water System.

SUPPLEMENTAL RESPONSE

The following information supplements and supersedes the response submitted as Attachment 14 to the letter from Mark McBurnett to Document Control Desk, "Response to Request for Additional Information," dated July 13, 2009, U7-C-STP-NRC-090064 (ML092740559).

The process described in Regulatory Guide 1.206, C.III.I.19 was used to screen proposed departures and changes to the information presented in the ABWR Design Control Document. The screening process is controlled by a project procedure, U7-P-RA02-001, Screening Process for Plant Changes. The attached white paper, "PRA Screening Process for Plant Changes from the DCD" was discussed with the NRC during the audit of the STP 3 & 4 PRA, September 22 and 23, 2009. As a result of the Screening Process, Table 19.2-2 will be modified as shown below to reflect consistent nomenclature for the screening results. The changes in Table 19.2-2 also reflect changes associated with Requests for Additional Information not yet incorporated into the STP 3 & 4 COLA.

For a number of Tier 2 departures, the order in which the departures are listed in Table 19.2-2 has been rearranged to group Tier 2 departures according to those requiring prior NRC approval. The changes associated with rearranging the order are not shown in gray highlight below, but will be shown in the revision to the COLA.

Table 19.2-2 PRA Assessments of STP COLA Departures from ABWR DCD

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
Tier 1 (T1) Changes			
STD DEP T1 2.1-2 Reactor Pressure Vessel System	RIP motor casings do not have cladding.	The RIP is a Toshiba design in which the motor casings have cladding near stretch-tube portion and end of casing.	No effect on PRA, not modeled.
STD DEP T1 2.2-1 Control Systems Changes to Inputs, Tests, and Hardware	The reference ABWR DCD Tier 1 Table 2.2.1 ITTAC Acceptance Criteria for Item 11 states the "test signals exists in only one control channel at a time."	Only the power supply associated with the one non-Class 1E uninterruptible power supply being tested will become inoperable and both of the dual-redundant controller channels remain operational when this testing is conducted. This change also provides detail power supply design of RCIS in COLA Section 7.7.1.2(5).	No effect on PRA, not modeled.
STD DEP T1 2.3-1 Deletion of MSIV Closure and Scram on High Radiation	Design included MSIV trip on high radiation in steam tunnel	No MSIV trip on high radiation in steam tunnel	No effect on PRA, not modeled.
STD DEP T1 2.4-1 Residual Heat Removal System and Spent Fuel Pool Cooling	The ABWR has two RHR loops connected to the Fuel Pool Cooling and Cleanup System (FPCCS) with normally closed interties to permit supplemental cooling during refueling outages.	The current design has three RHR loops connected to the FPCCS with normally closed interties to permit additional supplemental cooling during refueling outages to reduce outage time.	Increasing the number of RHR loops connected to FPCCS from two to three is judged to have a negligible impact on CDF. It is an improvement in outage management control for the spent fuel cooling system. [See 19L.6.5, 19L.6.6 19L.8, 19L.9, Table 19L-9, 19Q.4.1, 19Q.4.2, 19Q.7.6, 19Q.7.7.1]
STD DEP T1 2.4-2 Feedwater Line Break Mitigation	For feedwater line break, feedwater flow assumed to be unavailable when hotwell inventory depleted, no automatic isolation feedwater flow.	Class 1E Breakers to trip condensate pumps required based on containment pressure analysis from feedwater Line break.	No effect on PRA, not modeled. Feedwater line break mitigation not specifically modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP T1 2.4-3 RCIC Turbine/Pump	RCIC-Terry type turbine	RCIC integrated pump and turbine	The new RCIC system has been designed for operation with fewer support systems than the previous design. This reduction of operational dependencies is expected to improve reliability. No changes, other than editorial, to the PRA. [See 19.3, 19.9.12, 19.9.30, 19.11, 19.13, 19K.3, 19K.11.1, 19K Tables, 19M.6.3]
STD DEP T1 2.12-1 Electrical Breaker/Fuse Coordination and Low Voltage Testing	Electrical Power distribution interrupting devices are coordinated such that the interrupting device closest to the fault opens first.	The description of interruption device coordination has been modified to include acceptable industry practice with standards and codes (e.g. IEEE 141, IEEE 242, etc). Change is made to address the exception to DCD Tier 1 requirements for circuits feeding small loads in circuits with standard size breakers/fuses for use in 120 Vac and 125Vdc panel boards.	No effect on PRA, not modeled.
STD DEP T1 2.12-2 I&C Power Divisions	Three Divisions of ~~Class 1E AC~~ Safety-Related Interruptible Instrument Power (Division I, II, and III)	Four divisions of ~~Class 1E AC~~ Safety-Related Interruptible Instrument Power (Division I, II, III, and IV). Division IV powered from Division II 480V MCC. Division IV power supplied to the safety-related Distributed Control and Information System (DCIS) Division IV	~~No quantifiable effect on the model.~~ No effect on the PRA, not modeled. [See 19L.6.6, ~~Table 19L.8-4, 19N,~~ 19Q.4.4]
STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination	Contains two redundant hydrogen recombiners and safety related oxygen/hydrogen analyzers.	Hydrogen recombiners are eliminated and Hydrogen and Oxygen analyzers are maintained, however downgraded to non-safety related.	No effect on PRA, not modeled. [See 19A, 19B, 19E, 19M]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP T1 2.15-1 Radwaste Building Reclassification	The radwaste building structure is Seismic Category I.	The radwaste building structure is not classified as Seismic Category I, consistent with the design for previous nuclear plants and Regulatory Guide 1.143, Rev. 2.	No effect on PRA, not modeled. Editorial changes [See 19.4, 19H]
STD DEP T1 2.15-2 Diesel Generator Supplemental Cooling	ABWR DCD Tier 1 Subsection 2.15.5 describes the operation and setting of the Reactor Building Safety-Related DG HVAC System to control temperature in the diesel generator (DG) engine rooms during DG operation and states the maximum temperature limit in the room is 50º C.	This departure revises the DG engine room maximum temperature limit during DG operation to 60º C.	No direct effect on the PRA, not explicitly modeled. Equipment ~~assumed to be~~ will be qualified for the environment. DG control panels are cooled by reactor building HVAC in separate rooms and are not affected by this change.
STD DEP T1 3.4-1 Safety-Related I&C Architecture	The ABWR DCD inconsistently describes and ESF architecture that sometimes applies a dual train SLU structure for all ESF functions, while at other times applies it to a very limited set of ESF functions. The ABWR DCD also describes RMUs as strictly processing input and output signals, while CMUs (Control Room Multiplexing Units) strictly perform logic control.	The current design limits the application of the dual train SLU architecture of the limited set of ESF functions. It also allows Remote DLCs to perform some control logic functions. It also replaces the concept of CMUs in the control room with Voter Logic Units (VLUs) in the control room that perform all of the 2-out-of-4 voting trip logic.	~~A delta-PRA assessment was performed to assess the updates affect on the instrument fault trees and common cause failures of the EMUX and the Chapter 19D fault trees and Chapter 19N CCF.~~ A review was performed to assess the new proposed design effect on the instrument fault trees and common cause failures of the I& C system described in the Chapter 19 Appendix 19D fault trees and Appendix 19N CCF. Other than nomenclature changes for the functions modeled, no changes to the PRA I&C models were made. No change to the results or conclusions of the PRA were identified as a result of this review. [See 19.1, 19.3, 19.7, 19.8, 19.9.8, 19.11, 19K, 19M, Tables, 19N, 19Q, 19QC]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP T1 5.0-1 Site Parameters	Site parameters were chosen to bound most potential US sites.	The design basis flood level ~~increased in order to handle~~ is based on a main cooling reservoir failure. The maximum precipitation rate for rainfall is increased from 49.3 cm/hr to 50.3 cm/hr based on meteorology studies. The humidity as measured from wet bulb temperature has been increased. The STP site does not satisfy the minimum shear wave velocity of 305 m/s (1000 ft/s). The shear wave velocity varies horizontally within a soil strata and vertically with depth. A site-specific soil structure interaction (SSI) analysis has been performed to confirm that the site specific SSI is bound by the DCD SSI.	The design basis external flood is included in the PRA ~~evaluated in Chapter 19R~~. [See 19.3, 19.8, 19.9, 19.11, 19.13, 19K, 19Q, and 19R]. The humidity, precipitation rate, and shear wave velocity exceptions do not affect the PRA.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
Tier 2 (T2) Changes Affecting Technical Specifications, Prior NRC Approval Required			
STD DEP 6.2-2 Containment Analysis	DCD assumptions resulted in potentially non-conservative calculated containment temperature and pressure responses following a feedwater line or steam line break.	Design assumptions for Feedwater Line Break (FWLB) have been updated. ANSI/ANS 5.1 1979 sets forth methods for calculating decay heat power from fission products, U239 and Np239 following shutdown of light water reactors.	No effect on the PRA, not modeled.
STD DEP 7.3-12 Leak Detection and Isolation System Sump Monitoring	--	Technical Specification 3.4.3 (LCO, Actions B.1 and B.2, SR 3.4.3.1) and its associated Bases (Applicable Safety Analysis, LCO, Actions B.1 and B.2) are changed to show the new leakage values and the addition of an "increase in unidentified leakage" parameter.	No effect on the PRA, not modeled.
STD DEP 7.3-17 ADS Electrical Interface	--	This change clarifies that the control logic is only performed in Division I, II, and III, conforming with the three divisions of ECCS, however sensor signals come from all four divisions.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.5-1 Post-Accident Monitoring (Drywell Pressure)	The details of the Post Accident Monitoring System (PAM) and Post Accident Sampling System (PAS) do not fully comply with subsequent regulatory updated requirements related to RG 1.97.	The PAM and PAS will be designed to fully comply with RG 1.97.	No effect on the PRA, not modeled.
STD DEP 7.7-10 Control Rod Drive Control System Interface	--	The CRT display is replaced with the RCIS Dedicated Operator Interface, a flat panel touch screen. A discussion of the RAPI enforcing rod blocks based upon signals external and internal to the system is added.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.7-18 RCIS Operator Interface	--	New annunciation (alarms) for the RCIS - Rod insert block and RWM Trouble. Some status information is now shown on MCRP display. Logic and control actions available on the Dedicated Operator's Panel.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 8.3-1 Plant Medium Voltage Electrical System Design	Only 6.9kV; ESF busses fed directly from UAT and RAT.	Two medium voltage systems 13.8 kV/4.6 kV. PG buses changed to 13.8 kV. Class 1E and PIP buses changed to 4.16 kV. Class 1E 4.16kV still fed directly from Unit Auxiliary Transformers (UATs). Two Reserve Auxiliary Transformers (RATs). 13.8 kV Combustion Turbine Generator with increased rating (20 Mwe). Emergency Diesel Generator changed to 4.16 kV, rating increased to 7200 kW. Larger RATs and UATs. Larger capacity Main Power Transformer.	~~Yes, a~~A delta-PRA assessment was performed using system fault trees on Figures 19D6.11, 12, & 13~~; and~~. The only change other than editorial to the fault trees is the addition of several breakers from the 13.8 kV CTG to the 4.16kV Class 1E buses and 4.16kV PIP buses. Changes incorporated into various sections of Chapter 19 that refer to the condensate pump and condensate booster pump being able to connect to CTG. [See 19.3, 19.7.3, 19.11, 19B, 19K Tables, 19L.8, and Table 19L-9, 19Q]
STP DEP 8.3-3 Electrical Site-Specific Power and Other Changes	--	Site specific changes include diesel generator loading calculations for sizing and drawing single lines to add site-specific power centers and motor control centers.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 10.4-5 Condensate and Feedwater System	3 Variable Speed (ASD driven) Reactor FW Pumps (booster and main pump), 33-67% NBR capacity and a Flow Control Valve in HP Heater Bypass line for startup/shutdown reactor level control. Normal rated power operation is with all 3 MD Reactor FW Pumps operating. If one operating Reactor Feedwater Pump	4 Variable Speed (ASD driven) Reactor FW Pumps and 4 condensate booster pumps, Low Flow Control Valve that provides for level control during startup/shutdown. Normal rated power operation is with 3 MD Reactor FW Pumps operating and one in auto standby. If one operating Reactor FW Pump trips, the Reactor FW Pump in auto start is not	No direct effect on the PRA, editorial ~~Editorial~~ change to the PRA to reflect the addition of the condensate booster pumps. [See 19.1, 19.3, 19.9, 19L, and 19Q]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
	trips, the other 2 operating reactor FW pumps must increase speed and discharge flows to maintain rated power operation. FWCS design for DCD is based upon above FW system design.	successful, automatic power reduction (by recirculation runback) occurs to avoid reactor scram.	

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
Tier 2 (T2) Changes Requiring Prior NRC Approval (Other than Affecting Technical Specifications)			
STD DEP 3B-2 Revised Pool Swell	--	The COL applicant no longer has access to the analytical codes described in DCD Section 3B Reference 14, and an alternate method is used to perform the revised pool swell analysis. This alternate method utilizes a calculation approach that is similar to the DCD approach; however, it uses some different assumptions and different analytical software for implementation of the analysis. The use of this alternate method to assess the pool swell results for the changes in the containment pressure response provides accurate results that are used as input for the wetwell internals design, and assures that these components will be adequately designed for the appropriate loads.	No effect on the PRA. The change modifies an analysis ~~anaylsis~~ method used in containment design.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
Tier 2 (T2) Changes			
STD DEP 1.1-1 Type of License Required	ABWR DCD was submitted for Design Certification.	The COLA is submitted to receive a Class 103 Combined License under 10 CFR 52.	No effect on the PRA, not modeled.
STP DEP 1.1-2 Dual Units at STP 3 & 4	Single Unit site.	Dual Unit site with common fire protection system.	No direct effect on PRA, editorial changes for fire protection system [See 19I.3-1, 19L.8, 19M.6.3, and 19Q.4.4]
STD DEP 1.2-1 Control Building Annex	Control Rod Drive Motor-Generator sets in Control Building.	Control rod drive motor generators and supporting equipment moved to Control Building annex.	No direct effect on PRA, editorial changes in 19M.6.3 for Fire Hazard reduction.
STP DEP 1.2-2 Turbine Building	A natural draft cooling tower is used for the heat sink.	Turbine Generator differs dimensionally, the main cooling reservoir is used for the heat sink.	No direct effect on the PRA, editorial change in 19R for level monitors associated with condenser cooling water and 19M for the evaluation of the effects on turbine building FIVE analysis.
STD DEP 1.8-1 Tier 2 Codes, Standards, and Regulatory Guide Edition Changes	The Civil design based on ASME B&PV Code Section III Division 2-1989, ACI 349-1980, and 1991 Uniform Building Code.	The Civil design based on ASME B&PV Code Section III Division 2-2001 with 2003 Addenda, ACI 349-1997, and 2006 International Building Code.	No effect on the PRA, not modeled.~~; the PRA considers all components that impact plant risk. The quality class of the component generally does not affect the modeling of the component within the PRA.~~.
STD DEP 1.AA-1 Shielding Design Review	Appendix 1AA provides the integrated doses for environmental qualification of safety-related equipment.	Doses have been re-evaluated incorporating results of design detailing.	No effect on PRA, not modeled. [design dose rates typically not modeled. in a PRA].

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 2.2-5 CRAC2 and MAACS Codes	CRAC2 computer code is used for accident analysis.	MAACS computer code is used for accident analysis, an improvement over CRAC2.	No direct effect on the PRA, ~~slight change to~~ no change to generic siting consequence analysis of the DCD. Site-specific consequence analysis uses new MAACS consequence analysis code. [See FSAR 2.2.2 and 2.2.3 for COL License Information Item 2.42, the Environmental Report, Chapter 7.2, FSAR 19.2.4.4, 19.3.4, 19E, 19E.3]
STP DEP 3.5-1 Missile Protection	Not required for single unit design, favorable orientation.	Provides Site Specific information relating to main steam turbine orientation in relation to essential systems of adjoining units.	No effect on the PRA, not modeled.
STD DEP 3.6-1 Steam Tunnel Concrete Thickness	The main steam tunnel design specifies a concrete thickness of 2 meters.	The main steam tunnel design considers shielding and structural requirements for determining concrete thickness.	No effect on the PRA, not modeled.
STD DEP 3.8-1 Resizing the Radwaste Building	--	Due to process changes described in STD DEP 11.2-1 and 11.4-1, the dimensions and design analysis for the Radwaste Building has changed from the DCD, revising its minimum bearing capacity, shear wave velocity, and Poisson ratio to reflect the shallower Radwaste Building Embedment.	No effect on the PRA, not modeled.
STD DEP 3.9-1 Reactor Internals Materials	Code Case 580-1 material is applied.	Code Case N5280-2 material is used, a nickel-based alloy.	No effect on the PRA, not modeled.
STD DEP 3B-1 Equation Error in Containment Impact Load	The multiplying factor "W" dimensions are seconds/foot.	In analyzing containment impact loads, the multiplying factor "W" is corrected to 0.0052 seconds/meter.	No effect on the PRA, not modeled.
STD DEP 3H-1 Liner Anchor Material	ABWR DCD Tier 2 Subsection 3H.1.4.4.3 incorrectly identifies the Containment Liner Anchor material	This departure corrects the Containment Liner Anchor material identified in Subsection 3H.1.4.4.3 to	No effect on the PRA, not modeled. [Building design details are outside the scope.]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
	as ASTM A-633 Gr. C.	SA-36.	
STD DEP 3I-2 Environmental Qualifications - Radiation	--	The "Integrated Dose-Gamma & Beta" values for the main steam tunnel is revised and instrument rack rooms is returned to the DCD value based on current results of post-accident radiation calculations and analysis.	No effect on the PRA, not modeled.
STD DEP 3MA-1 Interfacing LOCA	--	The ISLOCA evaluation is inconsistent with STD DEP T1 2.4-1, 2.4-3 and the COLA P&IDs. This departure corrects inconsistencies between Appendix 3MA and P&IDs in Chapter 21.	No ~~effect on~~ change to the PRA, clarification to text. ~~screened from evaluation due to piping redesign. [See 19.8 and 19B.2.45 of DCD]~~
STD DEP 4.5-1 Reactor Materials	--	The description of the materials for the control rod drive (CRD) mechanisms in Section 4.5.1 and the reactor internals in Section 4.5.2 of the DCD have been revised (1) to reflect the materials successfully used in operating ABWR designs over the last 10 years; (2) to clarify some data and provide equivalent materials, as appropriate; and (3) to clarify some fabrication and material issues for reactor internals materials.	No effect on the PRA, not modeled.
STD DEP 4.6-1 FMCRD Friction Test Equipment	FM~~RCF~~CRD friction testing utilizes a special test fixture connected to the HCU test port. The test fixture contains a small pump and associated hydraulic controls to pressurize the underside of the hollow piston of the FMCRD.	Water for the test equipment is supplied from the CRD pump discharge line. With this design, the test fixture pump can be eliminated.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 5.2-2 PSI/ISI NDE of the Reactor Coolant Pressure Boundary	PSI and ISI of welds in Reactor Coolant System meet requirements of Regulatory Guide 1.150, Rev.1.	PSI and ISI of welds in Reactor Coolant System piping meet the requirements of ASME Section XI, Appendix VIII as mandated by 10 CFR 50.55a.	No effect on the PRA, not modeled.
STD DEP 5.3-1 Reactor Pressure Vessel Material Surveillance Plan	--	Site specific supplement per COL License Information Item 5.5 in DCD 5.3.4.2.	No effect on the PRA, not modeled.
STD DEP 5.4-1 Reactor Water Cleanup System	Two 50% RWCU pumps (approximately 1% feedwater flow).	Flow capacity of pumps and filter demineralizers increased by 100% (approximately 2% feedwater flow). Pump discharge head increased.	~~Not explicitly modeled in the PRA.~~ Modeled in the shutdown PRA, no quantifiable effect in the PRA, operator action dominates the function, additional sources of shutdown cooling available. [See 19L.6.6, 19L.8, 19L.6.4, 19Q.4.1, 19Q.7.6, 19Q.7.7.1, and 19QB]
STD DEP 5.4-2 Reactor Recirculation System	--	Revised design of the RIP's cable box allows improved serviceability and maintainability because of smaller cable boxes and plug in power connector.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 5.4-3 Residual Heat Removal System Interlock	(1) The RHR IBD diagram includes an interlock that will close the wetwell spray valve in the low pressure flooder (LPF) mode. The statement that the wetwell spray can be operated with the system in the LPF mode is incorrect. (2) Table 5.4-3 indicates the open logic for the minimum flow valve is "pump running AND low loop flow signal", logic diagram indicates "pump discharge pressure high AND low loop flow"; (3) several pressure relief valves in Table 5.4-5 indicate relief pressure is 3.44 MPaG, however design pressure is 3.43MPaG.	Items (1) and (2) logic inconsistencies corrected; item (3) In Table 5.4-3, relief pressure for E11-F028A-C and E11-F051A-C are changed from 3.44 MPa to 3.43 MPa.	No effect on the PRA, not modeled.
STD DEP 5.4-4 RMC Heat Exchanger	Section 5.4.1 describes that the materials for the RMHX shell, shell tube sheet, and water box are carbon steel.	Stainless steel will be used for the RMHX shell, shell tube sheet, and water box.	No effect on the PRA, not modeled.
STD DEP 5.4-5 Reactor Head Vent Line (GI-195)	--	A vent line from the Reactor Water Cleanup System Reactor Pressure Vessel (RPV) head-spray line to the Reactor Head Vent Line is added in response to GI-195.	No effect on the PRA. The change eliminates a potential failure mode in the head vent line.
STD DEP 5A-1 Delete Appendix on Complying with Regulatory Guide	Text is included in Appendix 5A on complying with RG 1.150 which covers PSI and ISI welds in the Reactor Coolant System.	The text of Appendix 5A on complying with RG 1.150 has been deleted because PSI and ISI will be conducted in accordance with ASME Section XI.	No effect on the PRA, not modeled.
STD DEP 5B-1 Residual Heat Removal Flow and Heat Capacity Analysis	A factor related to RHR heat removal rate is 0.3705 MW/°C with an associated UHS water temperature of 29.4°C.	To support reduced outage times, the factor related to RHR heat removal rate is increased to 0.427 MW/°C and UHS water temperature is increased to 35°C.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 6.2-3 Containment Penetrations and Isolation	--	From first-of-a-kind efforts, further design details are included in Tables 6.2-7, 6.2-8, and 6.2-9 related to containment isolation valves, primary containment penetrations, and potential leakage paths. Based on equipment procurement, containment isolation valves associated with the ABWR Primary Containment have been adjusted. ABWR Primary Containment Penetrations have been modified to meet US mechanical and electrical separation requirements. Potential leakage paths from Primary Containment to the environment are included in Table 6.2-9.	~~No direct effect on PRA.~~ No change to the PRA. Clarification to text. This departure corrects primary containment penetration errors and inconsistencies in Section 6.2 of the reference ABWR DCD and provides additional design detail that was not present in the reference ABWR DCD.
STD DEP 6.6-1 Pre-service and Inservice Inspection and Testing of Class 2 and Class 4 Components and Piping	RHR heat exchanger nozzles are required to have 100% accessibility for PSI during fabrication. The use of some piping system configurations is restricted to ensure that accessibility for ISI is maintained.	The 100% accessibility for PSI of heat exchanger nozzles during fabrication is no longer applicable. Additionally, an evaluation is required to insure ISI accessibility is provided if some restricted piping system configurations are used.	No effect on the PRA, not modeled.
STD DEP 6C-1 Containment Debris Protection of ECCS Strainers	Conical strainer installed in DCD.	The model of strainer changed from conical suction strainer to CCI cassette type strainer which satisfies the requirements of Regulatory Guide 1.82, Rev. 3.	No change to the ABWR PRA, no change in function or failure data. [See 19L-8, 19Q.4.2]. Potentially an improvement for the plant-specific PRA.
STD DEP 7.1-1 References to Setpoints and Allowable Values	The Technical Specifications are formatted to include Allowable Values, Setpoints, and other calculations.	The NRC changed requirements for Technical Specifications to only include Allowable Values; the correct reference is to the methods for calculating the setpoints and margins.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.1-2 ATWS DB for Startup Range Neutron Monitoring	Miscellaneous changes to DCD descriptions.	(1) Section 7.1.2.4.1(2)(d) clarified description of power to the stepping motor driver modules derive their power from a bus that automatically receives power from EDG if necessary. (2) The SRNM subsystem provides ATWS permissive signals to the ESF logic and control system. (3) The APRM subsystem provides ATWS permissive signal to the ESF logic and control system.	No effect on the PRA, not modeled.
STD DEP 7.2-2 Description of Scram Actuating Relays	Relay logic contact status is specified as normally closed for air header dump valve solenoids.	Air header dump valve solenoid relay logic contact status is specified as "open" when the coil is "energized."	No change to the PRA. Clarification to text. ~~Correction to description. No effect on PRA, not modeled.~~
STD DEP 7.2-4 Manual Scram Monitoring	Two manual scram switches and the reactor mode switch provide means to manually initiate a reactor trip. Additionally, one bypass initiating variable is monitored in addition to the scram initiating variables.	No statement about monitoring initiating variables is included to eliminate possible misinterpretation.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.2-6 RPS Instrumentation Ranges	--	New specifications for Reactor Protection System Instrumentation (Reactor Vessel High Pressure, Drywell High Pressure, Reactor Vessel Low Water Level 3, Low Charging Pressure to Rod Control HCU Accumulators, Turbine Control Valve Fast Closure) are provided with ranges to optimize performance.	No effect on the PRA, not modeled.
STD DEP 7.3-1 Time Intervals for Accident Analysis	--	To insure consistency in information, input variables used in LOCA analysis are referenced to a table.	No ~~effect on~~ change to the PRA. Clarification to text.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.3-2 Automatic Depressurization Subsystem (ADS)	Actuation of the automatic safety/relief valves is described as "with electrical power."	Actuation of the automatic safety/relief valves utilizes pneumatics for the relieving function, but the operating air is introduced electrically through a solenoid valve.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-4 ADS Logic	The DCD does not clearly describe the logic and sequencing for the ADS.	The logic and sequencing for the ADS is fully described eliminating possible misinterpretations.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-5 Water Level Monitoring	The DCD describes the equipment design for the ADS and RHR/LPFL I&C using the terms "Low" and :Low-Low" when describing initiating inputs from the reactor water level instrumentation.	Nomenclature related to water level initiating inputs is clarified using terminology based on nominally quantified levels (terms such as "Level 1.5" and "Level 1" instead of "Low" and "Low-Low").	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-6 SRV Position Indication	In the main control room, position indication for safety/relief valves provides lights when solenoid-operated pilot valves are energized to open using LVDTs mounted on the valves.	Indication of safety/relief valve position is provided by a limit switch, giving a direct indication of the valve's position.	Not explicitly modeled in the PRA. Potentially a beneficial effect for the plant-specific PRA.
STD DEP 7.3-7 ADS Manual Control	ADS inhibit switch is a keylock type.	The ADS inhibit and SRV control switches are no longer the keylock type and the ADS manual actuation is now initiated by a single pushbutton.	Not explicitly modeled in the PRA. Potentially a beneficial effect for the plant-specific PRA.
STD DEP 7.3-9 Shutdown Cooling Operation	--	Clarifications are provided in describing RHR Shutdown Cooling Mode valve alignment during Low Pressure Flooder (LPFL) actuation signal.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-10 ESF Logic and Control System (ELCS) Mode	The operator may control the RHR pumps and injection valves manually after LPFL initiation to use RHR capabilities in other modes if the core is being cooled by other emergency core cooling systems.	An expanded description of mode switches in the main control room is provided. To reduce operator burden and support the displays, RHR has specific mode operation capability. Additionally, ELCS mode automatic	Not explicitly modeled in the PRA. Beneficial effect for the plant-specific PRA.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
		logic changes are implemented to insure that the HPCF "C" diverse hard-wired manual initiation function has priority over the normal automatic initiation logic for HPCF "C".	
STD DEP 7.3-11 Leak Detection and Isolation System Valve Leakage	Two sets of asbestos packing rings are provided with a leak-off line from the chamber between packing rings routed to a collection sump where leakage is identified.	Valves use one set of expanded graphite packing to seal the valve stem penetration, eliminating the need for a leakage detection system.	No effect on the PRA, not modeled.
STD DEP 7.3-13 Containment Spray Logic	If Containment Spray has been initiated, then the system automatically realigns to the LPFL Mode if Reactor Vessel Water Level falls below Level 1.	The LPFL mode has precedence over Containment Spray when below Level 1. Clarifications are provided in how Drywell and Wetwell Sprays can be initiated as well as the interlocks associated with this mode of RHR operation.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-14 Residual Heat Removal Suppression Pool Cooling	--	This departure corrects an inconsistency between COLA subsection 7.3.1.1.4 and ABWR DCD Tier 2 subsection 5.4.7.1.1.5 and Figure 7.3-4.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.3-15 Reactor Building Service Water Logic Interfaces	Divisions I and II provide flow signals to the Main Control Rooms for the Reactor Coolant Water controls.	All three divisions provide flow signals to the main control rooms.	~~Not explicitly~~ No effect on the PRA, not modeled. Potentially a beneficial effect for the plant-specific PRA.
STD DEP 7.3-16 Testing Safety Relief Valve Solenoid Valves	SRV pilot solenoid valves can only be tested when the reactor is not pressurized.	Improved testing capabilities have been incorporated into the ABWR design which allows testing to be performed at any pressure.	~~Not explicitly~~ No effect on the PRA, not modeled. Potentially a beneficial effect for the plant-specific PRA.
STD DEP 7.4-1 Alternate Rod Insertion	--	Multiple clarifications are made describing implementation of the ARI function.	No ~~effect on~~ change to the PRA. Clarification to text.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.4-2 Residual Heat Removal Alarm	DCD alarm name " RHR Logic Power Failure."	The alarm is replaced with the more general alarm "ELCS Out of Service." The "Manual Initiation Armed" alarm is clarified to only activate when the RHR system is in LFPL mode of operation.	No effect on the PRA, not modeled.
STD DEP 7.6-1 Oscillation Power Range Monitor (OPRM) Logic	Oscillation Power Range Monitor (OPRM) trip logic performed separately from the APRM trip logic.	OPRM trip logic decisions are made within the OPRM unit and provided to the RPS separately from the APRM trips.	No effect on the PRA, not modeled.
STD DEP 7.6-2 SPTM Subsystem of Reactor Trip and Isolation System	--	The SPTM system is clarified as part of the Reactor Trip and Isolation System.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.6-3 SPTM Sensor Arrangement	DCD Tier 2 Section 7.6.1.7.3(2) states that, "Each SRV in direct sight of two sets of temperature sensors within 9 meters."	Clarifies that the SRV discharge line quenchers are in direct sight.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.6-4 Range of Power Range Neutron Monitoring Operability	The PRNM provide information for monitoring average power level of the reactor core and monitoring the local power when the reactor power is in the power range (above approximately 15%).	For the PRNM to provide information, the power range begins at approximately 5%.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-1 RPV Water Level Instrumentation	All instrument lines are flushed even when they do not need to be.	Condensable gas build-up in reactor vessel reference leg water level instrument lines is addressed by using CRD water to continually flush instrument lines having condensing chambers.	~~Not explicitly modeled in the~~ No effect on the PRA, not modeled. Potentially a beneficial effect for the plant-specific PRA.
STD DEP 7.7-2 SRV Discharge Pipe Temperature Data Recording	--	Discharge temperatures of all the safety/relief valves are shown on an historian function in the control room.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.7-3 Feedwater Turbidity	Measurement of Feedwater turbidity is discussed.	Feedwater turbidity is not discussed; it is not considered to have any safety significance and no practical method has been developed for measurement.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-4 Automatic Power Regulator/Rod Control	--	The APR is clarified as the direct controlling system that interfaces with the RCIS for accomplishing automatic rod movement mode and the PGCS interfaces only with APR for initiating various reactor power change control tasks.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-5 Rod Control and Information System (RCIS) Display	--	Detailed information about available display information at the RCIS dedicated operator interface on the main control panel is provided.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-6 RCIS Commands	--	Redundant "command signals" are provided from RFCS to RCIS for the ARI function.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-7 RCIS Design	--	RCIS design details pertaining to the organization, classification, and/or terminology of component groupings have been modified. Additionally, a more complete design description is provided.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-9 Selected Control Rod Run-In (SCRRI) Function	--	As a secondary function, the SCRRI function provides mitigation of loss of a feedwater heating event.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-11 Rod Withdrawal Sequence Restrictions	--	Ganged Rod movement and ganged withdrawal sequence restrictions are expanded.	No effect on the PRA, not modeled.
STD DEP 7.7-12 RCIS Indication	--	Provides detailed design information including the reference rod pull sequence, RCIS capability, RCIS providing feedback signals, generation of a rod withdrawal block	No ~~effect on~~ change to the PRA. Clarification to text.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
		signal, and an audible alarm at the operators panel for a RRPS violation.	
STD DEP 7.7-13 Optical Isolation	Discusses the details of a specific technology that can be used for achieving optical isolation. However, the description is overly restrictive in describing a specific type of optical technology to be used for meeting the optical isolation.	The detailed description of the specific type of technology used for optical isolation is removed to prevent restricting the type of technology that can be used for achieving suitable optical isolation.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-14 RCIS Bypass	--	Clarification in design details for RCIS.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-20 Recirculation Flow Control Logic	The Recirculation Flow Control System automatically operates when above 70% power.	Information is provided concerning manual and automatic operation for other rod patterns and power levels; operation below 25% has been described and load follow capability has been enhanced.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-22 ATLM Description	--	The description of the ATLM setpoint and rod block action has been expanded to further describe the interface of the systems and the applications.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-23 Automatic Traversing Incore Probe (ATIP) Function	Gain adjustment factors for Local Power Range Monitoring uses inputs from the "Automatic Fixed Incore Probe (AFIP)."	Gain adjustment factors for local power range monitoring are provided by an Automatic Traversing Incore Probe (ATIP).	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 7.7-24 Steam Bypass and Pressure Control Interfaces	An external signal interface for the Steam Bypass and Pressure Control (SB&PC) System is narrow range dome pressure signals from SB&PC System to the Recirculation Flow Control System.	Narrow range dome pressure signals are replaced by "Validated dome pressure signals." Based on pressure demand, the SB&PC System calculates position error and servo current for each turbine valve.	No ~~effect on~~ change to the PRA. ~~Corrections~~Clarification to text.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 7.7-27 RCIS Table Deletion	Table 7.7-1 provides the environmental conditions for the Rod Control and Information System (RCIS) module operation environment.	Table 7.7-1 is deleted because its information is duplicated elsewhere in the FSAR.	No ~~effect on~~ change to the PRA. Clarification to text.
STP DEP 8.2-1 Electrical Equipment Numbering	--	The non-safety and safety-related medium voltage buses numbering conventions were changed. Figure 8.2-1, Sheets 1-7, have been revised to show the new bus numbers and equipment location in the turbine building.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 8A-1 Regulatory Guidance for Lightning Protection	--	This change acknowledges availability of SRP and regulatory guidance for the lightning protection system.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 9.1-1 Fuel Handling Cranes and Equipment	Tier 2 (FSAR/DCD) - Paragraph 9.1.2.1.2 fuel storage racks provided in spent fuel storage for 270% of one full core fuel load, which is equivalent to a minimum of 2354 fuel storage positions (assembles).	Fuel storage racks in spent fuel pool shall be 270% of one full core fuel load, which is equivalent to a minimum of 2354 assemblies. Pool design is capable of 3072 assemblies and at STP's option more racks can be provided as extra scope. DCD should be the basis for minimum racks.	No effect on the PRA, not modeled.
STD DEP 9.2-1 Reactor Building Cooling Water System	RCW heat exchanger design capacity for divisions A and B of 47.73 GJ/h; the capacity for division C is 44.38 GJ/h.	RCW heat exchanger design capacity for divisions A and B of 50.1 GJ/h; the capacity for division C is 46.1 GJ/h. These increased capacities are based on meeting LOCA heat loads with a margin of 20% to allow for fouling.	No ~~direct effect on~~ change to the PRA. ~~Clarification to text.~~This engineering change supports increased heat removal capacity and corrects inconsistencies in Section 9.2.11.2.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP 9.2-2 Makeup Water Preparation System	--	Changes specific to the operation of the Makeup Preparation Water (MWP) System including flow capacity, storage capacity, rate for providing dematerialized water, supply makeup water to the ultimate heat sink, etc.	No effect on the PRA, not modeled.
STP DEP 9.2-3 Turbine Building Cooling Water System	The heat removal capacity of each of the three heat exchangers in the Turbine Building Cooling Water System is 68.7 GJ/h with a flow rate of 3405 m3/h.	The heat removal capacity of each of the three heat exchangers in the Turbine Building Cooling Water System is increased to 114.5 GJ/h, using the increased flow rate of 4550 m3/h.	No effect on the PRA, not modeled.
STP DEP 9.2-5 Reactor Service Water (RSW) System	In the DCD, only the portion of the RSW in the CB was described. Remaining portion is not defined in the DCD (Paragraph 9.2.15).	RSW system design reflects new location of RSW pump house and increased system flow and discharge pressure necessary to meet the increased heat removal requirements of the reactor cooling water system. Cooling tower fans are added to the UHS [See site-specific Requirement for the UHS].	Included in the delta PRA [See 19.3, ~~19.4, 19.8,~~19.9~~, 19.11, 19K.3, 19L.11.1, Tables 19K-1 through 19K-4, 19Q, and 19R~~].
STD DEP 9.2-7 HVAC Normal Cooling Water System	--	This design change corrects inconsistencies in Tables 9.2-6 and 9.2-7, and Figure 9.2-2 such that the non-safety-related HVAC Normal Cooling Water (HNCW) system waterside heat removal rate is greater than or equal to the airside cooling duty heat loads.	No effect on the PRA, not modeled.
STP DEP 9.2-8 Potable and Sanitary Water System	--	The potable water subsystem is capable of supplying both STP 3&4 and the sewage treatment subsystem is capable of treating sanitary wastes collected from all four units located at the site.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP 9.2-9 HVAC Normal Cooling Water	--	Departure reduces equipment, piping, valve sizes, and electrical power for better maintainability, and changes return temperature from 12ºC to 14.7ºC.	No effect on the PRA, not modeled.
STP DEP 9.2-10 Turbine Service Water	--	Turbine Service Water (TSW) system interface requirements are revised to reflect site specific information.	No direct effect on PRA. Turbine building flooding tables associated with TSW modified to reflect site specific information. [See Table 19R-1]
STD DEP 9.3-1 Radwaste Drain Materials	Carbon steel pipe for majority of K11 Radioactive Drain System.	Stainless Steel for entire K11 Radioactive Drain system.	No effect on the PRA, not modeled.
STD DEP 9.3-2 Separate Breathing Air System	Breathing air system is included in service air system (P51).	Separate breathing air system (P81) from service air system (P51).	No effect on the PRA, not modeled.
STD DEP 9.3-3 Reactor Building Sampling Station	CRD water sampling is described in the DCD.	Because CRD system water is supplied from condensate water, CRD system sampling can be substituted by condensate system sampling. The condensate system monitors oxygen and conductivity. Process samples from the CRD are not needed.	No effect on the PRA, not modeled.
STP DEP 9.4-1 Service Building HVAC System	--	The HVAC System is revised to remove the provisions for toxic gas monitors and the TSC alarm for high toxic gas concentration.	No effect on the PRA, not modeled.
STD DEP 9.4-2 Control Building HVAC System	--	The control building HVAC system smoke removal mode is revised to include control room main air supply duct bypass lines around the air-handling unit with two motor operated dampers for each of the two control room habitability area HVAC divisions and each of the three safety-related equipment HVAC areas.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 9.4-3 Service Building HVAC System	Service building HVAC system has two subsystems, the clean air HVAC System and the Controlled Area HVAC System.	Subsystems are deleted and consolidated to supply air to both the Clean Area and the Controlled Area.	No effect on the PRA, not modeled.
STD DEP 9.4-4 Turbine Island HVAC System	--	Design changes include: additional supply/exhaust air flow, relocated electrical building into turbine building, increase in equipment quantities, additional condensate booster pumps, etc.	No effect on the PRA, not modeled.
STD DEP 9.4-5 Radwaste Building Ventilation	--	Eliminated HVAC equipment supporting the radwaste incinerator which was deleted. A dedicated air conditioning system for electrical, HVAC equipment rooms and other areas was added as a result of design evolution. Operation control of the exhaust air system form radwaste process area is augmented to automatically route the exhaust air through filtration equipment upon detection of airborne radioactivity.	No effect on the PRA, not modeled.
STD DEP 9.4-6 Control Building HVAC System	One flow element/flow switch in the common discharge duct of each emergency filtration unit.	A flow element/flow switch is to be installed on the discharge side of each emergency filtration unit fan using a two out of two logic signal for automatic switchover.	No effect on the PRA, not modeled.
STD DEP 9.4-7 Control Building Annex HVAC	MG set rooms are ventilated by C/B safety-related equipment area HVAC; cooling is provided by non-safety-related MG set room air handling unit.	MG set room air handling unit is independent of from C/B safety related equipment area HVAC.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP 9.4-8 Reactor Building HVAC System	--	Configuration of fans and air conditioning units (ACU) in Figure 9.4-3 modified because current configuration is inconsistent with Tier 1 Figure 2.15.5j. Fire damper is stated in Tier 2 9.4.5.5.2, but Tier 1 Figure 2.15.5i has no Fire Damper-the statement of Fire Damper in Tier 2 is eliminated.	No ~~effect on~~ change to the PRA. Clarification to text.
STD DEP 9.4-9 Turbine Building HVAC	--	The Turbine Building's exhaust system is changed and its HVAC recirculation duct is deleted.	No effect on the PRA, not modeled.
STD DEP 9.5-1 Diesel Generator Jacket Water Cooling Water System	Inspection and Testing requirements for the diesel generator jacket cooling water system conformed to RG 1.108.	The requirements have been integrated onto RG 1.9 Rev.4, endorsing IEEE-387, which addresses qualification and periodic testing of the diesel generators.	No effect on the PRA, not modeled. The effect of standards included in base failure data.
STD DEP 9.5-2 Lower Drywell Flooder Fusible Plug Valve	Contains specific design details about fusible plugs based on an old design concept and patent application; however the actual fusible plugs have never been built and tested.	The fusible plugs are described in generic terms of the design requirements and incorporate design experience from actual design and test results. Clarifications are made specifying the fusible plug ~~opening temperature,~~ lower drywell isolation valve details, etc.	No direct effect on the PRA, but described in Chapter 19. The change incorporates design experience which should decrease the likelihood of failure. [See 19E, 19E.2.8.2.1, 19E.2.8.2.6]
STD DEP 9.5-3 System Description - Reactor Internal Pump Motor	MG sets and adjustable speed drives described in DCD 9.5.10.2 and 7.7.1.3.	Several changes to the technical description of the non-safety Motor-Generator (MG) sets and ASD descriptions.	No effect on the PRA, not modeled.
STD DEP 9.5-4 Lighting and Servicing Power Supply System	Mercury lamps are provided for use for high ceilings, except where breakage could introduce mercury into the reactor coolant system.	The mercury lights are replaced with high pressure sodium (HPS) lamps.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
~~STD~~STP DEP 9.5-6 Diesel Generator Fuel Oil Storage and Transfer System	--	The sample connection for the Fuel Oil Storage Tank is relocated slightly above grade elevation, fill connection is relocated at grade elevation and vent is extended to an elevation exceeding maximum flood level. The Fuel Oil Storage Tanks are relocated in concrete vaults underground, with piping routed underground. Locked, closed isolation valves have been added to the fill and sample lines, and a second transfer pump for the Diesel Generator Fuel Oil system has been added.	Not explicitly modeled in the PRA. Potential beneficial effect (two fuel oil transfer pumps) for plant-specific PRA.
STP DEP 9.5-7 Fire Protection - House Boiler Area of the Turbine	The house boiler is a fuel oil-heated boiler.	The house boiler is an electrically heated boiler.	No effect on PRA fire modeling. Slight ~~positive effect~~ improvement in Turbine Building fire frequency.
STP DEP 10.1-1 Turbine Pressure Description	Inlet pressure at the turbine main steam valves is controlled by the pressure regulator such that turbine inlet pressure varies linearly with reactor power level.	The inlet pressure at the turbine main steam valves reflects reactor power, steam line flow and pressure regulator programming, but never exceeds the pressure for which the turbine components and steam lines are designed."	No effect on the PRA, not modeled.
STP DEP 10.1-2 Steam Cycle Diagram	Steam and power conversion system consists of four condensate pumps, two heater drain tanks, a typical multi-pressure condenser design, and a main turbine with the single stage reheat.	Four condensate booster pumps are added to this system, with three filters and six demineralizers, four reactor feed pumps, four heater drain pumps, one heater drain tank, and a turbine design with two stages of reheat.	No effect on PRA, not modeled. [See STD DEP 10.4-5]
STP DEP 10.1-3 Rated Heat Balance	--	Modified to reflect turbine manufacturer.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STP DEP 10.1-4 Valves Wide Open Heat Balance	--	Modified to reflect turbine manufacturer.	No effect on the PRA, not modeled.
STP DEP 10.2-1 Turbine Design	--	Modified to reflect turbine manufacturer, revised ISI and IST inspection intervals based on design.	No effect on the PRA, not modeled.
STP DEP 10.2-2 Turbine Rotor Design	--	Modified to reflect turbine manufacturer.	No effect on the PRA, not modeled. Turbine missile generation likelihood decreased.
STP DEP 10.2-3 Turbine Digital Control	--	Significant advancements in reliability and machine protection result through the use of a digital turbine control system.	No effect on the PRA, not modeled. Turbine trip function reliability enhanced.
STP DEP 10.2-4 Bulk Hydrogen Storage	Bulk hydrogen is stored near the turbine building.	Bulk hydrogen is stored well away from the power block buildings.	No effect on the PRA, not modeled.
STD DEP 10.3-1 Main Steam Line Drains	The drains from the steam lines inside containment are connected to the steam lines outside the containment to permit equalizing pressure across the MSIVs during startup and following steam line isolation.	The main steam system also serves as the "alternate leakage path" to contain the radioactive steam with passes the main steam isolation valve before they close to isolate the reactor under emergency conditions.	No effect on the PRA, not modeled.
STD DEP 10.4-1 Turbine Gland Seal Steam	--	A non-safety-related Gland Seal Evaporator (GSE) is added to the Turbine Gland Steam System to supply sealing steam to the main turbine shaft seal glands and various turbine valve stems, including the turbine bypass and main turbine stop-control valve stems.	No effect on the PRA, not modeled.
STP DEP 10.4-2 Main Condenser	MC utilizes three independent multi-pressure single-pass shells, with each shell containing at least two tube bundles, and series circulating water flow.	MC utilizes three condenser shells cross-connected to equalize pressure, with each shell containing four tube bundles, and parallel circulating water flow. Number of circulating water pumps increased to 4, flow	No effect on PRA, not modeled. Editorial changes in Chapter 19. [See 19R.4.3 and 19R.5.3]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
		rates modified.	
STP DEP 10.4-3 Main Condenser Evacuation System	Auxiliary boiler steam used for steam jet air ejectors during startup.	An additional vacuum pump is added and changes to the source of motive steam supplying the steam jet air ejectors during power operation.	No effect on the PRA, not modeled.
STD DEP 10.4-6 Load Rejection Capability	ABWR Standard design has a turbine bypass capacity of 33% of nuclear boiler rated flow.	A clarification is made in regards to reactor trip resulting from turbine trip or generator load rejection from power levels above 33%.	No effect on the PRA, not modeled.
STD DEP 10.4-7 Turbine Bypass Hydraulic Control	--	Indication for the use of valve position transmitters, one hydraulic accumulator for each bypass valve, the addition of the fast-acting solenoid valve, and the interface between the steam Bypass and Pressure Control System for positioning of the bypass valves.	No effect on the PRA, not modeled.
STD DEP 11.2-1 Liquid Radwaste Process Equipment	--	Information is replaced completely due to a change in the design of the liquid radioactive waste system. The liquid radwaste system is composed of three subsystems designed to collect, treat, and recycle or discharge different categories of waste water; the low conductivity subsystem, high conductivity subsystem, and detergent waste subsystem.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 11.3-1 Gaseous Waste Management System	Off-gas is exhausted along with SJAE discharge pressure, (needing the addition of vacuum pumps for stable exhaust during plant operation). Additionally, an integrated recombiner (combining the preheating unit and condensate unit) is applied.	Off-gas is exhausted along with SJAE discharge pressure, using vacuum pumps to stabilize exhaust during plant operation. Additionally, the recombiner has a preheating unit and condensate unit (each as a separate unit).	No effect on the PRA, not modeled.
STD DEP 11.4-1 Radioactive Solid Waste Update	--	Solidification System and the incinerator system are deleted because equipment operations and maintenance difficulties negatively impact the effectiveness of these processes. A second spent resin storage tank is added for separating two different resins. The SWMS mobile system consists of equipment modules, complete with all subcomponents, piping and instrumentation and controls necessary to operate the subsystem.	No effect on the PRA, not modeled.
STP DEP 11.5-1 Process and Effluent Radiation Monitoring and Sampling System	--	Implementation of specific equipment is vendor-based. Specific detector types will be selected at a later date based on the state of art and availability. Many additional changes have been made.	No effect on the PRA, not modeled.
STD DEP 12.3-1 Cobalt Content in Stainless Steel	--	Vendors supplying the materials cannot reasonably achieve the cobalt limits in all cases, so a graded approach is used to specify locations receiving the least.	No effect on the PRA, not modeled.
STD DEP 12.3-2 Deletion of CUW Backwash Tank Vent Charcoal Filter	The CUW vent for CUW backwash is fitted with a charcoal filter canister to reduce the omission of radioiodines into the plant	The CUW system contains charcoal filter on its vent. The CUW backwash tank is vented into the reactor building HVAC System	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
	atmosphere.	exhaust, exiting the plant via the plant stack as monitored release.	
STD DEP 12.3-3 Steam Tunnel Blowout Panels	The blowout panels for the steam tunnel are located in the relatively inaccessible section of the RHR heat exchanger shielded cubicle which are controlled access areas.	The design does not have blowout panels in the steam tunnel. The main steam tunnel is vented to the turbine building.	No effect on the PRA, not modeled.
STD DEP 12.3-4 Alarm Capability for Area Radiation Monitors (ARMs)	--	The ARMs will have alarm capability and five additional monitors are required in the Reactor Building.	No effect on the PRA, not modeled.
STD DEP 14.2-1 Control Rod Drive Friction Testing Requirements	--	Normal control rod positioning is accomplished by an electrical motor. Mechanical binding of a CRD will result in blade separation from the ball nut which would be detected by permanently installed instrumentation. The CRDs are easily monitored for performance degradation during normal withdrawal; therefore periodic friction testing is not required.	No effect on the PRA, not modeled.
STD DEP 16.2-1 thru STD DEP 16.5-46 Technical Specifications Changes	See COLA Part 7 for changes.	See COLA Part 7 for changes.	No effect on the PRA, not specifically modeled.
STD DEP 18.4-1 Main Generator Synchronization Control Relocation	--	The controls required for the synchronization of the main generator have been relocated from the control console to the main control panel.	No effect on the PRA, not modeled.

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP 19.3-1 Evaluation of Common Cause Failures	ABWR SSAR Chapter 19D.8.6 documents the results of a PRA sensitivity analysis on common cause failure of selected mechanical systems performed by GE.	Common cause factors were added to the ABWR plant model used to quantify the effects of plant-specific factors for South Texas Project Units 3 & 4 PRA.	Included in delta PRA assessment. [See Chapter 19.3] [This is not a departure from the design certified in the DCD]
STD DEP 19.7-1 Control Rod Drive Improvements	The FMCRD brake design has to be fully testable on an annual basis.	The FMCRD electro-mechanical brake is a Class 1E safety related component with a 10-year Environmental Qualification replacement life; brake performance characteristics testing is performed every two ~~to~~ years when a replacement brake is installed. It is recommended ~~Hitachi recommends~~ that approximately 20 motor sub-assembly units, including the brake, to be tested during the refueling outages.	No effect on PRA, not modeled. Editorial change to Chapter 19. [See Chapter 19.7.2]
STD DEP 19I.7-1 Atmospheric Control System Bypass Analysis	The seismic margins PRA for the Atmospheric Control System 50 mm crosstie valves requires the opening of two normally closed motor-operated valves to create a containment bypass path.	The analysis has been modified by replacing motor-operated valves with air-operated valves.	No effect on PRA, not modeled. Editorial change to Chapter 19. [See Chapter 19I.7]
STP DEP 19R-1 RSW Pump House Redesign	ABWR design, vertical RSW pumps, pump rooms protected from flooding from other pump rooms, pumps above water level in UHS.	STP design, RSW pumps located below UHS, pump rooms are protected by watertight doors between trains.	Control building flooding assessment is unaffected, RSW design modified for new RSW pump house design, vacuum breakers removed. [See 19R]. A delta-PRA assessment for flooding in redesigned RSW pump house [See 19.4, 19.7, 19.8, 19.9, 19.10, 19.11, 19K, 19Q, and 19R]

Departure Number	~~Certified~~ Design Basis ~~(DCD)~~	US ABWR/STP Design Bases	Potential Impact on PRA [STP COLA Section]
STD DEP VENDOR Vendor Replacement	The reference ABWR DCD was developed with numerous statements that activities during construction and startup would be performed in accordance with GE approval or oversight. The intent of these statements was to ensure that the designer was appropriately involved in startup testing or construction activities.	This standard departure replaces the terms such as GE, GEH, and General Electric with the generic term NSSS Vendor, with an alternative vendor specified, or in some cases has eliminated the term altogether. This departure also replaces General Electric Company's product references such as NEDEs and NEDOs with the corresponding reference of another ABWR vendor whose reference has been approved by the NRC for use in this application.	No effect on PRA. Editorial changes in references.

OTHER			
Site Specific Requirement UHS System Design	Spray Pond UHS with specific RBCW/TBCW, etc., in/out temperatures given based on generic site.	The UHS function is provided by mechanical draft cooling towers, which are sized to satisfy the results of temperature studies to confirm they are within envelopes specified in ABWR DCD design. One UHS and RSW pump house for each unit.	~~No direct effect in the PRA.~~ Forced draft fans (2 per division) included with RSW system model in site-specific PRA. [See STP DEP 19R-1 for RSW pump house flooding]

PRA Screening Process for Plant Changes from DCD

Background

Regulatory Guide 1.206, Combined License Applications for Nuclear Power Plants, in Section C.III.I.19 describes a process for developing the plant-specific PRA from the design-certification PRA including evaluating design changes and departures from the certified design. The recommended process is:

Identify any design changes or departures from the certified design.

- Map the design changes and departures onto specific PRA elements, recognizing that some design changes and departures may be unrelated to any PRA element (i.e., have no potential for affecting the results of the PRA).

- Develop screening criteria to determine which of the remaining design changes and departures should be included in the plant-specific PRA model. 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. Similarly, certain changes or deviations from the certified design or the certified design PRA need not be reflected in the plant-specific PRA as long as it can be shown that (1) they are not important changes or deviations, and (2) do not have a significant impact on the PRA results and insights.

- Develop the plant-specific PRA model by revising the design certification PRA to reflect the remaining design changes and departures.

- Develop revised results, including revised risk insights, for the plant-specific PRA.

The screening process used in developing the plant-specific PRA model for South Texas Project Units 3 and 4 followed this process in determining which proposed changes and departures required evaluation in the ABWR PRA.

Table 19.2-2 presents a summary of the screening process for proposed changes to and departures from the DCD. Thirteen Tier 1 departures, nine Tier 2 departures that affect Technical Specifications and require NRC review, one Tier 2 departure that changes a method of analysis and requires NRC review, one hundred and twenty eight Tier 2 departures (not including the Departures to Technical Specifications described below), and one site specific information change related to Ultimate Heat Sink Design are summarized in this Table. A PRA evaluation was performed for potential changes and departures that did not screen.

Departures to Technical Specifications, STD DEP 16.2-1 thru STD DEP 16.5-4, do not involve out of service times or surveillance test intervals and are screened from further analysis.

Design changes and departures unrelated to any PRA element are identified as “No effect on the PRA, Not Modeled” in Table 19.2-2 and screened from further analysis. Additional text may be included in Table 19.2-2 for clarification if the function or equipment is included in the Chapter 19 DCD or SSAR text or tables but not included in ABWR PRA. (92 changes)

Those changes and departures that result from clarifying statements in the DCD are identified as “No Change to the PRA. Clarification to text.” and screened from further analysis. (36 changes)

Design changes and departures that affect descriptions of functions, equipment, etc., but have no effect on PRA model elements, results or conclusions are identified as “No direct effect on the PRA” or “Not explicitly modeled” with additional clarifying remarks and a reference to Chapter 19 text, if necessary, that discusses the change in Table 19.2-2, and screened from further analysis. (12 changes)

Eleven changes remain after the preliminary screening described above. One additional change, the site-specific Ultimate Heat Sink (UHS) design, although not a departure from the certified design, is included in the evaluation process.

STD DEP T1 2.4-1, Residual Heat Removal System and Spent Fuel Pool Cooling. This departure is screened from further evaluation. Increasing the number of RHR loops connected to FPCCS from two to three is judged to have a negligible impact on core damage frequency (CDF) during Low Power and Shutdown. It is an improvement in outage management control for the spent fuel cooling system.

STD DEP T1 2.4-3, Reactor Core Isolation Cooling (RCIC) Turbine/Pump. The new RCIC system has been designed for operation with fewer support systems than the previous design. This reduction of operational dependencies is expected to improve reliability. Data from some operating utilities with the new monoblock turbine/pump design was obtained and reviewed. No change to failure data, or maintenance unavailabilities was made to the PRA model. The deleted support equipment are identified in the affected Chapter 19 sections.

STD DEP T1 3.4-1, Safety-Related I&C Architecture. A review was performed to assess the new proposed design effect on the instrument fault trees and common cause failures (CCFs) of the I& C system described in the Chapter 19 Appendix 19D fault trees and Appendix 19N CCF. Other than nomenclature changes for the functions modeled, no changes to the PRA I&C models were made. No change to the results or conclusions of the PRA were identified as a result of this review.

STP DEP T1 5.0-1, Site Parameters. The design basis external flood from a breach of the main cooling reservoir is evaluated in Appendix 19R. All other site parameter departures do not affect the PRA described in Chapter 19. Those site parameters that remained within the defined parameters of Tier 1 Chapter 5.0 (e.g., extreme wind and tornado) are not reevaluated in the site-specific PRA.

STD DEP 5.4-1, Reactor Water Cleanup System. This function is modeled in the shutdown PRA, however, there is no quantifiable effect in the PRA. Operator action dominates the failure of the function, removal of the single train maintenance unavailabilities improves the reliability of the function, but the effect of additional sources of shutdown cooling that are available, would mask the slight improvement in function reliability.

STD DEP 6C-1, Containment Debris Protection of ECCS Strainers. The model of strainer changed from conical suction strainer to CCI cassette type strainer which satisfies the requirements of Regulatory Guide 1.82, Rev. 3. There is no change to the ABWR PRA, as there is no change in function or in failure data. This change potentially affects the plant specific model that will satisfy 10CFR 50.71(h).

STD DEP 8.3-1, Plant Medium Voltage Electrical System Design. Two medium voltage systems 13.8 kV/4.6 kV. Plant Generation buses changed to 13.8 kV. Class 1E and Plant Investment Protection buses changed to 4.16 kV. Class 1E 4.16kV still fed directly from unit Auxiliary Transformers (UATs) through two breakers. There are two Reserve Auxiliary Transformers (RATs). The 13.8 kV Combustion Turbine Generator has an increased rating (approximately 20 Mwe). The Emergency Diesel Generators changed to 4.16 kV, and the rating increased to approximately 7200 kW. Larger capacity RATs and UATs, and a larger capacity Main Power Transformer are also included. A delta-PRA assessment was performed using system fault trees on Figures 19D6.11, 12, & 13. The only change other than editorial to the fault trees is the additional of several breakers from the 13.8 kV CTG to the 4.16kV Class 1E 4.16kV buses and PIP buses. There was no significant effect on PRA results or conclusions. Changes are incorporated into various sections of Chapter 19 that refer to the condensate pump and condensate booster pump being able to connect to CTG.

STD DEP 9.2-5, Reactor Service Water (RSW) System. The RSW system design reflects new location of RSW pump house and increased system flow and discharge pressure necessary to meet the increased heat removal requirements of the reactor cooling water system. Cooling tower fans are added to the UHS. These changes are Included in the delta PRA. There was no significant change to the PRA results and conclusions.

STD DEP 9.5-7, Fire Protection - House Boiler Area of the Turbine. The house boiler is now an electrically heated boiler. This change has no effect on PRA fire modeling. There is a potential slight improvement in Turbine Building fire frequency described in SSAR. No changes to the conclusions of the fire screening analysis.

STD DEP 19.3-1, Evaluation of Common Cause Failures. This change incorporates an identified error in the ABWR PRA in Appendix 19D.8.6 of the SSAR. This is not a plant change. The error was previously evaluated and described in the SSAR in a sensitivity evaluation, but was not incorporated into the ABWR PRA. This change is included in the delta-PRA assessment. [This is not a departure for the design certified in the DCD].

STP DEP 19R-1, RSW Pump House Redesign. The ABWR describes the conceptual UHS as a pond with the RSW pump house on the side. The RSW piping goes above the level of the pond in the pump house, creating a potential siphon if a significant leak were to occur in the lowest level of the Control Building, the associated RSW pumps were tripped on high sump level in the room, but the discharge motor-operated valve (MOV) fails to close. Siphon breakers were added to the conceptual RSW design to limit the amount of water drained to the Control Building lowest level. In addition, the length of the RSW piping to and from the Control Building was limited to less than 2000 meters to limit the effects of this internal flood scenario.

In the STP design, the RSW pumps are located below UHS, and the pump rooms and electrical distribution rooms in the pump house are protected by watertight doors between trains. The siphon breakers are removed and an additional MOV is automatically closed in each RSW train to limit the effect of internal flooding in the Control Building. A pump house internal flood evaluation was performed and is described in Appendix 19R.

The UHS function is provided by mechanical draft cooling towers, which are sized to satisfy the results of temperature studies to confirm they are within envelopes specified in ABWR DCD design. There is one UHS and RSW pump house for each unit. Each UHS division (3 divisions) contains two fans, one operating and one in standby. The UHS fans are included in the site-specific PRA as part of the RSW system. The UHS fans are added to the set of risk-significant equipment in Chapter 19 Appendix K.