CHAPTER 19

PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

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ACRONYMS AND ABBREVIATIONS

ac	alternating current
ANS	American Nuclear Society
ANSI	American National Standards Institute
CCW	component cooling water
CCWS	component cooling water system
CDF	core damage frequency
CFR	Code of Federal Regulation
COL	Combined License
CPNPP	Comanche Peak Nuclear Power Plant
CTW	cooling tower
dc	direct current
DCD	Design Control Document
ESW	essential service water
ESWP	essential service water pump
ESWS	essential service water system
FSAR	Final Safety Analysis Report
IPE	individual plant examination
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPSD	low-power and shutdown
LRF	large release frequency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment
RCP	reactor coolant pump
RG	Regulatory Guide
RMTS	risk-managed technical specifications
RY	reactor-year
SA	severe accident
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guidance
SFCP	Surveillance Frequency Control Program
SSC	structure, system and components
T/B	turbine building

ACRONYMS AND ABBREVIATIONS (Continued)

UHS	ultimate heat sink
WOG	Westinghouse Owners Group

19.0 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

This section of the referenced Design Control Document (DCD) is incorporated by reference with no departures or supplements.

19.1 PROBABILISTIC RISK ASSESSMENT

CP COL 19.3(8)

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. Cross-references between PRA programs, risk-informed applications and FSAR program descriptions are tabulated in Table 19.1-207.

19.1.1.2.1 Uses of Probabilistic Risk Assessment in Support of Licensee Programs

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CP COL 19.3(4)
CP COL 19.3(8)

Replace the second paragraph in DCD Subsection 19.1.1.2.1 with the following.

The probabilistic risk assessment (PRA) is updated to assess site-specific information and associated site-specific external events. A systematic process is used to develop the site-specific PRA from the design certification PRA. This process includes the following activities:

- 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.
- 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.

Site-specific information is reviewed to identify information related to the assumptions used in the PRA and having a potential effect on the PRA insights. Identification of the site-specific design is described in Table 1.8-1R in Section 1.8. These site-specific design issues, except essential service water system (ESWS) and ultimate heat sink (UHS), are considered having no potential influence to the results of the PRA. PRA screening assessment are shown in Subsections 19.1.4 through 19.1.6.

The Licensee programs that could be impacted are described in Subsections 19.1.7.1, 19.1.7.4, 19.1.7.5, 19.2.5 and Chapter 18.

19.1.1.3.1 Uses of Probabilistic Risk Assessment in Support of Licensee Programs

CP COL 19.3(8) Add the following text to the first paragraph in DCD Subsection 19.1.1.3.1.

The PRA in the construction phase will be updated and upgraded as necessary to support implementation of the Maintenance Rule (Subsection 19.1.7.2) and the Reactor Oversight Process (Subsection 19.1.7.3) prior to fuel load.

19.1.1.3.2 Risk-Informed Applications

CP COL 19.3(8) Replace the content of DCD 19.1.1.3.2 with the following.

The PRA in the construction phase will be updated and upgraded as necessary to support implementation of risk informed Technical Specifications (Risk Managed Technical Specifications and Surveillance Frequency Control Program) described in Subsection 19.1.7.6 prior to fuel load.

19.1.1.4 Operational Phase

CP COL 19.3(8) Replace the content of DCD Subsection 19.1.1.4 with the following.

The uses of PRA in support of licensee programs and description of risk-informed applications being implemented during the operational phase are described in the following subsections.

19.1.1.4.1 Uses of Probabilistic Risk Assessment in Support of Licensee Programs

CP COL 19.3(8) Replace the content of DCD Subsection 19.1.1.4.1 with the following.

The PRA will be used in the operational phase to support licensee programs such as the human factors engineering program (Chapter 18), the severe accident management program (Subsection 19.2.5), the maintenance rule (Subsection 19.1.7.2), and the reactor oversight program (Subsection 19.1.7.3).

The PRA models and results provide input to such as the preventive maintenance basis program and other related maintenance and reliability programs including the motor-operated valve and air-operated valve reliability and testing programs.

19.1.1.4.2 Risk-Informed Applications

CP COL 19.3(8) Replace the content of DCD Subsection 19.1.1.4.2 with the following.

The PRA will be updated to reflect the risk-informed technical specifications in accordance with RG 1.174 and RG 1.177, including Initiative 4b, RMTS, in accordance with NEI 06-09 (Reference 19.1-11) and Initiative 5b, risk-informed method for control of surveillance frequencies in accordance with NEI-04-10 (Reference 19.1-201), as described in Subsection 16.1.1.2.

19.1.2.3 PRA Technical Adequacy

CP COL 19.3(1) Replace the content of DCD Subsection 19.1.2.3 with the following.

The quality of the methodologies, processes, analyses, and personnel associated with the site-specific PRA comply with the provisions for nuclear plant quality assurance. Toward this end, the PRA adheres to the recommendations provided in RG 1.200 pertaining to quality and technical adequacy. The US-APWR incorporates the technical elements of an acceptable PRA shown in Table 1 of RG 1.200 (Reference 19.1-9), and is consistent with the technical characteristics and attributes given in Table 2 through Table 10 of RG 1.200.

A peer review against the technical elements of the ASME/ANS RA-Sa-2009 PRA standard and associated addenda as clarified by Regulatory Guide 1.200 will be performed prior to use of the PRA to support risk-informed applications or before initial fuel load.

19.1.2.4 PRA Maintenance and Update

CP COL 19.3(9) Replace the third paragraph in DCD Subsection 19.1.2.4 with the following.

Changes to PRA inputs and discovery of new information will be evaluated to determine whether a PRA maintenance or upgrade is warranted. Changes to the PRA impacting risk insights or key assumptions will be prioritized to ensure that the most significant changes are incorporated as soon as practical and associated

documentation is updated accordingly. Other changes will be incorporated during the next PRA update.

19.1.4.1.2 Results from the Level 1 PRA for Operations at Power

CP COL 19.3(4) Add the following text after the first sentence in DCD Subsection 19.1.4.1.2.

The only site-specific design that has potential effect on level 1 PRA for operation at power is the site-specific UHS.

Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4 use cooling towers (CTWs) as the UHS for the ESWS. Discharged cooling water from the heat exchangers of the ESWS is sprayed into the CTW basin, while the standard US-APWR design simply indicates that the UHS is an assured source of water, without reference to type of source, cooling or discharge.

The UHS consists of four 50 percent capacity mechanical draft CTWs, one for each ESWS train, and four 33-1/3 percent capacity basins to supply cooling water more than 30 days. Each CTW consists of two cells with fans and motors, drift eliminators, film fills, risers, and water distribution system all enclosed and supported by a seismic category I reinforced concrete structure. Each basin includes an ESWP intake structure that contains one 50 percent capacity ESWP and one 100 percent capacity UHS transfer pump, and associated piping and components. The fan motors are powered from the Class 1E normal ac power system. The UHS transfer pump located in each basin is powered from the Class 1E bus, which is independent from the one to power associated ESWP.

Adoption of CTWs to the UHS for the ESWS raises additional failure modes for the ESWS, which are associated with the failure of the CTW fans and the drain valves. Failure of the CTW fans or drain valves would cause degradation of heat release from the ESWS to the atmosphere, which would result in an increase of the ESWS temperature in the faulted train. Failure of both fans and drain valve in a single CTW train is considered a potential failure mode of the ESWS.

Failures of CTW fans and drain valves were modeled in ESWS fault tree to address the effect of site-specific UHS. The reliability of ESWS affects both the initiating event frequency of loss of CCW and the reliability of ESWS after the initiating event. Therefore, the initiating event frequency given later in this subsection based on the US-APWR design was re-quantified based on the site-specific ESWS designs along with re-quantification of post-initiating event ESWS reliability.

Assumptions and important design features regarding the UHS and ESWS are as follows:

- A drain line is provided as an overfill protection from overfilling the basin and failing the pump(s).
- There are adequate low-level and high-level alarms to provide rapid control room annunciation of a level problem and to allow adequate time to confirm the level and take effective action to address it.
- On failure of the fans during normal plant operation, operating status of each fan is indicated in the main control room (MCR).
- Should the plant trip, two basins are effective in removing decay heat for more than 24 hours without replenishment or transferring water from another basin.
- The transfer line is a high integrity line, regularly tested and inspected for corrosion.
- Failure of the transfer line will not drain any CTW basin.
- The basin water is tested regularly and maintained in a condition to preclude corrosion and organic material from plugging strainers.
- Ventilation of the ESWP room is sufficiently reliable that the availability of the ESWP is not degraded.
- In operating trains, cooling tower drain valves are locked closed. For trains where the ESWP is not operating, water in exposed safety-related ESW piping in the cooling tower is drained to the basin through the drain line by opening the drain valve manually prior to the onset of temperatures that could cause freezing. After draining, the operator closes and locks the drain valve.

The internal event core damage frequency (CDF) was found to be numerically the same as reported later in this subsection with an actual increase in the CDF due to the site-specific designs of less than 1 percent. The initiating event frequency for loss of component cooling water (CCW), as reported later in this subsection in Tables 19.1-2 and 19.1-23, increases from 2.4E-05/reactor-year (RY) to 2.6E-05/RY due to the site-specific ESWS designs. The effect of the site-specific ESWS designs on the internal CDF is very small. Therefore, any discrepancy of cutsets, basic event importances of the standard design SSCs and operator actions, and dominant sequences from that documented for the standard US-APWR design is considered negligible. Changes in importance are the basic events related to the site-specific design shown in Table 19.1-204. The results described below are considered sufficient and applicable.

CP COL 19.3(10) Add the following text at the end of the second to last paragraph in DCD Subsection 19.1.4.1.2.

The site-specific PRA will evaluate and address the key sources of uncertainty and key assumptions listed in DCD Table 19.1-38. Walkdowns during

construction will be used to assess and update as needed (i) key insights and assumptions (identified in DCD Table 19.1-119), (ii) routing and locations of piping and cables assumed in the internal fire and flooding events, and (iii) fragility values used in the seismic margin analysis that are important to the risk profile of the facility; the site-specific PRA will confirm that this information is accurately reflected in the as-built design and construction. Differences between the as-built plant and the design used as the basis for the US-APWR PRA will be reviewed to determine whether there is significant impact on PRA results.

19.1.4.2.2 Results from the Level 2 PRA for Operations at Power

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STD COL 19.3(4) Add the following text after the first sentence in DCD Subsection 19.1.4.2.2.

The only site-specific design that has potential effect on level 2 PRA is the site-specific UHS.

As is the case of the Level 1 PRA for operations at power (Subsection 19.1.4.1.2), modeling of the site-specific UHS results in small effect on the reliability of the component cooling water system (CCWS) for internal events. There is only small increase of CDF resulting from loss of CCW initiating events, also the contribution of total loss of CCW initiation event to the large release frequency (LRF) for operations at power is considered insignificant. It has been therefore determined that consideration of the site-specific UHS would have no discernible effect on the Level 2 PRA results that are based on the standard US-APWR design. Therefore, the results described below are considered sufficient and applicable.

19.1.5 Safety Insights from the External Events PRA for Operations at Power

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CP COL 19.3(4) Replace the second and third paragraphs in DCD Subsection 19.1.5 with the following.

The last three events listed above receive detailed evaluation in the following subsections. The first four events are subject to the screening criteria consistent with the guidance of ASME/ANS RA-Sa-2009, taking into consideration the features of advanced light water reactors.

The assessment of the other external events is provided below:

The screenings for other external events are performed using the following steps taking into consideration the features of advanced light water reactors. At first, qualitative screenings are performed using the analysis reported in Chapter 2 in accordance with the guidelines of ASME/ANS RA-Sa-2009. Section 6-2 of the standard defined the initial preliminary screening criteria as supporting technical requirement EXT-B1. The five qualitative screening criteria are:

1. Equal or lesser damage potential than the events for which the plant has been designed.
2. Lower event frequency of occurrence than another event
3. Cannot occur close enough to the plant to have an affect
4. Included in the definition of another event
5. Sufficient time to eliminate the source of threat or to provide an adequate response

If the external event cannot be screened on the qualitative screening criteria, quantitative screenings are performed. The supporting technical requirement EXT-C1 of ASME/ANS RA-Sa-2009, Criterion C, for conservative analysis allows for the use of a bounding or demonstrably conservative analysis with a mean frequency < 10^{-6}/year.

To support the goal that new reactor designs would have a substantially lower risk profile, Comanche Peak Units 3 and 4 use a value of <10^{-7}/year for the CDF determined by bounding or conservative analysis to quantitatively screen external events if the external event cannot be screened qualitatively.

The qualitative and quantitative screenings are performed using the analysis reported in the FSAR Sections 2.2, 2.3 and 2.4, and Section 3.5. The summary of the screenings is described in Table 19.1-205. Only tornado events are not screened because the probability of expected maximum tornado wind speed on the site is close to 10^{-7}/year.

<u>High Winds, Tornado Winds, and Hurricane Winds</u>

For high winds, tornado winds and hurricanes, tornadoes are evaluated using level 1 PRA as a bounding analysis from the discussion in Subsection 2.3.1.2.3.

The following sections show the results of the tornado PRA elements (1) tornado hazards, (2) plant vulnerabilities, (3) accident scenario, and (4) quantification.

- Tornado hazard

  A tornado wind speed hazard curve for CPNPP Units 3 and 4 was developed following NUREG/CR-4461 which also forms the basis for

NRC Regulatory Guide 1.76. The tornado hazard methodology developed in NUREG/CR-4461 fully meets the requirements of ASME/ANS RA-Sa-2009.

The CPNPP Units 3 and 4 are near Glen Rose, Texas and are located at 32° 17’ latitude and 97° 47’ longitude. The tornado hazard curve has been developed based on data reported in NUREG/CR-4461 for the 2° box surrounding the site, which recorded 655 tornado occurrences from 1950 through 2003. The hazard curve produced for the CPNPP Units 3 and 4 is shown in Figure 19.1-201. Strike and exceedance frequencies for tornadoes categorized in enhanced F-scale intensity are shown in Table 19.1-201.

- Plant vulnerabilities

Components significant to the internal events PRA were reviewed to identify component vulnerability during tornadoes. Component failures that could cause initiating events were also reviewed.

All systems and components essential for safe shutdown and for maintaining the integrity of the reactor coolant pressure boundary are located within seismic category I buildings, which are designed to withstand the loading of a design basis tornado. The design basis tornado is described in Subsection 3.3.2 and in Table 19.1-202.

Based on a review of components, the following were identified as potential vulnerabilities during tornadoes with intensities below the design basis tornado.

- Plant switchyard
- Fire protection water tank and associated piping of the fire protection water supply system
- CTW for the non-essential chilled water system and associated pipings
- Permanent buses of the non-safety power system
- Main steam supply system downstream of the main steam isolation valves
- Main feedwater system upstream of the main feedwater isolation valves

Structures, systems, and components (SSCs) will be designed using the site-specific basic wind speed of 96 mph or higher. Within this analysis, plant vulnerabilities located outdoors that are not seismic category I or II structures are assumed to be damaged for tornado strikes of intensity enhanced F-scale 1 and greater. In this analysis, the following systems are

assumed to be damaged for tornado strikes of intensity enhanced F-scale 1 and greater:

- Plant switchyard
- Fire protection water supply system
- Non-essential chilled water system

Seismic category II structures are designed to withstand a basic wind speed of 155 mph. The seismic category II structure that contains PRA related equipment is the turbine building (T/B). Tornado induced failure of the T/B is conservatively assumed to have an effect on the operability of alternate ac power system. In this analysis, the following systems are assumed to be damaged by tornado strikes resulting in failure of the T/B:

- Plant switchyard
- Fire protection water supply system
- Non-essential chilled water system
- Non-safety electric power system
- Alternate ac power supply system

Direct damage to the seismic category I structures and the components within the structure can be caused by tornadoes exceeding the design basis tornado. In this analysis safety related systems are assumed to be damaged for tornado strikes of a design basis tornado or greater (wind speed ≥ 230 mph).

- Accident scenario

  When a tornado strikes the plant, there is a probability that a tornado initiated accident scenario may be induced with some mitigation functions inoperable due to damage from a tornado strike. Based on plant vulnerabilities identified in the previous section, the internal events PRA was reviewed to identify initiating events or degradation of mitigation functions that may be caused by a tornado strike. The following internal events accident initiators may be caused by a below design basis tornado strike:

  - Loss of offsite power (LOOP)
  - Main steam line break downstream of main steam isolation valves
  - Loss of feedwater flow
  - Feedwater line break upstream of the main feedwater isolation valves

  The following mitigation and support systems may be degraded by tornado-induced failures from a below design basis tornado strike:

- Alternate CCW utilizing the fire protection water supply system
- Alternate CCW utilizing the non-essential chilled water system
- Non-safety electric power system
- Alternate ac power supply system (this is a mitigation system for LOOP events, which is initiating event potentially caused by a tornado strike)

LOOP is the most severe initiating event for tornado strikes with enhanced F-scale intensity of F3 or greater and dominates the plant risk profile. LOOP event is applied to the tornado PRA as the most limiting case.

Based on the results of the plant vulnerability analysis and the discussion above, tornado-induced accident scenarios were categorized into three scenarios as shown in Table 19.1-203. The frequency of each scenario derived from the hazard fragility analysis of the T/B is also shown.

- Quantification

For the tornado induced accident scenarios, the CDF was calculated based on the internal event PRA results. The dominant core damage scenarios were the following:

- Enhanced F-scale intensity F1 and F2 tornado strike-induced LOOP and plant switchyard damaged combined with failure of all four CCW or ESW pumps.

The plant switchyard is assumed to be damaged by the tornado strike of enhanced F-scale intensity F1 and F2. A LOOP occurs and CCW or ESW pumps fail to re-start due to common cause failure. Since there is no function to cool reactor coolant pump (RCP), RCP seal loss-of-coolant accident (LOCA) occurs, which results in the core damage. The CDF for this scenario is 2.9E-08/RY.

- Enhanced F-scale intensity of F3, F4 and F5 tornado strike-induced LOOP and T/B damage combined with failure of all four emergency gas turbine generators.

The plant switchyard and the T/B are assumed to be damaged by the tornado strike with wind speed between 136 mph and 230 mph. A LOOP occurs and the emergency gas turbine generators fail to operate due to common cause failure. The alternative power source is unavailable since the T/B is damaged and total loss of ac power occurs. Offsite power cannot be recovered due to damage of the T/B. RCP seal LOCA occurs and eventually the core is damaged. The CDF for this scenario is 2.3E-08/RY.

- Failure of all safety systems by a beyond design basis tornado. This event leads directly to core damage. This CDF for this scenario is 2.5E-08/RY.

The total CDF caused by a tornado strike during at-power operation is less than 8E-08/RY. Tornado induced CDF is one order of magnitude lower than the total CDF for internal events and internal flood and internal fire events. A bounding screening assessment for extreme winds has been performed. The results show that the CDF due to extreme winds is less than 1.0E-7 per year.

The CDF from tornadoes during LPSD does not contribute more than ten percent of the total shutdown CDF and total shutdown LRF compared to the US-APWR DCD PRA. Tornado events during LPSD do not have significant contribution to risk. A bounding screening assessment for extreme winds has been performed. The results show that the CDF due to extreme winds is less than 1.0E-7 per year.

<u>External Flooding</u>

Subsection 2.4.2 systematically considers the various factors that can contribute to the incident of external flooding. The deterministic PMP flood described in Section 2.4 of the CPNPP FSAR, screens under Criterion #1 of EXT-B1 of ASME/ANS RA-Sa-2009 since the event is of equal or lesser damage potential than the events for which the plant has been designed.

<u>Transportation and Nearby Facility Accidents</u>

These events consist of the following:

- Hazards associated with nearby industrial activities, such as manufacturing, processing, or storage facilities
- Hazards associated with nearby military activities, such as military bases, training areas, or aircraft flights
- Hazards associated with nearby transportation routes (aircraft routes, highways, railways, navigable waters, and pipelines)

In Subsection 2.2.3.1, design basis events internal and external to the nuclear power plant are defined as those events that have a probability of occurrence on the order of about 10^{-7}/RY or greater and potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR Part 100 could be exceeded. The following categories are considered for the determination of design basis events: explosions, flammable vapor clouds with a delayed ignition, toxic chemicals, fires, collisions with the intake structure, and liquid spills.

The effects of these events on the safety-related components of the plant are insignificant as discussed in Subsection 2.2.3.1. These events meet the preliminary screening criteria of ASME/ANS RA-Sa-2009. The applicable preliminary screening criteria for each events are described in Table 19.1-205.

Aircraft Crash

As described in Subsection 3.5.1.6, the probability of aircraft-related accidents for CPNPP Units 3 and 4 is less than the order of 10^{-7} per year for aircraft, airway, and airport information reflected in Subsection 2.2. This event meets the quantitative screening criteria and is not addressed further.

19.1.5.1.1 Descriptions of the Seismic Risk Evaluation

CP COL 19.3(4) Replace the last sentence of the first paragraph after the first bullet "Selection of review level earthquake" in DCD Subsection 19.1.5.1.1 with the following.

The seismic margin analysis of the DCD is incorporated by reference although the RLE of CPNPP is less than the DCD RLE of 0.5g, which is 1.67 times the SSE (0.3g).

CP COL 19.3(5) Add the following paragraph after the description of the bullet item “Fragility analysis."

There are no site specific deviations from the HCLPF values or other assumptions in the seismic margins evaluation provided in the DCD Subsection 19.1.5.1. Seismic fragility will be re-evaluated considering the site-specific designs before the first fuel load. Seismic fragilities of the structures are developed using the methodology in Reference 19.1-204.

19.1.5.1.2 Results from the Seismic Risk Evaluation

CP COL 19.3(5) Add the following text at the beginning of DCD Subsection 19.1.5.1.2.

The site-specific design that has potential effect on seismic risk is the site-specific UHS.

The UHS is designed with sufficient inventory to provide cooling for at least 30 days following the most limiting design basis accident without makeup water in accordance with the guidance of RG 1.27. No credit is taken for the availability of makeup water during the design basis accident. Therefore, the possibility of loss of CWT function caused by seismic failure of makeup water is negligible.

The design of the UHS consists of reinforced concrete structures that are directly founded on the Glen Rose Formation limestone Layer C, and does not include any earth embankments for side wall support. Additionally, the layout design of the site-specific seismic Category I SSCs ensures that there are no adjacent non-seismic Category I structures that may adversely affect site-specific seismic Category I SSCs including the UHS structures. Accordingly, seismic Category I SSCs are not exposed to the possible impact of a failure or collapse of non-seismic Category I SSCs. Therefore, the presence of the subject retaining wall and adjacent backfill slopes do not have any adverse effect on the UHS structures.

The intake (makeup) piping layout precludes draining of cooling tower basin water from failed non-seismic intake piping. The pumping equipment and cooling fans are higher than the elevation of the basin wall and the ground elevation, and are enclosed by a concrete wall as is shown in Figures 3.8-208 and 3.8-209. The pumping equipment and cooling fans are protected from flooding due to the failure of the non-seismic intake piping to the UHS.

Based on these design features, seismically driven common failures are not significant in the CPNPP Units 3 and 4 SMA.

CP COL 19.3(4) Add a paragraph after the last paragraph in DCD Subsection 19.1.5.1.2 with the following.

The plant-specific HCLPFs of CPNPP Units 3 and 4 that are not less than 1.67 times SSE will be confirmed using the design specific in-structure response and the results of the stress analysis of the US-APWR standard design.

19.1.5.2.2 Results from the Internal Fires Risk Evaluation

STD COL 19.3(4) Add the following text at the beginning of DCD Subsection 19.1.5.2.2.

The only site-specific design that has potential effect on internal fires risk is the site-specific UHS.

Four-train separation is maintained in the site-specific UHS design. Modeling of the site-specific UHS shows a small effect on the reliability of CCWS for internal fire events. As was the case with the results of the Level 1 PRA for operations at power (Subsection 19.1.4.1.2), it has been determined that consideration of the site-specific UHS would have no discernible effect on the fire PRA results that are based on the standard US-APWR design. Therefore, the results described below are considered sufficient and applicable.

19.1.5.3.2 Results from the Internal Flooding Risk Evaluation

STD COL 19.3(4) Add the following text at the beginning of DCD Subsection 19.1.5.3.2.

The only site-specific design that has potential effect on internal flooding risk is the site-specific UHS.

Four-train separation is maintained in the site-specific UHS design. Modeling of the site-specific UHS shows a small effect on the reliability of CCWS for internal flooding events. As was the case with the results of the Level 1 PRA for operations at power (Subsection 19.1.4.1.2), it has been determined that consideration of the site-specific UHS would have no discernible effect on the internal flooding PRA results that are based on the standard US-APWR design. Therefore, the results described below are considered sufficient and applicable.

19.1.6.2 Results from the Low-Power and Shutdown Operations PRA

STD COL 19.3(4) Add the following text at the beginning of DCD Subsection 19.1.6.2.

The only site-specific design that has potential effect on low-power and shutdown risk is the site-specific UHS.

As was the case with the Level 1 PRA for operations at power (Subsection 19.1.4.1.2), modeling of the site-specific UHS shows a small effect on the reliability of CCWS for internal events. Considering the small increase of loss of CCW initiating event frequency, it has been determined, that consideration of the site-specific UHS would have no discernible effect on the low-power and shutdown (LPSD) results that are based on the standard US-APWR design. Therefore, the results described below are considered sufficient and applicable.

19.1.7 PRA-Related Input to Other Programs and Processes

CP COL 19.3(8) Add the following sentence to the first paragraph of Subsection 19.1.7.

The implementation of the specific programs and risk informed applications are delineated in Table 19.1-207.

19.1.7.1 PRA Input to Design Programs and Processes

STD COL 19.3(4) Replace the last sentence of DCD Subsection 19.1.7.1 with the following.

Key insights and assumptions are summarized in Table 19.1-119 and specified pages replaced by Table 19.1-119R. Site-specific key assumptions are summarized in Table 19.1-206.

19.1.7.6 PRA Input to the Technical Specification

CP COL 19.3(1) Replace the last paragraph in DCD Subsection 19.1.7.6 with the following.

The PRA needed for implementation of RMTS, SFCP, and peer review will be available one year prior to fuel load.

19.1.9 References

CP COL 19.3(4) Add the following references after the last reference in DCD Subsection 19.1.9.

19.1-201 *Risk-Informed Method for Control of Surveillance Frequencies*, NEI 04-10, Rev. 1, Nuclear Energy Institute, Washington DC, April 2007.

19.1-202 *Climatology Models for Extreme Hurricane Winds Near the United States*, Thomas H. Jagger and James B. Elsner, January 19, 2006.

19.1-203 *A Simple Empirical Model for Predicting the Decay of Tropical Cyclone Winds after Landfall*, John Kaplan and Mark Demaria, JOURNAL OF APPLIED METEOROLOGY, Volume 34, November, 1995.

CP COL 19.3(5) 19.1-204 *Seismic Fragility Application Guide*, EPRI TR-1002988, Electric Power Research Institute, Palo Alto, CA, 2002

CP COL 19.3(6)

Table 19.1-119R Key Insights and Assumptions (Sheet 1 of 50)

Key Insights and Assumptions	Dispositions
Design features and insights	
1. High Head Safety Injection System	
- The high head safety injection system consists of four independent and dedicated SI pump trains.	6.3.2.1.1
- The SI pump trains are automatically initiated by ECCS actuation signal, and supply borated water from the RWSP to the reactor vessel via direct vessel injection line.	6.3.2.1.1
- Each SI pump is connected to a dedicated direct vessel injection nozzle for injection into the reactor downcomer region.	6.3.2.1.1
- SI pump suction isolation valves (SIS-MOV-001A/B/C/D) remain open during normal and emergency operations. These valves are remotely closed by operator action from MCR or RSC to isolate RWSP to terminate leak or if pump/valve maintenance requires it.	6.3.2.2.6.1
- This system provides the safety injection function during LOCA events and feed and bleed operation.	6.3.3 19.2.5 13.5.2
- During plant shutdown, safety injection provides RCS makeup function in loss of RHRS. In the case of failure of operable SI pump, the pumps that are locked out for LTOP compliance can be used if available.	5.2.2.1.2 5.2.2.2.2.2 19.2.5 13.5.2
- SI pump can be manually actuated by DAS from MCR.	7.8.1.1.1 Table 7.8-5
- SI pumps are operable regardless of HVAC system of the safeguard component area within mission time.	Table 19.1-180

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Table 19.1-119R Key Insights and Assumptions (Sheet 3 of 50)

Key Insights and Assumptions	Dispositions
3. Chemical and Volume Control System	
- The CVCS provides a means to maintain a programmed inventory of reactor coolant during all phases of plant operation.	9.3.4.1.2.1
- The CVCS continuously supplies seal water to the reactor coolant pump seals, as required by the reactor coolant pump design.	9.3.4.1.2.4 9.3.4.2.7.2
- The charging pumps are arranged in parallel with common suction and discharge headers. Each pump provides full capability for normal makeup.	9.3.4.2.6.1
- Charging injection is provided by the CVCS. One CVCS charging pump is capable of maintaining normal RCS inventory with small system leak if the leakage rate is less than that from a break of a pipe 3/8 inch in inside diameter.	9.3.4.2.7.4
- Normally, one charging pump is operating and takes suction from the VCT, supplies charging flow to the RCS and seal water to the reactor coolant pumps. The flow rate of the charging pump is controlled by the flow control valve located in the charging line and the flow control valve located in the reactor coolant pump seal injection line	9.3.4.2.1 9.3.4.2.6.1 9.3.4.2.7.2
- The pump can take suction from the VCT, the reactor makeup control system, the refueling water storage auxiliary tank and the spent fuel pit.	9.3.4.2.6
- During normal operation, the VCT water level is controlled by automatic makeup. In case the automatic makeup fails to actuate and the water level in the VCT decreases, low VCT water level is detected and actuates a low-low level signal that opens the stop valves in the refueling water storage auxiliary tank supply line, and closes No. 1 and No. 2 stop valves in the VCT outlet to provide emergency makeup.	9.3.4.2.1 9.3.4.5.4.1
- Two centrifugal boric acid transfer pumps are utilized for the transfer and circulation of the boric acid solution in the two boric acid tank.	9.3.4.2.3.1 9.3.4.2.6.2 9.3.4.2.6.9
- During plant shutdown, when the RHR system is in operation, the RHR system provides reactor coolant to the CVCS, upstream of the letdown heat exchanger in the letdown line.	9.3.4.2.7.3
- During plant shutdown, charging injection provides RCS makeup function in loss of RHRS. In the case of failure of operable charging pump, the pumps that are locked out for LTOP compliance can be used if available.	5.2.2.1.2 5.2.2.2.2.2 19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 4 of 50)

Key Insights and Assumptions	Dispositions
4. Containment Spray System / Residual Heat Removal System	
- The containment spray system (CSS) and the residual heat removal system (RHRS) share major components which are containment spray/residual heat removal (CS/RHR) pumps and heat exchangers.	5.4.7.1 5.4.7.2.1 6.2.2 6.2.2.1 6.2.2.2
- The CSS/RHRS consists of four independent subsystems, each of which receives electrical power from one of four safety buses. Each subsystem includes one CS/RHR pump and one CS/RHR heat exchanger, which have functions in both the CS system and the RHRS.	6.2.2 5.4.7.2.1
- All four CS/RHR pumps automatically start to supply water in RWSP and containment spray header isolation valves are open automatically on the receipt of a containment spray signal.	6.2.2.2.1 6.2.2.2.7.2
- CSS/RHRS provides multiple functions such as,	3.2.2
(1) containment spray to decrease pressure and temperature in the containment, (2) alternate core cooling in case all safety injection systems fails during LOCA in conjunction with a fast depressurization of the RCS by using the EFW pumps to remove heat through the SGs and by manually opening the MSDVs especially in high RCS pressure sequences, (3) RHR operation for long term core cooling, (4) heat removal function for long term containment cooling, (5) providing water to flood the reactor cavity and (6) fission product removal. (7) During plant shutdown, RHRS provides function to remove decay heat from the RCS.	6.2.2 6.2.2.1 6.2.5 5.4.7.1 5.4.7.2.1 5.4.7.2.3.3 19.2.5 13.5.2
- The RHRS is designed and equipped with pressure relief valves to prevent RHRS over-pressurization and low temperature over-pressurization.	5.4.7.1
- Two motor operated valves in series on the RHR suction line with power lockout capability during normal power operation minimize the probability of RCS pressure entering the RHR system. Even if both these valves are opened during normal power operation, the RHR system is designed to discharge the RCS inventory to the in-containment RWSP. The RHRS is designed to prevent an interfacing system LOCA by having a design rating of 900 lb. The RHR 900 lb. design rated system can withstand the full RCS pressure. The current values are in accordance with Section III of the ASME Code for Service Level A.	6.3.1.4 5.4.7.1 5.4.7.2.1 5.4.7.2.2 Table 5.4.7-2

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Table 19.1-119R Key Insights and Assumptions (Sheet 12 of 50)

Key Insights and Assumptions	Dispositions
10. Reactor Coolant System High Point Vents	
- Safety depressurization valves (SDVs) and depressurization valves (DVs) are provided at top head of the pressurizer in order to cool the reactor core by feed and bleed operation when loss of heat removal from steam generator occurs.	5.4.12.2 19.2.5 13.5.2
- RCS depressurization system dedicated for severe accident is provided to prevent high pressure melt ejection. The location of release point from the valve is in containment dome area.	5.4.12.2
- Safety depressurization valves can be manually actuated by DAS.	7.8.1.1.1 Table 7.8-5
11. Main Steam Supply System	
- The system consists of MSRV, MSDV, MSSVs, and MSIV in each main steam line and TBVs.	10.3 10.3.1.1
- Six MSSVs are provided per each main steam line and are located in the main steam piping upstream of the MSIVs. The MSSVs have the three kind of set pressure.	10.3.2.3.2 Table 10.3.2-2
- One air-operated MSRV and one motor-operated MSDV are installed on each main steam line piping.	10.3.2.3.3
- MSIVs are installed in each of the main steam lines to (1) limit uncontrolled steam release from one steam generator in the event of a steam line break, and to (2) isolate the faulted SG in the event of SGTR.The valve is designed to fully close by receipt the signal such as low main steam line pressure.	10.3.2.1 10.3.2.3.4
- In LOCA event with failure of all HHISs, operators open MSDVs to depressurize the RCS for alternate core injection.	19.2.5 13.5.2
- During shutdown operation, when the RCS is mid-loop state with the closed state, operators open MSDVs for heat removal via SGs.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 13 of 50)

Key Insights and Assumptions	Dispositions
12. Component Cooling Water System	
- The CCWS consists of two independent subsystems. One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four trains. Each train has one CCW pump and CCW heat exchanger. Each subsystem is served by one CCW surge tank.	9.2.2.1.1 9.2.2.2
- The CCWS is designed to withstand leakage in one train without loss of the system's safety function.	9.2.2.1.1
- Two motor operated valves are located at the CCW outlet of the RCP thermal barrier Hx and close automatically upon a high flow rate signal at the outlet of this line in the event of in-leakage from the RCS through the thermal barrier Hx, and prevents this in-leakage from further contaminating the CCWS.	9.2.2.2.1.5
- During normal operation, heat loads of the CCWS are RCP, charging pump, letdown heat exchanger, instrument air, spent fuel pit cooling heat exchanger, etc.	9.2.2.1.2.1
- Normally open header tie line isolation valves, which are motor-operated valves, is automatically closed upon detection of ECCS actuation signal and under voltage signal or containment spray signal to separate each subsystem into two independent trains.	9.2.2.2.1.5
- CS/RHR heat exchanger outlet valves, which are motor-operated valves, are normally closed and automatically are opened by ECCS actuation signal.	9.2.2.2.1.5
- During normal operation, at least one train in each subsystem is operable. Total of two CCWP and two CCW heat exchangers are in operation. During accident, all CCWPs are automatically actuated by ECCS actuation signals.	9.2.2.2.2.1 9.2.2.2.2.4
- During a severe accident event, it is assumed that the containment fan cooler unit fans are non-operable and that the non-essential chilled water system is unavailable. Valves are provided to manually align the CCW to the containment fan cooler unit cooling coils. This supplies CCW to the cooling coils in the containment fan cooler unit for long term containment cooling.	9.4.6.2.1 19.2.5 13.5.2
- In the case of loss of CCW, a non-essential chilled water system or a fire system is able to connect to the CCWS in order to cool the charging pump and maintain RCP seal water injection.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 14 of 50)

Key Insights and Assumptions	Dispositions
13. Essential Service water system	
- The ESWS is arranged into four independent trains (A, B, C, and D). Each train consists of one ESWP, two 100% strainers in the pump discharge line, one 100% strainer upstream of the CCW HX, one CCW HX, one essential chiller unit, and associated piping, valves, instrumentation and controls.	9.2.1 9.2.1.2.1 9.2.1.2.3.1 13.5.2
- The case where ESW pump motors are air-cooled has a small impact on PRA results because the HVAC system for the ESW pump room is reliable due to operator backup.	9.2 13.5.2
- In the case where ESW pump motors are air-cooled, backup actions can avoid excessive room heat up in the event of loss of ESW pump room ventilation. Operational procedures to avoid excessive room heat up will be prepared.	9.2.1.2.2.1 9.2 13.5.2
- During normal operation, two trains are operating and at least one other train is on standby.	9.2.1.2.3.1
- The motor-operated valve provided at the discharge of each ESW pump actuates in conjunction with the pump operation. The discharge valves are opened after the ESW pump start.	9.2.1.2.2.6
- During normal operation, two ESW trains are operating and at least one train is on standby.	9.2.1.2.3.1
- The motor-operated valve is provided at the ESWP discharge of each pump. While the ESW pump is running, the valve remains open. The valve position is monitored in the control room.	9.2.1.2.2.6 9.2.1.2.3.1
- All valves except the pump discharge valves in the flow path are locked open.	9.2.1.2.3.1
- When one ESW train is unavailable due to failure of the discharge line valve to open, operators start the standby ESWP, monitoring pump discharge pressure.	9.2.1.2.3.1 19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 15 of 50)

Key Insights and Assumptions	Dispositions
14. Onsite Electric Power System	
- The onsite Class 1E electric power systems comprise four independent and redundant trains, each with its own power supply, buses, transformers, and associated controls.	8.3.1.1 8.3.1.1.2.1 8.3.1.1.3
- One independent Class 1E GTG is provided for each Class 1E train.	8.3.1.1.2.1
- Non-Class 1E 6.9kV permanent buses P1 and P2 are also connected to the non-Class 1E A-AAC GTG and B-AAC GTG, respectively. The loads which are not safety-related but require operation during LOOP are connected to these buses.	8.3.1.1.1
- In the event of SBO, power to one Class 1E 6.9kV bus can be restored manually from the AAC GTG.	8.3.1.1.1 8.3.1.1.2.2 8.3.1.1.2.3 19.2.5 13.5.2
- Common cause failure between Class 1E GTG and non-Class 1E GTG is minimized by design characteristics. The AAC power source engine and generator are designed by a different manufacturer than the Class 1E EPS engine and generator, and have diverse starting systems, independent and separate auxiliary and support systems .	8.3.1.1.1 8.4.1.3
- The non-Class 1E GTG can be started manually when connecting to the Class 1E bus in the event of SBO.	8.4.1.3
- Power to the shutdown buses can be restored from the AAC sources within 60 minutes	8.4.1.3
- The GTG does not need cooling water system. Cooling of GTG is achieved by air ventilation system	8.3.1.1.3 8.3.1.1.3.10
- GTG combustion air intake and exhaust system for each of the four GTGs supply combustion air of reliable quality to the gas turbine and exhausts combustion products from the gas turbine to the atmosphere. The air intake also provides ventilation/cooling air to the GTG assembly.	9.5.5 9.5.8

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Table 19.1-119R Key Insights and Assumptions (Sheet 17 of 50)

Key Insights and Assumptions	Dispositions
16. Containment System	
- The containment prevents or limits the release of fission products to the environment.	3.1.2.7 3.8.1
- Hydrogen control system that consists of igniters is provided to limit the combustible gas concentration. Twenty igniters are required to mitigate a challenge to containment integrity from the spectrum of potential hydrogen detonation scenarios for both at-power and LPSD severe accident sequences. The igniters start with the ECCS actuation signal and are powered by two non-Class 1E buses with non-Class 1E GTGs as well as dedicated batteries to 11 strategically located igniters.	6.2.5.2
- Alternate containment cooling system using the containment fan cooler units is provided to prevent containment over pressure even in case of containment spray system failure.	9.4.6.2.1 19.2.5 13.5.2
- Reactor cavity flooding system by firewater injection is provided to enhance heat removal from molten core ejected into the reactor cavity. This system is available as a countermeasure against severe accidents even in case of fire.	9.5.1.2.2 19.2.5 13.5.2
- The FSS is also utilized to promote condensation of steam. The FSS is lined up to the containment spray header when the CSS is not functional, and provides water droplet from top of containment. This will temporarily depressurize containment.	9.5.1.2.2 19.2.5 13.5.2
- A set of drain paths from SG compartment to the reactor cavity is provided in order to achieve reactor cavity flooding. Spray water which flows into the SG compartment drains to the cavity and cools down the molten core after reactor vessel breach.	3.4.1.5.1
- Reactor cavity has a core debris trap area to prevent entrainment of the molten core to the upper part of the containment.	3.8.1 19.2.3.3.4
- Reactor cavity is designed to ensure thinly spreading debris by providing sufficient floor area and appropriate depth.	3.8.1 19.2.3.3.3
- Reactor cavity floor concrete is provided to protect against challenge to liner plate melt through.	3.8.1 19.2.3.3.3
- Main penetrations through containment vessel are isolated automatically with the containment penetration signal even in case of SBO.	6.2.4

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Table 19.1-119R Key Insights and Assumptions (Sheet 19 of 50)

Key Insights and Assumptions	Dispositions
18. Main equipments and instrumentations used for severe accident mitigation are designed to perform their function in the environmental conditions such as containment overpressure and temperature rise following hydrogen combustion.	19.2.3.3.7
19. Instrumentations for detecting core damage with high reliability are provided.	5.3.3.1
20. Risk significant SSCs are identified for the RAP.	17.4
21. The in-core instrumentation is inserted through nozzles located in the RV closure head. No penetrations through the RV are located below the top of the reactor core. This minimizes the potential for a loss of coolant accident by leakage from the reactor vessel, allowing the reactor core to be uncovered.	5.3.3.1
22. Check valves in accumulator, high head injection system, and other systems are in diverse configuration because: - The accumulator does not have any pumps to drive upon a failed closed check valve but other systems have pumps so the forces acting on the valves to open them (even if the valves are similar) are different - The duty cycles in the systems are different. They are cycled at different times when the systems are tested. - Maintenance practices including testing may also be different. Common cause failure between the check valves in accumulator and HHIS is therefore not model in the PRA.	19.1.4.1 Table 19.1-38
23. Surveillance test interval and refueling outages are consistent with Technical Specifications.	Chapter 16
24. The availability and reliability of all trains of safety related systems will be controlled by the maintenance and configuration risk management programs. Availability goals will be set for each train of all safety related systems and their availability will be tracked and compared to these goals.	17.6
25. Administrative controls to ensure the availability of AAC as a back up function to the Class 1E GTGs will be implemented.	13.5.2
26. Administrative controls to ensure the availability of demineralized water storage tank as a back up function to the EFW pits will be implemented.	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 20 of 50)

Key Insights and Assumptions	Dispositions
Operator actions (At Power)	
1. Operator actions modeled in the PRA are based on symptom oriented procedures. Risk significant operator actions identified in the PRA will be addressed in plant operating procedures including abnormal operating procedure (AOP), emergency operating procedure (EOP), etc.	19.2.5 13.5.2
2. In the operational VDU of US-APWR, the layout of controllers & monitoring alignment in each window are different and this feature would make the operator perceive them as different locations.	18.4 19.2.5 13.5.2
3. In the case of loss of CCW, operators connect a non-essential chilled water system or a fire protection water supply system to the CCWS in order to cool the charging pump and maintain RCP seal water injection. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.1.4 19.2.5 13.5.2
4. When station blackout occurs, operators connect the alternate ac power to Class 1E bus in order to recovery emergency ac power. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2
5. If emergency feed water pumps cannot feed water to two intact SGs, operators will attempt to open the cross tie-line of EFW pump discharge line in order to feed water to two more than SGs by one pump.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 21 of 50)

Key Insights and Assumptions	Dispositions
6. The CS/RHR System has the function to inject the water from RWSP into the cold leg piping by switching over the CS/RHR pump lines to the cold leg piping if all safety injection systems failed (Alternate core cooling operation). In high RCS pressure sequences, a fast depressurization of the RCS by using the EFW pumps to remove heat through the SGs and by manually opening the MSRVs allows alternate core cooling injection using the CS/RHR pumps. Alternate core cooling operation may be required under conditions where containment protection signal is valid. In such cases, alternate core cooling operation is prioritized over containment spray, because prevention of core damage would have higher priority than prevention of containment vessel rupture.	19.2.5 13.5.2
7. When any two EFW pumps that commonly utilize at EFW pit have failed, operators supply water to operating EFW pumps from alternate EFW pit or demineralized water storage pit in order to ensure the water source.	19.2.5 13.5.2
8. In the case of failure to isolate failed SG, but success to sufficiently depressurize RCS by secondary side cooling and Safety depressurization valve in SGTR event, operators do RCS pressure control in order to prepare to early RHR cooling in order to ensure long term heat removal. (RCS pressure control means stopping SI safety injection and starting charging pump. RCS pressure under SI injection remains higher for connecting RHR system. Charging pump is back up for failure of RHR cooling after stopping SI injection.)	19.2.5 13.5.2
9. In the case of above, if operators fail to move RHR cooling after SI injection control, operators start to bleed and feed operation. Operators open safety depressurization valve and start the safety injection pump (if standby) in order to ensure long term heat removal.	19.2.5 13.5.2
10. When the main steam isolation valve fail to close in SGTR event, with status signal of this valve, operators try to close this valve in order to stop leakage of RCS coolant from the failed SG.	19.2.5 13.5.2
11. In the case of loss of failed SG isolation function in SGTR event, with SG pressure indication after above operation, operators open main steam depressurization valve of intact SG loop in order to promote SG heat removal and to depressurize RCS and move to cool down and recirculation operation.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 22 of 50)

Key Insights and Assumptions	Dispositions
12. In the case of loss of secondary side cooling function by emergency feedwater system in transient events including turbine trip, load loss event etc., with emergency feedwater pump flow rate, operators start to recover main feedwater system in order to maintain secondary side cooling.	19.2.5 13.5.2
13. In the case of loss of SI injection function entirely in LOCA event, with SI flow rate and RCS temperature indication, operators provide secondary side cooling to reduce RCS pressure and temperature by opening the main steam depressurization valves manually and supplying water from the emergency feedwater system in order to enable low pressure injection with containment spray system / residual heat removal system.	19.2.5 13.5.2
14. In the case of loss of containment spray system function, alternate containment cooling operation is implemented utilizing CV natural recirculation in order to remove heat from CV. This preparation contains CCW pressurization with N2 gas, disconnection heat load of non-safety chiller and CRDM etc. and connection to containment fan cooler units. This operation is implemented when the containment pressure reaches the design pressure.	19.2.5 13.5.2
15. In the case of leakage of the RWSP water from HHIS piping, CSS/RHRS piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	19.2.5 13.5.2
16. When the containment isolation signal fail to automatically actuate, with CV pressure abnormally high signal, operators manually actuate the containment isolation signal in order to remove heat from the containment vessel.	19.2.5 13.5.2
17. RCS is depressurized through operating the depressurization valve after onset of core damage and before reactor vessel breach. This operation prevents events due to high pressure melt ejection.	19.2.5 13.5.2
18. Operation of firewater injection to reactor cavity is implemented to flood reactor cavity in case of containment spray system failure, after onset of core damage and before reactor vessel breach.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 23 of 50)

Key Insights and Assumptions	Dispositions
19. When the running charging pump is unavailable, operators start the standby charging pump.	19.2.5 13.5.2
20. Operators manually start SI pumps by DAS by detection of DAS alarm in the software CCF for recovery of the automatic injection using SI pump.	19.2.5 13.5.2
21. Operators manually open SDVs by DAS by detection of DAS alarm in the software CCF for RCS depressurization.	19.2.5 13.5.2
22. When reactor trip fails (i.e., ATWS event), operators initiate boric acid transfer to maintain the adequate boron concentration in the RCS using CVCS. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2
23. When containment pressure is abnormally high due to failure of automatic containment spray actuation, operators manually actuate containment spray by opening containment spray isolation valve and CS/RHR heat exchanger cooling line valves and starting CS/RHR pumps.	19.2.5 13.5.2
24. When incoming breakers fail to automatically open in the loss of offsite power case, operators manually open the breakers to isolate Class 1E 6.9kV ac switchgears from the faulted offsite power.	19.2.5 13.5.2
25. After onset of core damage prior to reactor vessel breach, operators open the depressurization valves for RCS depressurization in order to prevent the breach caused by high pressure meld ejection.	19.2.5 13.5.2
26. Operation of fire injection to reactor cavity is implemented to flood reactor cavity in case of containment spray system failure, after onset of core damage and before reactor vessel breach.	19.2.5 13.5.2
27. Operators calibrate the EFW pit water level sensor, which is applied to changeover water source of EFW pump or to supply demineralized water to the EFW pit.	19.2.5 13.5.2
28. Operators calibrate CCW surge tank pressure sensor which is used to pressurize CCWS for alternate containment cooling.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 24 of 50)

Key Insights and Assumptions	Dispositions
29. Operators calibrate containment pressure sensors used for ESF actuation signals (safety) and for alternate containment cooling (non-safety).	19.2.5 13.5.2
30. Action to open Unlocked motor-operated valve is performed in series through the communication between operators in electrical room and in main control room.	18.6
31. MCR crew members consists of the following team members at all times during the evolution of an accident scenario: - Reactor operator (RO) - Senior reactor operator (SR) - Shift technical advisor (STA) The RO operates the plant during normal and abnormal situations, and SRO and STA check the action of the RO. If the RO commits an error during the operation, SRO or STA would correct the circumstances. However, when there is not enough available time to take corrective action, recovery credit is not considered.	19.2.5 13.5.2
32. For operator actions at local area (action that takes place outside control room) auxiliary operators (licnsed and non-licensed) are available: - Auxiliary operator 1 - Auxiliary operator 2 Normally the auxiliary operators are stational in the MCR. If the local manipulation of equipment is required to mitigate accidents or to prevent core damage, the auxiliary operator moves to the appropriate area in the reactor building or auxiliary building, to access equipment such as manual valves. It is assumed that auxiliary operator 1 operates equipment and auxiliary operator 2 checks the actions of auxiliary operator 1. If auxiliary operator 1 commits an error during the operation, auxiliary operator 2 corrects it.	19.2.5 13.5.2
33. Misalignment of remote-operated valves (e.g. motor-operated valves, air-operated valves), pumps and gas turbine generators after test and maintenance will be fixed before initiating events occur. Remote-operated valve open/close positions and control switch positions are monitored in the main control room, so they will be detected in a short time	19.2.5 13.5.2
34. The controls and displays available in the US-APWR control room are superior to conventional control room HSIs and, therefore, human error probabilities in the US-APWR operation would be less than those in conventional plants.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 25 of 50)

Key Insights and Assumptions	Dispositions
35. Misalignment of remote-operated valves (e.g. motor-operated valves, air-operated valves), pumps and gas turbine generators after test and maintenance will be fixed before initiating events occur. Remote-operated valve open/close positions and control switch positions are monitored in the main control room, so they will be detected in a short time.	19.1.4 19.1.5 13.5.2
36. The controls and displays available in the US-APWR control room are superior to conventional control room HSIs and, therefore, human error probabilities in the US-APWR operation would be less than those in conventional plants.	Chapter 18 19.1
37. In the SGTR event, operators perform at least one action to equalize primary and secondary pressure after the ruptured SG isolation. - Open safety depressurization valves - Start pressurizer auxiliary spray - Open depressurization valves for severe accident - Actuate pressurizer spray by restarting RCPs	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 26 of 50)

Key Insights and Assumptions	Dispositions
Operator actions (LPSD)	
1. Operator actions modeled in the PRA are based on symptom oriented procedures. Risk significant operator actions identified in the PRA will be addressed in plant operating procedures including AOP, EOP, etc.	19.2.5 13.5.2
2. Maintenance procedures indicate to check valve positions from the main control room after outages or testing. Valves that have been aligned in the wrong position will be detected and fixed to the correct position within a short period of time.	19.2.5 13.5.2
3. In the operational visual display unit (VDU) of US-APWR, the layout of controllers & monitoring alignment in each window are different and this feature would make the operator perceive them as different locations.	18.4 19.2.5 13.5.2
4. When the RCS is at atmospheric pressure, gravity injection from SFP is effective. Operator will perform the gravity injection by opening the injection flow path from SFP to RCS cold legs, and supplying water from RWSP to SFP.	19.2.5 13.5.2 5.4.7.2.3.6
5. When station blackout occurs, operators connect the alternative ac power with alternate gas turbines to Class 1E bus in order to recover emergency ac power. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2
6. In the case of loss of CCW/ESW, operators connect the fire suppression system to the CCWS and start the fire suppression pump in order to cool the charging pump and maintain injection to RCS. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2
7. In the case of loss of decay heat removal functions by RHRS and SGs operators start the charging pump in order to recover water level in the RCS. If water level in the RWSAT, which is the water source of charging pumps, indicates low level the operator will supply RWSP water to the RWSAT by the refueling water recirculation pump. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 27 of 50)

Key Insights and Assumptions	Dispositions
8. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves and stop leakage of RCS coolant from RHRS where LOCA occurs.	19.2.5 13.5.2
9. In case the RCS water level decreases during mid-loop operation and the failure of automatic isolation valve occurs, operator will perform the manual isolation of low-pressure letdown line.	19.2.5 13.5.2
10. When over-draining occurs and the automatic isolation valve fails, with RCS water level – low, operators close the valve on the letdown line in order to stop draining.	19.2.5 13.5.2
11. In the case of loss of decay heat removal functions by RHRS and SGs, operators start the safety injection pump in order to maintain RCS water level. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 13.5.2
12. In the case of failure of running RHRS, with RHR flow rate – low, operators open the valves on the standby RHR suction line and discharge line and start the standby RHR pump in order to maintain RHR operating.	19.2.5 13.5.2
13. In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	19.2.5 13.5.2
14. In the case of failure of running CCWS, with CCW flow rate – low, operators start the standby CCW pump in order to maintain CCWS operating.	19.2.5 13.5.2
15. When ESW strainer plugs up, with ESW pump pressure – normal, ESW flow rate – low and differential pressure – significant, operators switch from plugged strainer to standby strainer in order to maintain ESWS operating.	19.2.5 13.5.2
16. In the case of loss of decay heat removal functions from RHR, with RCS temperature – high or RCS water level – low, operators feed water to SGs by motor-driven EFW pump, open main steam depressurization valves and close the pressurizer spray vent valve (if the valve is opened) in order to remove decay heat from RCS.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 28 of 50)

Key Insights and Assumptions	Dispositions
17. In the case of failure of feed or steam line associated with available motor-driven EFW pump during secondary side cooling, operators open the EFW tie-line valves in order to feed water to multiple SGs.	19.2.5 13.5.2
18. When incoming breakers fail to automatically open in the loss of offsite power case, operators manually open the breakers to isolate Class 1E 6.9kV ac switchgears from the faulted offsite power	19.2.5 13.5.2
19. When running CS/RHR pumps are tripped due to loss of offsite power, operators restart the CS/RHR pumps to maintain the RHR operation.	19.2.5 13.5.2
20. Operators manually start charging pump and safety injection pump as a local action when the software CCF occurs.	19.2.5 13.5.2
21. Action to open Unlocked motor-operated valve is performed in series through the communication between operators in electrical room and in main control room.	18.6
22. In the event of decreasing RCS water level, operator actions to trip the CS/RHR pumps before cavitation and to restart the pumps after water level is restored will improve the reliability of RHR recovery. This operator action is important to reduce risk during shutdown.	5.4.7.2.3.6 13.5.2
23. In the event of decreasing RCS water level, operators trip CS/RHR pumps before pump cavitation occurrence. After recover the water level, operators restart the pump. The action to restart the pump has high reliability, which reduces the risk during shutdown operation.	5.4.7.2.3.6 13.5.2
24. MCR crew members consists of the following team members at all times during the evolution of an accident scenario: - Reactor operator (RO) - Senior reactor operator (SR) - Shift technical advisor (STA) The RO operates the plant during normal and abnormal situations, and SRO and STA check the action of the RO. If the RO commits an error during the operation, SRO or STA would correct the circumstances. However, when there is not enough available time to take corrective action, recovery credit is not considered.	19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 29 of 50)

Key Insights and Assumptions	Dispositions
25. For operator actions at local area (action that takes place outside control room) auxiliary operators (licnsed and non-licensed) are available: - Auxiliary operator 1 - Auxiliary operator 2 Normally the auxiliary operators are stational in the MCR. If the local manipulation of equipment is required to mitigate accidents or to prevent core damage, the auxiliary operator moves to the appropriate area in the reactor building or auxiliary building, to access equipment such as manual valves. It is assumed that auxiliary operator 1 operates equipment and auxiliary operator 2 checks the actions of auxiliary operator 1. If auxiliary operator 1 commits an error during the operation, auxiliary operator 2 corrects it.	19.2.5 13.5.2
26. Misalignment of remote-operated valves (e.g. motor-operated valves, air-operated valves), pumps and gas turbine generators after test and maintenance will be fixed before initiating events occur. Remote-operated valve open/close positions and control switch positions are monitored in the main control room, so they will be detected in a short time.	19.1.6 13.5.2
27. The controls and displays available in the US-APWR control room are superior to conventional control room HSIs and, therefore, human error probabilities in the US-APWR operation would be less than those in conventional plants.	Chapter 18 19.1

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Table 19.1-119R Key Insights and Assumptions (Sheet 30 of 50)

Key Insights and Assumptions	Dispositions
Operator actions (Severe Accidents)	
1. Operators manually initiate severe accident mitigation systems in accordance with the instructions from the technical support centre staff.	13.5.2
2. In the loss of support system sequences, operators will attempt to recover CCW/ESW or ac power while suppressing containment overpressure with firewater injection into spray header.	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 31 of 50)

Key Insights and Assumptions	Dispositions
LPSD assumptions	
1. Freeze plug may not be used for US-APWR because the isolation valves are installed considering maintenance and CCWS has been separated individual trains. Therefore, the freeze plug failure is excluded from the potential initiator.	13.5.2
2. Redundant narrow range water level instrument and a mid-range water level instrument are provided to measure mid-loop water level. Installation of a redundant water narrow level instrument enhances reliability of the mid-loop operation. A temporary mid-loop water level sensor that measures the RCS water level with reference to pressure at the reactor vessel head vent line and cross over leg is installed in addition to these permanent water level sensors to cope with surge line flooding events.	5.4.7.2.3.6 Figure 5.1-2
3. When the RCS is mid-loop operation with the closed state, the reflux cooling with the SGs is effective.	19.1.6 19.2.5 13.5.2
4. Various temporary equipment will be possible in the containment during LPSD operation for maintenance. However, it is unlikely that these materials reach the RWSP because debris interceptors are installed over the SG compartment floor openings and within the header compartment (see Chapter 6, Subsection 6.2.2). Therefore, potential plugging of the suction strainers due to debris is excluded from the PRA modeling.	6.2.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 32 of 50)

Key Insights and Assumptions	Dispositions
5. Low-pressure letdown line isolation valves are installed. One normally closed air-operated valve is installed in each of two low-pressure letdown lines that are connected to two of four RHR trains. During normal plant cooldown operation, these valves are opened to divert part of the normal RCS flow to the CVCS for purification and the RCS inventory control. These valves are automatically closed and the CVCS is isolated from the RHRS by the RCS loop low-level signal to prevent loss of RCS inventory at mid-loop operation during plant shutdown. There are no features that automate the response to loss of RHR.	5.4.7.2.2.3 5.4.7.2.3.6 7.6.1.7 19.2.5 13.5.2 TS 3.4.8 TS 3.9.6
6. The time when loss of RHR occur were set to be 12 hours after plant trip, which is the time POS 4 (mid-loop operation) is entered after plant trip, since this condition gives the most severe condition for mid-loop operation from a decay heat perspective. The pressurizer spray-line vent line with 3/4 inch diameter is assumed to be open at the initial condition. One hour after loss of RHR function, the operator is assumed to perform the following actions: - Close pressurizer spray line vent, - Start emergency feed water (EFW) pump, and - Open main steam depressurization valve. POS 8 (mid-loop operation) assumes vacuum venting equipment vents air from the RCS through the SDVs. After loss of RHR, the operator is assumed to perform the following actions. - Close valves installed in line to vacuum venting equipment such as SDVs - Start EFW pump, and - Open main steam depressurization valve.	19.2.5 13.5.2
7. Nitrogen will not be injected in the SG tubes to speed draining in the US-APWR design. The SG tubes will be filled with air during midloop operation.	19.2.5 13.5.2
8. Operator actions assumed in the PRA will be considered in the shutdown response guideline, which will be developed satisfying NUMRAC 91-06 and following other recent guidelines such as INPO 06-008.	19.2.5 13.5.2
9. Cleanliness, housekeeping and foreign material exclusion areas are administrative controls and programs to be developed by any applicant referencing the certified US-APWR design for construction and operation	6.2 Table 6.2.2-2 19.2.5 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 33 of 50)

Key Insights and Assumptions	Dispositions
10. The reactivity insertion event due to boron dilution has been judged to be insignificant to risk because of the following factors: - Strict administrative controls are in place to prevent boron dilution - Boron dilution events are highly recoverable - The CVCS design inherently limits te maximum boron duration rate. - The consequences of re-criticality are minor unless they continue for very long.	15.4.6.2 19.2.5 13.5.2
11. Administrative controls ensure the RCS water level, temperature and pressure indication are available during shutdown.	19.2.5 13.5.2
12. Either at least three pressurizer safety valves or the pressurizer manway is removed to prevent potential damage of the SG nozzle dams and loss of RCS inventory caused by loss of RHR function and subsequent pressurization while SG nozzle dams and reactor vessel head are in place.	5.4.7.2.3.6
13. Maintenance rule process is implemented to evaluate the risk of configurations being entered during shutdown. These practices assure that removing a number of related systems from service at the same time is carefully considered and virtually never done when the conditional risk impacts are high.	17.6
14. The SG nozzle dam installation level for the US-APWR is higher than in most conventional operating plants. The installation and removal of SG nozzle dams are done when the RCS water level is above the top of the main coolant piping (MCP).	5.4.7.2.3.6
15. The de-tensioning and tensioning of RV head stud bolts are performed at an RCS water level between the flange and the top of the MCP.	5.4.7.2.3.6
16. The installation and removal of the in-core instrumentation system (ICIS) is not done at mid-loop operation but is done when the RCS water level is above the top of the MCP.	5.4.7.2.3.6
17. Loss of SFP cooling is also progress the phenomena and has sufficient time to recovery because of large coolant inventory in the pool.	

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Table 19.1-119R Key Insights and Assumptions (Sheet 34 of 50)

Key Insights and Assumptions	Dispositions
18. Surge line flooding may occur if decay heat removal function is lost during plant operating states where the pressurizer manway is the only vapor release pass from the RCS. Water held up in the pressurizer can erroneous readings of water level indicators measured with reference to the pressurizer. This phenomenon can also prevent gravity injection from the SFP. Measures to prevent accident evolution caused by surge line flooding are important. Adoption of both measures listed below can reduce risk from surge line flooding event. - Installation of an temporary RCP water level sensor that measure the MCP water level with reference to pressure at the reactor vessel head vent line and cross over leg when the RCS is vented at a high elevation. - Operational procedures to perform continuous RCS injections when loss of RHR occurs under conditions where the pressurizer manway is the only vapor release pass from the RCS. The temporary water level will satisfy the following specifications. - Water level can be read outside the containment vessel (CV) in order to be effective during events which involve harsh environment in the CV - Tygon tubing monometer will not be used - Instrumentation piping diameter will be sufficient enough to prevent delay in response	5.4.7.2.3.6 19.2.5 13.5.2
19. Two types of instruments are provided in US-APWR design to measure the temperature representative of the core exit whenever the reactor vessel head is located on top of the reactor vessel. The first one is core exit thermocouples located inside the RV. The second is resistance temperature detectors in the reactor coolant hot leg. These two independent instruments will be available whenever the RCS is in a mid-loop condition and the reactor vessel head is located on top of the reactor vessel.	5.4.7.2.3.6
20. Technical Specification controls to ensure the OPERABILITY of a train of the SIS and associated water source (i.e., RWSP and refueling cavity) as an RCS makeup function during cold shutdown in reduced inventory conditions and during refueling with water level <23 ft above the top of reactor vessel flange.	TS 3.4.8, TS 3.9.6
21. Operating procedural controls to ensure the availability of the refueling cavity level instrument and alarm while the refueling cavity is flooded.	13.5.2
22. Operating procedural controls to ensure the availability of at least one safety-related pump, e.g., a CS/RHR pump, to provide makeup to the refueling cavity when the cavity is flooded.	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 35 of 50)

Key Insights and Assumptions	Dispositions
23. Administrative controls to ensure that the availability of equipment necessary to achieve containment isolation, specifically, the equipment hatch hoist, lifting rig, and AACs while the containment remains open.	13.5.2
24. All containment penetrations are closed immediately after a loss of all RHR trains.	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 36 of 50)

Key Insights and Assumptions	Dispositions
Expeditious actions outlined in GL 88-17 The following actions described as expeditious actions in Generic Letter 88-17 (Reference 19.1-54) are important to plant safety and should be implemented prior to operating in a reduced inventory condition. The expeditious actions applicable to the US-APWR design are the followings:	
1. Discuss the Diablo Canyon event, related events, lessons learned, and implications with appropriate plant personnel. Provide training shortly before entering a reduced inventory condition.	13.5.2
2. Implement procedures and administration controls that reasonably assure that containment closure will be achieved prior to the time at which a core uncover could result from a loss of decay heat removal coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory. These procedures and administrative controls should be active and in use prior to entering a reduced RCS inventory condition. Procedures should reflect that the containment is capable of being closed prior to reaching 200 °F in the RCS.	13.5.2
3. Provide at least two independent, continuous temperature indications that are representative of the core exit conditions whenever the RCS is in a mid-loop condition and the reactor vessel head is located on top of the reactor vessel.	13.5.2
Two types of instruments provided in the US-APWR design to measure RV temperature are core exit thermocouples located inside the RV and the resistance temperature detectors in the reactor coolant hot leg.	7.5.1.1.3.1 7.5.1.1.3.3
4. Provide at least two independent, continuous RCS water level indications whenever the RCS is in a reduced inventory condition.	13.5.2
Redundant narrow range level instruments are provided to meet this requirement.	5.4.7.2.3.6
5. Implement procedures and administrative controls that generally avoid operations that deliberately or knowingly lead to perturbations to the RCS and/or to systems that are necessary to maintain the RCS in a stable and controlled condition while the RCS is in a reduced inventory condition.	13.5.2
6. Provide at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal DHR systems.	13.5.2
Means of adding inventory to the RCS in the US-APWR design can be safety injection pumps, charging pump and gravity injection from the SFP.	6.3.2.1.1 5.4.7.2.3.6 9.3.4.2.6.1

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Key Insights and Assumptions	Dispositions
7. Implement procedures and administrative controls that reasonably assure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the RV. As noted in GL 88-17 (Reference 19.1-54), there is a possibility of rapid loss of RCS inventory by ejection of water through the cold leg SG manways in the event of a loss of RHR and subsequent RCS pressurization. To minimize this potential, an RCS vent path is required in accordance with GL 88-17. Whenever a cold leg opening is made without the associated cold leg nozzle dam installed, a hot leg SG manway and its associated nozzle dam will be kept open to provide an adequate vent path. Consistent with guidance provided in IN 88-36 (Reference 19.1-55), a hot leg SG manway will be the first manway opened and a hot leg nozzle dam will be the last dam to be installed.	5.4.7.2.3.6 13.5.2
8. Plant personnel calculate a plant-specific time to reach 200 °F in the RCS and time to hatch closure in order to determine if the hatch is "capable of being closed" prior to reaching harsh environment in containment in the event of loss of RHR.	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 42 of 50)

Key Insights and Assumptions	Dispositions
Internal fire assumption	
1. All fire doors serving as fire barriers between redundant safety train fire compartments are normally closed.	9.5.1
2. For transient combustibles, “three Airline trash bags” has been assumed in each fire compartment.	9.5.1
3. Transient combustibles with total heat release capacity of 93,000 Btu (obtained from NUREG/CR-6850, “AppendixG-table-7LBL-Von Volkinburg, Rubbish Bag” Test results) is assumed for Fire ignition source within Containment Vessel.	9.5.1
4. The Heat Release Rate of various items as specified in Chapter-11 of NUREG/CR-6850 is used.	9.5.1
5. Damage temperature of thermoplastic cables as shown in Appendix-H of NUREG/CR-6850 is used as the target damage temperature.	9.5.1
6. Human error probabilities of post-fire operator actions are assumed as follows.	9.5.1
- No credit has been taken for the operator actions of any equipment in the fire compartment affected by fire.	
- The Fire Brigade is provided to meet the requirements of Regulatory Guide 1.189. Higher stress levels of human actions post-fire are not assumed.	
- The HEP for operations at the remote shutdown console is assumed as 0.1.	
7. One of RCS letdown isolation valves and one of RCS vent line isolation valves are locked close by administrative controls	13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 43 of 50)

Key Insights and Assumptions	Dispositions
8. Stress levels following post-fire safe shutdown do not become higher following a fire for the following reasons: - The communication system will remain active during a fire due to the redundant nature of the plant communication systems which are installed with a minimum of two verbal communication paths between all plant locations. - The lighting system will remain operable during a fire because the emergency lighting system has a redundant power source for each fire area. - Heat and smoke propagation to adjacent fire compartments are assumed to be extremely low due to the installation of the fire dampers installed in series within each HVAC duct passing through the fire compartment, therefore it is reasonable to ignore the stress increase due to their heat and smoke.	9.5.2 9.5.3
9. It has been assumed that a “Challenging fire” for MCR shown in Table C-4 of appendix C of NUREG/CR-6850 causes an adverse operational environment in the MCR. Therefore, it has been assumed that a MCR “Challenging fire” may force the operators to abandon MCR and evacuate to the RSC room.	9A.2.7.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 47 of 50)

Key Insights and Assumptions	Dispositions
Internal flood assumption	
1. Drain systems are designed to compensate with flood having flow rate below 100 gpm. Flood with flow rate below 100 gpm will not propagate to other areas due to the drain systems.	3.4.1.3
2. R/B is separated in two divisions (i.e. east area and west area). This design is prevents loss of all safety systems though postulated major floods that leak water over the capacities of flood mitigation systems. East side and west side of reactor building (R/B) are physically separated by flood propagation preventive equipment such as water tight doors. Therefore, flood propagation between east side and west side in the reactor building is not considered.	3.4.1.3 19.2.5 13.5.2
3. Watertight doors are provided for the boundaries between R/B and A/B in the bottom floor and between R/B and T/B in flood area 1F. This measure prevents flood propagation from non-safety building to R/B.	3.4.1.3
4. Flooding of ESW system can be isolated within 15 minutes. The leaking train can also be identified by low outlet flow from each CCW HX or decrease in the ESWS header pressure. The leaking ESWS trains are then isolated by shutting down the corresponding ESWS.	
5. Four trains of ESW system have physical separation and flooding in one train does not propagate to other trains.	9.2.1.2.1 13.5.2
6. The components that are environmentally qualified are considered impregnable to spraying or submerge effects. Also component failure by flooding will not result in the loss of an electrical bus.	
7. Penetrations within the boundaries between the restricted area and non-restricted area are sealed and doors or dikes are provided for openings. Therefore, flood propagation, except for major flood events is not considered.	3.4.1.3
8. A water leak in the break room that adjoins the MCR would be isolated immediately by the operators in the MCR.	19.5.3.1
9. Internal flooding PRA is developed based on internal events PRA models. However, operator actions in flooded areas are not assumed.	19.1.4.1
10. Operator actions to isolate or mitigate flood source are not assumed except the actions in the MCR and break room.	19.1.4.1

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Table 19.1-119R Key Insights and Assumptions (Sheet 48 of 50)

Key Insights and Assumptions	Dispositions
11. The administrative controlled flood barriers that separated the reactor building between the east side and the west side are effective. The other water tight doors may be opened during maintenance.	13.5.2 19.2.5 (RAI 19-50)
12. The outage states of mitigation systems are important for LPSD risk From the insight of flooding risk, one train of mitigation system on each side in R/B should be available. So that assumed the available safety injection pumps trains A and C are available during POS 8-1. B and D pumps are assumed out of service.	13.5.2 19.2.5

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Table 19.1-201

Tornado Strike and Exceedance Frequency for the Comanche Peak Site

Enhanced F-Scale Tornado Intensity	Wind Speed (mph)	Description	Strike Frequency (/yr)	Strike Exceedance Frequency (/yr)
F0	65-85	Light Damage	1.3E-04	2.8E-04
F1	86-110	Moderate Damage	1.0E-04	1.5E-04
F2	111-135	Considerable Damage	3.7E-05	5.1E-05
F3	136-165	Severe Damage	1.2E-05	1.4E-05
F4	166-200	Devastating Damage	2.1E-06	2.4E-06
F5	200-230	Incredible Damage	2.0E-07	2.3E-07
F5	230>	Beyond Design Base	2.5E-08	2.5E-08

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Table 19.1-202

Parameters of the Design Basis Tornado

Parameter Description	Parameter
Tornado maximum wind speed	230 mph
Tornado maximum pressure drop	1.2 psi
Tornado-generated missile spectrum and associated velocities	15 ft long schedule 40 steel pipe moving horizontally at 135 ft/s. 4000 lb automobile moving horizontally at 135 ft/s. 1 in diameter steel sphere moving horizontally at 26 ft/s.

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Table 19.1-203

Tornado Accident Scenarios

Wind Speed (mph)	Assumed Impact on Plant	Frequency (/yr)	CCDP	CDF (/RY)
86-135 (F1 and F2 scale)	Loss of Offsite Power with - loss of alternate CCW	1.4E-04	2.1E-04	2.9E-08
135-230 (F3, F4 and F5 scale)	Loss of Offsite Power with - loss of alternate CCW, and - loss of alternate ac power supply	1.4E-05	1.7E-03	2.3E-08
>230 mph (F5 scale)	Failure of safety related systems Assumed guaranteed core damage	2.5E-08	1	2.5E-08

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Table 19.1-204 (Sheet 1 of 3)
Important Basic Event Related to the Site-Specific Design

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
1	SWSCF8CTYR-FF	COOLING TOWER FAN FAIL TO RUN (RUNNING) (CCF)	5.8E-09	3.3E-05	5.7E+03
2	SWSCF8CTBD-R-ALL	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	1.3E-06	4.1E-04	3.3E+02
3	SWSCF8CTBD-R-457	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
4	SWSCF8CTBD-R-147	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
5	SWSCF8CTBD-R-345	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
6	SWSCF8CTBD-R-578	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
7	SWSCF8CTBD-R-138	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
8	SWSCF8CTBD-R-358	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
9	SWSCF8CTBD-R-134	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
10	SWSCF8CTBD-R-178	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	1.2E-06	2.2E+01
11	SWSCF8CTBD-R-456	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
12	SWSCF8CTBD-R-124	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
13	SWSCF8CTBD-R-168	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
14	SWSCF8CTBD-R-146	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
15	SWSCF8CTBD-R-128	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
16	SWSCF8CTBD-R-245	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
17	SWSCF8CTBD-R-258	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01
18	SWSCF8CTBD-R-568	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	8.5E-07	1.5E+01

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Table 19.1-204 (Sheet 2 of 3)
Important Basic Event Related to the Site-Specific Design

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
19	SWSCF8CTBD-R-257	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
20	SWSCF8CTBD-R-167	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
21	SWSCF8CTBD-R-127	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
22	SWSCF8CTBD-R-356	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
23	SWSCF8CTBD-R-567	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
24	SWSCF8CTBD-R-235	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
25	SWSCF8CTBD-R-136	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
26	SWSCF8CTBD-R-123	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	6.7E-07	1.2E+01
27	SWSCF8CTBD-R-368	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
28	SWSCF8CTBD-R-467	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
29	SWSCF8CTBD-R-247	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
30	SWSCF8CTBD-R-278	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
31	SWSCF8CTBD-R-678	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
32	SWSCF8CTBD-R-238	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
33	SWSCF8CTBD-R-346	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
34	SWSCF8CTBD-R-234	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	4.2E-07	8.0E+00
35	SWSCF8CTBD-R-67	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	1.2E-06	4.4E+00
36	SWSCF8CTBD-R-36	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	1.2E-06	4.4E+00

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Table 19.1-204 (Sheet 3 of 3) Important Basic Event Related to the Site-Specific Design

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
37	SWSCF8CTBD-R-27	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	1.2E-06	4.4E+00
38	SWSCF8CTBD-R-23	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	1.2E-06	4.4E+00
39	SWSCF8CTBD-R-237	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	2.0E-07	4.4E+00
40	SWSCF8CTBD-R-236	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	2.0E-07	4.4E+00
41	SWSCF8CTBD-R-267	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	2.0E-07	4.4E+00
42	SWSCF8CTBD-R-367	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	2.0E-07	4.4E+00
43	SWSCTYR002C	COOLING TOWER FAN UHS-MFN-002C FAIL TO RUN (RUNNING)	1.4E-05	2.5E-05	2.7E+00
44	SWSCTYR001C	COOLING TOWER FAN UHS-MFN-001C FAIL TO RUN (RUNNING)	1.4E-05	2.5E-05	2.7E+00
45	SWSCF8CTBD-R-18	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	5.4E-07	2.5E+00
46	SWSCF8CTBD-R-14	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	5.4E-07	2.5E+00
47	SWSCF8CTBD-R-58	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	5.4E-07	2.5E+00
48	SWSCF8CTBD-R-45	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	3.6E-07	5.4E-07	2.5E+00
49	SWSCF8CTBD-R-145	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	9.1E-08	2.5E+00
50	SWSCF8CTBD-R-458	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	9.1E-08	2.5E+00
51	SWSCF8CTBD-R-148	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	9.1E-08	2.5E+00
52	SWSCF8CTBD-R-158	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START(RUNNING) (CCF)	6.0E-08	9.1E-08	2.5E+00

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Table 19.1-205 (Sheet 1 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
Nearby Industrial, Transportation and Military Facilities	Explosion	2.2.3.1.1	- Transportation Routes (2.2.3.1.1.1) The nearest commercial traffic is FM 56, which passes approximately 1.4 mi west-southwest of the nearest safety-related structure of CPNPP Units 3 and 4. An evaluation performed for materials with a TNT equivalency of 224 percent and using the maximum cargo for two trucks determined the safe distance to be 0.52 mi. There is considerable margin between the required safe distance and the actual distance to the nearest safety-related structure (1.4 mi). Also there are no navigable waterways used for commercial shipping within 5 mi of the CPNPP Units 3 and 4 site, and there are no main railroad lines within 5 mi of CPNPP Units 3 and 4 . - Nearby Industrial Facilities (2.2.3.1.1.2) Subsection 2.2.2.1 identifies the following facilities located within 5 mi of CPNPP Units 3 and 4, along with any potential hazardous material stored at those locations: the IESI Somervell County Transfer Station; Wolf Hollow 1, LP; the Glen Rose Medical Center; the Glen Rose WWTP; the Texas Department of Transportation Maintenance Station; and Cleburne Propane. Subsection 2.2.1 identifies six registered petroleum storage tanks within 5 mi of the CPNPP Units 3 and 4 sites. The contents, capacities, and locations of the tanks relative to CPNPP Units 3 and 4 are summarized in Table 2.2-201. .	3 3	None	No

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Table 19.1-205 (Sheet 2 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			- On-site Explosion Hazards (2.2.3.1.1.3) Gas explosions from on-site sources outside containment at CPNPP Units 3 and 4 are not credible sources of missile generation per DCD Subsection 3.5.1.1.3. The frequency of this event is calculated to be $<10^{-7}$ (criterion 6). The chemicals used for the Makeup Water Treatment System are not flammable or explosive. - Gas Wells - Explosion (2.2.3.1.1.4) One technigue used to control wellhead fires is the use of explosives to remove the oxygen from the air and thereby suffocate the fire. Potential wellhead fires in the Barnett Shale formation do not have sufficient flow rates to warrant the use of explosives to extinguish them. Therefore, the likelihood of utilizing this technique is negligible.	6 3		
	Flammable Vapor Clouds	2.2.3.1.2	- Transportation Routes (2.2.3.1.2.1) For the evaluation of the potential effects of accidents on FM 56, a single tanker truck volume of 9600 gal was assumed along with assumed rupture sizes of 4.5 square meters (m^2) and 1 m^2 located at the bottom of the tank. The release rates, puddle formation, and evaporation rates were calculated by the ALOHA code. These evaluations determined that for all cases there is a negligible overpressure at the site resulting from ignition of a vapor cloud, and the concentrations remain below the lower explosive limit at CPNPP Units 3 and 4.	3	None	No

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Table 19.1-205 (Sheet 3 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			- Industrial Facilities (2.2.3.1.2.2) There are five possible sources that may release propane into the environment from Cleburne Propane (four tanks and three trucks). Of these sources, the largest volume of propane is housed in an 18,000-gal tank. Large rupture sizes of 5 m^2 and 1 m^2 were examined for this faciliy. The release rates were calculated by the ALOHA code. The evaluation determined that there is a negligible overpressure in the area of CPNPP Units 3 and 4 resulting from a delayed ignition of a vapor cloud, and the concentrations at the CPNPP Units 3 and 4 site are negligible.	3		
			- Pipeline (2.2.3.1.2.3) Table 2.2-213 provides detailed information on the pipelines that were evaluated. These pipelines bound the potential effects to CPNPP Units 3 and 4.For the natural gas pipelines, the gas releases were calculated using the ALOHA code assuming each pipeline was connected to an infinite source so that gas escapes from the broken end of the pipeline at a constant rate for an indefinite period of time. The ALOHA results demonstrate that there is a negligible overpressure in the area of CPNPP Units 3 and 4 resulting from ignition of the gas cloud and that the concentration of the natural gas at the CPNPP Units 3 and 4 site remains below 2260 parts per million (ppm), which is well below the lower flammability limit of 44,000 ppm. For the Sunoco crude oil pipeline, both large breaks and small breaks were analyzed. The resulting overpressure at the nearest safety-related structure is 0.274 psi, which is much less than the 1 psi acceptance criteria. The vapor concentration at the CPNPP Units 3 and 4 control room intake is less than 8600 ppm, which is less than the LEL of 13,000 ppm.	3		

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Table 19.1-205 (Sheet 4 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			For the small breaks, a leak rate of 0.62 cfs was assumed for a period of 32 hours (hr). The concentration at the CPNPP Units 3 and 4 control room intakes is below 8680 ppm, which is below the LEL of 13,000 ppm. The Sunoco crude oil pipeline does not represent an explosion or flammable vapor cloud hazard at CPNPP Units 3 and 4. - Gas Wells (2.2.3.1.2.4) The closest functioning natural gas well, owned and operated by XTO Energy Inc., is 1.2 mi from the center point of CPNPP Units 3 and 4. For the purposes of evaluating the consequences of breaching a well, a gas release rate of 15.6 million cu ft/day was assumed.The results show that the maximum concentration at the CPNPP Units 3 and 4 control room intakes is 346 ppm, which is well below the LEL concentration of 44,000 ppm. The maximum overpressure at the closest safety-related structure resulting from ignition of the natural gas cloud is negligible. The analysis shows that, at the assumed release rate, the area of flammability is less than 0.1 mi downwind from a gas well release. The analysis also shows the overpressure from a gas explosion does not exceed 1 psig at a distance less than 0.1 mi from the cloud. It is concluded that the delayed ignition of vapor clouds from nearby transportation routes, pipelines, and facilities does not pose a hazard to CPNPP Units 3 and 4.	3		

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Table 19.1-205 (Sheet 5 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
	Toxic Chemicals	2.2.3.1.3 6.4.4.2	For releases of hazardous chemicals from stationary sources or from frequently shipped mobile sources in quantities that do not meet the screening criteria, detailed analyses for main control room habitability are discussed in Section 6.4.	1	None	No
			Mobile Sources (2.2.3.1.3.2.1) Of the three mobile sources (road, railroad, and waterway), only roadways are within 5 mi of the site; neither railroads nor waterways need be considered further based on the distance criteria prescribed in Regulatory Guide 1.78. Based on a postulated chlorine release, the quantity of hazardous material that may transverse FM 56 is greater than the acceptable quantity as identified in Regulatory Guide 1.78. The frequency of a hazardous chemical release on roads was also examined. Results show the total frequency for a road-based hazardous material release is higher than the 1.0E-6 screening frequency of Regulatory Guide 1.78. Therefore, a more detailed main control room habitability analysis is necessary for roadway transportation. Table 2.2-214 summarizes the chemical, quantity, and distance to the nearest CPNPP Units 3 and 4 MCR inlets to be considered for the main control room habitability analysis in Section 6.4.	1	None	No

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Table 19.1-205 (Sheet 6 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[(1)]	Freq. (/yr)	Site Appl.
			- Stationary Sources (2.2.3.1.3.2.2) The fixed facilities that could not be initially screened out based on the chemicals stored at the facility are: Wolf Hollow I, LP; Cleburne Propane; DeCordova SES; and Glen Rose WWTP. Table 2.2-214 summarizes the chemicals that do not meet the Regulatory Guide 1.78 screening criteria, and the quantity and distance to the nearest CPNPP Units 3 and 4 MCR inlets to be considered for the control room habitability analysis in Section 6.4. Section 6.4.4.2 performed the analysis on the design based control room habitability to specific toxic chemicals of mobile and stationary sources. Using conservative assumptions and input data for chemical source term, CPNPP Units 3 and 4 control room parameters, site characteristics, and meteorology inputs, postulated chemical releases are analyzed for maximum value concentration to the MCR using the HABIT code, version 1.1. RG 1.78 specifies the use of HABIT 1.1 software for evaluating control room habitability. Instrumentation to detect and alarm a hazardous chemical release in the vicinity of CPNPP Units 3 and 4, and to automatically isolate the control room envelope (CRE) from such releases is not required based on analyses described in Subsection 6.4.4.2. No hazardous chemicals concentrations in the MCR exceeded the IDLH criteria of RG 1.78. Thus, the main control room is habitable for toxic chemicals from mobile or stationary sources because no hazardous chemical concentration in the main control room exceeds the criteria of RG 1.78 (criterion 1).			

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Table 19.1-205 (Sheet 7 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
	Fires	2.2.3.1.4	Fires originating from accidents at any of the facilities or transportation routes discussed previously would not endanger the safe operation of the station because of the distance between potential accident locations and CPNPP Units 3 and 4. The location of CPNPP Units 3 and 4 is at least 0.25 mi away from any potential accident location. The nuclear island is situated sufficiently clear of trees and brush. The distance exceeds the minimum fuel modification area requirements of 30 ft. per NFPA-1144. NFPA 1144 minimum setback distance in the Owner Controlled Area (OCA) will be procedurally maintained. Also, the OCA adjacent to the isolation zone will be cleared of any concentration of vegetation for security reasons as well. There is no threat from brush or forest fires. Based on the CPNPP Units 3 and 4 site configuration, the Protected Area distance from the perimeter fence to the power block, the security isolation zone of 20 feet and the setback distance in accordance with the guidance in NFPA 1144 of minimum 30 feet, a wildfire in the vicinity of the site will not continue to propagate onto the Protected Area. Furthermore, this combined distance will ensure that the power block will not experience temperatures from a wildfire that would affect the CDF established in the PRA.	1, 3	None	No

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Table 19.1-205 (Sheet 8 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			Fire and smoke from accidents at nearby homes, industrial facilities, transportation routes, or from area forest or brush fires, do not jeopardize the safe operation of the plant due to the distance of potential fires from the plant. Any potential heavy smoke problems at the MCR air intakes would not affect the plant operators. A potential gas well fire was analyzed using the ALOHA code. This heat flux is sufficiently low as to not result in exceeding any of the thermal acceptance criteria of the structures .			

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Table 19.1-205 (Sheet 9 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
			On-site fuel storage facilities are designed in accordance with applicable fire codes, and plant safety is not jeopardized by fires or smoke in these areas. A detailed description of the plant fire protection system is presented in DCD Subsection 9.5.1. Plant structures containing safety-related equipment are constructed of non-combustible materials such as steel reinforced concrete with fire resistant roof designs to meet the International Building Code (Criterion 1). At CPNPP Units 3 and 4, the closest safety-related buildings to the Protected Area perimeter are the four Essential Service Water Buildings (ESW)/Ultimate Heat Sink (UHS) Basins located at the north side of each unit. The corners of the farthest east and farthest west UHS water basins are approximately 50 ft from the Protected Area perimeter. Although the UHS basins are safety-related, they consist of a large basin of water located within the ground where a wildfire is not anticipated to adversely affect their integrity or function. The safety-related components of the ESW Building are located in the southern half of the building placing them farther from the Protected Area perimeter. This distance is approximately 64 ft farther into the protected Area. The overall distance of the set back distance (30 ft), the security isolation zone (20 ft) and the distance the building is from the Protected Area perimeter (approx. 114 ft) is the minimum combined distance (approximately 164 ft) between the closest building and the flame front of a wildfire (Criterion 3). Refer to FSAR Figure 1.2-1R.			

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Table 19.1-205 (Sheet 10 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Collision with Intake Structure	2.2.3.1.5 2.4.8	The ESWS and the CWS draw water from the intake structure on Lake Granbury. The ESWS is supplied with water from the ultimate heat sink (UHS) and returns water to the UHS. The UHS is designed to assure sufficient cooling water inventory to mitigate the consequences of a design basis accident for a minimum of 30 days without makeup. The intake structure is not safety related. Thus, collision with the intake structure is of equal or lesser damage potential than the events for which the plant has been designed, bounded by the impact from Loss of Cooling Water Reservoirs (criterion 1).	1	None	No
	Liquid spills	2.2.3.1.6	The accidental release of petroleum products into Lake Granbury, the most likely material released, would not affect operation of the plant. The normal water level in Lake Granbury is El. 693.00 ft, with the pump intake screen at 656.00 ft. Liquids with a specific gravity less than unity, such as petroleum products, would float on the surface of the lake and are not likely to be drawn into the makeup water system. Liquids with a specific gravity greater than unity would disperse and be diluted before reaching the pump intake (criterion 3). Furthermore, liquid spills drawn into the makeup water system are bounded by the impact from complete Loss of Cooling Water Reservoirs (criterion 1).	1, 3	None	No

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Table 19.1-205 (Sheet 11 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
	Aircraft Hazards	3.5.1.6	The probability of aircraft-related hazards for CPNPP Units 3 and 4 is less than 10-7 per year (criterion 6). There are no commercial airports within 5 mi of CPNPP site. Only one military training route, Victor air route VR-158, passes within 10 mi of CPNPP site. The probability of an aircraft crashing into the plant (PFA) is estimated in the following manner: $PFA = C \times N \times A/w$ Where C = In-flight crash rate per mile for aircraft using the airway (4×10^{-10}) w = Width of airway, plus twice the distance from the airway edge to the site, conservatively provided in statute miles, equals 10 statute miles + (2 x 2 statute miles) N = Estimated annual number of aircraft operations A = Effective area of plant in square miles (0.0990) In order to maintain PFA less than the order of 10^{-7} , the above equation is rearranged to solve for N using values of C, A, and w given above: $N = PFA/(C \times A/w)$ = 17,600 operations per year	6	$<10^{-7}$	No

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Table 19.1-205 (Sheet 12 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The annual number of aircraft operations on military training route VR-158 are less than 17,600 operations per year. Thus the probability of aircraft-related hazards for CPNPP Units 3 and 4 is less than 10^{-7} per year (criterion 6).			
	Site Proximity Missle	3.5.1.5	Externally initiated missiles considered for design are based on tornado missiles and hurricane missiles as described in DCD Subsection 3.5.1.4. As described in Section 2.2, no potential site-proximity missile hazards are identified except aircraft, which are evaluated in Subsection 3.5.1.6. Thus, no site proximity missile hazard is identified (criterion 1).	1	None	No
	Turbine Missile	3.5.1.3.1 3.5.1.3.2	The CPNPP site plan shows the location of CPNPP Units 3 and 4 is such that no postulated low trajectory turbine missiles from CPNPP Units 1 and 2 can affect CPNPP Units 3 and 4. The probability of turbine missile accidents for CPNPP Units 3 and 4 is less than 10^{-7} per year is analyzed in FSAR Subsection 3.5.1.3.2. Mathematically, *P4 = P1 x P2 x P3*, where RG 1.115 considers an acceptable risk rate for *P4* as less than 10^{-7} per year. For unfavorably oriented T/Gs determined in Subsection 3.5.1.3, the product of *P2* and *P3* is estimated as 10^{-2} per year, which is a more conservative estimate than for a favorably oriented single unit. The probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, *P1*, as less than 10^{-5} per year. CPNPP Units 3 and 4 procedures will require inspection intervals and a turbine valve test frequency to maintain *P1* within acceptable limits. The acceptance risk rate *P4 = P1 x P2 x P3* is therefore maintained as less than the frequency of Tornado Missiles, 10^{-7} per year (criterion 6).	6	$<10^{-7}$	No

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Table 19.1-205 (Sheet 13 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[(1)]	Freq. (/yr)	Site Appl.
Meteorology	Hurricanes	2.3.1.2.2	The Gulf of Mexico and the Atlantic Coast areas are the most susceptible to tropical cyclones. The number of tropical storms passing within 50 statute mi of the CPNPP site are listed on Table 2.3-208 and shown on Figure 2.3-213. These data, obtained from the NOAA Coastal Services Center, show that only one hurricane, in 1900, passed within 50 mi of the site during the period 1851 – 2006. After a hurricane or tropical storm makes landfall, it begins to break apart, although remnants of the storm can continue moving inland. These remnants have been known to bring heavy precipitation, high winds, and tornadoes to locations near the CPNPP Site. Tropical cyclones including hurricanes lose strength rapidly as they move inland, and the greatest concern is potential damage from winds or flooding due to excessive rainfall. Figure 2.3-214 shows the decay of tropical cyclone winds after landfall. As seen, only the fastest moving storms will maintain any significant wind speed by the time they reach the CPNPP site. From this figure, a tropical cyclone with 86 mph winds traveling at 18 mph will have dissipated to less than 40 mph at the CPNPP site. The Probable Maximum Hurricane (PMH) for the CPNPP site, the PMH sustained (10-minute average) wind speed at 30 ft aboveground is 81 mph.	1	None	No

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Table 19.1-205 (Sheet 14 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The determination of the frequency that hurricanes, with wind speed above 90 mph, could reach the CPNPP site depends on the frequency of hurricanes striking this section of the Texas coast, the hurricane wind speed at landfall, the attenuation of wind speed while traveling inland, and the probability of a hurricane striking the CPNPP site. As stated in FSAR Subsection 2.3.1.2.2, thirty-nine tropical storms or hurricanes have struck the Texas coast between 1899 through 2006. For major hurricanes (Category 4 or higher), the return period is 12.5 yr (annual frequency of $8x10^{-2}$). The minimum wind speed for a Category 4 hurricane on the Saffir/Simpson scale is 131 mph. FSAR Figure 2.3-212 gives the number of hurricanes as a function of wind speed. These results were based on the entire U. S. coast not only the Gulf coast. As expected, the hurricane frequency of occurrence decreases as wind speed increases. For a wind speed of 125 knots (144 mph) the return period is given as 10 yr. The shape of the wind speed versus return period curve in Figure 2.3-212 shows that there is a maximum probable wind speed. This has been investigated by Jagger and Elsner (Reference 19.1-202) who determined that the maximum possible near-coastal hurricane wind speed is estimated to be 183 kt (211 mph) using a maximum likelihood approach and 208 kt (240 mph) using a Bayesian approach. The Gulf coast model presented in this paper gives a mean 1000-year return level of 173 kt (199 mph) with a 95% confidence limit of 191 kt (220 mph). In the following evaluations, the hurricane wind speed will be assumed to be the maximum possible wind speed of 240 mph with a recurrence interval of zero.			

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Table 19.1-205 (Sheet 15 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			In a paper by Kaplan and Demaria (Reference 19.1-203), the decay of tropical cyclone winds after landfall was evaluated. The wind speed after landfall is given by the following inland wind decay model: $V(t) = V_b + (RV_o - V_b)e^{-\alpha t} - C$ Where: V(t) is the wind speed as a function of time, V_b is 26.7 kt, R is 0.9, α is 0.095 hr^{-1}, t is the time after landfall, and C is a correction factor to account for the inland distance. Where: $C = m\left[ln\left(\frac{D}{D_0}\right)\right] + b$ Where: D in the inland distance in kilometers, D_o is 1 km, $m = c_1*t(t_0 - t)$, $b = d_1*t(t_0 - t)$, c_1 = 0.0109 kt/hr^2, d_1 = -0.0503 kt/hr^2, and t_o = 50 hr.			

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Table 19.1-205 (Sheet 16 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			Assuming a maximum landfall wind speed or 208 kt (~240 mph), a translational velocity of 16 kt (18.4 mph), and a distance of 400 miles from the CPNPP site to Galveston, gives a maximum possible wind speed of 61 mph at the CPNPP site. This should be considered as the upper bound of possible hurricane wind speed at the CPNPP site. The design-basis hurricane parameters used in the design and operation of CPNPP are based on Rev. 0 of RG 1.221. Figure 1 of RG 1.221 indicates that a design-basis hurricane wind speed of 145 mph applies to the CPNPP site. This value is a nominal 3-second gust at 33 ft. above ground over open terrain and has an exceedance probability of 10^{-7} per year (10 million year return period). Only one hurricane, in 1900, passed within 50 mi of the site during the period 1851 – 2006. This gives a frequency of 1/156 yr = 6.4×10^{-3} per yr of a hurricane striking the CPNPP site. As shown above, the probability of a major hurricane striking the Texas coast is small (8×10^{-2} per year) and the probability of a major hurricane passing within 50 miles of the CPNPP site is also small (6.4×10^{-3} per yr). The probability of design-basis hurricane is 10^{-7} per yr. Even if a major hurricane is assumed to strike the CPNPP site, the maximum wind speed would be 145 mph based on Rev. 0 of RG 1.221. Therefore, hurricane winds can be screened out as not risk significant because the frequency of hurricanes reaching the CPNPP site with a wind speed above the tornado design wind speed of 230 mph is exceedingly small and thus bounded by the tornado design criteria (criterion 1).			

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Table 19.1-205 (Sheet 17 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Tornadoes	2.3.1.2.3	The tornadoes reported during the years 1950-2006 in the vicinity of the site (Bosgue, Erath, Hood, and Johnson Counties) are shown in Tables 2.3-209 and 2.3-210. During this period, a total of 158 tornadoes touched down in these counties that have a combined area of 3414 sq mi. These local tornadoes have a mean path area of 0.21 sq mi excluding tornadoes with a zero length or without a length specified. The site recurrence frequency of tornadoes can be calculated using the point probability method as follows: Total area of tornado sightings =3414 sq mi Average annual frequency =158 tornadoes/56. 58 yr =2.79 tornadoes/yr Annual frequency of a tornado striking a particular point P =([0.21 mi2/tornado] [2.79 tornadoes/yr])/3414 sq mi = 0.00017 yr-1 Mean recurrence interval =1/P =5883 yr	Not screened - (bounding analysis conducted)	Close to 10^{-7}	Yes (Section 19.1.5)

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Table 19.1-205 (Sheet 18 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The corresponding expected maximum tornado wind speed and upper limit (95 percentile) of the expected wind speed based on a 2 degree longitude and latitude box centered on the CPNPP site are given below with the associated probabilities. Probability / Expected maximum tornado wind speed (mph) / Upper limit (90 percent) of the expected tornado wind speed (mph) 10-5 / 141 / 146 10-6 / 178 / 184 10-7 / 205 / 217 The design basis tornado parameters used in the design and operation of CPNPP are based on Revision 1 of Regulatory Guide 1.76. For Region I, as described in RG 1.76, the design parameters are listed below: Translational Speed 46 mph (21 meter/sec) Rotational Speed 184 mph (82 meters/sec) Maximum Wind Speed (sum of the translational and rotational speed) 230 mph (103 meters/sec) Radius of Maximum Rotational Speed 150 ft (45.7 meters) Maximum Pressure Drop 1.2psi (83mb) Rate of Pressure Drop .5psi/sec (37mb/sec)			

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Table 19.1-205 (Sheet 19 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			Compliance with Regulatory Guide 1.76 is discussed in Section 1.9. Tornado loadings are discussed in Subsection 3.3.2. It is easily lost when stand alone. This event is not screened out. Perform a bounding analysis.			
	Thunderstorm	2.3.1.2.4	Thunderstorms, from which damaging local weather can develop (tornadoes, hail, high winds, and flooding), occur about 16 days each year based on data from the counties surrounding the site. The maximum frequency of thunderstorms and high wind events occurs from April to June, while the months from November through February have few thunderstorms. The distributions of thunderstorms and high wind events by county are displayed in Table 2.3-211. Impacts (criterion 1) and design criteria (criterion 4) from thunderstorms at CPNPP are bounded by and included in the tornado design criteria	1, 4	Not determined	No
	Lightning	2.3.1.2.5 8.1.5	The annual mean number of thunderstorm days in the site area is conservatively estimated to be 48 based on interpolation from the isokeraunic map; therefore it is estimated that the annual lightning stroke density in the CPNPP site area is 25 strikes/sq mi/yr. Recent studies based on data from the National Lightning Detection Network (NLDN) indicate that the above strike densities are upper bounds for the CPNPP site. Impact of lightning (criterion 1) has been accounted for in the design of the plant. The impacts due to lightning are bounded by the impact contained in the CPNPP hurricanes and tornado events.	1, 4	None	No

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Table 19.1-205 (Sheet 20 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
	Hail	2.3.1.2.6 3.1.4.2	Almost all localities in Texas occasionally experience damage from hail. While the most commonly reported hailstones are 1/2 to 3/4 inch in diameter, hailstones 3 to 3-1/2 inch in diameter are reported in Texas several times a year. Fortunately, recurrence of damaging hail at a specific location is very infrequent. The total number of large-hail occurrences (3/4 in diameter or larger) for the fire county area around the CPNPP site is given in Table 2.3-212. Impact of hail (criterion 1) is bounded by the tornado missile spectrum design criteria for CPNPP. The loading on the structures (criterion 1) is bounded by the probable maximum winter precipitation evaluation.	1, 4	None	No
	Air Pollution Potential	2.3.1.2.7 6.4.4.2 9.4	The Clean Air Act, which was last amended in 1990, requires the U.S. Environmental Protection Agency (EPA) to set National Air Quality Standards for pollutants considered harmful to the Public health and the environment. The EPA Office of Air Quality Planning and Standards has set National Ambient Air Quality Standards for six principle pollutants, which are called "Criteria" pollutants. The newly promulgated EPA 8-hour ozone standard (62 FR 36, July 18, 1997) is 0.08 ppm in accordance with 40 CFR 50.10 (Reference 2.3-226). Somervell County is in attainment for all criteria pollutants (carbon monoxide, lead, nitrogen dioxide, particulate matter ([PM_{10}, particulate matter less than 10 micron], [$PM_{2.5}$, particulate matter less than 2.5 micron]), ozone, and sulfur oxides. The main control room is designed according to the habitability criteria defined in RG 1.78 allowing the control room to function in the event of extreme air pollution (criterion 1).	1	None	No

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Table 19.1-205 (Sheet 21 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
			The ventilation rate is a significant consideration in the dispersion of pollutants. Higher ventilation rates are better for dispersing pollution than lower ventilation rates. The atmospheric ventilation rate is numerically equal to the product of the mixing height and the wind speed within the mixing layer. Conditions in the region generally favor turbulent mixing. Two conditions which reduce mixing, increasing the air pollution potential, are surface inversions and stable air layers aloft. The surface inversion is generally a short-term effect and surface heating on most days creates a uniform mixing layer by mid-afternoon. The air stagnation trend for this general area is negative (Figure 2.3-246) over the 50-yr period of record. Thus, air pollution is not a significant site hazard (criterion 1).			
	Precipitation	2.3.1.2.8 2.3.2.1.5 3.8.4.3.4.2	Probable Maximum Precipitation (PMP), sometimes called maximum possible precipitation, for a given area and duration is the depth which can be reached but not exceeded under known meteorological conditions. For the site area, using a 100-yr return period, the PMP for 6, 12, 24, and 48 hours is 6.9, 8.3, 9.5, and 11.0 in, respectively (Table 2.3-217).	1	None	No

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Table 19.1-205 (Sheet 22 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The annual average and maximum 24-hour snowfall for these stations are given below: Station / Annual Average Snowfall (in) / Maximum 24-hr Snowfall (in) and Yr Fort Worth / 2.5 / 12.1 (1964) Dallas Love Field / 1.7 / 6.0 (1978) Mineral Wells / 1.8 / 4.0 (1978) Glen Rose / 1.8 / 4.5 (1973) To estimate the weight of the 100-yr snowpack at the CPNPP site, the maximum reported snow depths at Dallas Fort Worth Airport were determined. Table 2.3-202 shows that the greatest snow depth over the 30-yr record is 8 in. The 100-yr recurrence snow depth is 11.2 in using a factor of 1.4 to convert from a 30 yr recurrence interval to 100-yr interval. In the CPNPP site area, snow melts and/or evaporates quickly, usually within 48 hours, and does so before additional snow is added; thus, the water equivalent of the snowpack can be considered equal to the water equivalent of the falling snow as reported hourly during the snowfall. A conservative estimate of the water equivalent of snowpack in the CPNPP site area would be 0.20 in of water per inch of snowpack. Then, the water equivalent of the 100-yr return snowpack would be 11.2 in snowpack x 0.2 in water equivalent/inch snowpack =2.24 in of water. The 100-yr return period snow and ice pack for the area in which the plant is located, in terms of snow load on the ground and water equivalent, is listed below: • Snow Load =11.7 lb/ft^2 • Ice Load =5.06 in * 5.20 lb/ft^2/in =26.1 lb/ft^2			

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Table 19.1-205 (Sheet 23 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
			As stated in the US-APWR DCD Subsection 3.4.1.2, if PMWP was to occur, US-APWR safety-related systems and components would not be jeopardized. US-APWR seismic category I building roofs are designed as a drainage system capable of handling the probable maximum winter precipitation (PMWP). The US-APWR DCD also states that seismic category I structures have sloped roofs designed to preclude roof ponding. This is accomplished by channeling rainfall expeditiously off the roof. Also in subsection 3.4.1.2, the design-basis flooding level (DBFL) listed in Section 2.4, and adequate sloped site grading and drainage prevents flooding caused by probable maximum precipitation (PMP) or postulated failure of non safety-related, non seismic storage tanks located on site. Thus, the US-APWR design criteria bound the static and dynamic loadings (criterion 1).			
	Dust Storms	2.3.1.2.9 9.4	Blowing dust or sand may occur occasionally in West Texas where strong winds are more frequent and vegetation is sparse. While blowing dust or sand may reduce visibility to less than five mi over an area of thousands of sq mi, dust storms that reduce visibility to one mi or less are quite localized and depend on soil type, soil condition, and vegetation in the immediate area. The NCDC Storm Event database did not report any dust storms in Somervell County between January 1,1950 and August 31,2007. The HVAC is designed to provide filtering of particulates which minimizes the potential for impact from dust storms on plant equipment (Criterion 1).	1	None	No

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Table 19.1-205 (Sheet 24 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[1]	Freq. (/yr)	Site Appl.
	Ultimate Heat Sink	2.3.1.2.10 2.3.2.1.3	The performance of the ultimate heat sink is discussed in Subsection 9.2.5. The wet bulb design temperature for the ultimate heat sink was selected to be 80°F based on 30 yr (1977 -2006) of climatological data obtained from National Climatic Data Center/National Oceanic and Atmospheric Administrator for Dallas/Fort Worth International Airport Station in accordance with RG 1.27. The worst 30 day period was selected from the above climatological data between June 1, 1998 and June 30, 1998, with an average wet bulb temperature of 78.0°F. A 2°F margin was added to the maximum average wet bulb temperature for conservatism. These are not significant impact to ultimate heat sink.	1	None	No
	Extreme Winds	2.3.1.2.11 3.3.1.1	As with all external events, the risk impact of the hazard entails the loss of components based on their fragility with respect to the hazard. Due to the relatively high frequency and wide distribution of wind speeds, a conservative analysis was performed using actual industry failure rate experience with the most fragile component - the off site power grid supplying the plant. Multiple sources for the frequency that a loss of off site power would occur were evaluated. For the analyses, the most conservative available best estimate frequencies for at power and shutdown plant states were used. The analysis then conservatively assumes that off site power is not recovered within the mission time of the PRA analysis, even though industry experience has been recovery of off site power within the mission time of the PRA analysis. The PRA model also conservatively assumed failure of systems not located inside of structures not protected against tornado wind loadings.	6	None	No

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Table 19.1-205 (Sheet 25 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria[(1)]	Freq. (/yr)	Site Appl.
			The design wind has a basic speed of 155 mph, corresponding to a 3-second gust at 33 ft above ground for exposure category C (open terrain). For all seismic category I and II SSCs, the basic wind speed is multiplied by an importance factor of 1.15 correlating to essential facilities in hurricane-prone regions as defined in ASCE/SEI 7-05 Tables 1-1 and 6-1. All seismic category I and II SSCs including fire suppression systems are designed for the wind load and are not damaged by the extreme winds. Loss of offsite power is the limiting hazard due to extreme winds. A bounding assessment determined that the risk from extreme winds is not significant since the CDF due to the hazard is less than 1.0E-7.			

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Table 19.1-205 (Sheet 26 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Surface Winds	2.3.2.1.2	Annually, the prevailing surface winds in the region are from the south to southeast while the average wind speed is about 10 mi per hour (mph) based on-site data from 2001-2004 and 2006. As shown on Figures 2.3-208 through 2.3-210, the annual resultant wind vectors for the Dallas Fort Worth Airport, Mineral Wells, and CPNPP are 149°, 138°, and 153°, respectively. The annual average wind speeds for Dallas Fort Worth Airport, Mineral Wells, and CPNPP are 10.3, 9.0, and 9.8 mi per hour, respectively. In winter there is a secondary wind direction maximum from the north to northwest due to frequent outbreaks of polar air masses (Figures 2.3-274 and 2.3-306). Monthly and seasonal wind roses for the lower level CPNPP data are provided on Figures 2.3-278 through 2.3-293. On a monthly basis, these figures show the dominant south south-southeast wind direction. The seasonal wind rose plots show a significant additional north and north-northwest component in the winter and fall. The annual wind rose plot for CPNPP is provided on Figure 2.3-210. Monthly and seasonal wind roses for the upper level CPNPP data are provided on Figures 2.3-294 through 2.3-309. On a monthly basis, these figures show the dominant south-southeast wind direction. The seasonal wind rose plots show that the only significant north and north-northwest component is in the winter. The annual wind rose plot for CPNPP is provided on Figure 2.3-310. Thus, surface winds cannot severely affect the plant because of the insignificant potential hazards in comparison to dynamic loads resulting from tornadoes (criterion 1).	1	None	No

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Table 19.1-205 (Sheet 27 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
Hydrologic Engineering	Floods	2.4.2 2.4.10 3.4	The maximum flood level at CPNPP Units 3 and 4 due to PMF is elevation 794.09 ft NAVD 88. This elevation would result from a probable maximum precipitation (PMP) on the Squaw Creek watershed. Coincident wind waves would create maximum waves of 17 ft (trough to crest), resulting in a flood elevation of 811.09 ft NAVD 88 due to PMF. CPNPP Units 3 and 4 safety-related plant elevation is 822 ft NAVD 88, providing more than 10 ft of freeboard. The maximum water surface elevation at CPNPP Units 3 and 4 due to local intense precipitation is 820.93 ft NAVD 88, which is more than 1 ft below the safety-related plant elevation. Floods screen on Criterion 1, since the flood elevation evaluated in Chapter 2 is less than the design elevation of the plant. The Probable Maximum Precipitation (PMP) distributions used as input to the determination of the Probable Maximum Flood (PMF) for the CPNPP Units 3 and 4 were developed using Hydrometerological Report (HMR) 51 and HMR 52. The PMP distributions were calculated for the following scenarios: • Overall PMP for storm centers within the Squaw Creek watershed • Overall PMP for storm centers within the Paluxy River watershed • Squaw Creek Reservoir PMP for storm centers within the Squaw Creek watershed.	1		No

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Table 19.1-205 (Sheet 28 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The critical storm center within the Paluxy River watershed (Basin 4) results in the maximum PMP for the overall watershed (Basins 1, 2, 3 and 4 combined) at the confluence of Paluxy River and Squaw Creek. Additionally, when the storm center was kept in the Squaw Creek watershed (Basin 1) it resulted in a higher PMP for the Squaw Creek watershed. A higher PMP for the Squaw Creek watershed can result in a higher water surface elevation at CPNPP Units 3 and 4. The PMP for the critical storm center for each basin for the above mentioned scenarios was analyzed individually to determine the resulting peak runoff and the water surface elevation. No. of PMP Events			
	Probable Maximum Flood	2.4.3 2.4.10 3.4	The probable maximum flood (PMF) was determined for the Squaw Creek watershed and routed through the Squaw Creek Reservoir (SCR) to determine a water surface elevation of 794.09 ft NAVD 88. The CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft NAVD 88. Therefore, PMF on rivers and streams does not present any potential hazards for CPNPP Units 3 and 4 safety-related facilities. The PMF and maximum coincident wind wave activity results in a flood elevation of 811.09 ft NAVD 88. The top elevation of the retaining wall is 817 ft NAVD 88. The CPNPP Units 3 and 4 safety-related structures are located at elevation 822 ft NAVD 88 and are unaffected by flood conditions and coincident wind wave activity. Thus, the probable maximum flood cannot affect the plant because of the insignificance of the potential hazards (criterion 1).	1		No

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Table 19.1-205 (Sheet 29 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Dam Failures	2.4.4 2.4.8	Qualitative analysis considers both existing and future conditions and is performed based on comparison of distance from the confluence of the Paluxy River with the Brazos River, reservoir storage, dam height, and drainage area. Domino-type failures and simultaneous failures are postulated when applicable. The qualitative analysis resulted in two potential scenarios that were evaluated further by quantitative analysis. The quantitative analysis results in the critical dam failure event of the assumed domino-type failure of Fort Phantom Hill Dam, the proposed Cedar Ridge Resevoir, Morris Sheppard Dam, and De Cordova Bend Dam. In addition, Lake Stamford Dam is assumed to fail simultaneous with the Cedar Ridge Reservoir Dam. Dam failures are assumed coincident with the PMF. The resulting water surface elevation at the confluence of the Paluxy River and the Brazos River is 768.69 ft NAVD 88. CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 NAVD 88. There are no safety-related structures that could be affected by flooding due to dam failures. Thus, there are no safety-related structures that could be structurally affected by flooding due to dam failures because the displacing inventory released from a dam failure upstream of CPNPP would be sufficiently dispersed and retained in the remaining reservoir holdup volumes (criterion 3). Furthermore, the resulting impact from a dam failure on safety-related systems relying on the heat sink are bounded by a loss of makeup accident.	1, 3	None	No

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Table 19.1-205 (Sheet 30 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Surge and Seiche Flooding	2.4.5	CPNPP Units 3 and 4 are located approximately 275 mi inland from the Gulf of Mexico. CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft NAVD 88. A surge due to a probable maximum hurricane (PMH) event would not cause flooding at the site. SCR does not connect directly with any of the water bodies considered for such meteorological events associated with surge and seiche flooding. Because of the inland location and elevation characteristics, CPNPP Units 3 and 4 safety-related facilities are not at risk from surge and seiche flooding. Thus, surge and seiche flooding cannot affect the plant because of the location (criterion 3).	3	None	No
	Tsunami	2.4.6	CPNPP Units 3 and 4 are located approximately 275 mi inland from the Gulf Coast. CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft NAVD 88. Because of their inland location and elevation, CPNPP Units 3 and 4 safety related facilities would not be at risk from tsunami flooding. Thus, tsunami cannot affect the plant because of the safe distance (criterion 3).	3	None	No

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Table 19.1-205 (Sheet 31 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
	Ice Effects	2.4.7	The USACE ice jam database reports that Brazos River was obstructed by rough ice at Rainbow near Glen Rose, Texas, on January 22-23 and January 25-28, 1940, with flood stage of 20 ft. CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft NAVD 88. The SCR spillway elevation is 775 ft NAVD 88. The maximum water surface elevation during a probable maximum flood event is at 811.09 ft NAVD 88, which is more than 10 ft below the CPNPP Units 3 and 4 safety-related facilities. The possibility of inundating CPNPP Units 3 and 4 safety-related facilities due to an ice jam is remote. The climate and operation of SCR prevent any significant icing on the Squaw Creek. There are no safety related facilities that could be affected by ice induced low flow. Thus, ice effects cannot affect the plant because of the location (criterion 3).	3	None	No
	Cooling Water Canals and Resevoirs	2.4.8 2.4.9	There are no current or proposed safety-related cooling water canals or reservoirs required for CPNPP Units 3 and 4. The ultimate heat sink (UHS) is part of the essential (sometimes called emergency) service water system (ESWS). The UHS does not rely on cooling water canals or reservoirs and is not dependent on a stream, river, estuary, lake, or ocean. The UHS has the capacity to remove heat for at least 30-days in the instance of the (bounding) event of loss of makeup (criterion 1).	1	None	No
	Channel Diversions	2.4.9	There is no evidence suggesting there have been significant historical diversions or realignments of Squaw Creek or the Brazos River. The topography does not suggest potential diversions. The streams and rivers in the region are characterized by traditional shaped valleys with no steep, unstable side slopes that could contribute to landslide cutoffs or diversions. There is no evidence of ice-induced channel diversion.	1	None	No

CP COL 19.3(4)

Table 19.1-205 (Sheet 32 of 32)
Comanche Peak, Units 3 and 4 External Events Screening and Site Applicability

Category	Event	FSAR Section Disposition	Description	Screening and Applicability		
				Criteria(1)	Freq. (/yr)	Site Appl.
			The UHS has the capacity to remove heat for at least 30-days in the instance of the (bounding) event of loss of makeup (criterion 1).			
	Low Water	2.4.8 2.4.11	There are no safety-related facilities that could be affected by low-flow or drought conditions, since the UHS does not rely on the rivers and streams as a source of water (criterion 3). The UHS has the capacity to remove heat for at least 30-days in the instance of the (bounding) event of loss of makeup (criterion 1).	1, 5	None	No
	Groundwater	2.4.3 2.4.10 2.4.12 3.4	Groundwater is not used as an operational or safety-related source of water for CPNPP Units 3 and 4. CPNPP Units 3 and 4 are to be constructed on the Glen Rose Formation. According to the Design Control Document (DCD) for the US-APWR, the design maximum groundwater elevation is 1 ft below plant grade. The CPNPP plant grade elevation is 822 ft msl; therefore, the design maximum groundwater elevation is 821 ft msl relative to the current elevation of the Glen Rose Formation. Rainfall data presented was collected from the Opossum Hollow rain gauge located approximately 3.4-mi southwest of the CPNPP Unit 3 and 4 site. Overall, the hydrographs show that water levels in the deeper Glen Rose Formation (C-Zone) do not fluctuate and remain at a constant level near the base of the well or depict a steadily increasing water level, indicating the wells were dry (no groundwater infiltration into the well) or exhibiting slow recharge with the static water level not in equilibrium with the groundwater within the formation (criterion 6).	6	None	No

NOTES
(1) Screening criteria categories
"1" The event is of equal or lesser damage potential than the events for which the plant has been designed
"2" The event has a significantly lower frequency than another event and cannot result in worse consequences than this other event
"3" Cannot occur close enough to the plant to have an affect
"4" Included in the definition of another event
"5" Sufficient time to eliminate the threat or to provide an adequate response
“6” Meets the quantitative screening criteria < 10^{-7}/year

CP COL 19.3(4)
CP COL 19.3(5)

Table 19.1-206 (Sheet 1 of 2)

Site-specific Key Assumptions

Key Insights and Assumptions	Disposition
Site-Specific Design Features and Assumptions Design features and assumptions that contribute to high reliability of continuous operation after the 24 hour mission time are the followings.	
- The normal makeup water to the UHS inventory is from Lake Granbury via the circulating water system.	FSAR 9.2.5.2.2
- UHS transfer pumps and the ESW pumps located in each basin are powered by the different Class 1 E buses. UHS transfer pump operates to permit the use of three of the four basin water volumes.	FSAR 9.2.5.2.2, 9.2.5.3
- The transfer line is a high integrity line, regularly tested and inspected for corrosion.	FSAR 9.2.1.2.1, 9.2.5.4
- There are adequate low-level and high-level alarms to provide rapid control room annunciation of a level problem and to allow adeguate time to confirm the level and take effective action to address it.	FSAR 9.2.5.5
- Two basins contain enough water to supply water to remove decay heat for at least 24 hours after plant trip.	FSAR 9.2.5.1
- In operating trains, cooling tower drain valves are locked closed. For trains where the ESWP is not operating, water in exposed safety-related ESW piping in the cooling tower is drained to the basin through the drain line by opening the drain valve manually prior to the onset of temperatures that could cause freezing. After draining, the operator closes and locks the drain valve.	FSAR 9.2.1.3 Figure 9.2.5-1R
Overfill protection will be provided to prevent overfilling the basin and failing the pump(s). This feature is important to prevent degradation of the ESWS when the basin is overfilled due to failure in the transfer pump or circulation system.	FSAR 13.5 Prepare operational procedures to monitor the water level of basin at main control room.
Plant specific SSCs that potentially impact plant safety are seismically designed and thus will not impact the plant HCLPF. HCLPF values for the plant specific SSCs, such as cooling towers, will be confirmed with calculation using EPRI TR-1002988 methodology after completion of seismic design and stress analysis of the SSCs.	DCD 19.1.2.4 FSAR 19.1.5.1.1 DCD Tier 1 ITAAC #24
The UHS is designed with sufficient inventory to provide cooling for at least 30 days following the most limiting design basis accident without makeup water. The possibility of loss of CWT function caused by seismic failure of makeup water is negligible.	FSAR 9.2.5

CP COL 19.3(4)
CP COL 19.3(5)

Table 19.1-206 (Sheet 2 of 2)

Site-specific Key Assumptions

Key Insights and Assumptions	Disposition
The design of the UHS consists of reinforced concrete structures that are directly founded on the Glen Rose Formation limestone Layer C, and does not include any earth embankments for side wall support.	FSAR 2.5.5
The layout design of the site-specific seismic Category I SSCs ensures that there are no adjacent non-seismic Category I structures that may adversely affect site-specific seismic Category I SSCs including the UHS structures. The presence of the subject retaining wall and adjacent backfill slopes do not have any adverse effect on the UHS structures.	FSAR 3.7.2.8
The elevation of pumping equipment and cooling fans are higher than the elevation of the basin wall and the ground elevation, and are enclosed by a concrete wall. The pumping equipment and cooling fans are protected from flooding due to the failure of the non-seismic intake piping to the UHS.	FSAR 3.8.4.1.3.2
NFPA 1144 minimum setback distance in the Owner Controlled Area will be procedurally maintained. Also, the Owner Controlled Area adjacent to the isolation zone will be cleared of any concentration of vegetation for security reasons.	FSAR 9.5 NFPA 1144 minimum setback distance will be procedurally maintained
Administrative control will be in place to ensure that the truck bay entrance of the reactor building is closed when a tornado or hurricane is nearby or source of high wind is forecast for the immediate area.	FSAR 13.5
Adequate sloped site grading and drainage prevents flooding caused by probable maximum precipitation (PMP) or postulated failure of non safety-related, non seismic storage tanks located on site.	FSAR 3.4.1.2
All seismic Category 1 buildings and structures below-grade are protected against the effects of flooding, including ground water.	FSAR 3.4.1.2

CP COL 19.3(8)

Table 19.1-207 (Sheet 1 of 2)

Cross Reference of PRA Programs and Applications

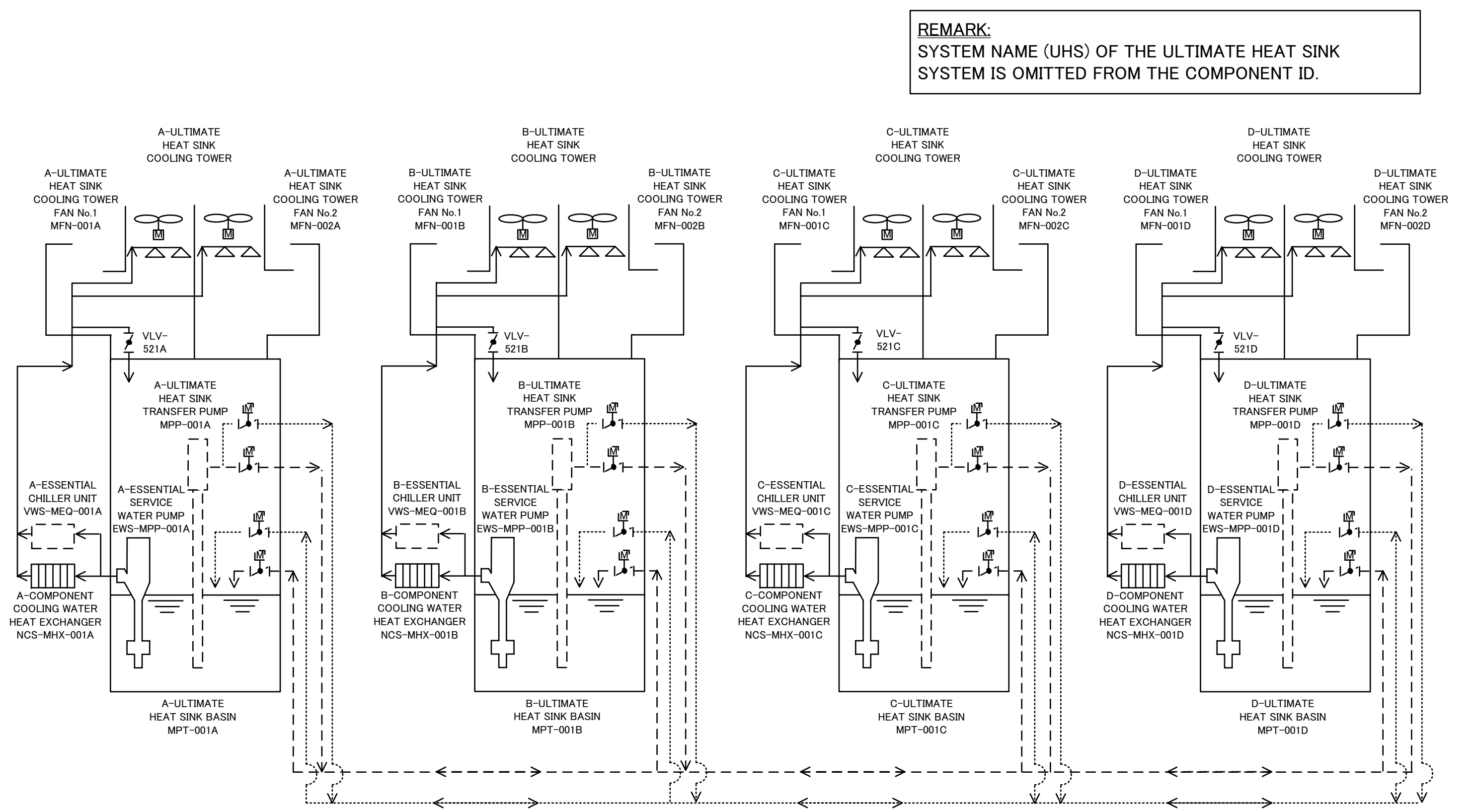
Application / Program	Design Phase FSAR 19.1.1.1	COL Phase FSAR 19.1.1.2	Construction Phase FSAR 19.1.1.3	Operational Phase FSAR 19.1.1.4	FSAR Cross Reference Section
Programs					
Input to design programs and processes	Determine risk / insights associated with design	Maintain assumptions / insights. Evaluate site specific aspects. (External events, SSCs beyond DCD)	Maintain assumptions / insights.	Maintain assumptions / insights.	Section 14.3.3.5, Section 19.1.7.1, Table 19-1-119R, Table 19.1-206, Section 19.2.5
Input to Maintenance Rule (MR) implementation (10CFR50.65)			Implement MR prior to initial fuel load	Provide inputs to MR for program	T.S. 5.5.18, Section 17.6, Section 19.1.7.2
Input to Reactor Oversight Process (ROP)			Implement ROP prior to initial fuel load	Provide inputs to ROP for program	Section 19.1.7.3
Input to Reliability Assurance Program (RAP)	Provide importance measures for RAP	Maintain assumptions / insights. Evaluate site specific aspects. (External events, SSCs beyond DCD)	Maintain assumptions / insights.	Maintain assumptions / insights.	Section 17.1, Section 17.2, Section 17.3, Section 17.4, Table 17.4-1, Table 17.4-201, Section 19.1.7.4

Table 19.1-207 (Sheet 2 of 2)

CP COL 19.3(8)

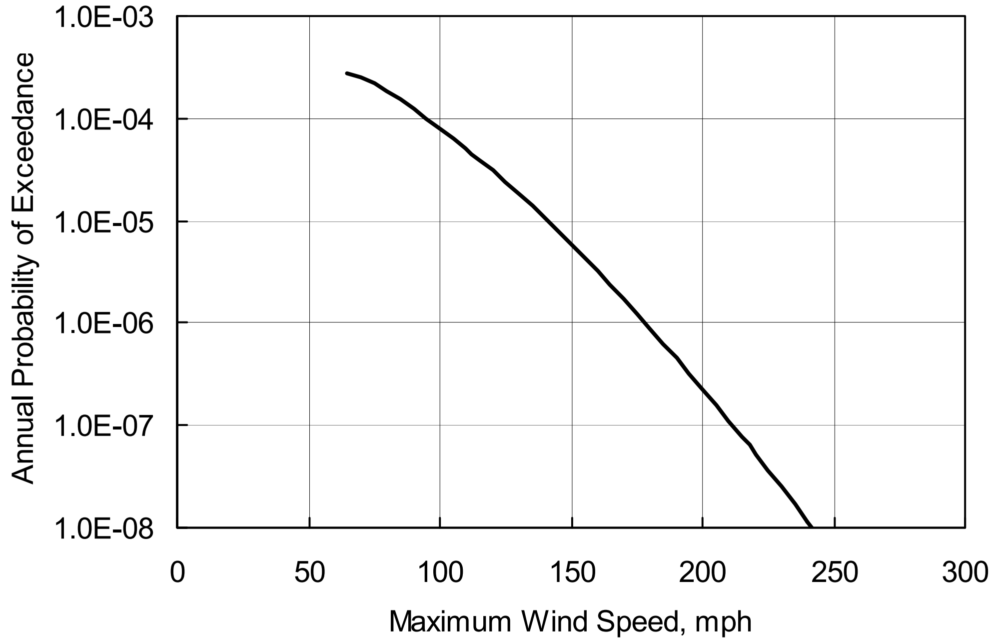
Cross Reference of PRA Programs and Applications

Application / Program	Design Phase FSAR 19.1.1.1	COL Phase FSAR 19.1.1.2	Construction Phase FSAR 19.1.1.3	Operational Phase FSAR 19.1.1.4	FSAR Cross Reference Section
Programs					
Input to regulatory treatment of Non-Safety-Related Systems Program	Provide importance measures for program	Maintain assumptions / insights. Evaluate site specific aspects. (External events, SSCs beyond DCD)	Maintain assumptions / insights.	Maintain assumptions / insights.	Section 19.1.7.5
Input to Human Factors Engineering (HFE) Program		Input to procedures and HFE program	Input to procedures and HFE program	Input to procedures and HFE program	Chapter 18
Applications					
Input to Technical Specifications (Risk Managed Technical Specifications, Initiative 4b)			Implement prior to initial fuel load	Provide inputs to Initiative 4b program	TS 5.5.18, Section 16.1.1.2, Section 19.1.7.6
Input to Technical Specifications (Surveillance Frequency Control Program, Initiative 5b)			Implement prior to initial fuel load	Provide inputs to Initiative 5b program	TS 5.5.19, Section 16.1.1.2, Section 19.1.7.6



CP COL 19.3(4)

Figure 19.1-2R Simplified System Diagram (Sheet 20 of 42) (Essential Service Water System [2of3])



CP COL 19.3(4)

Figure 19.1-201 Point Estimate Probability of Tornado Exceeding Maximum Wind Speed at the Comanche Peak Site

19.2 SEVERE ACCIDENT EVALUATION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.2.3.3.7 Equipment Survivability

STD COL 19.3(7) Replace the second-to-last paragraph in DCD Subsection 19.2.3.3.7 with the following.

Equipment survivability assessments will be performed prior to fuel load of the as-built equipment required to maintain safe shutdown and containment structural integrity to provide reasonable assurance that they will operate in the environmental conditions resulting from hydrogen burns associated with severe accidents for which they are intended and over the time span for which they are needed. The FSAR will be updated prior to fuel load to state that the assessments have been performed and document the conclusion that the as-built equipment meets the survivability requirements. These assessments are required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The ability of the as-built equipment to perform during severe accident hydrogen burns will be assessed using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354 (Reference 19.2-11).

19.2.5 Accident Management

STD COL 19.3(6) Add the following text after the last paragraph in DCD Subsection 19.2.5.

An accident management program will be developed, in which severe accident management procedures that capture important operator actions described in the severe accident management framework are included. The accident management program will incorporate the instructions provided in NEI 91-04 Revision 1 (Reference 19.2-201). Development of emergency operating procedures is addressed in Subsection 13.5.2.1. Training requirements will also be developed as part of the accident management program addressed in DCD Section 18.9, and training for operators will be completed prior to first fuel load.

19.2.6.1 Introduction

STD COL 19.3(4) Replace the content of DCD Subsection 19.2.6.1 with the following.

This section is prepared using site-specific PRA information to consider potential design improvements as required under 10 CFR 50.34(f) and follows content guidance provided in NRC Regulatory Guide 1.206. Information for this section is from Subsections 7.2 and 7.3 of the Environmental Report, Part 3 of the Combined License (COL) Application.

19.2.6.1.1 Background

STD COL 19.3(4) Add the following text after the last paragraphs in DCD Subsection 19.2.6.1.1.

Design or procedural modifications that could mitigate the consequences of severe accidents are known as severe accident mitigation alternatives (SAMAs). For design certification, SAMAs are known as severe accident mitigation design alternatives (SAMDAs), which focus on design changes and do not consider procedural modifications for SAMAs. For an existing plant with a well-defined design and established procedural controls, the normal evaluation process for identifying potential SAMAs includes four steps:

1. Define the base case -The base case is the dose-risk and cost-risk of severe accidents before implementation of any SAMAs. A plant's PRA is the primary source of data in calculating the base case. The base case risks are converted to a monetary value to use for screening SAMAs.

2. Identify and screen potential SAMAs - Potential SAMAs can be identified from the plant's individual plant examination (IPE), the plant's PRA, and the results of other plants' SAMA analyses. This list of potential SAMAs is assigned a conservatively low implementation cost based on historical costs, similar design changes, and/or engineering judgment, then compared to the base case screening value. SAMAs with higher implementation cost than the base case are not evaluated further.

3. Determine the cost and net value of each SAMA - A detailed engineering cost evaluation is developed using current plant engineering processes for each SAMA remaining after step 2. If the SAMA continues to pass the screening value, step 4 is performed.

4. Determine the benefit associated with each screened SAMA - Each SAMA that passes the screening in step 3 is evaluated using the PRA model to determine the reduction in risk associated with implementation of the proposed SAMA. The reduction in risk benefit is then monetized and compared to the detailed cost estimate. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

In the absence of a completed plant with established procedural controls, the current analysis is limited to demonstrating that a US-APWR located at the site is bounded by the DCD analysis, and determining what magnitude of plant-specific design or procedural modifications would be cost-effective. Determining the magnitude of cost effective design or procedural modifications is the same as step 1, "Define base case," for operating nuclear plants. The base case benefit value is calculated by assuming that the current dose risk of the unit could be reduced to zero, then assigning a defined dollar value for this change in risk. Any design or procedural change cost that exceeds the benefit value would not be considered cost-effective.

The dose-risk and cost-risk results (Section 7.2 of the Environmental Report) are monetized in accordance with methods established in NUREG/BR-0184. NUREG/BR-0184 presents methods for determination of the value of decreases in risk by using four types of attributes: public health, occupational health, off-site property, and on-site property. Any SAMAs in which the conservatively low implementation cost exceeds the base case monetization would not be expected to pass the screening in step 2. If the baseline analysis produces a value that is below that expected for implementation of any reasonable SAMA, no matter how inexpensive, then the remaining steps of the SAMA analysis are not necessary.

(Note: Hereafter where the word "SAMDA" appears in the DCD, it is replaced with "SAMA" in the Final Safety Analysis Report (FSAR) without any further notification.)

19.2.6.2 Estimate of Risk for Design

STD COL 19.3(4) Replace the last sentence of the first paragraph in DCD Subsection 19.2.6.2 with the following.

The second analysis is a Level 3 PRA analysis that integrates the Level 2 source term to quantify the consequences based on the site-specific parameters.

CP COL 19.3(4) Replace the third through the last sentences of the third paragraph and all of the fourth paragraph in DCD Subsection 19.2.6.2 with the following.

In the offsite dose risk quantification, three years of site-specific meteorological data are used. The 50-mile population distribution data are based on the projected population for calendar year 2056.

The total population dose risk is 3.0E-01 person-rem/reactor-year, and the largest contributor is from RC5 - containment failure condition including overpressure

failure after core damage, hydrogen combustion failure after core damage, hydrogen combustion long after reactor vessel failure and basemat melt-through (49 percent).

19.2.6.4 Risk Reduction Potential of Design Improvements

CP COL 19.3(4) Replace the last sentence in DCD Subsection 19.2.6.4 with the following.

The maximum averted cost is approximately $400k.

19.2.6.5 Cost Impacts of Candidate Design Improvements

STD COL 19.3(4) Replace the first sentence in the last paragraph in DCD Subsection 19.2.6.5 with the following.

SAMA cost evaluation results are described in Table 19.2-9R.

19.2.6.6 Cost-Benefit Comparison

CP COL 19.3(4) Replace the content of DCD Subsection 19.2.6.6 with the following.

The maximum averted cost-risk of approximately $400k for a single US-APWR unit at the CPNPP Unit 3 and 4 is so low that there are no design changes over those already incorporated into the US-APWR design that could be determined to be cost-effective. A sensitivity evaluation was performed with a conservative 3% discount rate and the valuation of the maximum averted cost is approximately $1,055K. The benefit of each SAMA at 3% and 7% discount rates was calculated and is presented in Table 19.2-9R. The cost of each SAMA exceeds the corresponding benefit.

Accordingly, further evaluation of design-related SAMAs is not warranted. Evaluation of administrative SAMAs would not be appropriate until the plant design is finalized, and plant administrative processes and procedures are developed. At that time, appropriate administrative controls on plant operations would be incorporated into the plant's management systems as part of its baseline.

19.2.7 References

CP COL 19.3(6) Add the following reference document after the last document in DCD Subsection 19.2.7.

19.2-201 Severe Accident Issue Closure Guidelines, NEI 91-04 Rev. 1, Nuclear Energy Institute, December 1994

CP COL 19.3(4)

Table 19.2-9R

SAMA Cost Evaluation Results

	Design Alternative	Cost Impact	Maximum Averted Cost	Sensitivity of each SAMA benefit	
				7% Discount rate (baseline)	3% Discount rate (Sensitivity)
1	Provide additional dc battery capacity.	$2,000k		$160k	$422k
2	Provide an additional gas turbine generator.	$10,000k		$160k	$422k
3	Install an additional, buried off-site power source.	$10,000k		$164k	$433k
4	Provide an additional high-pressure injection pump with independent diesel.	$1,000k		$208k	$549k
5	Add a service water pump.	$5,900k		$100k	$264k
6	Install an independent reactor coolant pump seal injection system with dedicated diesel.	$3,800k	$400k (Baseline) $1,055k (Sensitivity)	$188k	$496k
7	Install an additional component cooling water pump.	$1,500k		$100k	$264k
8	Add a motor-driven feed-water pump.	$2,000k		$140k	$369k
9	Install a filtered containment vent to remove decay heat.	$3,000k		$240k	$632k
10	Install a redundant containment spray system.	$870k		$19k	$50k

19.3 OPEN, CONFIRMATORY, AND COL ACTION ITEMS IDENTIFIED AS UNRESOLVED

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.3.3 Resolution of COL Action Items

Replace the content of DCD Subsection 19.3.3 with the following.

CP COL 19.3(1) ***19.3(1)*** *Update of PRA and SA evaluation for input to RMTS and peer review*

This COL item is addressed in Subsections 19.1.2.3 and 19.1.7.6.

19.3(2) *Deleted from the DCD.*

19.3(3) *Deleted from the DCD.*

CP COL 19.3(4)
STD COL 19.3(4) ***19.3(4)*** *Update of PRA and SA evaluation based on site-specific information*

This COL item is addressed in Subsections 19.1.1.2.1, 19.1.4.1.2, 19.1.4.2.2, 19.1.5, 19.1.5.1.1, 19.1.5.2.2, 19.1.5.3.2, 19.1.6.2, 19.1.7.1, 19.1.9, 19.2.6.1, 19.2.6.1.1, 19.2.6.2, 19.2.6.4, 19.2.6.5 and 19.2.6.6, Tables 19.1-201, 19.1-202, 19.1-203, 19.1-204, 19.1-205, 19.1-206 and 19.2-9R, and Figures 19.1-201 and 19.1-2R.

CP COL 19.3(5) ***19.3(5)*** *SSC fragilities*

This COL item is addressed in Subsections 19.1.5.1.1, 19.1.5.1.2, 19.1.9 and Table 19.1-206.

STD COL 19.3(6)
CP COL 19.3(6) ***19.3(6)*** *Accident management program*

This COL item is addressed in Subsections 19.2.5, 19.2.7 and Table 19.1-119R.

STD COL 19.3(7) ***19.3(7)*** *Equipment survivability assessment*

This COL item is addressed in Subsection 19.2.3.3.7.

CP COL 19.3(8) ***19.3(8)*** *Licensee programs and risk-informed applications*

This COL item is addressed in Subsections 19.1, 19.1.1.2.1, 19.1.1.3.1, 19.1.1.3.2, 19.1.1.4, 19.1.1.4.1, 19.1.1.4.2, and 19.1.7, and Table 19.1-207.

CP COL 19.3(9) ***19.3(9)*** *PRA Maintenance and upgrade programs*

This COL item is addressed in Subsection 19.1.2.4.

CP COL 19.3(10) ***19.3(10)*** *Confirmation of PRA insights and assumptions*

This COL item is addressed in Subsection 19.1.4.1.2.

APPENDIX 19A

US-APWR BEYOND DESIGN BASIS AIRCRAFT IMPACT ASSESSMENT

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19A US-APWR BEYOND DESIGN BASIS AIRCRAFT IMPACT ASSESSMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19B

SUMMARY OF PSMS RELIABILITY ANALYSIS IN PRA

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19B SUMMARY OF PSMS RELIABILITY ANALYSIS IN PRA

This section of the referenced DCD is incorporated by reference with no departure or supplements.