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

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

19.1.1.2.1 Uses of Probabilistic Risk Assessment in Support of Licensee Programs

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CP COL 19.3(4) 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.

19.1.1.4.1 Uses of Probabilistic Risk Assessment in Support of Licensee Programs

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STD SUP 19.1(1) Add the following text after the first paragraph in DCD Subsection 19.1.1.4.1.

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 SUP 19.1(2) 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.

19.1.4.1.2 Results from the Level 1 PRA for Operations at Power

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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 an additional failure mode for the ESWS, which is the failure of CTW fans. Failure of the CTW fans would cause degradation of heat release from the ESWS to the atmosphere, which would result increase of the ESWS temperature in the faulted train. Failure of both fans in a single CTW train is considered a potential failure mode of the ESWS.

Failures of CTW fans 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.

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, 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.

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 because they are easy to obtain lower risk from advanced reactors design features or site characteristics. The 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. Lower damage potential than a design basis event

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

Following the qualitative screenings, quantitative screenings are performed. The supporting technical requirement EXT-B2 of ASME/ANS RA-Sa-2009 states that the criteria provided in the 1975 Standard Review Plan can be used as an acceptable basis for the screening criteria of external events. The criteria are:

i. the contribution to core damage frequency (CDF) is less than 10^{-6}/year, or

ii. the design-basis event at annual frequencies of occurrence is between 10^{-7} and 10^{-6}.

For Comanche Peak Units 3 and 4, a value of 10^{-7} for the annual frequency of occurrence is used as a more conservative quantitative screening criterion. If an event frequency is greater than 10^{-7}/year, perform bounding analysis or PRA to confirm that the risk is sufficient lower for advanced light water reactors such as less than 1% of total CDF. The remaining external events which do not meet the above screening criteria are assessed using a bounding analysis.

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 and Tornadoes</u>

For high winds and tornadoes, 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.

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 event during LPSD does not have significant contribution to risk.

External Flooding

Subsection 2.4.2 systematically considers the various factors that can contribute to the incident of external flooding. Based on the discussions in this section, the contribution of such events to the total CDF is considered insignificant. These events meet the preliminary screening criteria of ASME/ANS RA-Sa-2009.

Transportation and Nearby Facility Accidents

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.

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. Thus, this event 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).

19.1.5.1.2 Results from the Seismic Risk Evaluation

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.

19.1.7.1 PRA Input to Design Programs and Processes

STD COL 19.3(4) Add the following text after the last sentence of DCD Subsection 19.1.7.1.

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(6)

Table 19.1-119R Key Insights and Assumptions (Sheet 1 of 46)

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 7.8.1.1.1 Table 7.8-5
- SI pump can be manually actuated by DAS from MCR.	

CP COL 19.3(6)

Table 19.1-119R Key Insights and Assumptions (Sheet 3 of 46)

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

CP COL 19.3(6)

Table 19.1-119R Key Insights and Assumptions (Sheet 4 of 46)

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

CP COL 19.3(6)

Table 19.1-119R Key Insights and Assumptions (Sheet 12 of 46)

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 46)

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 pool 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 46)

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
- 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.	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 46)

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 8.3.1.1.1 8.4.1.3
- Common cause failure between class 1E GTG and non-class 1E GTG supply is minimized by design characteristics. Different rating GTGs with diverse starting system, independent and separate auxiliary and support systems are provided to minimize common cause failure.	
- The non-safety 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
- 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 9.5.5 9.5.8
- 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.	

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Table 19.1-119R Key Insights and Assumptions (Sheet 17 of 46)

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. The igniters start with the ECCS actuation signal and are powered by two non-class 1E buses with non-class 1E GTGs.	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 lines 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 46)

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. Instrumentation piping are installed at upside of the RV. 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

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Table 19.1-119R Key Insights and Assumptions (Sheet 20 of 46)

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 46)

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 46)

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. When the CCW header tie-line isolation valves fail to automatically close with specific signals which contain ECCS actuation signal plus under-voltage signal, containment spray signal, and surge tank level low signal, operators manually close these valves in order to separate CCW header.	19.2.5 13.5.2
18. 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
19. 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 46)

Key Insights and Assumptions	Dispositions
20. When the running charging pump is unavailable, operators start the standby charging pump.	19.2.5 13.5.2
21. 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
22. Operators manually open SDVs by DAS by detection of DAS alarm in the software CCF for RCS depressurization.	19.2.5 13.5.2
23. 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
24. 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
25. 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
26. 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
27. 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
28. 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
29. 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 46)

Key Insights and Assumptions	Dispositions
30. Operators calibrate containment pressure sensors used for ESF actuation signals (safety) and for alternate containment cooling (non-safety).	19.2.5 13.5.2
31. 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
32. 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
33. 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 equipments 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
34. 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
35. 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 46)

Key Insights and Assumptions	Dispositions
36. 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
37. 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 26 of 46)

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 46)

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. In the case of failure of running ESWS, with CCW flow rate – low, operators start the standby ESW pump in order to maintain ESWS operating.	19.2.5 13.5.2
16. 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
17. 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 MSDVs and close pressurizer spray vent valve (if open) 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 46)

Key Insights and Assumptions	Dispositions
18. 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
19. 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
20. 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
21. Operators manually start charging pump and safety injection pump as a local action when the software CCF occurs.	19.2.5 13.5.2
22. 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
23. 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
24. 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
25. 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 46)

Key Insights and Assumptions	Dispositions
26. 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 equipments 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
27. 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
28. 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 46)

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 46)

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. Hydrogen peroxide addition is adopted instead of aeration because it decreases the duration of the mid-loop operation: hydrogen peroxide addition operation does not require mid-loop duration. As a result of adopting hydrogen peroxide addition which is done at a higher SG nozzle level, the mid-loop operation is needed only to drain the SG primary side water while, thus reducing overall duration mid-loop operation.	5.4.7.2.3.6
3. 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
4. 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
5. Various equipments will be possible temporary in the containment during LPSD operation for maintenance. However, there are few possibilities that these materials fall into the sump because the debris interceptor is installed on the sump of US-APWR. 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 46)

Key Insights and Assumptions	Dispositions
6. 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
7. 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.	19.2.5 13.5.2
8. 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
9. 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
10. 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
11. 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

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Table 19.1-119R Key Insights and Assumptions (Sheet 33 of 46)

Key Insights and Assumptions	Dispositions
12. Administrative controls ensure the RCS water level, temperature and pressure indication are available during shutdown.	19.2.5 13.5.2
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 46)

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

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Table 19.1-119R Key Insights and Assumptions (Sheet 35 of 46)

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.	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
Two types of instruments are provided in US-APWR design to measure RCS water level are the middle range RCS water level sensor and the narrow level middle range water level sensor .	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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Table 19.1-119R Key Insights and Assumptions (Sheet 36 of 46)

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. Pressurizer safety valves are removed to prevent the damage of SG nozzle dams caused by loss of RHR function while SG nozzle dams and reactor vessel head are placed. Removal of the pressurizer safety valves is done during the period between removal of the SG manyays and installation of the SG nozzle dams. Installation of the pressurizer safety valves is performed during a period between removal of the SG nozzle dams and installation of SG manways.	5.4.7.2.3.6 13.5.2

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Table 19.1-119R Key Insights and Assumptions (Sheet 40 of 46)

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. Operators are well trained in responding to fire event.	9.5.1
7. One of RCS letdown isolation valves and one of RCS vent line isolation valves are locked close by administrative controls	13.5.2
8. Each yard transformer is separated by a fire barrier.	19.1.5.2.1

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Table 19.1-119R Key Insights and Assumptions (Sheet 44 of 46)

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 to be isolated within 15 minutes.	
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. 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.	19.2.5 13.5.2
9. 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.	19.2.5 13.5.2
10. 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

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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 4)
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 (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	1.3E-06	2.3E+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	1.3E-06	2.3E+01
5	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	1.3E-06	2.3E+01
6	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	1.3E-06	2.3E+01
7	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	1.3E-06	2.3E+01
8	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	1.3E-06	2.3E+01
9	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	1.3E-06	2.3E+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	1.3E-06	2.3E+01
11	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	9.0E-07	1.6E+01
12	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	9.0E-07	1.6E+01
13	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	9.0E-07	1.6E+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	9.0E-07	1.6E+01
15	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	9.0E-07	1.6E+01
16	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	9.0E-07	1.6E+01
17	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	9.0E-07	1.6E+01
18	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	9.0E-07	1.6E+01

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Table 19.1-204 (Sheet 2 of 4) 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	7.3E-07	1.3E+01
20	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	7.3E-07	1.3E+01
21	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	7.3E-07	1.3E+01
22	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	7.3E-07	1.3E+01
23	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	7.3E-07	1.3E+01
24	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	7.3E-07	1.3E+01
25	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	7.3E-07	1.3E+01
26	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	7.3E-07	1.3E+01
27	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	4.4E-07	8.4E+00
28	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	4.4E-07	8.4E+00
29	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	4.4E-07	8.4E+00
30	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	4.4E-07	8.4E+00
31	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	4.4E-07	8.4E+00
32	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	4.4E-07	8.4E+00
33	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	4.4E-07	8.4E+00
34	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	4.4E-07	8.4E+00
35	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	1.26E-06	4.5E+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	1.26E-06	4.5E+00

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Table 19.1-204 (Sheet 3 of 4)
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-67	COOLING TOWER FAN UHS-MFN-001A,B,C,D,002A,B,C,D FAIL TO RE-START (RUNNING) (CCF)	3.6E-07	1.26E-06	4.5E+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	1.26E-06	4.5E+00
39	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	2.1E-07	4.5E+00
40	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	2.1E-07	4.5E+00
41	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	2.1E-07	4.5E+00
42	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	2.1E-07	4.5E+00
43	SWSCF4CTYRBD-ALL	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF)	1.8E-07	1.6E-07	4.3E+00
44	SWSCF4CTYRBD-13	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	1.2E-07	8.3E-08	4.3E+00
45	SWSCF4CTYRBD-24	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	1.2E-07	8.3E-08	4.3E+00
46	SWSCF4CTYRBD-23	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	1.2E-07	8.3E-08	4.3E+00
47	SWSCF4CTYRBD-123	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	6.0E-08	<1.0E-07	4.3E+00
48	SWSCF4CTYRBD-124	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	6.0E-08	<1.0E-07	4.3E+00
49	SWSCF4CTYRBD-134	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	6.0E-08	<1.0E-07	4.3E+00
50	SWSCF4CTYRBD-234	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	6.0E-08	<1.0E-07	4.3E+00
51	SWSCF4CTYRBD-14	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO RUN (RUNNING) (CCF	1.2E-07	8.3E-08	4.3E+00
52	SWSCTYR001C	COOLING TOWER FAN UHS-MFN-001C FAIL TO RUN (RUNNING)	1.4E-05	2.6E-05	2.8E+00
53	SWSCTYR002C	COOLING TOWER FAN UHS-MFN-001C FAIL TO RUN (RUNNING)	1.4E-05	2.6E-05	2.8E+00
54	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	6.0E-07	2.7E+00

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Table 19.1-204 (Sheet 4 of 4)
Important Basic Event related to the Site-Specific Design

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
55	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	6.0E-07	2.7E+00
56	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	6.0E-07	2.7E+00
57	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	6.0E-07	2.7E+00
58	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	1.0E-07	2.7E+00
59	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	1.0E-07	2.7E+00
60	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	1.0E-07	2.7E+00
61	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	1.0E-07	2.7E+00
62	SWSCF4CTBDBD-ALL	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	1.3E-06	1.3E-06	2.0E+00
63	SWSCF4CTBDBD-34	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	8.3E-07	8.7E-07	2.0E+00
64	SWSCF4CTBDBD-12	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	8.3E-07	8.7E-07	2.0E+00
65	SWSCF4CTBDBD-23	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	8.3E-07	8.7E-07	2.0E+00
66	SWSCF4CTBDBD-14	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	8.3E-07	8.7E-07	2.0E+00
67	SWSCF4CTBDBD-124	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	4.2E-07	4.4E-07	2.0E+00
68	SWSCF4CTBDBD-134	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	4.2E-07	4.4E-07	2.0E+00
69	SWSCF4CTBDBD-234	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	4.2E-07	4.4E-07	2.0E+00
70	SWSCF4CTBDBD-123	COOLING TOWER FAN UHS-MFN-001B,D,002B,D FAIL TO START (RUNNING) (CCF)	4.2E-07	4.4E-07	2.0E+00

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Table 19.1-205 (Sheet 1 of 33)
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 2.24 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 sites, 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 DeCordova SES; 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. Those are not to be volatile enough to represent a hazard at the CPNPP Units 3 and 4 sites because of the safe standoff distance or insignificant potential hazards.	1, 3	None	No

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Table 19.1-205 (Sheet 2 of 33)
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 3.5.1.1.2.1. 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. Thus, explosions from transportation routes, nearby industrial facilities, on-site explosion hazards and gas wells cannot affect the plant because of the safe distance (criterion 3) or the insignificance of the potential hazards (criterion 1).			
	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 sguare meters (m2) and 1 m2 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.	1, 3	None	No

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Table 19.1-205 (Sheet 3 of 33)
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 prolpane 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 m2 and 1 m2 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 valpor cloud. and the concentrations at the CPNPP Units 3 and 4 sites are negligible. - 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.			

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Table 19.1-205 (Sheet 4 of 33)
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 ogerated 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 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 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 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 vagor clouds from nearby transportation routes, pipelines, and facilities does not pose a hazard to CPNPP Units 3 and 4. Thus, flammable vapor clouds from transportation routes, nearby industrial facilities, pipelines and gas wells cannot affect the plant because of the safe distance (criterion 3) or insignificance of the potential hazards (criterion 1).			

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Table 19.1-205 (Sheet 5 of 33)
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 control room habitability are discussed in Section 6.4. - 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	1	None	No
	Toxic Chemicals	2.2.3.1.3 6.4.4.2	hazardous material release is higher than the 1.0E-6 screening frequency of Regulatory Guide 1.78. Therefore, a more detailed 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 control room habitability analysis in Section 6.4.	1	None	No

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Table 19.1-205 (Sheet 6 of 33)
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 33)
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 awall from any potential accident location. The nuclear island is situated sufficiently clear of trees and brush. The distance exceeds the minimum fuel modification area reguirements 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 33)
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 . 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. Thus, fire and smoke from accidents at nearby facilities and transportation routes, forest or brush fires, and on-site fuel storage facilities can not affect the plant because of the safe distance from (criterion 3) or the insignificance of the potential hazards (criterion 1).			
	Collision with Intake Structure	2.2.3.1.5	The ESWS and the CWS draw make up 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 (criterion 1).	1	None	No

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Table 19.1-205 (Sheet 9 of 33)
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.
	Liguid 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. 696.00 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. Thus, liquid spills cannot affect the plant because no potential for it to be drawn into the makeup water system. (criterion 1).	1, 3	None	No

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Table 19.1-205 (Sheet 10 of 33)
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	Thus, the probability of aircraft-related hazards for CPNPP Units 3 and 4 is less than 10-7 per year (criterion 2). 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 x N x A/w* Where *C* = In-flight crash rate per mile for aircraft using the airway ($4x10^{-10}$) *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 (0.0907) 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 x A/w)* = 19,300 operations per year	2	$<10^{-7}$	No

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Table 19.1-205 (Sheet 11 of 33)
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 19,300 operations per year. Thus the probability of aircraft-related hazards for CPNPP Units 3 and 4 is less than 10^{-7} per year (criterion 2).			
	Site Proximity Missle	3.5.1.5	Externally initiated missiles considered for design are based on tornado 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 3).	3	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 (Criterion 3). The probability of 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, $P4 = P1 \times P2 \times P3$, 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 $P4 = P1 \times P2 \times P3$ is therefore maintained as less than 10^{-7} per year (criterion 2).	2, 3	$<10^{-7}$	No

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Table 19.1-205 (Sheet 12 of 33) 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 trogical 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 13 of 33)
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 17.7 yr (annual frequency of 5.7×10^{-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. This figure gives a return period of 1000 years for a wind speed of 175 knots (201 mph). 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 14 of 33) 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 15 of 33)
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. 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 = $6.4x10^{-3}$ per yr of a hurricane striking the CPNPP site. As shown above, the probability of a major hurricane striking the Texas coast is small ($5.7x10^{-2}$ per year) and the probability of a major hurricane passing within 50 miles of the CPNPP site is also small ($6.4x10^{-3}$ per yr). Even if a major hurricane is assumed to strike the CPNPP site, the maximum wind speed would be 61 mph based on the maximum possible hurricane landfall wind speed. 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 90 mph is exceedingly small (criterion 1).			

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Table 19.1-205 (Sheet 16 of 33)
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 y	No screening	Close to 10^{-7}	Yes (Section 19.1.5)

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Table 19.1-205 (Sheet 17 of 33)
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 / 133 / 139 10-6 / 171 / 177 10-7 / 205 / 212 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)			

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Table 19.1-205 (Sheet 18 of 33)
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.			
	Thunder-storms	2.3.1.2.4	Thunderstorms, from which damaging local weather can develop (tornadoes, hail, high winds, and flooding), occur about eight 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 monthly and regional distributions of thunderstorms and high wind events are displayed in Table 2.3-211. Thus, thunder storms cannot affect the plant because of the insignificance of the potential hazards (criterion 1) and the impact is less than hurricanes or tornadoes (criterion 4).	1, 4	Not determined	No
	Lightnings	2.3.1.2.5	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/sg 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. The lightning cannot affect the plant because of the insignificance of potential hazards (criterion 1) , and the impact is less than that of hurricanes and tornadoes (criterion 4).	1, 4	None	No

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Table 19.1-205 (Sheet 19 of 33)
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.
	Hails	2.3.1.2.6	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 monthly and seasonal breakdown of large-hail occurrences (3/4 in diameter or larger) for the area around the CPNPP site is given in Table 2.3-212. Hail cannot affect the plant because of the insignificance of the potential hazard (criterion 1). Also, the impact is less than from hurricanes or tornadoes (criterion 4).	1, 4	None	No
	Air Pollution Potential	2.3.1.2.7	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 ([PM10, particulate matter less than 10 micron], [PM2.5, particulate matter less than 2.5 micron]), ozone, and sulfur oxides.	1, 4	None	No

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Table 19.1-205 (Sheet 20 of 33)
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), and is less severe than the impact from toxic chemicals (criterion 4).			
	Precipitation	2.3.1.2.8 2.3.2.1.5	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 21 of 33)
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: Annual Average Maximum 24-hr Snowfall (in) and Yr Snowfall (in) 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/ft2 • Ice Load =5.06 in * 5.20 lb/ft2/in =26.1 lb/ft2			

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Table 19.1-205 (Sheet 22 of 33)
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, precipitation cannot affect the plant because of the insignificant potential hazard (criterion 1).			
	Dust Storms	2.3.1.2.9	Blowing dust or sand may occur occasionally in West Texas where strong winds are more freguent and vegetation is sparse. While blowing dust or sand may reduce visibility to less than five mi over an area of thousands of sg 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. Thus, dust storms cannot affect the plant because of the insignificant potential hazard (criterion 1).	1	None	No

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Table 19.1-205 (Sheet 23 of 33)
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	Estimated extreme winds (fastest mile) for the general area based on the Frechet distribution are: Return Period (yiear) — Wind Speed (mi per hr) 2 — 51 10 — 61 50 — 71 100 — 76 Fastest mile winds are sustained winds, normalized to 30 ft aboveground and include all meteorological ghenomena excegt tornadoes.	1, 4	None	No

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Table 19.1-205 (Sheet 24 of 33)
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. Site-specific structures, systems, and components (SSCs) are designed using the site-specific basic wind speed of 90 mph, or higher. Therefore, the maximum wind speed by extreme winds is not greater than the F-scale intensity F1 of tornadoes for CPNPP. Also 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. Although only loss of offsite power is the hazardous potential by extreme winds, it is considered as the loss of offsite power (LOOP) event for internal event PRA as weather-related LOOP. Thus, extreme winds are insignificant potential hazards (criteria 1 and 4).			

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Table 19.1-205 (Sheet 25 of 33) 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 through 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 (criterion 1).	1	None	No

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Table 19.1-205 (Sheet 26 of 33)
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	The maximum flood level at CPNPP Units 3 and 4 is elevation 793.66 ft msl. This elevation would result from a probable maximum precipitation (PMP) on the Squaw Creek watershed. Coincident wind waves would create maximum waves of 4.59 ft (trough to crest), resulting in a maximum flood elevation of 810.64 ft msl. CPNPP Units 3 and 4 safety-related plant elevation is 822 ft msl, providing more than 11 ft of freeboard under the worst potential flood considerations. 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, 2	$< 10^{-7}$	No

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Table 19.1-205 (Sheet 27 of 33)
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 The overall frequency of a 6-hour, 25-inch PMP event for the U.S. is determined by dividing the number of events by the duration of the historical record. The areal frequency of a PMP event is calculated by dividing the U.S. PMP frequency by this total area over which a PMP could occur and multiplying this amount by the area of a PMP, 10 mi^2. Thus the PMP frequency is: $f_{PMP} = \left(\frac{No.\ of\ PMP\ Events}{Duration\ of\ Historical\ \mathrm{Record}}\right)\left(\frac{A_{PMP\ Event}}{A_{Total\ PMP\ Region}}\right)$ $= \left(\frac{3}{177\ yrs}\right)\left(\frac{10\ mi^2}{1{,}211{,}967\ mi^2}\right)$ $= 1.4\times10^{-7}/yr$			

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Table 19.1-205 (Sheet 28 of 33)
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.
			Therefore, the frequency of a PMP of 25 inches over a 10 square mile is estimated to be 1.4 X 10^{-7} per year. This is a conservative estimate of the frequency of the PMP that results in a PMF for CNNPP Units 3 and 4 because additional periods of significant rainfall must also occur in close temporal proximity to the 25-inch 6-hour rainfall event. Given the calculated PMF is not projected to reach the safety-related elevation of the plant (criterion 1) and the estimated PMP and PMF frequency of 1. 4 X 10^{-7}/year, the frequency of a flooding event that would reach the safety-related elevation of the plant is projected to be well below 10^{-7} per year (criterion 2).			
	Probable Maximum Flood	2.4.3	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 793.66 ft msl. The CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft msl. 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 810.64 ft msl. The top elevation of the retaining wall is 795 ft msl. The CPNPP Units 3 and 4 safety-related structures are located at elevation 822 ft msl 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) and the frequency of the PMP is less than 10^{-7} per year (criterion 2).	1, 2	< 10^{-7}	No

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Table 19.1-205 (Sheet 29 of 33)
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 retaining wall is located approximately 555 ft. northeast from the center point of CPNPP Unit 3 on the slopes of the Squaw Creek Reservoir. Above the retaining wall, a 2:1 (horizontal to vertical) slope continues up to elevation 820 ft. The coincident wind wave activity analysis result is based on the run up on a continuous vertical wall. Comparative analysis for run up on adjacent slopes concludes it is conservative to assume that run up above the top elevation of the retaining wall rises vertically, because run up evaluated for the 2:1 slope would result in a lower elevation. It is assumed that the PMF with coincident wind wave activity elevation of 810.64 ft is applicable to the entire rim of the Squaw Creek Reservoir. The estimated frequency of a PMF capable of reaching the plant grade elevation is estimated to be less than 10-7 per year. Consideration of the maximum coincident wind wave activity along with the PMF would tend to lower the overall frequency.			

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Table 19.1-205 (Sheet 30 of 33)
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	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 760.71 ft. CPNPP Units 3 and 4 safety-related facilities are located at elevation 822 ft. 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 affected by flooding due to dam failures (criteria 1 and 3).	1, 3	None	No

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Table 19.1-205 (Sheet 31 of 33)
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 msl. 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 msl. 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 32 of 33)
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 msl. The SCR spillway elevation is 775 ft msl. The maximum water surface elevation during a probable maximum flood event is at 793.66 ft msl, which is more than 28 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	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 (criterion 3).	3	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.	3	None	No

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Table 19.1-205 (Sheet 33 of 33)
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 is part of the ESWS. Each unit's ESWS consists of four wet mechanical draft cooling towers, each providing 50 percent cooling capacity. Therefore, channel diversion can not adversely affect CPNPP Units 3 and 4 safety-related structures or systems (criterion 3).			
	Low Water	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).	3	None	No
	Groundwater	2.4.12	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. Thus, ground water cannot affect the plant because of its location (criterion 3).	3	None	No

NOTES

(1) Screening criteria categories
"1" Lower damage potential than a design basis event
"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 threat or to provide an adequate response

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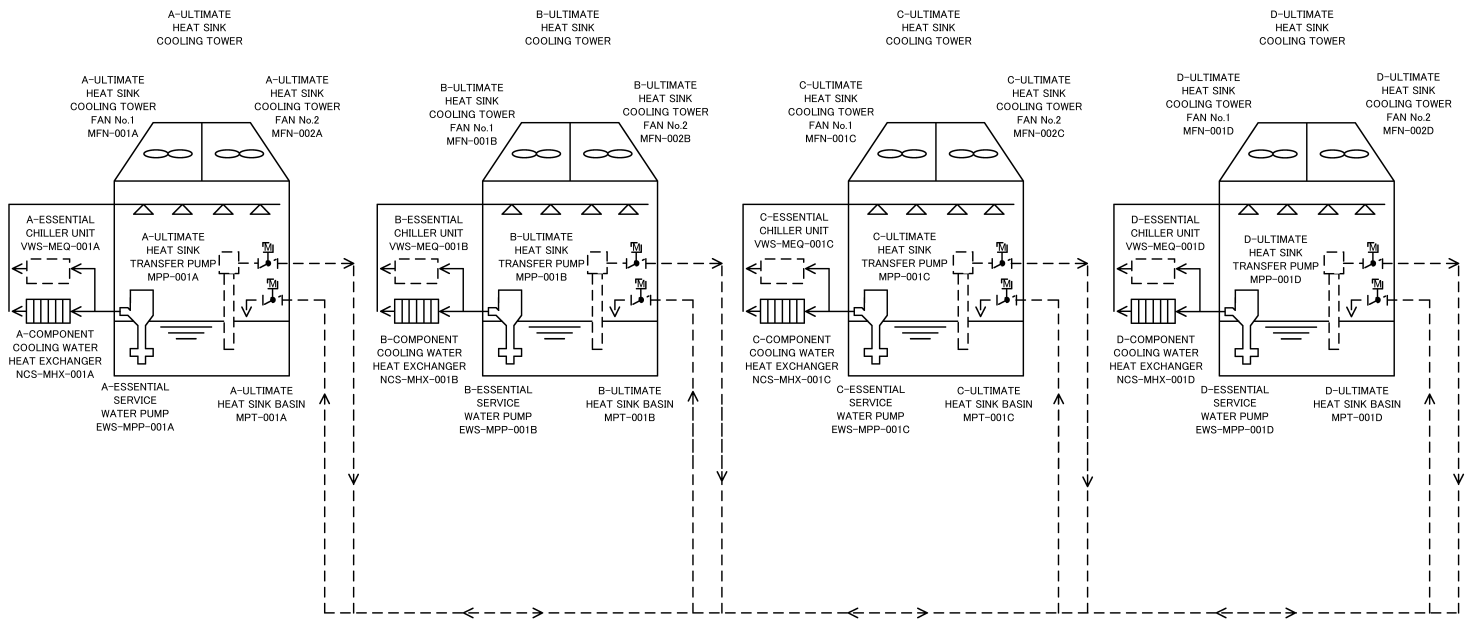
Table 19.1-206

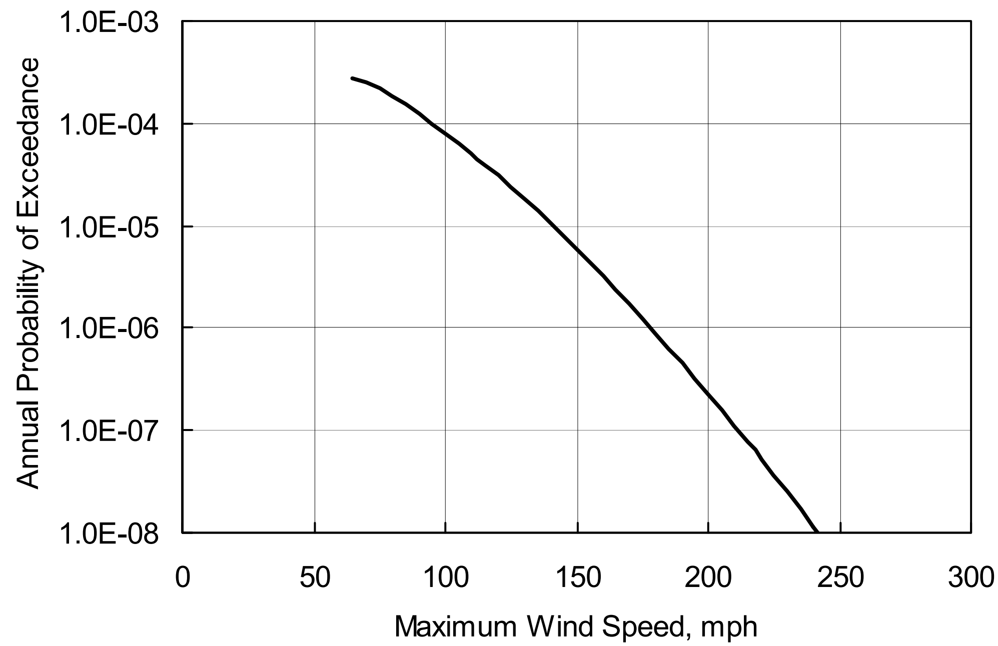
Site-specific Key Assumptions

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
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-103959 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
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 is nearby or source of high wind is forecast for the immediate area.	FSAR 13.5

REMARK:
SYSTEM NAME (UHS) OF THE ULTIMATE HEAT SINK SYSTEM IS OMITTED FROM THE COMPONENT ID.



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.

19.2.5 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 $305k.

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 less than $305k 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. Even with a conservative 3 percent discount rate, the valuation of the averted risk is less than $787k.

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
1	Provide additional dc battery capacity.	$2,000k		$122k	$315k
2	Provide an additional gas turbine generator.	$10,000k		$122k	$315k
3	Install an additional, buried off-site power source.	$10,000k		$125k	$323k
4	Provide an additional high-pressure injection pump with independent diesel.	$1,000k		$159k	$409k
5	Add a service water pump.	$5,900k		$76k	$197k
6	Install an independent reactor coolant pump seal injection system with dedicated diesel.	$3,800k	$305k	$143k	$370k
7	Install an additional component cooling water pump.	$1,500k		$76k	$197k
8	Add a motor-driven feed-water pump.	$2,000k		$107k	$275k
9	Install a filtered containment vent to remove decay heat.	$3,000k		$183k	$471k
10	Install a redundant containment spray system.	$870k		$14k	$37k

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 Subsection 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, 19.1.5.2.2, 19.1.5.3.2, 19.1.6.2, 19.1.7.1, 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.

19.3(5) *Deleted from the DCD.*

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 and Table 19.1-119R.

APPENDIX 19A

US-APWR BEYOND DESIGN BASIS AIRCRAFT IMPACT ASSESSMENT

TABLE OF CONTENTS

19A US-APWR BEYOND DESIGN BASIS AIRCRAFT IMPACT ASSESSMENT

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