

**Comanche Peak Nuclear Power Plant, Units 3 & 4
COL Application
Part 2, FSAR**

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

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UHS

ultimate heat sink

WOG

Westuinginghouse Owners Group

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**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.

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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

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

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

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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

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. The fluid system of ESWS is the same as the standard US-APWR design except that the essential service water pump (ESWP) motor is air-cooled. 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.

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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 in an 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 overflow protection from overflowing 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, the basins can be effective in removing decay heat more than 24 hours assuming nominal conditions but the hottest day of the year. This can be achieved by one fan per tower operating.
- 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 reliable not to significantly degrade the unavailability of ESWP.

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 19.1-20, increases from 2.3E-05/reactor-year (RY) to 2.4E-05/R Y 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 importance, cutsets, and dominant sequences from that documented for the standard US-APWR design is considered negligible and the results described below are considered sufficient and applicable.

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19.1.4.2.2 Results from the Level 2 PRA for Operations at Power

CP 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

CP COL 19.3(4) Replace the last two 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 ANSI/ANS-58.21-2007, taking into consideration the features of advanced light water reactors.

The assessment of the other external events is provided below:

High Winds and Tornadoes

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

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NRC Regulatory Guide 1.76. The tornado hazard methodology developed in NUREG/CR-4461 fully meets the requirements of ANSI/ANS 58.21-2007 (Reference 19.1-8).

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 suppression water tank and associated piping of the fire suppression system
- CTW for the non-essential chilled water system and associated pipings
- Selector circuit and breakers of the alternative ac power supply system
- Permanent buses of the non-safety power system
- Main steam system downstream of the main steam isolation valves
- Main feedwater system upstream of the main feedwater isolation valves

Structure, system, and components (SSCs) will be designed using the site-specific basic wind speed of 90 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

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

- Plant switchyard
- Fire suppression 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 has a potential effect on the selector circuit of the non-safety power source, which is located inside the T/B. When this selector circuit is damaged or accessibility to this equipment is prevented, the non-safety gas turbine generators cannot supply power to the safety bus. These non-safety gas turbine generators are located in the power source building, which is a seismic category I structure designed against the design basis tornado. In this analysis, the following systems are assumed to be damaged by tornado strikes resulting in failure of the T/B:

- Plant switchyard
- Fire suppression system
- Non-essential chilled water system
- Non-safety electric power system
- Alternative 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 \geq 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

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

Based on the results of the plant vulnerability analysis and the discussion above, tornado induced accident scenarios were categorized into four 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:

- 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. 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. Reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) occurs and eventually the core is damaged. The CDF for this scenario is 2.2E-08/R.Y.

- 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/R.Y.

The total CDF caused by a tornado strike is less than 6E-08/R.Y. Tornado induced CDF is one order of magnitude lower than the total CDF for internal events and internal flood and internal fire events.

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 ANSI/ANS-58.21-2007 (Reference 19.1-8).

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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 ANSI/ANS-58.21-2007 (Reference 19.1-8).

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 Description of the Seismic Risk Evaluation

CP COL 19.3(5) Replace the description of the bullet item "Fragility analysis" with the following.

Seismic fragility will be re-evaluated considering the site-specific designs before the first fuel load. Seismic fragilities of the structures are developed using the methodology in Reference 19.1-36.

19.1.5.2.2 Results from the Internal Fires Risk Evaluation

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CP 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

CP 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

CP 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.

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19.1.7.3 PRA Input to the Reactor Oversight Process

STD COL 19.3(3) Replace the content of DCD Subsection 19.1.7.3 with the following.

PRA input is provided to evaluate mitigating systems performance indicators and to develop the significant determination process. As part of the reactor oversight process, mitigating systems performance indicators will be evaluated based on the parameters and methodologies described in NEI 99-02 (Reference 19.1-202). PRA inputs to this process are as described in Appendix G of NEI 99-02. The significance determination process will use risk insights, where appropriate, to determine the safety significance of inspection findings.

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.

Luminant will update PRA and severe accident (SA) evaluation considering the site-specific design before the first fuel load, and the obtained PRA insights will be provided as required to implement the RMTS and SFCP.

19.1.9 References

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 *Regulatory Assessment Performance Indicator Guide*, NEI 99-02, Rev. 5, Nuclear Energy Institute, Washington DC, July 2007.
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CP COL 19.3(4)

**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
	230>	Beyond Design Base	2.5E-08	2.5E-08

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CP COL 19.3(4)

**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.

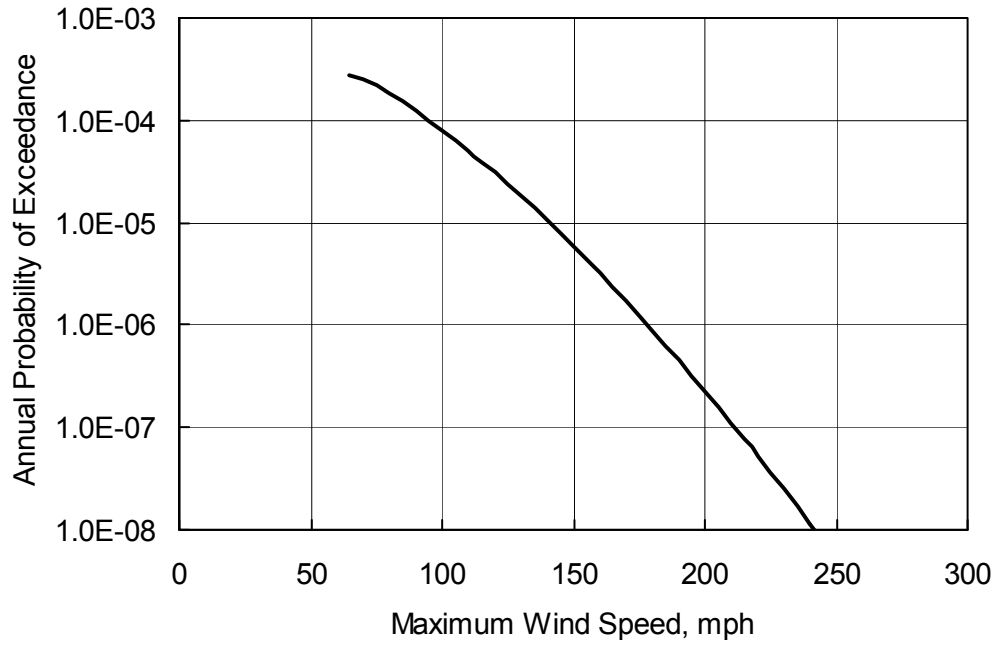
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CP COL 19.3(4)

**Table 19.1-203
Tornado Accident Scenarios**

Wind Speed	Assumed Impact on Plant	Frequency (/yr)	CCDP	CDF (/RY)
F1 scale	Loss of Offsite Power	1.0E-04	6.0E-05	6.0E-09
F2 scale	Loss of Offsite Power with - loss of alternate CCW	3.7E-05	1.5E-04	5.0E-9
F3 scale - 230 mph	Loss of Offsite Power with - loss of alternate CCW, and - loss of alternate ac power supply	1.4E-05	1.6E-03	2.2E-08
230 mph>	Failure of safety related systems Assumed guaranteed core damage	2.5E-08	1	2.5E-08

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CP COL 19.3(4)

Figure 19.1-201

**Point Estimate Probability of Tornado Exceeding
Maximum Wind Speed at the Comanche Peak Site**

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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

CP 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 based on such as the severe accident management guidance (SAMG) prepared by Westinghouse Owners Group (WOG). Important operator actions will be included in operating procedures, and training procedures will also be developed as part of the accident management program. Training for operators will be completed prior to the first fuel load.

19.2.6.1 Introduction

CP 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 the PRA, Subsection 19.1, and 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

CP 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:

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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 CPNPP 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.)

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19.2.6.2 Estimate of Risk for Design

CP 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 CPNPP site.

CP COL 19.3(4) Replace after the second sentence of the third 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

CP 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.

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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.

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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

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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*

This COL item is addressed in Subsection 19.1.7.6.

19.3(2) *Deleted from the DCD.*

STD COL 19.3(3) **19.3(3)** *PRA input to a reactor oversight process*

This COL item is addressed in Subsection 19.1.7.3.

CP 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, 19.2.6.1, 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 and 19.2-9R, and Figure 19.1-201.

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

This COL item is addressed in Subsection 19.1.5.1.1.

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

This COL item is addressed in Subsection 19.2.5.