CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF FIGURES

CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

19.1 INTRODUCTION

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

19.2 INTERNAL INITIATING EVENTS

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

19.3 MODELING OF SPECIAL INITIATORS

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

19.4 EVENT TREE MODELS

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

19.5 SUPPORT SYSTEMS

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

19.6 SUCCESS CRITERIA ANALYSIS

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

19.7 FAULT TREE GUIDELINES

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

19.8 PASSIVE CORE COOLING SYSTEM - PASSIVE RESIDUAL HEAT REMOVAL

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

19.9 PASSIVE CORE COOLING SYSTEM - CORE MAKEUP TANKS

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

19.10 PASSIVE CORE COOLING SYSTEM - ACCUMULATOR

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

19.11 PASSIVE CORE COOLING SYSTEM - AUTOMATIC DEPRESSURIZATION SYSTEM

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

19.12 PASSIVE CORE COOLING SYSTEM - IN-CONTAINMENT REFUELING WATER STORAGE TANK

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

19.13 PASSIVE CONTAINMENT COOLING

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

19.14 MAIN AND STARTUP FEEDWATER SYSTEM

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

19.15 CHEMICAL AND VOLUME CONTROL SYSTEM

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

19.16 CONTAINMENT HYDROGEN CONTROL SYSTEM

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

19.17 NORMAL RESIDUAL HEAT REMOVAL SYSTEM

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

19.18 COMPONENT COOLING WATER SYSTEM

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

19.19 SERVICE WATER SYSTEM

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

19.20 CENTRAL CHILLED WATER SYSTEM

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

19.21 AC POWER SYSTEM

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

19.22 CLASS 1E DC & UPS SYSTEM

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

19.23 NON-CLASS 1E DC & UPS SYSTEM

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

19.24 CONTAINMENT ISOLATION

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

19.25 COMPRESSED AND INSTRUMENT AIR SYSTEM

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

19.26 PROTECTION AND SAFETY MONITORING SYSTEM

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

19.27 DIVERSE ACTUATION SYSTEM

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

19.28 PLANT CONTROL SYSTEM

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

19.29 COMMON CAUSE ANALYSIS

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

19.30 HUMAN RELIABILITY ANALYSIS

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

19.31 OTHER EVENT TREE NODE PROBABILITIES

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

19.32 DATA ANALYSIS AND MASTER DATA BANK

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

19.33 FAULT TREE AND CORE DAMAGE QUANTIFICATION

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

19.34 SEVERE ACCIDENT PHENOMENA TREATMENT

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

19.35 CONTAINMENT EVENT TREE ANALYSIS

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

19.36 REACTOR COOLANT SYSTEM DEPRESSURIZATION

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

19.37 CONTAINMENT ISOLATION

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

19.38 REACTOR VESSEL REFLOODING

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

19.39 IN-VESSEL RETENTION OF MOLTEN CORE DEBRIS

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

19.40 PASSIVE CONTAINMENT COOLING

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

19.41 HYDROGEN MIXING AND COMBUSTION ANALYSIS

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

19.41.7 DIFFUSION FLAME ANALYSIS

Revise the last two paragraphs of DCD Subsection 19.41.7, Diffusion Flame Analysis to read as follows:

WLS DEP 6.2-1 In the event that ADS stage 4 fails to adequately direct hydrogen away from confined compartments, the compartment vents are designed to release the hydrogen at locations where it burns, but does not challenge the containment shell integrity.

Vents from the PXS and CVS compartments to the CMT room are located away from the containment shell and containment penetrations. Access hatches to the subcompartments that are near the containment shell are covered and secured closed such that they will not open as a result of a pipe break inside the compartment. Therefore, hydrogen releases to the CMT room from the subcompartments have been shown to not challenge the containment integrity.

19.42 CONDITIONAL CONTAINMENT FAILURE PROBABILITY DISTRIBUTION

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

19.43 RELEASE FREQUENCY QUANTIFICATION

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

19.44 MAAP4.0 CODE DESCRIPTION AND AP1000 MODELING

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

19.45 FISSION PRODUCT SOURCE TERMS

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

19.46 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.47 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.48 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.49 OFFSITE DOSE EVALUATION

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

19.50 IMPORTANCE AND SENSITIVITY ANALYSIS

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

19.51 UNCERTAINTY ANALYSIS

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

19.52 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.53 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.54 LOW POWER AND SHUTDOWN PRA ASSESSMENT

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

19.55 SEISMIC MARGIN ANALYSIS

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

19.55.6.3 Site Specific Seismic Margin Analysis

WLS COL 19.59.10-6

WLS DEP 2.0-1

Discussions regarding design foundation response spectra are presented in Subsection 3.7.1.1.1. The Lee site-specific seismic demand, characterized by the ground motion response spectrum (GMRS) (applicable to Unit 2) and foundation input response spectra (FIRS) A1 (applicable to Unit 1) are not enveloped by the AP1000 Tier 1 criteria for SSE, which combines both the CSDRS and Hard Rock High Frequency (HRHF) spectra. A site-specific analysis of the AP1000 was performed, similar to the analysis described in AP1000 DCD Appendix 3I, to demonstrate that these exceedances are within the seismic design margin of the AP1000 certified design and will not adversely affect the structures, systems, or components of the plant. Subsection 3.7.2.15 describes the confirmatory site-specific analyses of the nuclear island that demonstrates compliance with the AP1000 DCD. As part of this site-specific analysis, it has been confirmed that the high confidence of low probability of failure (HCLPF) values presented in Chapter 19 of the AP1000 DCD do not have to be adjusted for the Lee Nuclear Station, since the plant design is controlled by the CSDRS. It is therefore concluded that the Seismic Margin Assessment described in DCD Section 19.55 is applicable to the nuclear island of the Lee Nuclear Station.

WLS COL 19.59.10-6

The potential for relative displacement is described in Subsection 2.5.3.8, which addresses potential of permanent ground deformation (tectonic and non-tectonic).

- The potential for tectonic deformation at the site is negligible.
- The potential for non-tectonic surface deformation, including RIS, within the site area is negligible.

There is no information suggesting the potential for non-tectonic surface deformation within the site area.

Foundation bearing capacity for the nuclear island is described in Subsection 2.5.4.10.1.1. The Lee site specific average allowable bearing capacity for static and dynamic conditions, including appropriate safety factors, exceeds the DCD requirements by a factor of significantly greater than 1.67, the ratio of the Review Level Earthquake (RLE) to the SSE.

Similarly, the foundation bearing capacity for the non-safety related buildings adjacent to the nuclear island is described in Subsection 2.5.4.10.1.2. Tables 2.5.4-228 and 2.5.4-229 demonstrate that the Lee site-specific allowable bearing capacity for adjacent structures founded on granular fill also exceeds the

AP1000 standard design requirement by a factor of significantly greater than 1.67, the ratio of the RLE to the SSE.

As discussed in Subsection 2.5.4.8, a liquefaction hazard does not exist that could affect the Category I and seismic Category II portions of plant structures and facilities. Non-seismic portions of the annex buildings for Unit 1 and Unit 2 and the southern end of the non-seismic portion of the turbine building for Unit 2 may be founded on granular fill over saprolite. These areas will be highly resistant to liquefaction and will exhibit low to nil potential for liquefaction and related deformation, and low potential for adverse effects attributed to cyclic strain-softening or pore pressure build-up. These locations are also remote from the nuclear islands and thus have no potential for affecting the nuclear islands.

The stability of slopes is discussed in Subsection 2.5.5. Permanent slopes within a one-quarter mile distance of the nuclear island structures were evaluated to determine the potential hazard to the safety-related structures. It was concluded these slopes do not pose a hazard to these structures.

It is therefore concluded that the Seismic Margin Assessment analysis documented in DCD Section 19.55 is applicable to the Lee site, and that no site-specific conditions exist that have the potential to reduce the HCLPF values calculated for the certified design.

19.56 PRA INTERNAL FLOODING ANALYSIS

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

19.57 INTERNAL FIRE ANALYSIS

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

19.58 WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

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

19.58.3 CONCLUSION

Add the following information at the end of DCD Subsection 19.58.3:

WLS SUP 19.58-1

Table 19.58-201 documents the site-specific external events evaluation that has been performed for Lee Units 1 and 2. This table provides a general explanation of the evaluation and resultant conclusions and provides a reference to applicable sections of the COL where more detailed supporting information (including data used, methods and key assumptions) regarding the specific event is located. Based upon this evaluation, it is concluded that the Lee Units 1 and 2 site is bounded by the High Winds, Floods and Other External Events analysis documented in DCD Section 19.58 and APP-GW-GLR-101 (Reference 201) and no further evaluations are required at the COL application stage.

19.58.4 REFERENCES

201. Westinghouse Electric Company LLC, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations," Document Number APP-GW-GLR-101, Revision 1, October 2007.

WLS SUP 19.58-1

TABLE 19.58-201 (Sheet 1 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
High Winds	EF0 Tornado	A, C	Y	Cherokee County tornado activity is provided in Table 2.3-204 from 1950 through 2005. The event frequency was determined for each tornado category using the point probability method presented in Subsection 2.3.1.2.2. First, the average impacted area was calculated by averaging the area of each category of tornado activity (events with an area of zero value were conservatively disregarded in determining the average area). Second, the tornado frequency was calculated by dividing the total count of tornado events in each category including those with zero area by the measured duration (56 years). Third, the point probability of a tornado impacting a square mile (site area estimated as 1 mi^2) is calculated by taking the product of the average impacted area and the average tornado frequency and dividing by the total area of Cherokee County (392.7 $mi.^2$ per Subsection 2.3.1.2.2). This computation assumes that tornadoes with a zero path length have an area equal to the average area of the category. As shown in Table 2.3-204, there are no recorded category EF5 tornados in the region. A conservative event frequency of <1.00E-03 was assigned for EF5 tornado events, consistent with APP-GW-GLR-101 (Reference 201).	2.13E-05
	EF1 Tornado	A, C	Y		3.42E-05
	EF2 Tornado	A, C	Y		1.25E-05
	EF3 Tornado	A, C	Y		5.17E-05
	EF4 Tornado	A, C	Y		5.43E-05
	EF5 Tornado	D	Y		<1.00E-03

TABLE 19.58-201 (Sheet 2 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				These event frequencies are bounded by the limiting initiating event frequencies given in Table 3.0-1 of APP-GW-GLR-101. Therefore, the safety features of the AP1000 are unaffected and the CDFs given in APP-GW-GLR-101 Table 3.0-1 for these events are applicable to Lee Units 1 and 2.	
	Cat.1 Hurricane	D	Y	Historical data for tropical weather is archived by the National Coastal Services Center, and dates back to 1851. This data was used to analyze the occurrence of tropical weather traveling directly over Cherokee County, or near enough to Cherokee County to have a substantial impact (distance defined as 100 statute miles radius from plant). The resulting storms have been sorted to remove duplicate values. The event frequency is determined by dividing the measured duration (157 years) by the number of occurrences of tropical weather. Figure 6-1B of ASCE/SEI 7-05 shows the basic wind speed for the eastern part of the Gulf of Mexico, including the state of South Carolina. Lee is located in the northwest part of the state beyond the 90 mph contour. Thus, it is concluded that Lee is not located in a Hurricane Prone Region. There were no recorded events for Category 3, 4, or 5 hurricanes. However, a conservative event frequency of 1.00E-02 was assigned for these events, consistent with APP-GW-GLR-101 for Category 4 and 5 hurricanes (Reference 201).	1.27E-02
	Cat.2 Hurricane	A, C	Y		1.27E-02
	Cat.3 Hurricane	D	Y		1.00E-02
	Cat.4 Hurricane	D	Y		1.00E-02
	Cat.5 Hurricane	D	Y		1.00E-02
	Extratropical Cyclones	D	Y		9.55E-02

TABLE 19.58-201 (Sheet 3 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				These event frequencies are bounded by the limiting initiating event frequencies given in Table 3.0-1 of APP-GW-GLR-101. Therefore, the safety features of the AP1000 are unaffected and the CDFs given in APP-GW-GLR-101 Table 3.0-1 for these events are applicable to Lee Units 1 and 2. Winds below 74 mph (storms) are not considered to have an adverse impact of Lee Units 1 and 2 as the switchyard and non-safety buildings will be designed to function at a higher wind speed (96 mph). Therefore, no additional PRA considerations are required for winds below hurricane force.	
External Flood	External Flood	D	Y	As discussed in Subsection 2.4.2.2, specific analysis of Broad River flood levels resulting from surges, seiches, snowmelt, ice effects, flood-waves from landslides, and tsunamis is not required for the Lee Nuclear Station. As discussed in Subsections 2.4.2.2 and 2.4.2.3, the Probable Maximum Precipitation (PMP) event for the site (local intense precipitation) results in a flood elevation of 592.56 ft. The Lee Nuclear Station safety-related plant elevation is 593 ft. As discussed in Subsection 2.4.4, failure of the on-site reservoirs would not affect the safety-related facilities.	N/A

TABLE 19.58-201 (Sheet 4 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				As discussed in Subsections 2.4.1.2.2.6 and 2.4.4.3, the Probable Maximum Flood (PMF) event on the Make-Up Pond B watershed with the added effects of dam failure and coincident wind wave activity results in a flood elevation of 589.10 ft. The Lee Nuclear Station safety-related plant elevation is 593 ft. This result shows a margin of approximately 4 ft. between the calculated flood elevation and the point where safety-related SSCs could be impacted. As discussed in Subsection 2.4.4.3, the PMF event on the Broad River and inundated Make-Up Pond A, including effects of dam failures and the coincident wind wave activity, results in a flood elevation of 585.36 ft. Thus, the Make-Up Pond B event described above remains the bounding event for external flooding and provides reasonable assurance that the plant has adequate protection from external flooding. As discussed in Subsection 2.4.4.1, the Make-Up Pond C peak dam failure outflow was combined with the maximum historical flow recorded on the Broad River. The resulting combined peak outflow does not exceed the critical dam failure event for the Broad River watershed, and, even if routed to the Lee Nuclear Station without attenuation, the resulting water surface elevation would not exceed the elevation determined from the critical multiple dam failure scenario coincident with the Broad River watershed PMF. Thus, the consequences of the Make-Up Pond C failure event are bounded and would not adversely affect safety related structures.	

TABLE 19.58-201 (Sheet 5 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				The above discussion and results for "External Floods" are consistent with the evaluation presented in Section 4.0 of APP-GW-GLR-101 (Reference 201), which states that the AP1000 is protected against floods up to the 100 ft level (593 ft msl for Lee Nuclear Station). Therefore, it is concluded that this event frequency is bounded by the CDF of 5.85E-15 events per year given in APP-GW-GLR-101, Section 4.0 and the safety features of the AP1000 are unaffected.	
Transportation and Nearby Facility Accidents	Aviation (commercial/ general/ military)	A, B	Y	As discussed in Subsection 3.5.1.6, a calculation performed in accordance with the guidelines of Standard Review Plan (SRP) Section 3.5.1.6, determined the general aviation probability of aircraft accidents that hit safety related structures is less than 1.8E-7 per year. Note, the calculated event frequency is based entirely on the general aviation crash rate, including use of low altitude Airway V54. This event frequency is bounded by the limiting value of 1.21E-6 events/year for small aircraft in APP-GW-GLR-I01. As discussed in Subsection 3.5.1.6, no airports having more than 500 D^2 movements per year are located within 10 miles of the site, and no airports beyond 10 miles of the site have more than 1000 D^2 movements per year. Thus, the aircraft hazard probability does not need to be calculated because it is considered to be less than an order of magnitude of 1.0E-7 per year.	1.8E-07 (general aviation) <1.0E-7 (commercial aircraft)

TABLE 19.58-201 (Sheet 6 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				Subsection 2.2.2.7.2 identifies two high altitude airways used by commercial aircraft in the vicinity of the site. The centerline of the nearer, J208, is located approximately 9 miles from the site. Given the total width of the airway is 8 nautical miles (9.2 statute miles), the nearest edge of the airway more than 4 statute miles from the site, which exceeds the screening criterion of 2 statute miles given in SRP 3.5.1.6. Based on the above discussion, the commercial aircraft hazard probability is considered to be less than an order of magnitude of 1.0E-7 per year, and the APP-GW-GLR-101 criterion of an impact frequency of less than 1.0E-7 per year is met for the site. Thus, the commercial aircraft hazard for Lee Units 1 and 2 is bounded by the limiting initiating event frequencies given in subsection 5.1 of APP-GW-GLR-101.	
	Marine (ship/ barge)	E	N	As discussed in Subsection 2.2.3.1.3.1, the nearby Broad River is not navigable by barges and does not transport commercial traffic. No risk-important events related to marine transportation have been identified for Lee Units 1 and 2. This is consistent with the evaluation provided in Subsection APP-GW-GLR-101. Therefore, because no risk-important consequences were identified in the evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operations of Lee Units 1 and 2.	N/A

TABLE 19.58-201 (Sheet 7 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
	Pipeline (gas/ oil)	B	N	As discussed in Subsection 2.2.3.1.1.2, the releases from postulated rupture of the natural gas pipelines and refined petroleum pipelines within 5 miles of Lee do not pose a credible hazard to the site. As discussed in Subsection 2.2.3.1.2, unconfined vapor clouds with delayed ignition were evaluated for various energetic combustible materials, and determined to not result in any significant damage to the plant. Based upon the quantitative consequence evaluations performed, the limiting initiating event is conservatively estimated to be 1.0E-7 events per year. This is consistent with the evaluation provided is Subsection 5.3 of APP-GW-GLR-101. Thus, the pipeline accident hazard for Lee Units 1 and 2 is bounded by the limiting initiating event frequencies given in Subsection 5.3 of APP-GW-GLR-101.	N/A
	Railroad	D	N	As discussed in Subsection 2.2.3.1.1.1, the potential hazard resulting from railroad cars was evaluated using the methodology of RG 1.91. The maximum probable cargo based on RG 1.91 was used along with a conservative TNT equivalency, which resulted in a safe standoff distance that was less than the distance from the nearest approach of a railroad line to the site boundary. As discussed in Subsection 2.2.3.1.2, unconfined vapor clouds with delayed ignition were also evaluated for various energetic combustible materials, and determined to not result in any significant damage to the plant.	N/A

TABLE 19.58-201 (Sheet 8 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				Based upon the quantitative consequence evaluations performed, no risk-important events related to rail transportation have been identified for Lee Units 1 and 2. This is consistent with the evaluation provided in Subsection 5.4 of APP-GW-GLR-101. Therefore, because no risk-important consequences were identified in the evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operation Lee Units 1 and 2.	
	Truck	D	N	As discussed in Subsection 2.2.3.1.1.1, the potential hazard resulting from trucks was evaluated using the methodology of RG 1.91. The maximum probable cargo based on RG 1.91 was used along with a conservative TNT equivalency, which resulted in a safe standoff distance that was less than the distance from the nearest highway to the site boundary. As discussed in Subsection 2.2.3.1.2, unconfined vapor clouds with delayed ignition were also evaluated for various energetic combustible materials, and determined to not result in any significant damage to the plant. Based upon the quantitative consequence evaluations performed, no risk-important events related to truck transportation have been identified for Lee Units 1 and 2. This is consistent with the evaluation provided in Subsection 5.4 of APP-GW-GLR-101. Therefore, because no risk-important consequences were identified in the	N/A

TABLE 19.58-201 (Sheet 9 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
				evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operation of Lee Units 1 and 2.	
	Nearby Facility Accidents	D	Y	Subsection 2.2.3.1.1.3 discusses potential design basis events associated with accidents at nearby facilities. Subsection 2.2.3.2 concludes that the effects of events from these facilities on the safety-related components of the plant are insignificant. Therefore, because no risk-important consequences were identified, the potential for hazards from these sources are minimal and will not adversely affect safe operation of Lee Units 1 & 2.	N/A
Other events	A number of external events beyond those evaluated in DCD Subsection 19.58 and APP-GW-GLR-101 (Reference 201) were evaluated for the Lee site. These events are discussed below.			Based on the evaluations below, these events do not pose a credible threat to the safe operation of the station. Thus, these events are not considered to be risk-important and it can be concluded that the Lee Units 1 and 2 site is within the bounds of the Floods and Other External Events analysis documented in DCD Section 19.58 and APP-GW-GLR-101 (Reference 201).	

TABLE 19.58-201 (Sheet 10 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
	External Fires	D	Y	Subsection 2.2.3.1.4 discusses external fires and concludes that fires originating from accidents at nearby facilities or transportation routes, and brush and forest fires will not endanger the safe operation of the station. Additionally, this subsection states that 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 separation distance from the plant. Therefore, because no risk-important consequences were identified, the potential for hazards from external fires is minimal and will not adversely affect safe operation of Lee Units 1 & 2.	N/A
	Toxic Chemical Releases	D	Y	Based on the evaluations provided in Subsections 2.2.3.1.3 and 6.4.4.2, release of toxic chemicals from stationary industrial sources and mobile sources in the vicinity of Lee does not pose a credible threat to the control room operators. Based the quantitative consequence evaluations performed, toxic chemical release events at the Lee Nuclear Station are not considered to be risk-important. Therefore, because no risk-important consequences were identified in the evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operation of Lee Units 1 and 2.	N/A

TABLE 19.58-201 (Sheet 11 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
	On-Site Chemical Storage	D	Y	FSAR Subsection 2.2.3.1.3.2.1, “Stationary Sources,” states that there are no site-specific sources of airborne hazardous materials stored on the Lee Nuclear Station site in sufficient quantity to affect control room habitability. Based the quantitative consequence evaluations performed on-site chemical storage at the Lee Nuclear Station is not considered to be risk-important. Therefore, because no risk-important consequences were identified in the evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operation of Lee Units 1 and 2.	N/A

TABLE 19.58-201 (Sheet 12 of 12)
EXTERNAL EVENT FREQUENCIES FOR LEE

Category	Event	Evaluation Criteria (See Notes)	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency (Events/yr)
	Major Depots and Storage Areas Releases	D	Y	Based on the discussion in Subsection 2.2.2.2.4, none of the listed mines poses a credible threat to the site, because none of the mines uses explosives. Also, there are no military facilities within 5 mi, so no further evaluation is required. Based upon the quantitative consequence evaluations performed, no risk-important events related to major depots and storage areas releases have been identified for Lee Units 1 and 2. Therefore, because no risk-important consequences were identified in the evaluation, the potential for hazards from these sources are minimal and will not adversely affect safe operation Lee Units 1 and 2.	N/A

Notes:

1. All events that are physically possible are considered to be "applicable" and are discussed. Those events that are physically not possible are considered not applicable to the site.

Evaluation Criteria

A: The initiating event frequency (IEF) is less than the IEF in DCD Tier 2 Section 19.58 or Table 19.58-3 for the event.

B: IEF is less than 1.0E-07.

C: Core damage frequency (CDF) is less than 1.0E-08.

D: A specific event frequency for this event has not been determined. A deterministic quantitative consequence evaluation has been performed that has demonstrated that the event does not adversely impact the safe operation of Lee 1 and 2. Additional details are provided in the "Explanation of Applicability Evaluation" with references to the applicable FSAR Subsections.

E: The event is not physically possible for the site.

More than one screening note may apply to a given type of event.

19.59 PRA RESULTS AND INSIGHTS

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

19.59.10.5 Combined License Information

STD COL 19.59.10-1
STD COL 19.59.10-6

A review of the differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis will be completed prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design. A comparison of the as-built SSC high confidence, low probability of failures (HCLPFs) to those assumed in the AP1000 seismic margin evaluation will be performed prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis will be evaluated to determine if vulnerabilities have been introduced.

The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:

1. Specific minimum seismic requirements consistent with those used to define the AP1000 DCD Table 19.55-1 HCLPF values.

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of failure is expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the seismic margin insights of the site-specific PRA.The range of frequency response that is required for the equipment with its structural support is defined.

2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.

STD COL 19.59.10-2

A review of the differences between the as-built plant and the design used as the basis for the AP1000 PRA and DCD Table 19.59-18 will be completed prior to fuel load. The plant-specific PRA-based insight differences will be evaluated and the plant-specific PRA model modified as necessary to account for plant-specific design and design changes or departures from the design certification PRA.

As discussed in Subsection 19.58.3, it has been confirmed that the Winds, Floods and Other External Events analysis documented in DCD Section 19.58 is

applicable to the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Subsection 19.58 of the AP1000 DCD is applicable to this design.

STD COL 19.59.10-3

A review of the differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis will be completed prior to fuel load. Plant-specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes of departures from the certified design.

STD COL 19.59.10-4

The AP1000 Severe Accident Management Guidance (SAMG) from APP-GW-GLR-070, Reference 1 of DCD Section 19.59, is implemented on a site-specific basis. Key elements of the implementation include:

- SAMG based on APP-GW-GLR-070 is provided to Emergency Response Organization (ERO) personnel in assessing plant damage, planning and prioritizing response actions and implementing strategies that delineate actions inside and outside the control room.
- Severe accident management strategies and guidance are interfaced with the Emergency Operating Procedures (EOP's) and Emergency Plan.
- Responsibilities for authorizing and implementing accident management strategies are delineated as part of the Emergency Plan.
- SAMG training is provided for ERO personnel commensurate with their responsibilities defined in the Emergency Plan.

STD COL 19.59.10-5

A thermal lag assessment of the as-built equipment required to mitigate severe accidents (hydrogen igniters and containment penetrations) will be performed to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment will be performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The ability of the as-built equipment to perform during severe accident hydrogen burns will be assessed using the environment enveloping method or the test based thermal analysis method discussed in EPRI NP-4354 (DCD Section 19.59, Reference 3).

STD COL 19.59.10-6

As discussed in Subsection 19.55.6.3, it has been confirmed that the Seismic Margin Analysis (SMA) documented in DCD Section 19.55 is applicable to the Lee site. The site-specific effects (i.e., soil-related failure modes, etc.) have been

WLS COL 19.59.10-6

evaluated and it was concluded that the plant-specific plant-level HCLPF value is equal to or greater than 1.67 times the site-specific GMRS/FIRS A1 peak ground acceleration.

Add the following new information after DCD Subsection 19.59.10.5

STD SUP 19.59-1

19.59.10.6 PRA Configuration Controls

PRA configuration controls contain the following key elements:

- A process for monitoring PRA inputs and collecting new information.
- A process that maintains and updates the PRA to be reasonably consistent with the as-built, as operated plant.
- A process that considers the cumulative impact of pending changes when applying the PRA.
- A process that evaluates the impact of changes on currently implemented risk-informed decisions that have used the PRA.
- A process that maintains configuration control of computer codes used to support PRA quantification.
- A process for upgrading the PRA to meet PRA standards that the NRC has endorsed.
- Documentation of the PRA.

PRA configuration controls are consistent with the regulatory positions on maintenance and upgrades in Regulatory Guide 1.200.

Schedule for Maintenance and Upgrades of the PRA

The PRA update process is a means to reasonably reflect the as designed and as operated plant configurations in the PRA models. The PRA upgrade process includes an upgrade of the PRA plus a general review of the entire PRA model, and as applicable the application of new software that implements a different methodology, implementation of new modeling techniques, as well as a comprehensive documentation effort.

- During construction, the PRA is upgraded prior to fuel load to cover those initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to the scheduled date of the initial fuel load for a Level 1 and Level 2 PRA.
- Prior to license renewal the PRA is upgraded to include all modes of operation.

- During operation, PRA updates are completed as part of the upgrade process at least once every four years.
- A screening process is used to determine whether a PRA update should be performed more frequently based upon the nature of the changes in design or procedures. The screening process considers whether the changes affect the PRA insights. Changes that do not meet the threshold for immediate update are tracked for the next regulatory scheduled update. If the screening process determines that the changes do warrant a PRA update, the update is made as soon as practicable consistent with the required change importance and the applications being used.

PRA upgrades are performed in accordance with 10 CFR 50.71(h).

Process for Maintenance and Upgrades of the PRA

Various information sources are monitored to determine changes or new information that affects the model assumptions or quantification. Plant-specific design, procedure, and operational changes are reviewed for risk impact. Information sources include applicable operating experience, plant modifications, engineering calculation revisions, procedure changes, industry studies, and NRC information.

The PRA upgrade includes initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade.

This PRA maintenance and update incorporates the appropriate new information including significant modeling errors discovered during routine use of the PRA.

Once the PRA model elements requiring change are identified, the PRA computer models are modified and appropriate documents revised. Documentation of modifications to the PRA model include the changes as well as the upgraded portions clearly indicating what has been changed. The impact on the risk insights is clearly indicated.

PRA Quality Assurance

Maintenance and upgrades of the PRA are subject to the following quality assurance provisions:

Procedures identify the qualifications of personnel who perform the maintenance and upgrade of the PRA.

Procedures provide for the control of PRA documentation, including revisions.

For updates of the PRA, procedures provide for independent review, or checking of the calculations and information.

Procedures provide for an independent review of the model after an upgrade is completed. Additionally, after the PRA is upgraded, the PRA is reviewed by outside PRA experts such as industry peer review teams and the comments incorporated to maintain the PRA current with industry practices. Peer review findings are entered into a tracking system. PRA upgrades receive a peer review for those aspects of the PRA that are upgraded.

PRA models and applications are documented in a manner that facilitates peer review as well as future updates and applications of the PRA by describing the processes that were used, and provide details of the assumptions made and their bases. PRA documentation is developed such that traceability and reproducibility is maintained. PRA documentation is maintained in accordance with Regulatory Position 1.3 of Regulatory Guide 1.200.

Procedures provide for appropriate attention or corrective actions if assumptions, analyses, or information used previously are changed or determined to be in error. Potential impacts to the PRA model (i.e., design change notices, calculations, and procedure changes) are tracked. Errors found in the PRA model between periodic updates are tracked using the site tracking system.

PRA-Related Input to Other Programs and Processes

The PRA provides input to various programs and processes, such as the Maintenance Rule implementation, reactor oversight process, the RAP, and the regulatory treatment of non-safety systems program. The use of the PRA in these programs is discussed below, or cross-references to the appropriate FSAR sections are provided.

PRA Input to Design Programs and Processes

The PRA insights identified during the design development are discussed in DCD Subsection 19.59.10.4 and summarized in DCD Table 19.59-18. DCD Section 14.3 summarizes the design material contained in AP1000 that has been incorporated into the Tier 1 information from the PRA. A discussion of the plant features important to reducing risk is provided in DCD Subsection 19.59.9.

PRA Input to the Maintenance Rule Implementation

The PRA is used as an input in determining the safety significance classification and bases of in-scope SSCs. SSCs identified as risk-significant via the Reliability Assurance Program for the design phase (DRAP, Section 17.4) are included within the initial Maintenance Rule scope as high safety significance SSCs.

For risk-significant SSCs identified via DRAP, performance criteria are established, by the Maintenance Rule expert panel using input from the reliability and availability assumptions used in the PRA, to monitor the effectiveness of the maintenance performed on the SSCs.

The Maintenance Rule implementation is discussed in Section 17.6.

PRA Input to the Reactor Oversight Process

The mitigating systems performance indicators (MSPI) are evaluated based on the indicators and methodologies defined in NEI 99-02 (Reference 201).

The Significance Determination Process (SDP) uses risk insights, where appropriate, to determine the safety significance of inspection findings.

PRA Input to the Reliability Assurance Program

The PRA input to the Reliability Assurance Program is discussed in DCD Subsection 19.59.10.1.

PRA Input to the Regulatory Treatment of Nonsafety-Related Systems Programs

The importance of nonsafety-related SSCs in the AP1000 has been evaluated using PRA insights to identify SSCs that are important in protecting the utility's investment and for preventing and mitigating severe accidents. These investment protection systems, structures and components are included in the D-RAP/MR Program (refer to Subsection 17.4), which provides confidence that availability and reliability are designed into the plant and that availability and reliability are maintained throughout plant life through the maintenance rule. Technical Specifications are not required for these SSCs because they do not meet the selection criteria applied to the AP1000 (refer to Subsection 16.1.1).

MOV Program

The MOV Program includes provisions to accommodate the use of risk-informed inservice testing of MOVs (Subsection 3.9.6).

19.59.11 REFERENCES

Add the following text to the end of DCD Subsection 19.59.11:

201. NEI 99-02, Nuclear Energy Institute, "Regulatory Assessment Performance Indicator Guideline," Technical Report NEI 99-02, Revision 5, July 2007.

WLS DEP 6.3-1

TABLE 19.59-201 (Sheet 1 of 2)
AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1e. (cont.)	
Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.96
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1

WLS DEP 6.3-1

TABLE 19.59-201 (Sheet 2 of 2)
AP1000 PRA-BASED INSIGHTS

Insight	Disposition
The PRHR HX, in conjunction with the IRWST, the condensate return features and the PCS, can provide core cooling for greater than 14 days. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1

APPENDIX 19A
THERMAL HYDRAULIC ANALYSIS TO SUPPORT SUCCESS CRITERIA

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

APPENDIX 19B
EX-VESSEL SEVERE ACCIDENT PHENOMENA

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

APPENDIX 19C
ADDITIONAL ASSESSMENT OF AP1000 DESIGN FEATURES

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

APPENDIX 19D
EQUIPMENT SURVIVABILITY ASSESSMENT

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

APPENDIX 19E
SHUTDOWN EVALUATION

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

19E.2.3.2.6 Discussion of Safe Shutdown for AP1000

Replace the DCD Subsection 19E.2.3.2.6 with the following information:

WLS DEP 3.2-1 The functional requirements for the PXS specify that the plant be brought to a safe, stable condition using the PRHR HX for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the PRHR HX in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the PXS, in conjunction with the passive containment cooling system (PCS) and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system as identified in Subsection 7.4.1.1.

The CMTs automatically provide injection to the RCS after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The PXS can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system automatically actuates. However, after the initial plant cooldown following a non-LOCA event, operators assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition, such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the PRHR HX, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

In most sequences the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. For LOCAs and other postulated events, when the core makeup tank level reaches the automatic depressurization actuation setpoint, and other postulated events where the

PRHR HX operation is not extended or is exhausted, ADS may be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the RCS is nearly depressurized. For these conditions, the RCS depressurizes to saturated conditions at about 250°F within 24 hours. The PXS can maintain this safe shutdown condition as identified in Subsection 7.4.1.1.

The primary function of the PXS during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Analysis is provided in Subsection 19E.4.10.2 of this appendix that verifies the ability of the AP1000 passive safety systems to meet the safe shutdown requirements.

19E.2.7.2 Design Features to Address Shutdown Safety

Revise the third paragraph of DCD Subsection 19E.2.7.2 as follows:

WLS DEP 7.3-1 The safety analysis of boron dilution accidents is provided in Chapter 15 and is discussed in subsection 19E.4.5 of this appendix. For dilution events that occur during shutdown, the source-range flux-doubling signal closes the safety-related remotely operated CVS makeup line isolation valves to terminate the event. In addition, the signal is used to isolate the line from the demineralized water system to the makeup pump suction by closing the two safety-related remotely operated valves. The three-way pump suction control valve aligns the makeup pumps to take suction from the boric acid tank and, therefore, stops the dilution.

19E.4.10.2 Shutdown Temperature Evaluation

Replace DCD Subsection 19E.4.10.2 with the following information.

WLS DEP 3.2-1 As discussed in Subsection 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (DCD Reference 15) criteria.

As discussed in DCD Subsection 6.3.3 and DCD Subsection 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater

with a loss of ac power event demonstrates that the passive systems can bring the plant to a stable safe condition following postulated transients. A non-bounding, conservative analysis is represented in DCD Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in DCD Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in DCD Subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to ensure the results were conservative for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The WGOTHIC model provides the time-dependent condensate return rate which was incorporated into the LOFTRAN computer code described in DCD Subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power occurs (offsite and onsite), followed by the reactor trip. The PRHR HX is actuated on the low steam generator narrow range level coincident with low startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on Low cold leg temperature and the CMTs are actuated.

Once actuated, at about 2,700 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 6,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (DCD Figure 19E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 46,700 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from DCD Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced

to 420°F at about 52,900 seconds, while the core average temperature reaches 420°F at about 120,900 seconds (approximately 34 hours).

As discussed in DCD Subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

WLS DEP 6.3-1 As discussed in DCD Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this Section shows the PRHR HX is expected to maintain safe shutdown conditions for greater than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

WLS DEP 3.2-1

TABLE 19E.4.10-201
SEQUENCE OF EVENTS FOLLOWING A LOSS OF AC POWER
FLOW WITH CONDENSATE FROM THE CONTAINMENT SHELL
BEING RETURNED TO THE IRWST

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	60.6
Rods Begin to Drop	62.6
Low Steam Generator Water Level (Wide-Range) Reached	209.5
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	221.5
Low T_{cold} Setpoint Reached	2752
Steam Line Isolation on Low T_{cold} Signal	2764
CMTs Actuated on Low T_{cold} Signal	2764
IRWST Reaches Saturation Temperature	15,900
Heat Extracted by PRHR HX Matches Core Decay Heat	46,700
Cold Leg Temperature Reaches 420°F (loop with PRHR)	52,900
Core Average Temperature Reaches 420°F	120,900

APPENDIX 19F
MALEVOLENT AIRCRAFT IMPACT

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