Levy Nuclear Plant Units 1 and 2
COL Application
Part 2, Final Safety Analysis Report

CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

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

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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.

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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.

Add the following Subsection after DCD Subsection 19.55.6.2:

19.55.6.3 Site-Specific Seismic Margin Analysis

LNP COL 19.59.10-6 The LNP GMRS was developed as the Truncated Soil Column Surface Response (TSCSR) on the uppermost in-situ competent material at elevation 11 m (36 ft.) NAVD88 as described in Subsection 2.5.2.6. Since plant design grade will be established at elevation 15.5 m (51 ft.) NAVD88 by engineered fill above in-situ material as noted in Subsection 2.5.4.5, performance based surface horizontal and vertical response spectra (PBSRS) at the design grade scaled to meet 10 CFR Part 50 Appendix S requirements were developed as described in Subsection 2.5.2.6. Both the LNP scaled GMRS and the scaled PBSRS are enveloped by the AP1000 Certified Seismic Response Spectra as documented in Subsection 2.5.2.6. In addition, LNP site-specific SSI analysis was performed to evaluate the effect of the LNP unique foundation conditions on seismic demand. It was determined that the LNP site-specific seismic floor response spectra (FRS) at the six key locations are enveloped by the AP1000 CSDRS based FRS at the six key locations. In addition, the LNP maximum bearing pressure is less than the CSDRS based maximum bearing pressure of 24 ksf for soft rock sites. For the 24 ksf bearing pressure, the LNP site specific bearing factor of safety is greater than the acceptable factor of safety for static and dynamic loadings (Subsection 2.5.4.10.1.1). The LNP SSI analysis results are documented in Subsection 3.7.1.1.1. Thus, LNP site unique foundation conditions do not lower the High Confidence Low Probability of Failure (HCLPF) values calculated for the certified design.

As shown in Figures 2.5.2-355 and 2.5.2-357, both the CEUS SSC GMRS and the PBSRS are enveloped by the AP1000 CSDRS. As discussed in Subsection 3.7.1.1.2, the CEUS SSC LNP site specific floor response spectra (FRS) at the six key locations are bounded by the CSDRS FRS. In addition, the CEUS SSC LNP site specific nuclear island maximum bearing pressure is less than the 24 ksf design value. Thus, LNP site unique foundation conditions and CEUS SSC ground motions do not lower the High Confidence Low Probability of Failure (HCLPF) values calculated for the certified design.

The soils under the LNP 1 and LNP 2 nuclear islands (NI) foundations will be excavated to rock and backfilled with Roller Compacted Concrete (RCC), as discussed in Subsection 2.5.4.5.3. For the NI, this eliminates any potential site-specific effects such as seismically induced liquefaction settlements, slope stability, foundation failure or relative settlements that would lower the HCLPF values calculated for the certified design. As described in Subsection 2.5.4.8, the LNP site-specific soil conditions also do not affect the nuclear island sliding and overturning stability based on Westinghouse analysis. Thus, LNP site-specific soil conditions do not lower the HCLPF values calculated for the certified design.

As described in Subsection 2.5.4.8, LNP site-specific liquefaction analysis (for PBSRS) was performed for soil beyond the nuclear island perimeter which will be left in place. Based on the liquefaction analysis, it was concluded that liquefiable zones under the LNP 1 and 2 footprints are confined to the northwest corner of the Unit 2 Turbine Building and in isolated random pockets under the remaining LNP 1 and 2 footprints. The LNP earthwork design will incorporate vertical and horizontal drains that will prevent liquefaction in the northwest corner of the Unit 2 Turbine Building and in isolated random pockets under the remaining LNP 1 and 2 footprints. The extent of these horizontal and vertical drains is shown in Figures 2.5.4.8-205 and 2.5.4.8-206. Liquefaction analysis was also performed for 10^{-5} uniform hazard response spectra (UHRS) for soil beyond the nuclear island perimeter which will be left in place as is described in Subsection 2.5.4.8. Based on this liquefaction analysis, it can be concluded that liquefiable zones under the LNP 1 and 2 footprints for 10^{-5} UHRS are confined soil zones where LNP earthwork design will incorporate vertical and horizontal drains that prevent liquefaction (Figures 2.5.4.8-205 and 2.5.4.8-206). The 10^{-5} UHRS is greater than 1.67 times the LNP scaled GMRS and the scaled PBSRS developed using the updated EPRI SOG model, and the GMRS and the PBSRS developed using the CEUS SSC model and modified CAV filter. Thus, liquefaction potential of soil beyond the nuclear island perimeter which will be left in place has the potential to drive the plant level HCLPF; however the soil liquefaction HCLPF exceeds the 1.67*GMRS goal for the plant level HCLPF.

Seismic Category II structures (Annex Building [AB] and the first bay of the Turbine Building [TB]) and nonsafety-related structures (rest of the TB and Radwaste Building [RB]) adjacent to the NI will be supported on drilled shaft foundations. The Seismic Category II/I interaction issues between the adjacent drilled shaft supported structures and the NI have been addressed in Subsections 3.7.2.8.1, 3.7.2.8.2, and 3.7.2.8.3. The probable maximum relative displacements between the NI and the adjacent Turbine, Annex, and Radwaste Buildings' foundation mat for the PBSRS and the 10^{-5} UHRS are less than the 50 mm (2.0 inch) gap between the NI and the adjacent buildings' foundation mats. The 10^{-5} UHRS is greater than 1.67 times higher than the LNP scaled GMRS and the scaled PBSRS developed using the updated EPRI SOG model, and the GMRS and the PBSRS developed using the CEUS SSC model and modified CAV filter. Thus, Seismic Category II/I interaction between the NI and the adjacent buildings has the potential to drive the plant level HCLPF; however the HCLPF for Seismic Category II/I interaction between the NI and the adjacent buildings exceeds the 1.67*GMRS goal for the plant level HCLPF.

The LNP RCC bridging mat is designed to span the postulated (conservative) design basis karst void of 10 ft. The failure of the RCC bridging mat can result in displacement of the AP1000 nuclear island foundation in excess of the maximum 6 in. displacements specified in DCD Tier 1 Table 5.0-1. In the AP1000 PRA-based Seismic Margin Assessment, the RCC bridging mat failure is conservatively assumed to fall within the gross structural collapse event modeled in the hierarchical event tree discussed in DCD Section 19.55. As gross structural collapse is assumed to directly lead to core damage, failure of the RCC bridging mat has the potential to drive the plant level high confidence low probability of failure (HCLPF) value. The HCLPF capacity of the RCC mat was calculated as >0.14g using the conservative deterministic failure margin (CDFM) methodology of Reference 19.55.7-201. The >0.14g HCLPF capacity of the RCC bridging mat exceeds the overall plant HCLPF acceptance criteria of 1.67*scaled GMRS using the updated EPRI SOG model and the 1.67*GMRS developed using the CEUS SSC model and modified CAV filter.

Table 19.55-201 summarizes the HCLPF capacities of the LNP site-specific design features (e.g., RCC bridging mat, potential against soil liquefaction, and Seismic Category II/I interaction between the nuclear island and the adjacent buildings).

Thus, it can be concluded that the Seismic Margin Assessment analysis documented in Section 19.55 is applicable to the LNP site. Exceeding the HCLPF capacities for soil liquefaction and Seismic Category II/I interaction effects of buildings adjacent to the nuclear island will not affect the plant level HCLPF capacity. The RCC bridging mat HCLPF capacity, while potentially driving the plant-level HCLPF, exceeds the plant level HCLPF goal of 1.67*scaled GMRS using the updated EPRI SOG model and the GMRS developed using the CEUS SSC model and modified CAV filter.

19.55.7 REFERENCES

Add the following information at the end of DCD Subsection 19.55.7:

201. EPRI Report No. NP-6041-SL, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin”, Revision 1, August 1991.

Table 19.55-201
HCLPF Capacities for LNP Site Specific Design Features

LNP COL 19.59.10-6

Description	HCLPF Capacity[a]	HCLPF/GMRS[b]	Basis
Soil Liquefaction Potential under Adjacent Buildings	> 0.14g	> 1.67 GMRS	(c)
Seismic II/I Interaction Potential	> 0.14g	> 1.67 GMRS	(d)
RCC bridging mat	>0.14g	>1.67 GMRS	(e)

Notes:

a) LNP scaled Ground Motion Response Spectra (GMRS) peak ground acceleration (PGA) is 0.084g using updated EPRI SOG model (Subsection 2.5.2.6). The GMRS PGA using CEUS SSC model and modified CAV filter is 0.073g (Subsection 2.5.2.7).

b) HCLPF Capacity as a fraction of LNP updated EPRI SOG scaled GMRS PGA.

c) Liquefaction potential of soils under the adjacent buildings was evaluated for the LNP updated EPRI SOG 10^{-5} annual exceedance probability Uniform Hazard Response Spectra (10^{-5} UHRS). The LNP updated EPRI SOG 10^{-5} UHRS is greater than 1.67*scaled GMRS using the updated EPRI SOG model (Subsection 2.5.2.6) and the CEUS SSC GMRS with the modified CAV filter (Subsection 2.5.2.7).

d) Relative displacement between the NI and adjacent buildings for the LNP updated EPRI SOG 10^{-5} UHRS is less than the gap provided. The LNP updated EPRI SOG 10^{-5} UHRS is greater than 1.67*scaled GMRS using the updated EPRI SOG model (Subsection 2.5.2.6) and the CEUS SSC GMRS with the modified CAV filter (Subsection 2.5.2.7).

e) HCLPF capacity calculated using conservative deterministic failure margin method of Reference 19.55.7-201.

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:

LNP SUP 19.58-1 Table 19.58-201 documents the site-specific external events evaluation that has been performed for LNP 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 LNP 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

Add the following information at the end of DCD Subsection 19.58.4:

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

202. NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, February2007.

203. Not Used.

204. ASCE Standard ASCE/SEI 7-05, "Minimum Design Loads for Buildings and Other Structures,"2006.

205. NUREG/CR-6890, Volume 1, “Reevaluation of Station Blackout Risk at Nuclear Power Plants - Analysis of Loss of Offsite Power Events: 1986-2004

Table 19.58-201 (Sheet 1 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
High winds	EF0 Tornado	Notes 1, 3	From the data covering 57 years in FSAR Table 2.3.1-204, the number of each type of tornado as recorded by NOAA for the ten counties (total of 9230 mi2) containing and surrounding the Levy site was identified. For each type of tornado, the event frequency was estimated from the product of the number of tornadoes divided by the number of years and the expected area of a tornado from Table 2-14 of NUREG/CR-4461 divided by the total area of the counties.	7.72E-06
	EF1 Tornado	Notes 1, 3		3.56E-05
	EF2 Tornado	Notes 1, 3		5.08E-05
	EF3 Tornado	Notes 1, 3		4.13E-05
	EF4 Tornado	Notes 1, 3	There being no recorded occurrence of an EF4 or EF5 tornado in FSAR Table 2.3.1-204 or the NOAA National Climatic Data Center website, the event frequency was estimated to be the same as for an EF3 tornado.	4.13E-05
	EF5 Tornado	Notes 1, 3		4.13E-05
	Category 1 Hurricane	Note 3	From data covering 142 years on the NOAA Coastal Services Center website, the number of hurricanes of each category coming within 50 nautical miles of the Levy site was identified. The event frequency was estimated from number of hurricanes divided by the number of years.	1.06E-01
	Category 2 Hurricane	Notes 1, 3		1.41E-02
	Category 3 Hurricane	Notes 1, 3		2.82E-02
	Category 4 Hurricane	Notes 1, 3	There being no recorded occurrence of a Category 4 or Category 5 hurricane within 50 nautical miles of the Levy site in the data covering 142 years on the NOAA Coastal Services Center website, the event frequency was estimated based on the assumed occurrence of one such hurricane during the next 142 years.	3.52E-03
	Category 5 Hurricane	Notes 1, 3		3.52E-03
	Extratropical Cyclones	Note 3	The risk associated with extratropical cyclones is loss of off-site power (LOSP) due to high winds. Extreme straight-line winds associated with extratropical cyclones are included in the NCDC database (1950 – 2008). The	4.0E-03

Table 19.58-201 (Sheet 2 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			highest recorded wind speed for a thunderstorm in the NCDC database (1950 – 2008) is 80 knots (92 mph) for the ten county area around the LNP 1 and 2 site. The LOSP frequency, due to wind events, is presented in the data reported in NUREG/CR-6890, Volume 1, "Reevaluation of Station Blackout Risk at Nuclear Power Plants - Analysis of Loss of Offsite Power Events: 1986-2004". That report shows eight LOSP events due to high winds (defined in this report as wind speed less than 125 mph) during 1,984.7 reactor-years (Including both Critical and Non-critical conditions for all reactors in the United States). This yields a frequency of 4.0E-03 LOSP events per reactor-year due to high wind events with speeds less than 125 mph (enveloping Extratropical cylones, Category 1 and Category 2 hurricanes and, EF0 and EF1 tornados). Applying the 4.0E-03 LOSP events per reactor year probability to the "Extratropical Cyclone" subcategory of wind events in DCD Tier 2 Table 19.58-3 evaluation would reduce the CDF in DCD Tier 2 Table 19.58-3. The core damage frequency (CDF) for extra tropical cyclones is 3.9E-11. The total CDF for all severe winds and tornado events is 3.29E-09 which is below the 1.0E-08 CDF event screening criteria.	
External Flood	External Flood	Note 4	The plant grade floor elevation is Elevation 51 feet NVAD88. As stated in FSAR Subsection 2.4.2.3 the maximum water level due to probable maximum precipitation (PMP) is below the plant grade floor elevation of 51 feet. The conceptual design for the Levy Switchyard design requires that the maximum	N/A

Table 19.58-201 (Sheet 3 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			flood elevation, as determined by the PMP study, shall be considered during detailed design of Levy switchyard buildings floors/foundations, switchyard structure foundations, and switchyard equipment foundations. This ensures that flood sensitive switchyard components are not impacted by the PMP flood. Thus, plant structures, systems and components are not impacted by the PMP. The sensitivity analysis in DCD Tier 2 Subsection 19.58.2.2 for flooding-induced failure of the switchyard and non-safety structures was considered bounding for the LNP site. There is no impact on the following potential flooding mechanisms on Levy safety related structures: • Streams and River: probable maximum level concurrent with wind generated waves will not affect S/R structures at the Levy Site (FSAR Subsection 2.4.3) • Potential Dam Failure: Potential dam failure does not affect the Levy site (FSAR Subsection 2.4.4) • Probable Maximum Surge: Probable maximum hurricane (PMH) surge level including wave effect is less than the plant grade elevation (FSAR Table 2.4.5-215) • Seiche: potential for flooding at the site due to seiche effect is considered insignificant (FSAR Subsection 2.4.5.2.6) Probable maximum tsunami (PMT) flooding is below the plant grade floor elevation of 51 (See FSAR Table 2.4.6-208). Therefore, there is no impact of the PMT flood on	

Table 19.58-201 (Sheet 4 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			the LNP site.	
Transport-ation and Nearby Facility Accidents	Aviation Accident	Notes 2, 3	The probability of small aircraft crashing on seismic category I structures (i.e. Containment/Shield Building and Auxiliary Building) is calculated to be 7.01E-06 per year. This crash probability results a core damage frequency (CDF) of 0.41E-12 per year which is below the 1.0E-08 CDF event screening criteria. Therefore, small aircraft crash probability is acceptable. The probability of large aircraft crashing on seismic category I structures is calculated as 3.09E-8 per year. This meets the acceptance criteria of 1.0E-07 per year in Section 19.58.2.3.1 of DCD. Therefore, the probability of crash for large aircraft is acceptable.	7.01E-06 (small aircraft) 3.09E-08 (large aircraft)
	Marine Accident	Note 5	DCD Tier 2 Subsection 19.58.2.3.2 indicates that only sites with large waterways with ship and/or barge traffic that goes through or near the site need to consider marine accidents. FSAR Subsection 2.2.2.4 indicates that water traffic of the five navigable waterways near the site is limited to pleasure and/or fishing boats. Therefore, the key site-related assumptions in DCD Subsection 19.58.2.3.2 concerning marine accidents are not applicable to the Levy site.	N/A
	Pipeline Accident	Note 2	There are three natural gas pipelines in the area of the LNP site. As discussed in FSAR Subsection 2.2.3.2.3 the maximum downwind concentration of natural gas at LNP due to a postulated rupture of the pipeline is less than 1.0 percent. This is well below the lower flammability limit for natural gas of 4.8 percent in air. Therefore, there are no adverse effects due to the unlikely rupture of the gas	<1.0E-07

Table 19.58-201 (Sheet 5 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			pipelines at their closest approach to LNP and the key site-related assumptions in DCD Section 19.58.2.3.3 concerning damage from explosive material released from nearby pipeline accidents are not applicable to the Levy site. The initial event frequency of 1.0E-07 per year assumed in DCD Section 19.58.2.3.3 is considered valid for the Levy site.	
	Railroad and Truck Accidents	Note 2	FSAR Subsection 2.2.3.2.1 concludes that potential sources of explosions from nearby activities are limited to an explosion in highway transport. U.S. Highway 19/98 is located west of the center of the site and its nearest approach to the site is approximately 1974 m (6477 ft.). The highway is mainly used for local traffic and local commodity deliveries only. The safe distance for explosive material is 505 m (1658 ft.) for a pressure of 1 psi, this is well below the separation distance from U.S. Highway 19/98. Thus, there are no adverse effects on LNP due to the transport of explosives via roadway.	<1.0E-07
Other Events	A number of external events beyond those evaluated in DCD Subsection 19.58 were evaluated for the LNP 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 LNP 1 and 2 site is within the bounds of the Floods and Other External Events analysis documented in DCD Tier 2 Section 19.58	
	External Fires	Note 2	Fires originating from accidents at any facilities or transportation routes identified above do not have	<1.0E-07

Table 19.58-201 (Sheet 6 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			the potential to endanger the safe operation of LNP because the distances between potential accident locations and LNP are greater than 1.6 km (1 mi.). The closest potential source of a significant fire is the 76.2-cm (30-in.) natural gas line at 1769 m (5803 ft.) from LNP. The evaluation of the pipeline failure, discussed in FSAR Subsection 2.2.3.2.3, concludes that there are no adverse effects due to the unlikely rupture of the gas pipelines at their closest location to LNP. Therefore, because no risk important consequences were identified, the potential for hazards from external fires are minimal and will not adversely affect the safe operation of LNP 1 and 2.	
	Toxic Chemical Release	Note 4	Based on the discussion in FSAR Subsection 2.2.2.2, there are no manufacturing facilities in the vicinity that utilize or store products that are considered hazardous. The Town of Iglis water treatment plant (WTP) is located 3 miles from the LNP site. FSAR Table 2.2.2-202 provides the chemicals and quantities stored and used by the WTP. Per FSAR Subsection 2.2.3.3, the quantities are small and are not significant sources of airborne contamination even in the event of an accidental failure of the storage containers. Therefore, there are no sources of toxic chemicals within 8 km (5 mi.) of LNP that could pose a threat to LNP. There are no site-specific sources of hazardous materials stored on the site in sufficient quantity to affect control room habitability (FSAR Subsections 2.2.3.3 and 6.4.4.2). Thus, these events are not considered risk	N/A

Table 19.58-201 (Sheet 7 of 7)
External Event Frequencies

LNP SUP 19.58-1

Category	Event	Evaluation Criteria	Explanation of Applicability Evaluation	Event Frequency
			important.	
	Major Depots and Storage Areas Releases	Note 5	Based on the discussion in FSAR Subsection 2.2.2.2, there are no manufacturing facilities in the vicinity that utilize or store products that are considered hazardous. The Town of Iglis water treatment plant (WTP) is located 3 miles from the LNP site. FSAR Table 2.2.2-202 provides the chemicals and quantities stored and used by the WTP. Per FSAR Subsection 2.2.3.3, the quantities are small and are not significant sources of airborne contamination even in the event of an accidental failure of the storage containers. Per FSAR Subsection 2.2.3.6, there is no safety-related equipment located at the intake structure. Therefore, spills drawn into the intake structure do not pose a nuclear safety hazard. Per FSAR Subsection 2.2.1, there are no active military facilities within 8 km (5 mi.) of the LNP site. The only significant military facility is for a National Guard unit located 67.6 km (42 mi.) from the LNP site.	N/A

Note 1: 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.
Note 2: IEF is less than 1.0E-07.
Note 3: Core damage frequency (CDF) is less than 1.0E-08.
Note 4: 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 LNP 1 and 2.
Note 5: 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 any design changes or departures from the design certification PRA.

As discussed in Section 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, Section 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 analyses 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 or 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
LNP 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 site. The site-specific effects have been 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 peak ground acceleration.

In the AP1000 PRA-based Seismic Margin Assessment (SMA), the RCC bridging mat failure is conservatively assumed to fall within the gross structural collapse event modeled in the hierarchical event tree discussed in DCD Section 19.55. As gross structural collapse is assumed to directly lead to core damage, failure of the RCC bridging mat has the potential to drive the plant level high confidence low probability of failure (HCLPF) capacity.

The assessment of risk significance of the LNP RCC bridging mat is based on the assumption that events which result in demand beyond the CDFM HCLPF capacity of the RCC bridging mat will lead to gross structural collapse. A more realistic assessment is that an event beyond the conservative deterministic failure mode (CDFM) HCLPF capacity for the RCC bridging mat may result in some cracking within the RCC bridging mat which in turn may result in limited damage to the NI structures. Thus, exceeding the CDFM HCLPF capacity would only have a limited effect on the NI structure performance.

The CDFM HCLPF capacity for soil liquefaction potential is based on no liquefaction potential for the LNP 10^{-5} UHRS. A seismic event larger than the 10^{-5} UHRS is required for soil liquefaction. For the larger event, liquefaction will be confined to isolated areas under the adjacent Turbine and Annex buildings and may result in damage to these buildings which in turn may result in limited damage to the NI structures. For Seismic Category II/I interaction between the nuclear island and the adjacent buildings, the CDFM HCLPF capacity is based on calculated relative displacements between the NI and the adjacent buildings for the LNP 10^{-5} UHRS of less than one (1) in. A two (2) in. gap is provided between the NI and adjacent building foundations. A seismic event larger than the 10^{-5} UHRS seismic event is required for the relative displacement between the NI and the adjacent structures to exceed the 2 in. gap provided. For the larger event, impact between the NI and the adjacent Turbine and Annex buildings would occur and may result in some local damage to the NI structure.

The seismic interaction between the Turbine Building and the NI was evaluated as discussed in DCD Subsection 19.55.2.2.6 and it was determined that the results of the seismic margin assessment, the plant HCLPF capacity, and the insights derived from the seismic margin assessment are not affected. For SMA, the Annex Building and the Radwaste Building are assumed to have failed as described in DCD Subsection 19.55.3.3. Thus, exceeding the CDFM HCLPF capacity for soil liquefaction or for Seismic Category II/I interaction between the nuclear island and the adjacent buildings will not affect the plant level HCLPF capacity.

Table 19.59-201 summarizes the PRA-based insight for the RCC bridging mat (site-specific design feature).

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.

<u>Schedule for Maintenance and Upgrades of the PRA</u>

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 update 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 RTNSS 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 Section 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 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.

Table 19.59-201
PRA-Based Insights for Site-Specific SSCs

LNP COL 19.59.10-6

Insight	Disposition
RCC Bridging Mat In the Seismic Margin Assessment, the RCC bridging mat failure is conservatively assumed to fall within the gross structural collapse event. As gross structural collapse is assumed to directly lead to core damage, failure of the RCC bridging mat has the potential to drive the plant level HCLPF value.	
The RCC bridging mat serves two purposes: 1) replace the weakly cemented, undifferentiated Tertiary sediments that are present above elevation -7.3 m (-24 ft.) NAVD88, thereby, creating a uniform subsurface with increased bearing capacity; and 2) bridge conservatively postulated karst features. The RCC bridging mat has been designed to bridge a 3 m (10 ft.) air-filled cavity located immediately beneath the RCC (elevation -7.3 m [-24 ft.] NAVD88) at any plan location.	2.5.4.5.4
Roller Compacted Concrete Strength and Constructability Verification Program includes use of test results from large scale commercial projects, pre-COL RCC testing, post-COL RCC testing, and RCC testing during production construction. The RCC bridging mat will be constructed using construction specifications, non destructive testing and quality controls during construction, construction implementing procedures, and construction equipment that are comparable to those used on past successful RCC projects.	3.8.5.11

LNP DEP 6.3-1

Table 19.59-202
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.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours. 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.4.10.2 Shutdown Temperature Evaluation

LNP DEP 3.2-1

Revise the first and second paragraphs of DCD Subsection 19E.4.10.2 to read as follows:

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 (Reference 15) criteria.

As discussed in Subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the specified safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this condition following postulated transients. A non-bounding, conservative analysis is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in 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.

Add new paragraphs 3 and 4 to DCD Subsection 19E.4.10.2 to read as follows:

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 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 efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Revise the first sentence of the fifth paragraph (old third paragraph) of DCD Subsection 19E.4.10.2 to read as follows:

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip.

Revise paragraphs 6 and 7 (old paragraphs 4 and 5) of DCD Subsection 19E.4.10.2 to read as follows:

Once actuated, at about 2,400 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 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 34,500 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 Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 seconds, while the core average temperature reaches 420°F within 124,400 seconds (approximately 34.6 hours).

As discussed in 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.

LNP DEP 3.2-1
LNP DEP 6.3-1

Add a new eighth paragraph to DCD Subsection 19E.4.10.2 to read as follows:

As discussed in 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 more 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.

19E.9 REFERENCES

LNP DEP 3.2-1

14. Not used.

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

LNP DEP 3.2-1

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	≤ 60
Rods Begin to Drop	≤ 61
Low Steam Generator Water Level (Wide-Range) Reached	≤ 230
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	≤ 240
Low T_{cold} Setpoint Reached	≤ 2400
Steam Line Isolation on Low T_{cold} Signal	≤ 2400
CMTs Actuated on Low T_{cold} Signal	≤ 2400
IRWST Reaches Saturation Temperature	≤ 15,500
Heat Extracted by PRHR HX Matches Core Decay Heat	≤ 34,500
CMTs Stop Recirculating	--
Cold Leg Temperature Reaches 420°F (loop with PRHR)	≤ 48,600
Core Average Temperature Reaches 420°F	≤ 124,400

APPENDIX 19F MALEVOLENT AIRCRAFT IMPACT

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