CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

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CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

19.1 INTRODUCTION

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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 were developed as described in Subsection 2.5.2.6. Both the LNP GMRS and the 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.

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). As stated previously, the 10^{-5} UHRS is greater than 1.67 times the LNP GMRS and the PBSRS. Thus, liquefaction potential of soil beyond the nuclear island perimeter which will be left in place will not lower the HCLPF values calculated for the certified design.

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 GMRS and the PBSRS. Thus, there is no adverse Seismic Category II/I interaction between the NI and the adjacent buildings that would lower the HCLPF values calculated for the certified design.

Thus, it can be concluded that the Seismic Margin Assessment analysis documented in Section 19.55 is applicable to the LNP site.

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. Texas Tech University, Wind Science and Engineering Center, "A Recommendation for an Enhanced Fujita Scale (EF-Scale)," June 2004.

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		5.21E-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.

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 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.

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 no departures or supplements.

APPENDIX 19F MALEVOLENT AIRCRAFT IMPACT

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