CHAPTER 7
ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

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ACRONYMS AND ABBREVIATIONS

°C	degree Celsius
°F	degree Fahrenheit
ALWR	advanced light water reactor
AP1000	Westinghouse Electric Company, LLC, AP1000 Reactor
AP600	Westinghouse Electric Company, LLC, AP600 Reactor
BP	Containment Bypass
CFBC	Cross Florida Barge Canal
CDF	core damage frequency
CEDE	committed effective dose equivalent
CFE	Early Containment Failure
CFI	Intermediate Containment Failure
CFL	Late Containment Failure
CFR	Code of Federal Regulations
CI	Containment Isolation Failure
Ci	Curie
Ci/MTU	Curie per metric ton of uranium
Co-60	Cobalt-60
CREC	Crystal River Energy Complex
DBA	design basis accident
DCD	Westinghouse Electric Company, LLC, AP1000 Design Control Document for the certified design as amended
EAB	exclusion area boundary
EDE	effective dose equivalent
EIS	Environmental Impact Statement

ACRONYMS AND ABBREVIATIONS (CONTINUED)

ER	Environmental Report
ESP	Early Site Permit
ESRP	Environmental Standard Review Plan
GEIS	Generic Environmental Impact Statement
gpm	gallon per minute
IC	Intact Containment
IRWST	in-containment refueling water storage tank
km	kilometer
LNP	proposed Levy Nuclear Plant Units 1 and 2
LNP 1	proposed Levy Nuclear Plant Unit 1
LNP 2	proposed Levy Nuclear Plant Unit 2
LOCA	loss of coolant accident
LPGS	Liquid Pathway Generic Study
lpm	liter per minute
LPZ	low population zone
LWR	light water reactor
MACR	Maximum Averted Cost Risk
mi.	mile
MTU/yr	metric ton of uranium per year
MWd/MTU	megawatt day per metric ton of uranium
N/A	not applicable
NEPA	National Environmental Policy Act of 1969
NRC	U.S. Nuclear Regulatory Commission

ACRONYMS AND ABBREVIATIONS (CONTINUED)

PEF	Florida Power Corporation doing business as Progress Energy Florida, Inc.
person-rem	person-roentgen equivalent man
person-rem/yr	person-roentgen equivalent man per year
person-Sv/yr	person-Sievert per year
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RADTRAN	radioactive material transportation
rem	roentgen equivalent man
RRY	reference reactor year
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
sec/m^3	second per cubic meter
SER	Safety Evaluation Report
SG	steam generator
ST	Source Term
Sv	Sievert
TEDE	total effective dose equivalent
TRAGIS	Transportation Routing Analysis Geographic Information System
U-235	uranium-235
USDOT	U.S. Department of Transportation
USEPA	U.S. Environmental Protection Agency
Westinghouse	Westinghouse Electric Company, LLC

ACRONYMS AND ABBREVIATIONS (CONTINUED)

X/Q	atmospheric dispersion coefficient

7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter evaluates the environmental impacts of postulated accidents involving radioactive materials related to the operation of the proposed Levy Nuclear Plant Units 1 and 2 (LNP) and several appurtenant facilities. These appurtenant facilities include electric transmission lines, an electric switchyard, a water intake structure and associated pumphouse located on the Cross Florida Barge Canal (CFBC), and makeup and blowdown pipelines.

This chapter includes the following key components in the following sections of this Environmental Report (ER):

- ER Section 7.1 — Design Basis Accidents
- ER Section 7.2 — Severe Accidents
- ER Section 7.3 — Severe Accident Mitigation Measures
- ER Section 7.4 — Transportation Accidents

7.1 DESIGN BASIS ACCIDENTS

The purpose of this section is to provide a comparison of the off-site dose consequences and resulting health effects for design basis accidents (DBAs), as identified in the Westinghouse Electric Company, LLC (Westinghouse), AP1000 Design Control Document for the certified design as amended (DCD), and those contained in Section 15 of the Safety Evaluation Report (SER). The following sections contain information to meet the guidance specified in Chapter 7 of NUREG-1555.

7.1.1 SELECTION OF DESIGN BASIS ACCIDENTS

The DBAs considered in this subsection are from the DCD and are consistent with the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183. Table 7.1-1 lists the DBAs having the potential for releases to the environment and provides an initial evaluation of each accident. The radiological consequences of the DBAs listed in Table 7.1-1 are assessed to demonstrate that two new Westinghouse Electric Company, LLC, AP1000 Reactor (AP1000) units can be sited at the LNP site without undue risk to the health and safety of the public.

7.1.2 EVALUATION METHODOLOGY

Doses for the selected DBAs were evaluated at the LNP site exclusion area boundary (EAB) and low population zone (LPZ). The DCD presents the radiological consequences for the accidents identified in Table 7.1-1. The DCD DBAs are updated with LNP site data to demonstrate that the DCD analyses are bounding for the LNP site. The basic scenario for each accident is that some quantity of activity is released at the accident location inside a building and this activity is eventually released to the environment. The transport of activity within the plant is independent of the site and specific to the AP1000 design. Details about the methodologies and assumptions pertaining to each of the accidents are provided in the DCD. These doses must meet the site acceptance criteria in 10 Code of Federal Regulations (CFR) 50.34, provided as follows:

(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 [roentgen equivalent man] rem total effective dose equivalent (TEDE).

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation (for example, a large break loss of coolant accident [LOCA]). For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the additional acceptance criteria provided in Regulatory Guide 1.183. The dose acceptance criteria from Regulatory Guide 1.183, with one exception, are listed in Table 7.1-2. No dose limit is listed in the Regulatory Guide for the small line break outside containment. Therefore, the criterion was adopted from DCD Subsection 15.6.2 and is consistent with Environmental Standard Review Plan (ESRP) 15.6.2 of NUREG-0800. The dose limits ensure that the consequences of each DBA are acceptable from an overall risk perspective.

The dose to an individual located at the EAB or LPZ is calculated based on the amount of activity released to the environment, the atmospheric dispersion of the activity during the transport from the release point to the off-site location, the breathing rate of the individual at the off-site location, the time of exposure, and activity-to-dose conversion factors. The only site-specific parameter is atmospheric dispersion. The DCD doses are determined using time-dependent atmospheric dispersion coefficient (X/Q) values corresponding to the top 5^{th} percentile meteorology during the first two hours of the accident, meaning that conditions would be more favorable for dispersion 95 percent of the time. The doses evaluated herein are calculated based on the 50^{th} percentile site-specific X/Q values during the first two hours of the accident, reflecting more realistic meteorological conditions. Site-specific doses are obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (X/Q values). Because the site-specific X/Q values are bounded by the DCD X/Q values, this approach demonstrates that the site-specific doses are within those calculated in the DCD.

The LNP short-term X/Q values are calculated using Regulatory Guide 1.145 methods with site-specific meteorological data. The Regulatory Guide 1.145 methodology is implemented in the NRC-sponsored PAVAN computer program. This program computes X/Q values at the EAB and the LPZ for each combination of wind speed and atmospheric stability for each of 16 downwind direction sectors and then calculates overall (not direction-specific) X/Q values. For a given location, either the EAB or the LPZ, the 0- to 2-hour X/Q value is the 50^{th} percentile overall value calculated by PAVAN. For the LPZ, the X/Q values for all subsequent times are calculated by logarithmic interpolation between the 50^{th} percentile X/Q value and the annual average X/Q value. Releases of activity are assumed to be at ground level.

The accident doses are expressed as TEDE doses. The TEDE dose is the summation of the committed effective dose equivalent (CEDE) from inhalation of radioactive particles and the effective dose equivalent (EDE) from external exposure. The CEDE is determined using the dose conversion factors in the U.S. Environmental Protection Agency's (USEPA) Federal Guidance Report 11, while the EDE is based on the dose conversion factors in Federal Guidance Report 12 (References 7.1-001 and 7.1-002). As indicated in Regulatory Guide 1.183, the dose conversion factors in Federal Guidance Reports 11 and 12 are acceptable to the NRC staff. Appendix 15A of the DCD provides information about this methodology.

7.1.3 RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS

This subsection identifies the postulated accidents and provides a brief description of each accident used in the LNP site dose consequence assessments. A more detailed description of each accident is provided in DCD Chapter 15. An overall summary of the results of the LNP site-specific evaluated accident doses is presented in Table 7.1-2. Table 7.1-2 shows that the evaluated dose consequences meet the regulatory acceptance criteria.

The analysis approach for evaluating the AP1000 DBAs discussed in the following subsections is based upon the EAB and LPZ doses provided in DCD Chapter 15. The ratio of the LNP site X/Q values to the generic AP1000 site X/Q values for each post accident time period is given in Table 7.1-3. (Note: The X/Q value for 1.4 to 3.4 hours at the LNP site was not calculated. To calculate the EAB dose for the LOCA, the X/Q value for the period between 0 and 2 hours was used instead.)

7.1.3.1 Main Steam Line Break Outside Containment

The bounding AP1000 steam line break for the radiological consequence evaluation occurs outside containment. The facility is designed so that only one steam generator (SG) experiences an uncontrolled blowdown even if one of the main steam isolation valves fails to close. Feedwater is isolated after the rupture and the faulted SG dries out. The secondary side inventory of the faulted SG is released to the environs along with the entire amount of iodine and alkali metals contained in the secondary side coolant.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The TEDE doses for the preexisting iodine spike are shown in Table 7.1-4. The doses at the EAB and the LPZ are a small fraction of the 0.25 Sieverts (Sv) (25 rem) TEDE identified in 10 CFR 50.34. The doses for the accident-initiated iodine spike are shown in Table 7.1-5. These doses meet the TEDE dose guidelines of 10 CFR 50.34.

7.1.3.2 Locked Rotor

The AP1000 locked rotor event is the most bounding of several possible decreased reactor coolant flow events. This accident is postulated as an instantaneous seizure of the pump rotor in one of four reactor coolant pumps. The rapid reduction in flow in the faulted loop causes a reactor trip. Heat transfer of the stored energy in the fuel rods to the reactor coolant causes the reactor coolant temperature to increase. The reduced flow also degrades heat transfer between the primary and secondary sides of the SGs. The event can possibly lead to fuel cladding failure, which would result in an increase of activity in the coolant. The rapid expansion of coolant in the core combined with decreased heat transfer in the SG is assumed to cause the reactor coolant system pressure to increase dramatically.

Cool down of the plant by steaming off the SGs provides a pathway for the release of radioactivity to the environment. In addition, primary side activity is assumed to be carried over due to leakage in the SGs, mixes in the secondary side, and becomes available for release. The primary side coolant activity inventory is assumed to increase due to the postulated failure of some of the fuel cladding with the consequential release of the gap fission product inventory to the coolant. The significant releases from this event are the iodines, alkali metals, and noble gases. No fuel melting is assumed to occur.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The TEDE doses for the locked rotor accident, both with and without feedwater available, are shown in Table 7.1-6. The doses at the EAB and the LPZ are a small fraction of the TEDE limits identified in 10 CFR 50.34.

7.1.3.3 Control Rod Ejection

The control rod ejection accident is postulated as the gross failure of one control rod mechanism pressure housing resulting in ejection of the control rod cluster assembly and drive shaft. The failure is assumed to lead to a rapid positive reactivity insertion, potentially leading to localized fuel rod damage and significant releases of radioactivity to the reactor coolant.

Two activity release paths are assumed to contribute to this event. First, the equilibrium activity in the reactor coolant and the activity from the damaged fuel are blown down through the failed pressure housing to the containment atmosphere. The activity can leak to the environment over a relatively long period due to the containment's design basis leakage. Decay of radioactivity occurs during hold-up inside containment prior to release to the environs.

The second release path is from the release of steam from the SGs following the reactor trip. With a coincident loss of off-site power, additional steam must be released in order to cool down the reactor. The SG activity consists of the secondary side equilibrium inventory plus the additional contributions from reactor coolant leaks in the SGs. The reactor coolant activity levels are increased for this accident because the activity released from the damaged fuel mixes into the coolant prior to being leaked to the SGs. The iodines, alkali metals, and noble gases are the significant activity sources for this event. Noble gases entering the secondary side are quickly released to the atmosphere via the steam releases through the atmospheric relief valves. A small fraction of iodines and alkali metals in the flashed part of the leak flow are available for immediate release without benefit of partitioning. The unflashed portion mixes with secondary side fluids where partitioning occurs prior to the release as steam.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The doses at the EAB and the LPZ shown in Table 7.1-7 are well within the TEDE limits identified in 10 CFR 50.34.

7.1.3.4 Steam Generator Tube Rupture

The AP1000 SG tube rupture postulated accident assumes the complete severance of one SG tube. The accident is assumed to cause an increase in the secondary side activity due to reactor coolant flow through the ruptured tube. With the loss of off-site power, contaminated steam is released from the secondary system due to the turbine trip and dumping of steam via the atmospheric relief valves. Steam dump (and retention of activity) to the condenser is precluded due to the assumption of loss of off-site power. The release of radioactivity depends on the primary to secondary leakage rate, the flow to the faulted SG from the ruptured tube, the percentage of defective fuel in the core, and the duration and amount of steam released from the SGs.

The radioiodines, alkali metals, and noble gases are the significant nuclide groups released during a SG tube rupture accident. Multiple release pathways are analyzed for the tube rupture accident. The noble gases in the reactor coolant enter the ruptured SG and are assumed to be available for immediate release to the environment. In the intact loops, iodines and alkali metals leaked to the secondary side during the accident are partitioned as the intact SG is steamed down until switchover to the residual heat removal system occurs. In the ruptured SG, some of the reactor coolant flowing through the tube break flashes to steam while the unflashed portion mixes with the secondary side inventory. Iodines and alkali metals in the flashed fluid are not partitioned during steam releases while activity in the secondary side of the faulted generator is partitioned prior to release as steam.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The TEDE doses for the SG tube rupture accident with the accident-initiated iodine spike are shown in Table 7.1-8. The doses at the EAB and LPZ are a small fraction of the TEDE limits identified in 10 CRF 50.34. The preexisting iodine spike doses are shown in Table 7.1-9. These doses meet the TEDE dose guidelines of 10 CFR 50.34.

7.1.3.5 Failure of Small Lines Carrying Primary Coolant Outside of Containment

Small lines carrying reactor coolant outside the AP1000 containment include the reactor coolant system sample line and the chemical and volume control system discharge line to the radwaste system. These lines are not continuously used. The failure of the discharge line is neither significant nor analyzed. The flow (approximately 378 liters per minute [lpm] or 100 gallons per minute [gpm]) leaving containment is cooled below 60 degrees Celsius (°C) (140 degrees Fahrenheit [°F]) and has been cleaned by the mixed bed demineralizer. The reduced iodine concentration, low flow, and temperature make this break non-limiting with respect to off-site dose consequences.

The reactor coolant system sample line break is the more limiting break. This line is postulated to break between the outboard isolation valve and the reactor coolant sample panel. Off-site doses are based on an assumed break flow limited to 492 lpm (130 gpm) by flow restrictors with isolation occurring at 30 minutes. Radioiodines and noble gases are the only significant activities released. The source term is based on an accident-initiated iodine spike that increases the iodine release rate from the fuel by an assumed factor of 500 throughout the event. The activity is assumed to be released to the environment without decay or hold-up in the auxiliary building.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The results are shown in Table 7.1-10. The resulting dose at the EAB and LPZ is a small fraction of the TEDE limits identified in 10 CFR 50.34.

7.1.3.6 Large Break Loss of Coolant Accident

The core response analysis for the AP1000 demonstrates that the reactor core maintains its integrity for the large break LOCA. However, significant core degradation and melting is assumed in this DBA. The assumption of major core damage is intended to challenge various accident mitigation features and provide a conservative basis for calculating site radiological consequences. The source term used in the analysis is adopted from NUREG-1465 and Regulatory Guide 1.183 with the nuclide inventory determined for a three-region equilibrium cycle core at end of life.

The activity released consists of the equilibrium activity in the reactor coolant and the activity released from the damaged core. The AP1000 is a leak before break design; therefore, the coolant is assumed to blow down to the containment for 10 minutes. One-half of the iodine and the noble gases in the blowdown stream are released to the containment atmosphere.

The core release starts after the 10-minute blowdown of reactor coolant. The fuel rod gap activity is assumed to be released over the next half hour followed by an in-vessel core melt that lasts 1.3 hours. Iodines, alkali metals, and noble gases are released during the gap activity release. During the core melt phase, five additional nuclide groups are assumed to be released: the tellurium group, the noble metals group, the lanthanides group, the cerium group, and the barium and strontium group.

Activity is assumed to be released from the containment via the containment purge line at the beginning of the accident. After isolation of the purge line, activity continues to leak from the containment at its design basis leak rate. There is no emergency core cooling leakage activity because the passive core cooling system does not pass coolant outside the containment. A coincidental loss of off-site power has no impact on the activity release to the environment because of the passive designs for the core cooling and fission product control systems.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. Table 7.1-11 provides the EAB and LPZ doses. Both doses meet the TEDE dose guideline in 10 CFR 50.34. The activity released from the core melt phase of the accident is the greatest contributor to off-site doses. The EAB dose in Table 7.1-11 is given for the two-hour period during which the dose is greatest at this location. The initial two hours of the accident is not the worst two-hour period because of the delays associated with cladding failure and fuel damage.

7.1.3.7 Fuel Handling Accidents

The AP1000 fuel handling accident can occur inside containment or in the fuel handling area of the auxiliary building. The accident postulates the dropping of a fuel assembly over the core or in the spent fuel pool. The cladding of the fuel rods is assumed breached and the fission products in the fuel rod gaps are released to the reactor refueling cavity water or spent fuel pool.

The AP1000 DCD doses were reevaluated using the LNP site short-term accident dispersion characteristics. The resulting doses at the EAB and the LPZ are summarized in Table 7.1-12. The doses are applicable to FHAs inside containment and in the spent fuel pool in the auxiliary building. The doses are well within the TEDE guidelines in 10 CFR 50.34.

7.1.4 REFERENCES

7.1-001 U.S. Environmental Protection Agency, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, 1988.

7.1-002 U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, 1993.

Table 7.1-1
Selection of Accidents

ESRP/DCD Subsection	ESRP Description	DCD Description	Identified in NUREG-1555, Appendix A	Comment
15.1.5A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	Steam System Piping Failure	Yes	DCD Subsection 15.1.5
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	Feedwater System Pipe Break	Yes	Per DCD, bounded by Subsection 15.1.5 accident
15.3.3	Reactor Coolant Pump Rotor Seizure	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Yes	
15.3.4	Reactor Coolant Pump Shaft Break	Reactor Coolant Pump Shaft Break	Yes	Per DCD, bounded by Subsection 15.3.3 accident
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	Spectrum of Rod Cluster Control Assembly Ejection Accidents	No	Included for completeness
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Coolant Outside Containment	Failure of Small Lines Carrying Primary Coolant Outside Primary Containment	Yes	
15.6.3	Radiological Consequences of Steam Generator Tube Rupture (PWR)	Steam Generator Tube Failure	Yes	
15.6.5A	Radiological Consequences of a Design Basis LOCA Including Containment Leakage Contribution	LOCA Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Yes	DCD Subsection 15.6.5
15.6.5B	Radiological Consequences of a Design Basis LOCA: Leakage From Engineered Safety Feature Components Outside Containment	LOCA Resulting from a Spectrum of Postulated Piping Breaks Safety Feature Components Outside Containment Within the Reactor Coolant Containment Pressure Boundary	Yes	DCD Subsection 15.6.5
15.7.4	Radiological Consequences of Fuel Handling Accidents	Fuel Handling Accident	Yes	

Notes:

ESRP = Environmental Standard Review Plan
LOCA = loss of coolant accident
PWR = pressurized water reactor

Table 7.1-2
Summary of LNP Site-Specific Off-Site Doses Consequences

Accident	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)	Guideline Limit TEDE Sv (rem)
Main Steam Line Break			
Preexisting Iodine Spike	7.8E-04 (7.8E-02)	1.7E-04 (1.7E-02)	0.25 (25)
Accident-Initiated Iodine Spike	8.6E-04 (8.6E-02)	4.6E-04 (4.6E-02)	0.025 (2.5)
Reactor Coolant Pump Locked Rotor			
No Feedwater	6.2E-04 (6.2E-02)	8.2E-05 (8.2E-03)	0.025 (2.5)
Feedwater Available	4.7E-04 (4.7E-02)	1.7E-04 (1.7E-02)	0.025 (2.5)
Control Rod Ejection Accident	2.8E-03 (2.8E-01)	1.2E-03 (1.2E-01)	0.063 (6.3)
Steam Generator Tube Rupture			
Preexisting Iodine Spike	1.7E-03 (1.7E-01)	2.6E-04 (2.6E-02)	0.25 (25)
Accident-Initiated Iodine Spike	8.6E-04 (8.6E-02)	1.8E-04 (1.8E-02)	0.025 (2.5)
Small Line Break	1.6E-03 (1.6E-01)	2.2E-04 (2.2E-02)	0.025 (2.5)
Design Basis LOCA	3.7E-02 (3.7E+00)	1.1E-02 (1.1E+00)	0.25 (25)
Fuel Handling Accident	4.1E-03 (4.1E-01)	5.5E-04 (5.5E-02)	0.063 (6.3)

Notes:

Doses are based on Federal Guidance Report 11 and Federal Guidance Report 12 dose conversion.

TEDE guidelines are from Regulatory Guide 1.183. Small line break criteria based on Environmental Standard Review Plan 15.6.2.

EAB = exclusion area boundary
LOCA = loss of coolant accident
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-3
Ratio of LNP 50 Percent Accident Site X/Q Values to AP1000 DCD X/Q Values

			X/Q Ratio
Post Accident Time Period (Hours)	**LNP Site X/Q Values (sec/m^3)**	**AP1000 DCD X/Q Values (sec/m^3)**	**LNP Site / AP1000 DCD**
LOCA			
EAB			
1.4--3.4 hr [(a)]	7.81E-05	5.1E-04	1.53E-01
LPZ			
0--8 hr	1.06E-05	2.2E-04	4.82E-02
8--24 hr	7.81E-06	1.6E-04	4.88E-02
24--96 hr	4.01E-06	1.0E-04	4.01E-02
96--720 hr	1.54E-06	8.0E-05	1.93E-02
All Other Accidents			
EAB			
0--2 hr	7.81E-05	1.0E-03	7.81E-02
LPZ			
0--8 hr	1.06E-05	5.00E-04	2.12E-02
8--24 hr	7.81E-06	3.00E-04	2.60E-02
24--96 hr	4.01E-06	1.50E-04	2.67E-02
96--720 hr	1.54E-06	8.00E-05	1.93E-02

Notes:
a) The EAB X/Q value for the period 0 to 2 hours was used for the 1.4 to 3.4 hour period for the LNP site. The 1.4 to 3.4 hour period represents the worst 2-hour period for the EAB dose.

EAB = exclusion area boundary
LPZ = low population zone
sec/m^3 = seconds per cubed meter
X/Q = atmospheric dispersion coefficient

Table 7.1-4
Main Steam Line Break, 0 to 72 Hours, Preexisting Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	1.00E-02 (1.00E+00)	
0 to 8	--	5.81E-03 (5.81E-01)
8 to 24	--	7.18E-04 (7.18E-02)
24 to 96	--	1.08E-03 (1.08E-01)
Total	**1.00E-02 (1.00E+00)**	**7.61E-03 (7.61E-01)**
LNP Site		
0 to 2	7.81E-04 (7.81E-02)	
0 to 8	--	1.23E-04 (1.23E-02)
8 to 24	--	1.87E-05 (1.87E-03)
24 to 96	--	2.89E-05 (2.89E-03)
Total	**7.81E-04 (7.81E-02)**	**1.71E-04 (1.71E-02)**
Limit	0.25 (25)	0.25 (25)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-5
Main Steam Line Break, 0 to 72 Hours, Accident-Initiated Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	1.10E-02 (1.10E+00)	
0 to 8	--	1.02E-02 (1.02E+00)
8 to 24	--	3.77 E-03 (3.77E-01)
24 to 96	--	5.36E-03 (5.36E-01)
Total	**1.10E-02 (1.10E+00)**	**1.93E-02 (1.93E+00)**
LNP Site		
0 to 2	8.59E-04 (8.59E-02)	
0 to 8	--	2.16E-04 (2.16E-02)
8 to 24	--	9.81E-05 (9.81E-03)
24 to 72	--	1.43E-04 (1.47E-02)
Total	**8.59E-04 (8.59E-02)**	**4.58E-04 (4.58E-02)**
Limit	0.025 (2.5)	0.025 (2.5)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-6
Locked Rotor Accident, 0 to 1.5 Hours, Preexisting Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
No Feedwater		
AP1000 Tier 2		
0 to 1.5	8.00E-03 (8.00E-01)	3.89E-03 (3.89E-01)
LNP Site		
0 to 1.5	6.25E-04 (6.25E-02)	8.25E-05 (8.25E-03)
Locked Rotor Accident, 0 to 8 Hours, Preexisting Iodine Spike		
Feedwater Available		
AP1000 Tier 2		
0 to 2	6.00E-03 (6.00E-01)	--
0 to 8	--	7.94 E-03 (7.94E-01)
Total	**6.00E-03 (6.00E-01)**	**7.94E-03 (7.94E-01)**
LNP Site		
0 to 2	4.69E-04 (4.69E-02)	--
0 to 8	--	1.68E-04 (1.68E-02)
Total	**4.69E-04 (4.69E-02)**	**1.68E-04 (1.68E-02)**
Limit	0.025 (2.5)	0.025 (2.5)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-7
Control Rod Ejection Accident, 0 to 720 Hours, Preexisting Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	3.60E-02 (3.60E+00)	--
0 to 8	--	4.58E-02 (4.58E+00)
8 to 24	--	7.84E-03 (7.84E-01)
24 to 96	--	6.82E-04 (6.32E-02)
96 to 720	--	2.06E-04 (2.06E-02)
Total	**3.60E-02 (3.60E+00)**	**5.45E-02 (5.45E+00)**
LNP Site		
0 to 2	2.81E-03 (2.8E-01)	--
0 to 8	--	9.71E-04 (9.71E-02)
8 to 24	--	2.04E-04 (2.04E-02)
24 to 96	--	1.69E-05 (1.69E-03)
96 to 720	--	3.97E-06 (3.97E-04)
Total	**2.81E-03 (2.81E-01)**	**1.20E-03 (1.20E-01)**
Limit	0.063 (6.3)	0.063 (6.3)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-8
Steam Generator Tube Rupture, 0 to 24 Hours, Accident-Initiated Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	1.10E-02 (1.10E+00)	--
0 to 8	--	6.27E-03 (6.27E-01)
8 to 24	--	1.69E-03 (1.69E-01)
Total	**1.10E-02 (1.10E+00)**	**7.96E-03 (7.96E-01)**
LNP Site		
0 to 2	8.59E-04 (8.59E-02)	--
0 to 8	--	1.33E-04 (1.33E-02)
8 to 24	--	4.40E-05 (4.40E-03)
Total	**8.59E-04 (8.59E-02)**	**1.77E-04 (1.77E-02)**
Limit	0.025 (2.5)	0.025 (2.5)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-9
Steam Generator Tube Rupture, 0 to 24 Hours, Preexisting Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	2.20E-02 (2.20E+00)	--
0 to 8	--	1.16E-02 (1.16E+00)
8 to 24	--	7.24E-04 (7.24E-02)
Total	**2.20E-02 (2.20E+00)**	**1.23E-02 (1.23E+00)**
LNP Site		
0 to 2	1.72E-03 (1.72E-01)	--
0 to 8	--	2.46E-04 (2.46E-02)
8 to 24	--	1.88E-05 (1.88E-03)
Total	**1.72E-03 (1.72E-01)**	**2.65E-04 (2.65E-02)**
Limit	0.25 (2.5)	0.25 (2.5)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-10
Small Line Break Accident, 0 to 0.5 Hours, Accident-Initiated Iodine Spike

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 0.5	2.10E-02 (2.10E+00)	1.02E-02 (1.02E+00)
LNP Site		
0 to 0.5	1.64E-03 (1.64E-01)	2.16E-04 (2.16E-02)
Limit	0.025 (2.5)	0.025 (2.5)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-11
AP1000 Design Basis LOCA

Time (Hours)	EAB Dose TEDE Sv (rem) [a]	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
1.4 to 3.4	2.46E-01 (2.46E+01)	--
0 to 8	--	2.17E-01 (2.17E+01)
8 to 24	--	7.50E-03 (7.50E-01)
24 to 96	--	2.93E-03 (2.93E-01)
96 to 720	--	5.49E-03 (5.49E-01)
Total	**2.46E-01 (2.46E+01)**	**2.33E-01 (2.33E+01)**
LNP Site		
1.4 to 3.4	3.74E-02 (3.74E+00)	--
0 to 8	--	1.05E-02 (1.05E+00)
8 to 24	--	3.66E-04 (3.66E-02)
24 to 96	--	1.17E-04 (1.17E-02)
96 to 720	--	1.06E-04 (1.06E-02)
Total	**3.74E-02 (3.74E+00)**	**1.10E-02 (1.10E+00)**
Limit	0.25 (25)	0.25 (25)

Notes:

a) The EAB dose is for the worst 2-hour period.

EAB = exclusion area boundary
LOCA = loss of coolant accident
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

Table 7.1-12
Fuel Handling Accidents, 0 to 2 Hours

Time (Hours)	EAB Dose TEDE Sv (rem)	LPZ Dose TEDE Sv (rem)
AP1000 Tier 2		
0 to 2	5.20E-02 (5.20E+00)	2.59E-02 (2.59E+00)
LNP Site		
0 to 2	4.06E-03 (4.06E-01)	5.49E-04 (5.49E-02)
Limit	0.063 (6.3)	0.063 (6.3)

Notes:

EAB = exclusion area boundary
LPZ = low population zone
rem = roentgen equivalent man
Sv = Sievert
TEDE = total effective dose equivalent

7.2 SEVERE ACCIDENTS

ER Section 7.1 provides a comparison of the off-site dose consequences and resulting health effects for DBAs, as identified in the DCD and those contained in Section 15 of the SER. A direct comparison of the off-site dose consequences and health effects, as provided in NUREG-1555, is difficult. ER Section 7.1 provides quantitative results, whereas the results reported in this section are mostly expressed probabilistically. However, doses calculated for the EAB and LPZ in ER Section 7.1 from DBAs compare favorably to those calculated from severe accidents at a 0 to 80-kilometer (km) (0 to 50-mile [mi.]) radius (internal events only).

7.2.1 INTRODUCTION

This subsection evaluates the potential environmental impacts of severe accidents at the LNP site. As a class, severe accidents are considered less likely to occur and are not part of the design basis for the AP1000; however, because the consequences could be more severe, severe accidents are considered important both in terms of impact to the environment and off-site costs. Severe accidents are those accidents that are more severe than DBAs and result in substantial damage to the reactor core whether or not there are serious off-site consequences.

Westinghouse completed a probabilistic risk assessment (PRA) for the AP1000 design, as documented in the DCD, as part of the design certification process. The PRA included the development of a Level 3 PRA model to evaluate the consequences associated with severe accidents. The Westinghouse Level 3 PRA model used generic characteristics to represent site-specific attributes. This subsection presents an update of the generic PRA analysis of severe accidents to include Level 3 modeling of the site-specific characteristics of the LNP site.

In NUREG-1437, the NRC generically assesses the impacts of severe accidents during license renewal periods using the results of existing analyses and site-specific information to conservatively predict the environmental impacts of severe accidents for each plant during the renewal period. In NUREG-1437, assessment methodologies were developed to evaluate each of the three dose pathways (atmospheric, surface water, and groundwater) by which a severe accident may result in adverse environmental impacts. The assessment methodologies for the surface water and groundwater pathways provide additional assessment means beyond that available via Level 3 PRA modeling. Furthermore, the NUREG-1437 results, which are based on the assessment of many existing sites, support the assessment of other sites not specifically assessed in NUREG-1437 via site and site category comparisons. The results of this report are therefore used as a basis for evaluating the severe accident environmental impacts of a new nuclear power generating facility that may be built on the LNP site.

The results of LNP site-specific Level 3 analysis and the generic results of NUREG-1437 demonstrate that the potential impacts of a severe accident for the AP1000 design on the LNP site are of small significance, as defined by the NRC. The potential impact of severe accidents for the new reactor design is significantly lower than for current design reactors and is many orders of magnitude below the NRC safety goal numerical objectives. The Level 3 analysis results are also used to support the severe accident mitigation alternative (SAMA) analyses in ER Section 7.3.

7.2.2 SIGNIFICANCE CRITERIA FOR POTENTIAL SEVERE ACCIDENT RELEASES

Throughout this subsection, environmental impacts of the alternatives are assessed using the NRC three-level standard of significance: SMALL, MODERATE, or LARGE. This standard of significance was developed using the Council on Environmental Quality guidelines set forth in the footnotes to Table B-1 of 10 CFR 51, Subpart A, Appendix B:

- **SMALL** — Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the Commission has concluded that those impacts that do not exceed permissible levels in the Commission's regulations are considered SMALL.

- **MODERATE** — Environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

- **LARGE** — Environmental effects are clearly noticeable and are sufficient to destabilize any important attributes of the resource.

The impact categories evaluated in this chapter are the same as those used in the Generic Environmental Impact Statement (GEIS), NUREG-1437, Volumes 1 and 2.

In accordance with the National Environmental Policy Act of 1969 (NEPA) practice, potential additional mitigation is considered in proportion to the significance of the impact to be addressed (impacts that are SMALL receive less mitigative consideration than impacts that are LARGE).

In 1986, the NRC issued a Safety Goal Policy Statement (cited in NUREG-1811, Volume 1) outlining the following quantitative health objectives to be used in determining achievement of the safety goals for the operation of a reactor in the United States:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

These quantitative health objectives are translated into two numerical objectives as follows:

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, is about 5 x 10^{-4} per year. One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than 5 x 10^{-7} per reactor year.

- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or 2 x 10^{-3} per year. One-tenth of 1 percent of this implies that the risk of cancer to the population in the area near a nuclear power plant because of its operation should be limited to 2 x 10^{-6} per reactor year.

7.2.3 LNP SITE-SPECIFIC LEVEL 3 PRA ANALYSIS

This subsection updates the Westinghouse generic PRA analysis of severe accidents to include LNP site-specific attributes in the Level 3 modeling. The Level 3 PRA model uses the MACCS2 computer code, the same code used by Westinghouse. The MACCS2 dose pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground and skin, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water. The code also evaluates the extent of contamination to the surrounding area. The code does not specifically model groundwater pathways (for example, aquifers).

To assess human health impacts, the collective dose to the 80-km (50-mi.) population, number of latent cancer fatalities, and number of early fatalities associated with a severe accident were determined. Economic costs were also determined, including the costs associated with relocation of people, decontamination of property and equipment, and interdiction of food supplies.

7.2.3.1 LNP MACCS2 Input

The AP1000 PRA formed the foundation for the LNP MACCS2 analysis and is described in Chapter 19 of the DCD. The PRA identified six source term categories that may be used to represent the suite of potential severe accidents and the internal events accident frequency associated with each (that is, core damage frequency [CDF]). The six source term categories or accident classes are fully described in the DCD and are titled as follows:

- Intermediate Containment Failure (CFI).
- Early Containment Failure (CFE).
- Intact Containment (IC).
- Containment Bypass (BP).
- Containment Isolation Failure (CI).
- Late Containment Failure (CFL).

MACCS2 uses five input files to process numerous user-specified parameters. The input files include: ATMOS, MET, SITE, EARLY, and CHRONC. AP1000 design-specific and LNP site-specific parameters are used where appropriate. Otherwise, input parameters are consistent with the MACCS2 User's Guide, those provided in Sample Problem A (distributed with the MACCS2 code), or other recognized sources.

The ATMOS file includes inputs specific to the reactor and plume release and dispersion after an accident. AP1000-specific input includes the core inventory, reactor and associated building dimensions, and source terms including release fractions developed based on data provided by Westinghouse. Consistent with the Westinghouse modeling, releases were assumed to occur at the top of the containment building, and plume heat energy was neglected. Source term release fraction data used in the analysis are summarized in Table 7.2-1. Three plumes were modeled for each release category in the analysis.

The meteorological data used in the MACCS2 model MET file consisted of 1 year of hourly observations of wind speed, wind direction, stability class (derived from vertical temperature gradient), and precipitation. LNP site-specific meteorology data for February 2007 through January 2008 were obtained from the existing on-site meteorological monitoring station, as described in ER Sections 2.7 and 6.4. The MACCS2 code requires the MET input file hourly data to be complete, and therefore, no missing data are allowed. For periods where missing data were less than 6 consecutive hours, interpolation was utilized to estimate the missing data. For periods where the missing data were 6 or more consecutive hours, substitution was utilized using data from the same time of day, either previous to or after the data void.

The SITE file includes inputs specific to the region surrounding the reactor site. LNP site-specific parameters are used in the SITE file, which include year 2060 projected population, land fraction, watershed indices and ingestion factors, and economic data for the 80-km (50-mi.) region. Year 2060 population (approximate end of 40-year license period) was selected because it provides conservative results due to projected population growth. The 2060 population data is consistent with that presented in ER Section 2.5.

Land fraction (land versus water) and water shed indices (river systems versus ocean) were estimated from Figures 7.2-1 and 7.2-2, for the evaluation of surface water doses. As shown in Table 2.3-18, there are almost no surface water users (drinking water) in the 80-km (50-mi.) region of the LNP site. MACCS2 analyses assuming surface water users typically find that water ingestion dose from surface water contamination is very small (for example, less than 5 percent) compared to air pathway dose. Surface water users were assumed in the LNP MACCS2 analysis for evaluation purposes. The primary sources of public water supply in the region of the LNP site are underground aquifers, as discussed in ER Section 2.3. MACCS2 does not model groundwater pathways (for example, aquifers).

Site-specific land use and economic parameters for the region were developed using the 2002 Census of Agriculture, Bureau of Labor Statistics, and Bureau of Economic Analysis data (References 7.2-001, 7.2-002, and 7.2-003). The economic parameters used in the MACCS2 analysis are provided in Table 7.2-2. Economic values were escalated to December 2007 values using the consumer price index approach, as applicable. Specific land use and economic parameters used in the MACCS2 analysis for the 11 counties in the region are summarized in Table 7.2-3. Additional land use information is provided in ER Section 2.2.

The EARLY file includes input specific to the early time phase (1 week) after an accident, which is used to calculate early dose exposure and health effects. Protective action considerations are included in the input file using LNP site-specific inputs. Protective action considerations include the evacuation time estimates for the 16-km (10-mi.) emergency planning zone. Based on the evacuation time estimate study, an average evacuation speed of 1.1 meters per second (2.5 miles per hour) was modeled, with evacuation movement beginning 85 minutes following the declaration of general emergency by the site. Shielding and exposure factors are those used for Grand Gulf (provided in MACCS2 Sample Problem A). Ninety-five percent of the population was assumed to evacuate following the declaration of a general emergency.

The CHRONC file includes input specific to the long-term consequences of an accident. Input parameters in the CHRONC file are used to calculate long-term dose and health effect estimates, as well as off-site economic cost estimates associated with interdiction, decontamination, and land condemnation. LNP site-specific input includes updating generic economic cost inputs to the December 2007 value using the consumer price index approach, as well as calculating LNP site-specific farm and non-farm wealth values based on the 2002 Census of Agriculture, Bureau of Labor Statistics, and Bureau of Economic Analysis data (References 7.2-001, 7.2-002, and 7.2-003), which are also updated to December 2007 values.

7.2.3.2 LNP MACCS2 Results

The results of the LNP MACCS2 calculation and AP1000 internal event accident frequencies are used to calculate the risk from a severe accident for the region surrounding the LNP site. The risk is calculated as the product of the individual accident class frequency multiplied by the MACCS2 consequence associated with that accident class, such that the overall result represents the frequency weighted risk for the metric of interest (for example, population dose risk, early fatality risk, latent cancer fatality risk, and cost risk) caused by internal events.

The LNP MACCS2 summary results are provided in Table 7.2-4. The results associated with each accident category are provided in Tables 7.2-5 and 7.2-6. The results presented incorporate a variety of contributors such as evacuation costs, value of crops contaminated and condemned, value of milk contaminated and condemned, cost of decontamination of property, and indirect costs resulting from loss of use of the property and incomes derived as a result of the accident. Discussion of the results is presented in the following subsections.

Table 7.2-7 presents the average individual risk for early fatalities and latent cancer fatalities from severe accidents associated with the LNP. Table 7.2-8 presents the average individual risk of early fatality and latent cancer fatality as compared to the NRC safety goal numerical objectives and demonstrates that the LNP results are many orders of magnitude below the numerical objectives.

The following subsections address the MACCS2 analysis results as related to individual dose pathways of interest, specifically the atmospheric, surface water (fallout onto open bodies of water), and groundwater pathways.

7.2.3.3 MACCS2 Analysis Results for Atmospheric Pathway

Table 7.2-4 presents the population dose risk of 5.62E-04 person-Sieverts per year (person-Sv/yr) (5.62E-02 person-roentgen equivalent man per year [person-rem/yr]) calculated by MACCS2 for all pathways considered in MACCS2. The atmospheric pathway dose, however, is a large portion of the total population dose, so the total population dose is used here to conservatively represent the atmospheric dose risk.

The LNP MACCS2 population dose result is compared to the total population dose risk results of other studies in Table 7.2-9 (based on internal events). As can be seen, the population dose risk for the AP1000 at the LNP site is significantly lower than current design reactors. It is noted that the LNP population dose risk is slightly larger than that listed in the DCD for a generic AP1000 site. This is attributed to the fact that the AP1000 generic analysis is based on the 24-hour dose, while the LNP MACCS2 analysis (as well as the other studies) includes long-term dose contributors.

Based on the previous discussion, the consideration of the atmospheric pathway can be concluded to be a SMALL impact.

7.2.3.4 MACCS2 Analysis Results for Fallout onto Open Bodies of Water

Following a severe accident, a radiation hazard may exist from the deposition of airborne, radiological fallout onto open bodies of water. Depending on the type of water body, this hazard may lead to internal exposure from the ingestion of contaminated water or from consuming contaminated aquatic fauna. External exposure may result from swimming in the contaminated water or from recreational activities on the shoreline. The extent of the hazard is largely determined by the proximity of individuals to the reactor, the extent of contamination, and the ability for interdiction to reduce the exposure hazard. Of these various water-related pathways, MACCS2 calculates only the dose from drinking water.

As presented in Table 7.2-4, the LNP MACCS2 total population dose risk is 5.62E-04 person-Sv/yr (5.62E-02 person-rem/yr). The MACCS2 portion derived from drinking water is 1.39E-05 person-Sv/yr (1.39E-03 person-rem/yr), which is less than 3 percent of the total population dose. This is judged to represent a very SMALL impact. As previously noted, there are almost no surface water (drinking water) users in the 80-km (50-mi.) region of the LNP site. Therefore, this MACCS2 calculated drinking water dose is judged to be conservative for the LNP site.

Although the other surface water pathways (for example, consuming aquatic fauna and swimming) are not modeled by MACCS2, they have been previously evaluated in NUREG-1437 and are shown to be of SMALL significance for most sites, especially if interdiction is considered. For river sites these other surface water pathways have been found to result in uninterdicted population doses that are orders of magnitude lower than the atmospheric pathway. For coastal and estuary sites with large annual aquatic harvests, interdiction can provide dose reductions such that the population dose is essentially the same as the atmospheric pathway which is considered a SMALL impact. Because the LNP site is approximately 12.8 km (7.9 mi.) from the Gulf of Mexico, the dose associated with these other surface water pathways would be expected to fall between the typical river site and the typical coastal site. Coastal impacts from radiological releases associated with the LNP site would be expected to be less than other sites located on the coastline due to radiological deposition on the land prior to reaching the coastal environment. Thus, consideration of atmospheric fallout onto open bodies of water can be concluded to be a SMALL impact for the LNP site.

7.2.3.5 MACCS2 Analysis Results for Groundwater Pathways

Individuals can also receive a radiation exposure from groundwater pathways. For the groundwater pathway, the core is postulated to melt down, breach the reactor vessel, and fall onto the reactor building floor. As a result of chemical energy and decay heat, the melted fuel reacts with the concrete floor. Without the cooling water addition to the core debris, the basemat of the containment building may eventually breach, and molten core debris and radioactive water penetrate strata beneath the plant. The soluble radionuclides in the debris can be leached and transported with groundwater and contaminated reactor water to downgradient domestic wells used for drinking water or to surface water bodies used for drinking water, aquatic food, and recreation.

Groundwater pathways are not modeled by MACCS2. The AP1000 design has intentionally included design elements to minimize the potential for a severe accident to lead to core concrete interactions and an eventual breach of the containment building basemat. These design elements include in-vessel retention of core debris by external reactor vessel cooling (submerging the reactor vessel in water to facilitate cooling and thereby prevent vessel failure) and ex-vessel core debris cooling in the reactor cavity (providing a water-filled reactor cavity to receive core debris upon vessel failure). These design elements are discussed in more detail in the DCD. Should such a postulated event occur, however, it is noted that the general groundwater flow direction from the LNP site is west-southwest (see ER Subsection 2.3.1.5.4) toward the Gulf of Mexico. This flow direction would generally be expected to minimize the potential population impacts.

As identified in NUREG-1437, groundwater contamination caused by severe accidents has been evaluated generically in NUREG-0440, the Liquid Pathway Generic Study (LPGS). The LPGS evaluates the consequences, assuming that core melt with subsequent basemat melt-through occurs. The LPGS examines six generic sites, each representing a different site category, using typical or comparative assumptions on geology and adsorption factors. As noted in NUREG-1437, "The LPGS results are believed to provide generally conservative uninterdicted population dose estimates in the six generic plant-site categories. Five of these categories are site groupings in common locations adjacent to small rivers, large rivers, the Great Lakes, oceans, and estuaries . . . The sixth category is a 'dry' site located at either a considerable distance from surface water or where groundwater flow is away from a nearby surface water body." These generic analyses are judged to adequately bound the potential groundwater impacts associated with the LNP site, especially when the AP1000 specific design elements are considered.

According to NUREG-0440, the generic liquid pathway uninterdicted dose estimates are one or more orders of magnitude lower than those attributed to the atmospheric pathway. NUREG-1437 concludes that the risk from the groundwater exposure pathway generally contributes only a small fraction of that risk attributable to the population from the atmospheric pathway, but in a few cases, may contribute a comparable risk.

Because the atmospheric pathway has been found to have a SMALL impact, even a groundwater pathway of comparable risk would be considered a SMALL impact. Therefore, the consideration of groundwater pathways at the LNP site can be concluded to be a SMALL impact.

7.2.3.6 External Event Risk

The LNP MACCS2 results previously presented are based on internal events, consistent with the Level 3 risk results presented in the DCD. The DCD, however, does present the AP1000 CDFs associated with external events and internal flooding, as summarized in Table 7.2-10.

The combined internal flood and internal fire CDF contributions are only approximately 24 percent of the internal events CDF. Because the seismic CDF is not quantified for the AP1000, it was not evaluated quantitatively as a contributor.

To generically evaluate the potential risk impacts associated with internal flooding, fire, seismic, and any other external events (for example, hurricane risk), the internal events CDF may be multiplied by a factor of two, and the assumption made that the release category frequency proportions remain the same. Using these assumptions, the population dose risk for all at-power events would be 1.12E-03 person-Sv/yr (1.12E-01 person-rem/yr); that is, twice that calculated for internal events alone. This value is still very small and is significantly less than the risk associated with only internal events of current plant designs (presented in Table 7.2-9). Therefore, external event risk is judged to be acceptable.

7.2.3.7 Cumulative Risk

The LNP MACCS2 analysis examines the risk caused by internal events associated with a single AP1000 plant. It is noted that Florida Power Corporation doing business as Progress Energy Florida, Inc. (PEF) proposes constructing two AP1000 units at the LNP site. In consideration of the two units located on the LNP site, the cumulative population dose risk may be estimated by summing the individual dose risk associated with each unit, as provided in Table 7.2-11.

Table 7.2-11 demonstrates that the cumulative risk of constructing two new advanced AP1000 reactors at the LNP site are significantly lower than the risk associated with current plant designs listed in Table 7.2-9.

7.2.3.8 Comparison to Normal and Operational Releases

NUREG-1555 encourages the comparison of severe accident radiological release doses with those associated with normal and anticipated releases documented in ER Subsection 5.4.3. The calculated total body 80-km (50-mi.) population annual doses (using year 2020 population data) from the gaseous and liquid pathways total 6.15E-02 person-Sv/yr (6.15 person-rem/yr) based on Table 5.4-11. The MACCS2 calculated LNP severe accident 80-km (50-mi.) population dose risk (using the larger year 2060 population data) of 5.62E-04 person-Sv/yr (5.62E-2 person-rem/yr) is significantly less than the annual dose estimated for normal releases.

7.2.3.9 MACCS2 Analysis Conclusions

The LNP MACCS2 analysis of severe accidents for the AP1000 reactor design shows that the 80-km (50-mi.) population dose risk of 5.62E-04 person-Sv/yr (5.62E-02 person-rem/yr) is significantly lower than that for current reactor designs. Additionally, the severe accident risk is many orders of magnitude below the NRC safety goal numerical objectives.

This population dose is primarily attributable to the atmospheric pathway. MACCS2 does not specifically calculate population dose resulting from radioactive fallout onto open bodies of water except for doses associated with drinking water (external exposure from recreational activities like swimming in contaminated water or from consuming contaminated aquatic fauna is not calculated). The MACCS2 population dose derived from drinking water is less than 3 percent of the total population dose.

Due to the low severe accident risk, the potential impacts to the environment are determined to be of SMALL significance utilizing the NRC criteria.

Other metrics of interest, including early fatality risk, latent cancer fatality risk, affected land, and cost risk are presented. The calculated cost risk value of $408 per year is used in ER Section 7.3 for the SAMA analysis.

7.2.4 CONCLUSIONS

The LNP site-specific MACCS2 analysis of severe accidents for the AP1000 reactor design shows that the 80-km (50-mi.) population dose risk of 5.62E-04 person-Sv/yr (5.62E-02 person-rem/yr) is significantly lower than that for current reactor designs. Additionally, the severe accident risk is many orders of magnitude below the NRC safety goal numerical objectives. Due to the low severe accident risk, the potential impacts to the environment are determined to be of SMALL significance utilizing the NRC criteria.

7.2.5 REFERENCES

7.2-001 U.S. Department of Agriculture, “2002 Census of Agriculture, Florida State and County Data,” AC-02-A-9, Vol. 1, Part 9, June 2004.

7.2-002 U.S. Department of Labor, Bureau of Labor Statistics, “Consumer Price Index – All Urban Consumers, “ Website, www.bls.gov/data/, accessed February 11, 2008.

7.2-003 Bureau of Economic Analysis, Regional Economic Accounts, “Local Area Personal Income,” Website, www.bea.gov/bea/regional/reis/, accessed August 15, 2008.

Table 7.2-1
LNP MACCS2 Source Term Release Fractions

Release Category	Plume	Xe/Kr	I	Cs	Te	Sr	Ru	La	Ce	Ba
Intermediate Containment Failure	1	5.40E-1	3.19E-3	3.18E-3	4.18E-4	2.11E-2	9.11E-3	3.53E-3	2.64E-5	1.62E-2
	2	2.58E-1	1.35E-4	1.35E-4	1.67E-5	6.50E-4	1.68E-4	4.53E-3	1.68E-5	3.40E-4
	3	1.22E-1	0.00E+0	0.00E+0	6.04E-6	0.00E+0	0.00E+0	1.12E-2	4.06E-5	0.00E+0
Early Containment Failure	1	4.16E-1	5.53E-2	5.37E-2	1.23E-3	3.14E-3	1.16E-2	5.57E-5	9.54E-7	4.63E-3
	2	4.05E-1	1.26E-3	1.21E-3	1.61E-4	3.43E-4	2.58E-3	9.66E-6	4.56E-8	6.45E-4
	3	1.42E-1	0.00E+0	0.00E+0	6.04E-7	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Intact Containment	1	9.83E-4	1.20E-5	1.15E-5	8.04E-7	1.07E-5	1.31E-5	1.35E-6	5.85E-9	1.20E-5
	2	4.93E-4	0.00E+0	0.00E+0	4.83E-9	0.00E+0	0.00E+0	6.00E-9	3.20E-11	0.00E+0
	3	1.17E-3	0.00E+0	0.00E+0	1.81E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Containment Bypass	1	1.00E+0	1.69E-1	1.62E-1	6.27E-3	3.57E-3	4.48E-2	1.30E-4	3.19E-6	8.93E-3
	2	0.00E+0	4.64E-2	3.38E-2	3.12E-3	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.00E-6
	3	0.00E+0	2.34E-1	7.60E-2	6.89E-3	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.00E-6
Containment Isolation Failure	1	5.73E-1	4.56E-2	2.10E-2	1.64E-3	2.03E-2	4.04E-2	2.39E-4	2.97E-6	3.16E-2
	2	1.13E-1	0.00E+0	0.00E+0	1.15E-5	0.00E+0	0.00E+0	1.00E-7	0.00E+0	0.00E+0
	3	8.40E-2	0.00E+0	0.00E+0	9.37E-5	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Late Containment Failure	1	3.36E-4	1.20E-5	1.15E-5	1.00E-6	1.57E-5	1.68E-5	9.96E-7	7.41E-9	1.61E-5
	2	1.19E-3	5.00E-8	3.23E-8	1.75E-8	1.04E-6	2.90E-7	1.07E-5	4.05E-8	6.60E-7
	3	9.79E-1	2.13E-5	1.19E-5	3.67E-5	2.83E-3	1.42E-3	1.41E-1	5.34E-4	2.60E-3

Table 7.2-2
LNP MACCS2 Economic Parameters Inputs

Variable	Description	Base Case Value
DPRATE [a]	Property depreciation rate (per year)	0.2
DSRATE [a]	Investment rate of return (per year)	0.12
EVACST [b]	Daily cost for a person who has been evacuated (dollars per person per day)	\$51.73
POPCST [b]	Population relocation cost (dollars per person)	\$9580
RELCST [b]	Daily cost for a person who is relocated (dollars per person per day)	\$51.73
CDFRM0 [b] [c]	Cost of farm decontamination for various levels of decontamination (dollars per hectare of land)	\$1078 \$2395
TIMDEC [a] [c]	Decontamination time for each level	2 months 4 months
CDNFRM [b] [c]	Cost of non-farm decontamination per resident person for various levels of decontamination (dollars per person)	\$5748 \$15,328
DLBCST [b]	Average cost of decontamination labor (dollars per worker per year)	\$67,062
TFWKNF [a]	Time workers spend in contaminated areas	$^{1}/_{3}$ total time
VALWF0 [d]	Weighted average value of farm wealth (dollars per hectare of land)	\$9185
VALWNF [d]	Weighted average value of non-farm wealth (dollars per person)	\$189,167

Notes:

a) Based on NUREG/CR-4551 values.

b) These parameters for the LNP base case use the NUREG/CR-4551 value updated to 2007 using the consumer price index (U.S. Department of Labor, Bureau of Labor Statistics).

c) Two decontamination levels are modeled. The first base case value is associated with a dose reduction factor of 3. The second base case value is associated with a dose reduction factor of 15.

d) Developed from LNP county-specific values, updated to 2007 using the consumer price index (U.S. Department of Labor, Bureau of Labor Statistics).

Table 7.2-3
County-Specific Land Use and Economic Parameters

County	Fraction Farm	Fraction Dairy	Farm Sales (per Hectare of Land)	Property Value (per Hectare of Land)	Non-Farm Property Value (per Person)
Alachua	0.40	0.095	$651	$9407	$215,844
Citrus	0.13	0.249	$346	$8192	$186,411
Dixie	0.07	0.318	$517	$5471	$134,903
Gilchrist	0.37	0.725	$1351	$5400	$166,789
Hernando	0.21	0.283	$821	$11,682	$183,958
Lake	0.30	0.020	$2440	$12,911	$206,033
Levy	0.25	0.461	$1140	$5267	$156,978
Marion	0.27	0.097	$799	$13,905	$191,316
Pasco	0.35	0.123	$1233	$11,921	$186,411
Putnam	0.20	0.123	$1245	$6830	$154,525
Sumter	0.54	0.098	$404	$6967	$154,525

Table 7.2-4
LNP MACCS2 Results
(0 to 80-Km [50-Mi.] Radius, Internal Events Only)

Plant Design	80-Km (50-Mi.) Dose Risk (Person-Sv/yr [Person-rem/yr])	Cost Risk (Dollars per Year)	Fatality Risk (per Year)	
			Early	Latent Cancer
AP1000	5.62E-04 (5.62E-02)	$408	2.39E-09	2.84E-05

Notes:

person-rem/yr = person-roentgen equivalent man per year
person-Sv/yr = person-Sieverts per year

Table 7.2-5
LNP MACCS2 Consequence Results by Source Term
(0 to 80-Km [50-Mi.] Radius, Internal Events Only)

Source Term	Frequency (per Year)	Dose (Person-Sv [Person-rem])	Dose Risk (Person-Sv/yr [Person-rem/yr])	Early Fatalities	Early Fatality Risk (per Year)	Latent Cancer Fatalities	Latent Cancer Fatality Risk (per Year)	Total Cost (Dollars)	Cost Risk (Dollars per Year)
ST1 – CFI	1.89E-10	1.56E+04 (1.56E+06)	2.95E-06 (2.95E-04)	1.56E-02	2.95E-12	6.73E+02	1.27E-07	$8.83E+09	$1.67E+00
ST2 – CFE	7.47E-09	1.65E+04 (1.65E+06)	1.23E-04 (1.23E-02)	1.48E-01	1.11E-09	8.71E+02	6.51E-06	$9.12E+09	$6.81E+01
ST3 – IC	2.21E-07	4.50E+01 (4.50E+03)	9.95E-06 (9.95E-04)	0.00E+00	0.00E+00	2.02E+00	4.46E-07	$1.39E+06	$3.07E-01
ST4 – BP	1.05E-08	3.87E+04 (3.87E+06)	4.06E-04 (4.06E-02)	1.20E-01	1.26E-09	1.91E+03	2.01E-05	$3.13E+10	$3.29E+02
ST5 – CI	1.33E-09	1.50E+04 (1.50E+06)	2.00E-05 (2.00E-03)	1.31E-02	1.74E-11	9.85E+02	1.31E-06	$7.25E+09	$9.64E+00
ST6 – CFL	3.45E-13	9.19E+03 (9.19E+05)	3.17E-09 (3.17E-07)	0.00E+00	0.00E+00	2.96E+02	1.02E-10	$1.81E+10	$6.24E-03
Total	**2.41E-07**	**--**	**5.62E-04 (5.62E-02)**	**--**	**2.39E-09**	**--**	**2.84E-05**	**--**	**$4.08E+02**

Notes:

BP = Containment Bypass
CFE = Early Containment Failure
CFI = Intermediate Containment Failure
CFL = Late Containment Failure
CI = Containment Isolation Failure
IC = Intact Containment
person-rem/yr = person-roentgen equivalent man per year
person-Sv/yr = person-Sieverts per year
ST = Source Term

Table 7.2-6
Affected Land Results by Source Term
(0 to 80-Km [50-Mi.] Radius)

Source Term	Decontaminated Land (Hectares)	Condemned Land (Hectares)
ST1 – CFI	13,400	1540
ST2 – CFE	14,700	662
ST3 – IC	3	0
ST4 – BP	48,500	2650
ST5 – CI	10,100	1410
ST6 – CFL	31,700	252
Worst Case	**48,500**	**2650**

Notes:

BP = Containment Bypass
CFE = Early Containment Failure
CFI = Intermediate Containment Failure
CFL = Late Containment Failure
CI = Containment Isolation Failure
IC = Intact Containment
ST = Source Term

Table 7.2-7
LNP AP1000
Average Individual Risk of Early Fatality and Latent Cancer Fatality

Source Term	Frequency (per Year)	Population Weighted Early Fatalities (1.6 Km [1 Mi.])	Frequency Weighted Early Fatality Risk	Early Fatality Contribution (%)	Population Weighted Latent Cancer Fatalities (16 Km [10 Mi.])	Frequency Weighted Latent Cancer Fatality Risk	Latent Cancer Fatality Contribution (%)
ST1 – CFI	1.89E-10	9.30E-05	1.76E-14	0.14	1.92E-03	3.63E-13	0.94
ST2 – CFE	7.47E-09	8.48E-04	6.33E-12	51.83	2.73E-03	2.04E-11	52.63
ST3 – IC	2.21E-07	0.00E+00	0.00E+00	0.00	1.13E-05	2.50E-12	6.45
ST4 – BP	1.05E-08	5.49E-04	5.76E-12	47.16	9.89E-04	1.04E-11	26.80
ST5 – CI	1.33E-09	7.98E-05	1.06E-13	0.87	3.84E-03	5.11E-12	13.18
ST6 – CFL	3.45E-13	0.00E+00	0.00E+00	0.00	1.17E-04	4.04E-17	0.00
Total	**2.41E-07**	--	**1.22E-11**	--	--	**3.87E-11**	--

Notes:

BP = Containment Bypass
CFE = Early Containment Failure
CFI = Intermediate Containment Failure
CFL = Late Containment Failure
CI = Containment Isolation Failure
IC = Intact Containment
ST = Source Term

Table 7.2-8
Comparison between the Average Individual Risk and the Safety Goal

Consequence Metric	LNP MACCS2 Results [a]	Safety Goal
Early Fatalities [b]	1.2E-11	< 5E-07 [d]
Latent Cancer Fatalities [c]	3.9E-11	< 2E-06 [d]

Notes:

a) Frequency weighted for each source term (based on internal events only).

b) Population weighted early fatality risk within 1.6 km (1 mi.) includes evacuation.

c) Population weighted latent cancer fatality risk within 16 km (10 mi.) includes evacuation.

d) Individual and societal risk consequence goals are based on the NRC safety goal policy statement and developed into numerical goals by the NRC staff in NUREG-1811, Volume 1, December 2006.

Table 7.2-9
Mean Annual Dose Risk for Several Sites
(Internal Events Only)

Plant	Population Dose Risk (80-Km [50-Mi.]) (Person-Sv/yr [Person-rem/yr])
LNP AP1000 [(a)]	5.62E-04 (5.62E-02) [(a)]
Zion	5.47E-01 (5.47E-01) [(b)]
Grand Gulf	5.20E-03 (5.20E-01) [(c)]
Surry	5.80E-02 (5.80E-00) [(d)]
DCD AP1000	4.32E-04 (4.32E-02) [(e)]

Notes:

a) Located at the LNP site.

b) Table 5.1-1 in NUREG/CR-4551, Volume 7, Revision 1.

c) Table 5.1-1 in NUREG/CR-4551, Volume 6, Revision 1.

d) Table 5.1-1 in NUREG/CR-4551, Volume 3, Revision 1.

e) Table 1B-1 in the DCD, located at a generic site, 24-hour emergency phase dose only.

person-rem/yr = person-roentgen equivalent man per year
person-Sv/yr = person-Sieverts per year

Table 7.2-10
AP1000 PRA CDF Results

Events	Core Damage Frequency (per Year) (At-Power)
Internal Events	2.41E-7
Internal Flood	8.82E-10
Internal Fire	5.61E-8
Seismic	-- [a]
Total	2.97E-7

Notes:

Data are based on Table 19.59-15 of the DCD.

a) Seismic risk CDF is not quantified for the AP1000. The seismic margin method was used.

CDF = core damage frequency
PRA = probabilistic risk assessment

Table 7.2-11
Mean Annual Cumulative Dose Risk
(Internal Events Only)

Plant	Population Dose Risk (80-Km [50-Mi.]) (Person-Sv/yr [Person-rem/yr])
LNP 1 (AP1000) [(a)]	5.62E-04 (5.62E-02)
LNP 2 (AP1000) [(a)]	5.62E-04 (5.62E-02)
Total	**1.12E-03 (1.12E-01)**

Notes:

a) Proposed Levy Nuclear Plant Unit 1 (LNP 1) and proposed Levy Nuclear Plant Unit 2 (LNP 2) located at the LNP site.

person-rem/yr = person-roentgen equivalent man per year
person-Sv/yr = person-Sieverts per year

7.3 SEVERE ACCIDENT MITIGATION MEASURES

A severe accident mitigation design alternative (SAMDA) evaluation was performed for the AP1000 plant design and is presented in the DCD, Appendix 1B. The evaluation was performed to identify potential safety beneficial design alternatives and to evaluate whether the safety benefit of the alternative design candidates outweighs the costs associated with implementation. Because the AP1000 is an advanced reactor design that incorporates many safety features, the SAMDA analysis did not find any additional design alternatives to be cost beneficial. The AP1000 SAMDA analysis was based on data representing a generic site.

This section updates the Westinghouse SAMDA analysis based upon the LNP site-specific MACCS2 model results presented in ER Section 7.2 to determine if the DCD conclusions remain valid (that is, none of the identified design alternatives are cost beneficial).

7.3.1 THE SAMA ANALYSIS PROCESS

Design or procedural modifications that could mitigate the consequences of a severe accident are known as SAMAs. In the past, SAMAs were known as SAMDAs, which primarily focused on design changes and did not consider procedural changes. The DCD analysis is a SAMDA analysis.

For an existing plant with a well-defined design and established procedural controls, the normal evaluation process for identifying potential SAMAs includes the following steps:

1. Define the Baseline — The plant’s PRA results are used to calculate the population dose risk and cost risk associated with severe accidents in the baseline plant configuration (that is, before implementation of any SAMAs). The NRC-approved methodologies are used to calculate the monetary value of unmitigated severe accident risk. This monetary value, sometimes termed the Maximum Averted Cost Risk (MACR), reflects the monetary value of eliminating all severe accident risk, and therefore, provides a conservative baseline screening value for the SAMA candidates.

2. Identify and Screen Potential SAMAs — Potential SAMA candidates are identified from the plant’s Individual Plant Examination, insights from the plant’s PRA, and the results of other plants’ SAMA analyses. A conservatively low implementation cost for each SAMA candidate is estimated based on historical costs, similar design changes, and engineering judgment. The estimated implementation costs are then compared against the baseline screening value MACR. SAMA candidates whose implementation cost exceeds the MACR can be screened and not evaluated further.

3. Develop Detailed Cost Benefit Estimates — For each SAMA remaining following the screening process, a less conservative, more detailed engineering cost estimate may be developed using current plant engineering processes. If a SAMA candidate's detailed cost estimate is below the MACR, the candidate is retained for further detailed evaluation. Alternatively, or in combination with a more detailed cost estimate, a less conservative, more detailed benefit estimate may be developed. For each SAMA remaining unscreened, the PRA model may be used to determine the risk reduction associated with implementation of the proposed SAMA. The benefit risk reduction is then monetized, and the cost benefit is evaluated. Those SAMA candidates that remain cost beneficial may be further evaluated for implementation considering other aspects such as operational disadvantages.

The scope of the plant PRA available is often limited to internal events. However, external events (for example, seismic events and fire events) have been identified by the nuclear industry as small, but non-negligible contributors to plant risk. SAMA assessments generally address the potential impact of external events through either their inclusion quantitatively (where frequency data is available), through quasiquantitative inclusion (for example, using a common multiplier factor on the internal event inputs or the MACR result), through sensitivity studies, qualitative assessment, or a combination of all of these.

7.3.2 AP1000 DCD SAMDA ANALYSIS

The AP1000 SAMDA evaluation is presented in Appendix 1B of the DCD. A list of SAMDA candidates was developed based on a review of SAMDAs evaluations for other plant designs, including the Westinghouse Electric Company, LLC, AP600 Reactor (AP600), and PRA results. Fifteen candidate design alternatives were selected for further evaluation for the AP1000 design. Table 7.3-1 (based on DCD Table 1B-5) identifies the 15 candidate design alternatives considered for the AP1000 and the estimated implementation costs for each. Additional discussion of each design alternative is presented in the DCD.

An evaluation of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative was conservatively maximized. As part of this process, it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the implementation cost estimate for each alternative was intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

Using the cost benefit calculation methodology of NUREG/BR-0184, the MACR was calculated using the dose risk and cost risk values developed for a generic site. The calculated MACR value was $21,000.

A comparison of the implementation costs for each SAMDA to the MACR value of $21,000 found that none of the SAMDAs would be cost effective. The least costly SAMDA, self-actuating containment isolation valves, had an implementation cost of approximately $33,000, with the others having costs at least an order of magnitude greater. The self-actuating containment isolation valve SAMDA candidate was further evaluated and found to result in minimal risk reduction achievement, thereby confirming its status as not cost beneficial.

7.3.3 LNP SAMA ANALYSIS

For the LNP site, the DCD SAMDA evaluation is reperformed incorporating the LNP MACCS2 analysis results to determine if the DCD conclusions remain valid.

The principal inputs to the baseline calculation are the internal events CDF (reported in ER Section 7.2), population dose risk and cost risk (reported in Table 7.2-4), exposure cost value ($2000/person-rem/yr, as provided in NUREG/BR-0184), licensing period (40 years), and economic discount rate (7 percent).

For the LNP analysis, the MACR value based on internal events was calculated to be approximately $13,000. To account for external events, this MACR value was multiplied by a factor of two to achieve an MACR value of $26,000. As discussed in ER Section 7.2 and presented in Table 7.2-10, the combined internal flood and internal fire CDF contributions are only approximately 24 percent of the internal events CDF. The seismic CDF is not quantified for the AP1000, and it was not evaluated quantitatively as a contributor. To generically evaluate the potential impacts associated with internal flooding, as well as all other external events (for example, fire, seismic, hurricane), a factor of two is applied to the MACR result, which is equivalent to applying a factor of two to the MACCS2 population dose risk and cost risk results. The MACR results are presented in Table 7.3-2, showing the various contributors.

The 15 SAMDA candidates identified in the DCD form an initial list of potential cost-beneficial plant modifications. In consideration of additional potential candidates for the LNP SAMA analysis, it is noted that the NRC previously evaluated additional potential design candidates for the AP1000 SAMDA, as documented in NUREG-1793, including those candidates evaluated for the AP600 which might have applicability to the AP1000. NUREG-1793 states that “the staff’s review of the more than 120 candidate design alternatives considered for the AP600 did not identify any new alternatives more likely to be cost-beneficial than those included in the AP1000 design alternative evaluations.” Regarding the NRC review of the AP1000 candidates, NUREG-1793 states that “the staff’s review did not reveal any additional design alternatives that obviously should have been given consideration by the applicant.” Based on the previous extensive review for additional design candidates, no new design candidates are identified.

In the absence of a completed plant with established procedural and administrative controls, the LNP analysis can only evaluate physical plant modifications. Evaluation of administrative SAMAs would not be appropriate until a plant design is finalized, and plant administrative processes and procedures are being developed. At that time, appropriate administrative controls on plant operations will be incorporated into the plant's management systems as part of its baseline. PEF will consider "risk insight" when developing SAMA procedures and will implement them prior to initial fuel load.

The implementation cost estimates developed by Westinghouse for the AP1000 SAMA candidates have been reviewed by the NRC for reasonableness, including comparisons with cost estimates developed for other plant designs, such as the advanced boiling water reactor and combustion engineering System 80+, as documented in NUREG-1793. The NRC concluded that the approximate cost estimates developed by Westinghouse are adequate for the purposes of the cost-benefit evaluation. Therefore, no implementation cost estimate revisions are required for the LNP SAMA analysis.

When the LNP site MACR is compared to the implementation costs of the AP1000 SAMDA candidate design alternatives presented in Table 7.3-1, no candidates are found to be potentially cost beneficial. Only one alternative is reasonably close to being potentially cost beneficial. Alternative 3 (self-actuating containment isolation valves) has a cost of $33,000, which is near the MACR value of $26,000. The remaining alternatives are over an order of magnitude more costly (the next lowest cost alternative is Alternative 14, a more reliable diverse actuation system, with an estimated implementation cost of $470,000). Because Alternative 3 is reasonably close to the MACR value and some sensitivity cases (presented below) show this design alternative could be cost beneficial if certain assumptions were revised, it is further evaluated.

The DCD further examines Alternative 3 and notes that this alternative provides almost no benefit in reducing the plant CDF, and the benefit related to release can be estimated by assuming the modification eliminated all the CI release category. Using these assumptions, the DCD finds that the benefit is of the order of a few thousand dollars, and therefore, is not cost beneficial. The LNP MACCS2 analysis (Table 7.2-5) shows that the CI release category contributes less than 4 percent to the total population dose risk and cost risk, such that there would be a negligible quantified benefit. The LNP MACCS2 analysis thus confirms the DCD conclusions that this SAMA candidate is not cost beneficial.

A number of SAMA sensitivity cases were examined to assess the impact of key inputs and assumptions. The results of the sensitivity cases are presented in Table 7.3-3. The sensitivity cases examined are similar to those conducted in the AP1000 SAMDA. The results indicate that there is significant margin in the conclusions of the SAMA analysis and that none of the SAMA candidates are cost beneficial for the AP1000 plant located at the LNP site.

7.3.4 CONCLUSIONS

For the LNP site, the AP1000 SAMDA evaluation has been reperformed, incorporating the LNP MACCS2 analysis results. The results showed that the DCD conclusions remain valid. No SAMA candidates are found to be cost beneficial.

This conclusion is consistent with the NRC AP1000 SAMDA review conclusions presented in NUREG-1793, which states the following:

> The staff concurs with the applicant's conclusion that none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. It further concluded that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations because the estimated CDFs would remain very low on an absolute scale.

Table 7.3-1
AP1000 SAMDA Candidate Design Alternatives

Number	Design Alternative	Implementation Cost
1	Upgrade Chemical, Volume, and Control System for Small LOCA	$1,500,000
2	Containment-Filtered Vent	$5,000,000
3	Self-Actuating Containment Isolation Valves	$33,000
4	Safety Grade Passive Containment Spray	$3,900,000
5	Active High Pressure Safety Injection System	N/A - (Not consistent with passive system design objectives)
6	SG Shellside Heat Removal	$1,300,000
7	SG Relief Flow to IRWST	$620,000
8	Increased SG Pressure Capability	$8,200,000
9	Secondary Containment Ventilation with Filtration	$2,200,000
10	Diverse IRWST Injection Valves	$570,000
11	Diverse Containment Recirculation Valves	N/A - (Already implemented in the AP1000 design)
12	Ex-Vessel Core Catcher	$1,660,000
13	High-Pressure Containment Design	$50,000,000
14	More Reliable Diverse Actuation System	$470,000
15	Locate Residual Heat Removal System Inside Containment	N/A - (Negligible achievable risk reduction)

Notes:

IRWST = in-containment refueling water storage tank
LOCA = loss of coolant accident
N/A = not applicable
SAMDA = severe accident mitigation design alternative
SG = steam generator

Table 7.3-2
LNP SAMA Baseline Costs

Off-Site Exposure Cost	$861
Off-Site Economic Cost	$3126
On-Site Exposure Cost	$124
On-Site Cleanup Cost	$3557
Replacement Power Cost	$5046
Summed Cost (Based on Internal Events)	$12,716
Total Cost [(a)]	$26,000

Notes:

a) Total cost = summed cost multiplied by 2 to account for external events and rounded up to the next thousand dollar value

SAMA = severe accident mitigation alternative

Table 7.3-3
Cost Benefit Sensitivity Results

	Case Studied	Cost (Dollars)
Base Case	7-Percent Discount Rate	$26,000
S-1	3-Percent Discount Rate	$51,000
S-2	High Dose (10 Multiplied by the Base Case)	$42,000
S-3	50-Percent CDF	$13,000
S-4	Twice the Base CDF	$51,000
S-5	10 Times the Benefit (10 Multiplied by MACR)	$255,000

Notes:

CDF = core damage frequency
MACR = Maximum Averted Cost Risk

7.4 TRANSPORTATION ACCIDENTS

The advanced light water reactor (ALWR) technology being considered for the LNP and the alternative sites (Crystal River Energy Complex [CREC], Dixie, Putnam, and Highlands [refer to ER Subsection 9.3.2]) is the AP1000. The configuration for this new nuclear power generating facility is two units. A single AP1000 unit was used to evaluate transportation impacts in ER Section 3.8 and the accidents from transportation in this section relative to the reference light water reactor (LWR) in WASH-1238.

Subparagraphs 10 CFR 51.52(a)(1) through (5) delineate specific conditions the reactor licensee must meet to use Table S-4 (reproduced in this ER as Table 3.8-1) as part of its ER. For reactors not meeting all of the conditions in paragraph (a) of 10 CFR 51.52, paragraph (b) of 10 CFR 51.52 requires a further analysis of the transportation effects.

The conditions in paragraph (a) of 10 CFR 51.52 establishing the applicability of Table S-4 are reactor core thermal power, fuel form, fuel enrichment, fuel encapsulation, average fuel irradiation, time after discharge of irradiated fuel before shipment, mode of transport for unirradiated fuel, mode of transport for irradiated fuel, radioactive waste form and packaging, and mode of transport for radioactive waste other than irradiated fuel.

Based on comparison of the AP1000 characteristics to the criteria listed in 10 CFR 51.52(a), the AP1000 does not meet the following two evaluation criteria (as discussed in ER Subsections 3.8.1.3 and 3.8.1.5, respectively):

- Subparagraph 10 CFR 51.52(a)(2) requires that the reactor fuel have a uranium-235 (U-235) enrichment not exceeding 4 percent by weight. As noted in DCD Table 4.1-1, for the AP1000, the enrichment of the initial core, varies by region from 2.35 to 4.45 percent, and the average for reloads is 4.51 percent. The AP1000 fuel exceeds the 4 percent U-235 condition.

- Subparagraph 10 CFR 51.52(a)(3) requires that the average burnup not exceed 33,000 megawatt days per metric ton of uranium (MWd/MTU). According to the DCD, the AP1000 has an average maximum burnup of 60,000 MWd/MTU for the peak rod. The extended burnup is 62,000 MWd/MTU. Therefore, the AP1000 does not meet this subsequent evaluation condition.

Because the AP1000 does not meet all criteria set forth in Table S-4, a subsequent analysis was performed for the LNP and the alternative sites that is used as the supporting basis for ER Section 3.8 and this section.

ER Section 3.8 addresses issues associated with the transportation of radioactive materials from the LNP and alternative sites. This section addresses accidents associated with the shipment of unirradiated and spent fuel.

7.4.1 TRANSPORTATION OF UNIRRADIATED FUEL

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 CFR 51.52(a) (see Table 3.8-1). The consequences of accidents that are severe enough to result in a release of unirradiated particles to the environment from fuel for ALWR fuels are not significantly different from those for current generation LWRs. The fuel form, cladding, and packaging are similar to those LWRs analyzed in WASH-1238. Consequently, as described in the NRC's assessment of environmental impacts at the North Anna, Clinton, and Grand Gulf Early Site Permit (ESP) sites (NUREG-1811, NUREG-1815, and NUREG-1817, respectively), the NRC concluded that the overall transportation accident risks associated with advanced reactor spent fuel shipments are likely to be SMALL and are consistent with the risks associated with transportation of spent fuel from current generation reactor.

7.4.2 TRANSPORTATION OF SPENT FUEL

In its assessments of the proposed ESP sites, the NRC used the radioactive material transportation (RADTRAN) 5 computer code to estimate impacts of transportation accidents involving spent fuel shipments (Reference 7.4-001). As provided in Draft NUREG-1872, "RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences (e.g., "fender benders") to those with low frequencies and high consequences (i.e., accidents in which the shipping container is exposed to severe mechanical and thermal conditions)."

The NRC conducted a screening analysis on the inventories reported in an Idaho National Engineering and Environmental Laboratory document entitled, "Early Site Permit ER Sections and Supporting Documentation," to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations (Reference 7.4-002). The screening identified the radionuclides that would contribute more than 99.999 percent of the dose from inhalation and the results are reported in NUREG-1811, NUREG-1815, and NUREG-1817.

Radionuclide inventories are important parameters in the calculation of accident risks. The radionuclide inventories used in this analysis were taken directly from NUREG-1811, NUREG-1815, and NUREG-1817, with the exception of Cobalt-60 (Co-60), which is discussed below.

Co-60 inventories were taken directly from NUREG/CR-6672. The following discussion is from Section 7.2.3.5 of NUREG/CR-6672 and provides a discussion regarding the importance of including Co-60 in the overall source term:

> During reactor operation, corrosion products formed in the reactor's primary cooling system deposit on fuel assembly surfaces where elements in these deposits are activated by neutron bombardment. The resulting radioactive deposits are called CRUD. Due to vibratory loads

> during incident free transportation, impact loads during collision accidents, and thermal loads during accidents that lead to fires, portions of these radioactive deposits may spall from the rods. Then, if some of these spalled materials become airborne during an accident, their release to the atmosphere could contribute to the radiation exposures caused by the accident. Although CRUD contains a number of radionuclides, only Co-60 would contribute significantly to these radiation exposures. Since the CRUD deposits on typical [pressurized water reactor] PWR spent fuel rods typically contain 0.2 [Curies] Ci of Co-60 per rod and the generic PWR assemblies for which ORIGEN inventories were calculated contain respectively 289 spent fuel rods, the amounts of Co-60 produced by activation of deposits on assembly surfaces is 57.8 Ci for the generic PWR assembly (115.6 [Curies per metric ton of uranium] Ci/MTU based on 0.5 MTU/assembly).

The spent fuel inventory used in this analysis for the AP1000 is presented in Table 7.4-1.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71. Spent fuel shipping casks must be certified Type B packaging systems, meaning they must withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability. As noted in Draft NUREG-1872, "the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). The staff assumed that shipping casks for Westinghouse AP1000 reactor spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo."

The NRC performed the RADTRAN 5 accident risk calculations using unit radionuclide inventories (Ci/MTU) for the spent fuel shipments from the ALWRs. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (metric tons of uranium per year [MTU/yr]) to derive estimates of the annual accident risks associated with spent fuel shipments from each potential ALWR. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantities: 24 MTU/yr for the AP1000. This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2. Information on how these values were calculated is presented in ER Section 3.8.

In the NRC's assessment of the proposed ESP sites, the NRC used the release fractions for current generation LWR fuels to approximate the impacts from the ALWR spent fuel shipments. This assumed that the fuel materials and containment systems (cladding and fuel coatings) behave similarly to current LWR fuel under applied mechanical and thermal conditions. For this assessment, the same release fractions were used to approximate the impacts from the AP1000 reactor spent fuel shipments.

The shipping distances and population distribution information for the routes from the LNP and alternative sites were the same as those used for the "incident-free" transportation impacts analysis (described in ER Subsection 3.8.2).

Table 7.4-2 presents unit accident risks associated with transportation of spent fuel from the LNP and alternative sites to the proposed Yucca Mountain repository. The accident risks are provided in the form of a unit collective population dose (person-roentgen equivalent man [person-rem]). The table also presents estimates of accident risk per reference reactor year (RRY) normalized to the reference LWR analyzed in WASH-1238.

The estimated shipping distances from the LNP and alternative sites to the spent fuel disposal facility are presented in ER Section 3.8.

7.4.3 NONRADIOLOGICAL IMPACTS

Nonradiological impacts are calculated using accident, injury, and fatality rates from published sources. The rates (that is, impacts per vehicle-km traveled) are then multiplied by estimated travel distances for workers and materials. The general formula for calculating nonradiological impacts is as follows:

Impacts = (unit rate) x (round-trip shipping distance) x (annual number of shipments)

In this formula, impacts are presented in units of the number of accidents, number of injuries, and number of fatalities per year. Corresponding unit rates (impacts per vehicle-km traveled) are used in the calculations.

The general approach used in this document to calculate nonradiological impacts of unirradiated and spent fuel shipments is based on the approach used in the Yucca Mountain Supplemental Environmental Impact Statement, which used adjusted state-level accident, injury, and fatality statistics, as shown in Table 7.4-3 (References 7.4-003 and 7.4-004). The round-trip distances between the proposed advanced reactor sites and the fuel fabrication facility (assumed to be located in Columbia, South Carolina, and Lynchburg, Virginia) and Yucca Mountain, Nevada (Table 7.4-4) provided the data for the last part of the equation. State-by-state shipping distances were obtained from the Web-TRAGIS output file and combined with the annual number of shipments and accident, injury, and fatality rates by state (References 7.4-003 and 7.4-004), to calculate nonradiological impacts. The results are shown in Table 7.4-4. The values presented in Table 7.4-5 were calculated from the values reported in Table 7.4-4 multiplied by the applicable number of shipments for unirradiated and spent fuel. Table 7.4-5 values were then compared to those reported in Table S-4 of 10 CFR 51.52. It should be noted that because of the larger round-trip distances and greater number of shipments, the nonradiological impacts of shipping spent nuclear fuel comprise more than 95 percent of the total nonradiological impacts, which include the impacts associated with shipping unirradiated fuel.

7.4.4 CONCLUSION

Considering the uncertainties in the data and computational methods, the NRC concluded that the overall transportation accident risks associated with ALWR unirradiated and spent fuel shipments are likely to be SMALL and are consistent with the transportation risks from current generation reactors presented in Table S-4 of 10 CFR 51.52. The same conclusion is true of the transportation accident risks associated with the spent fuel from the proposed new reactors at the LNP site and the alternative sites.

7.4.5 REFERENCES

7.4-001 Neuhauser, K. S. and F. L. Kanipe, "RADTRAN 5 User Guide," Sandia National Laboratories, SAND2003-2354, July 2003.

7.4-002 Idaho National Engineering and Environmental Laboratory, "Early Site Permit ER Sections and Supporting Documentation," Engineering Design File Number 3747, July 2003.

7.4-003 Saricks, C.L. and M.M. Tompkins, "State-Level Accident Rates of Surface Freight Transportation: A Reexamination," Argonne National Laboratory, ANL/ESD/TM-150, April 1999.

7.4-004 Blower, Daniel and Anne Matteson, Center for National Truck Statistics, "Evaluation of the Motor Carrier Management Information System Crash File, Phase 1," UMTRI 2003-6, prepared for Federal Motor Carrier Safety Administration, March 2003.

Table 7.4-1
Radionuclide Inventory Used in Transportation Accident Risk Calculations for the AP1000

Radionuclide	AP1000 Inventory (Ci/MTU)
Am-241	7.27E+02
Am-242m	1.31E+01
Am-243	3.34E+01
Ce-144	8.87E+03
Cm-242	2.83E+01
Cm-243	3.07E+01
Cm-244	7.75E+03
Cm-245	1.21E+00
Cs-134	4.80E+04
Cs-137	9.31E+04
Co-60 [(a)]	1.20E+02
Eu-154	9.13E+03
Eu-155	4.62E+03
Pm-147	1.76E+04
Pu-238	6.07E+03
Pu-239	2.55E+02
Pu-240	5.43E+02
Pu-241	6.96E+04
Pu-242	1.82E+00
Ru-106	1.55E+04
Sb-125	3.83E+03
Sr-90	6.19E+04
Y-90	6.19E+04

Notes:

The “m” next to an isotope indicates a metastable state.

a) Co-60 is the key radionuclide constituent of fuel assembly crud.

Ci/MTU = Curies per metric ton uranium

Table 7.4-2
Spent Fuel Transportation Accident Risks for the AP1000

Site	Unit Population Dose (person-rem) [(a)]	Shipments per Year [(b)]	Population Dose (person-rem per RRY) [(c)]
LNP/CREC	2.27E-06	39	8.85E-05
Dixie	2.24E-06	39	8.74E-05
Highlands	2.32E-06	39	9.05E-05
Putnam	2.28E-06	39	8.89E-05
Table S-4	--	--	SMALL

Notes:

a) The inventory in RADTRAN calculations was adjusted for the 0.5 MTU per shipment.

b) Calculations are based on 39 normalized shipments per year.

c) Values are the product of unit population dose multiplied by normalized shipments per year.

person-rem = person-roentgen equivalent man
RRY = reference reactor year

Table 7.4-3 (Sheet 1 of 2)
Adjusted Accident, Injury, and Fatality Rates for the United States

	Accidents/Trucks (km)		Fatalities/Trucks (km)		Injuries/Trucks (km)	
State/Parameter	**Interstate**	**Total**	**Interstate**	**Total**	**Interstate**	**Total**
Alabama	4.63E-07	6.19E-07	1.35E-08	3.45E-08	1.78E-07	2.56E-07
Arizona	2.17E-07	1.76E-07	1.48E-08	1.48E-08	1.4E-07	1.1E-07
Arkansas	2.2E-07	2.43E-07	9.76E-09	3.5E-08	1.18E-07	1.49E-07
California	2.63E-07	1.36E-07	1.1E-08	5.67E-09	1.49E-07	7.68E-08
Colorado	7.32E-07	7.12E-07	1.8E-08	2.76E-08	3.78E-07	3.64E-07
Connecticut	1.48E-06	1.45E-06	2.28E-08	3.01E-08	7.36E-07	7.39E-07
Delaware	8.5E-07	1.19E-06	8.82E-09	3.7E-08	4.1E-07	6.13E-07
Florida	1.13E-07	1.46E-07	1.21E-08	1.69E-08	6.6E-08	8.52E-08
Georgia	N/A	1.1E-06	N/A	3.07E-08	N/A	5.51E-07
Idaho	4.84E-07	6.48E-07	5.98E-09	3.92E-08	3.68E-07	4.73E-07
Illinois	3.64E-07	4.86E-07	1.31E-08	1.73E-08	1.8E-07	1.97E-07
Indiana	3.69E-07	2.77E-07	1.06E-08	1.35E-08	1.68E-07	1.38E-07
Iowa	1.84E-07	2.43E-07	1.48E-08	2.11E-08	1.03E-07	1.36E-07
Kansas	4.66E-07	6.29E-07	8.19E-09	3.61E-08	3.05E-07	4.14E-07
Kentucky	5.09E-07	8.5E-07	2.02E-08	3.61E-08	2.65E-07	4.33E-07
Louisiana	N/A	3.63E-07	N/A	1.45E-08	N/A	2.21E-07
Maine	7.2E-07	6.76E-07	1.43E-08	1.23E-08	3.74E-07	4E-07
Maryland	8.86E-07	1.22E-06	1.02E-08	3.13E-08	5.51E-07	7.27E-07
Massachusetts	1.41E-07	2.54E-07	1.26E-09	5.98E-09	6.12E-08	1.25E-07
Michigan	4.64E-07	3.53E-07	1.69E-08	1.69E-08	3.13E-07	2.64E-07
Minnesota	2.81E-07	2.89E-07	4.72E-09	1.89E-08	1.01E-07	1.45E-07
Mississippi	7.88E-08	1.03E-07	3.94E-09	5.35E-09	4.68E-08	6.84E-08
Missouri	7.62E-07	8.8E-07	1.95E-08	3.1E-08	3.77E-07	4.38E-07
Montana	1.02E-06	9.54E-07	2.14E-08	3.2E-08	3.07E-07	3.1E-07
Nebraska	5.24E-07	7.12E-07	2.16E-08	2.95E-08	2.36E-07	3.11E-07
Nevada	3.69E-07	4.02E-07	1.04E-08	1.4E-08	1.78E-07	1.94E-07
New Hampshire	4.32E-07	6.25E-07	N/A	1.86E-08	1.96E-07	2.81E-07
New Jersey	9.27E-07	8.09E-07	1.91E-08	1.12E-08	4.69E-07	4.55E-07
New Mexico	1.85E-07	1.77E-07	1.86E-08	1.73E-08	1.38E-07	1.3E-07
New York	N/A	5.66E-07	N/A	1.95E-08	N/A	2.22E-07
North Carolina	5.68E-07	5.48E-07	2.35E-08	2.55E-08	3.8E-07	3.79E-07
North Dakota	4.96E-07	5.61E-07	1.61E-08	1.75E-08	2.27E-07	3.04E-07
Ohio	2.69E-07	1.9E-07	6.14E-09	6.14E-09	1.68E-07	1.28E-07
Oklahoma	4.4E-07	4.53E-07	2.09E-08	2.32E-08	3.47E-07	3.42E-07
Oregon	N/A	3.54E-07	N/A	3.21E-08	N/A	1.63E-07
Pennsylvania	8.44E-07	1.11E-06	2.13E-08	3.83E-08	4.6E-07	6.4E-07
Rhode Island	N/A	N/A	N/A	N/A	N/A	N/A
South Carolina	N/A	7.7E-07	N/A	4.09E-08	N/A	3.96E-07
South Dakota	3.82E-07	3.76E-07	9.61E-09	2E-08	2.06E-07	1.91E-07

Table 7.4-3 (Sheet 2 of 2)
Adjusted Accident, Injury, and Fatality Rates for the United States

State/Parameter	Accidents/Trucks (km) Interstate	Accidents/Trucks (km) Total	Fatalities/Trucks (km) Interstate	Fatalities/Trucks (km) Total	Injuries/Trucks (km) Interstate	Injuries/Trucks (km) Total
Tennessee	2.02E-07	2.61E-07	1.57E-08	2.05E-08	1.1E-07	1.52E-07
Texas	9.85E-07	1.08E-06	2.05E-08	4.25E-08	6.57E-07	6.45E-07
Utah	4.76E-07	5.58E-07	1.87E-08	2.19E-08	3.04E-07	3.41E-07
Vermont	3.09E-07	4.89E-07	N/A	1.53E-08	1.82E-07	2.64E-07
Virginia	6.45E-07	4.35E-07	2.54E-08	1.83E-08	3.72E-07	2.59E-07
Washington	4.35E-07	3.36E-07	2.83E-09	8.35E-09	2.16E-07	1.68E-07
West Virginia	2.82E-07	3.53E-07	2.65E-08	4.38E-08	1.34E-07	1.68E-07
Wisconsin	7.37E-07	9.04E-07	1.43E-08	3.5E-08	4E-07	4.92E-07
Wyoming	1.11E-06	1.11E-06	1.7E-08	1.95E-08	3.88E-07	3.88E-07

Notes:

km = kilometer
N/A = not available

Sources: References 7.4-003 and 7.4-004

Table 7.4-4
Nonradiological Impacts, Per Shipment, Resulting from Shipment of Unirradiated and Spent Nuclear Fuel

	Unirradiated Fuel				Spent Nuclear Fuel			
	Round-trip distance (km)	Accidents	Injuries	Fatalities	Round-trip distance (km)	Accidents	Injuries	Fatalities
LNP/CREC	2303	1.39E-03	7.64E-04	6.24E-05	8939	3.60E-03	2.16E-03	1.55E-04
Dixie Site	2244	1.39E-03	7.60E-04	6.18E-05	8725	3.58E-03	2.15E-03	1.53E-04
Highlands Site	2608	1.43E-03	7.84E-04	6.62E-05	9574	3.68E-03	2.21E-03	1.63E-04
Putnam Site	1985	1.36E-03	7.43E-04	5.86E-05	8933	3.61E-03	2.16E-03	1.55E-04

Table 7.4-5
Nonradiological Impacts Resulting from the Total Amount of Shipments of Unirradiated and Spent Nuclear Fuel for a RRY, Normalized to Reference LWR

Site	Accidents per RRY[(a)]	Injuries per RRY[(a)]	Fatalities per RRY[(a)]
LNP/CREC	1.47E-01	8.80E-02	6.35E-03
Dixie	1.46E-01	8.76E-02	6.27E-03
Highlands	1.51E-01	9.00E-02	6.68E-03
Putnam	1.47E-01	8.79E-02	6.33E-03
Table S-4	--	1.00E-01	1.00E-03

Notes:

a) The values in the table have been calculated from the values presented in Table 7.4-4 based on 4.9 shipments per year of unirradiated fuel and 39 shipments per year of spent fuel ([(unirradiated fuel accidents – 1.39E-03) x (4.9 shipments)] + [(spent fuel accidents – 3.60-E-03) x (39 shipments)] = Accidents per RRY – 1.47E-01).

km = kilometer
RRY = reference reactor year