CHAPTER 7, POSTULATED ACCIDENTS

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7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter assesses the environmental impacts of postulated accidents involving radioactive materials. Section 7.1 evaluates design basis accidents (DBAs), Section 7.2 considers the impact of severe accidents, Section 7.3 addresses severe accident mitigation alternatives (SAMA), and Section 7.4 pertains to transportation accidents.

7.1 DESIGN BASIS ACCIDENTS

7.1.1 SELECTION OF ACCIDENTS

The DBAs considered in this section are from the *AP1000 DCD* (Westinghouse 2008). Table 7.1-1 lists the DBAs having the potential for releases to the environment and shows the NUREG-0800 Standard Review Plan (SRP) section numbers and accident descriptions as well as the corresponding accidents as defined in the AP1000 DCD. The radiological consequences of the accidents listed in Table 7.1-1 are assessed to demonstrate that new units can be sited at the VCSNS site without undue risk to the health and safety of the public.

7.1.2 EVALUATION METHODOLOGY

The AP1000 DCD presents the radiological consequences of the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with VCSNS site data to demonstrate that the DCD analyses are bounding for the VCSNS 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, such as activity release pathways and credited mitigation features, are provided in the DCD.

The dose to an individual located at the exclusion area boundary (EAB) or the low population zone (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 offsite location, the breathing rate of the individual at the offsite location, and activity-to-dose conversion factors. The only site-specific parameter is atmospheric dispersion. Site-specific doses are obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (X/Q values). Since the site-specific X/Q values are bounded by the DCD XQ values, this approach demonstrates that the site-specific doses are within those calculated in the DCD.

The LPZ boundary for Units 2 and 3 is the same as the LPZ boundary for Unit 1 and consists of the area within a 3-mile radius of Unit 1 (see Figure 2.5-1).

The DCD uses conservative assumptions to perform bounding safety analyses that substantially overstate the environmental impact of the identified accidents. Among the conservative assumptions in the DCD is the use of time-dependent X/Q values corresponding to the top fifth percentile meteorology during the first two hours of the accident, meaning that conditions would be more favorable for dispersion 95% of the time. The doses in this environmental report are calculated based on the fiftieth percentile site-specific X/Q values during the first two hours of the accident, reflecting more realistic meteorological conditions. The X/Q values are calculated using the methodology of Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Revision 1 (Regulatory Guide 1.145) with site-specific meteorological data. As indicated in Subsection 2.7.5, 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 (nondirection-specific) X/Q values. For a given location, either the EAB or the LPZ, the 0–2 hour X/Q value is the fiftieth percentile overall value calculated by PAVAN. For the LPZ, the X/Q values for all subsequent times are calculated by logarithmic interpolation between the fiftieth percentile X/Q value and the annual average X/Q value. Releases are assumed to be at ground level, and the shortest distances between the power block and the offsite locations are selected to conservatively maximize the X/Q values.

The accident doses are expressed as total effective dose equivalent (TEDE), consistent with 10 CFR 50.34. The TEDE consists of the sum of the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from external exposure. The CEDE is determined using the dose conversion factors in Federal Guidance Report 11 (U.S. EPA 1988), while the EDE is based on the dose conversion factors in Federal Guidance Report 12 (U.S. EPA 1993). Appendix 15A of the AP1000 DCD provides information on the methodologies used to calculate CEDE and EDE values. As indicated in Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000 (Regulatory Guide 1.183), the dose conversion factors in Federal Guidance Reports 11 and 12 are acceptable to the NRC staff.

7.1.3 SOURCE TERMS

The design basis accident source terms in the AP1000 DCD are calculated in accordance with Regulatory Guide 1.183, based on 102% of the rated core thermal power of 3400 MW. The time-dependent isotopic activities released to the environment from each of the evaluated accidents are presented in Tables 7.1-2 through 7.1-10.

7.1.4 RADIOLOGICAL CONSEQUENCES

Environmental report DBA doses are evaluated based on more realistic meteorological conditions than in the DCD. For each of the accidents identified in Table 7.1-1, the site-specific dose for a given time interval is calculated by multiplying the AP1000 DCD dose by the ratio of the site X/Q value, developed in Subsection 2.7.5.2, to the DCD X/Q value. The time-dependent DCD X/Q values and the time-dependent site X/Q values and their ratios are shown in Table 7.1-11. Because all site X/Q values are bounded by DCD X/Q values, site-specific doses for all accidents are also bounded by DCD doses. The total doses are summarized in Table 7.1-12, based on individual accident doses presented in Tables 7.1-13 through 7.1-22. For each accident, the EAB dose shown is for the two-hour period that yields the maximum dose, in accordance with Regulatory Guide 1.183.

The results of the VCSNS site analysis contained in the referenced tables demonstrate that all accident doses meet the site acceptance criteria of 10 CFR 50.34. The acceptance criteria in 10 CFR 50.34 applies to accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. For events with a higher probability of occurrence, more restrictive dose limits are specified in Regulatory Guide 1.183. Where applied, the more restrictive dose limit is either 10% or 25% of the 10 CFR 50.34 limit of 25 rem TEDE. Although conformance to these more restrictive dose limits is not required for an environmental report, they are shown in the tables for comparison purposes.

The TEDE dose limits shown in Tables 7.1-12 through 7.1-22 are from Regulatory Guide 1.183, Table 6, for all accidents except Reactor Coolant Pump Shaft Break (SRP Section 15.3.4) and Failure of Small Lines Carrying Primary Coolant Outside Containment (SRP Section 15.6.2). Although Regulatory Guide 1.183 does not address these two accidents, NUREG-0800 indicates a dose limit of 2.5 rem for these accidents. All doses are within the acceptance criteria. Because the dose criteria of 10 CFR 50.34 is intended to provide assurance of low risk to the public under postulated accidents, any health effects resulting from the DBAs are considered to be negligible.

Section 7.1 References

1. U.S. EPA 1988, *Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, U.S. EPA, EPA-520/1-88-020, 1988.

2. U.S. EPA 1993, *Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil,* U.S. EPA, EPA-402-R-93-081, 1993.

3. Westinghouse 2008, AP1000 Document APP-GW-GL-700, *AP1000 Design Control Document,* Revision 17, Westinghouse Electric Company, 2008.

Table 7.1-1
Selection of Accidents

SRP/DCD Section	SRP Description	DCD Description	Identified in NUREG-1555, Section 7.1, Appendix A	Comment
15.1.5A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	Steam System Piping Failure	Yes	Addressed in DCD Section 15.1.5
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	Feedwater System Pipe Break	Yes	In the DCD, this is bounded by Section 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	In the DCD, this is bounded by Section 15.3.3 accident
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	Spectrum of Rod Cluster Control Assembly Ejection Accidents	No	Evaluated for completeness
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Failure of Small Lines Carrying Primary Coolant Outside Containment	Yes	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	Steam Generator Tube Rupture	Yes	
15.6.5A	Radiological Consequences of a Design Basis Loss of Coolant Accident Including Containment Leakage Contribution	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Yes	Addressed in DCD Section 15.6.5
15.6.5B	Radiological Consequences of a Design Basis Loss of Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Yes	Addressed in DCD Section 15.6.5
15.7.4	Radiological Consequences of Fuel Handling Accidents	Fuel-Handling Accident	Yes	

Table 7.1-2
Activity Releases for Steam System Piping Failure with Preexisting Iodine Spike

	Activity Release (Ci)				
Isotope	0–2 hour	**0–8 hour**	**8–24 hour**	**24–72 hour**	**Total**
Kr-85m	6.86E-02	1.83E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	2.82E-01	1.13E+00	2.25E+00	6.69E+00	1.01E+01
Kr-87	2.76E-02	4.10E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	1.12E-01	2.50E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	1.28E-01	5.07E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	1.59E-01	6.09E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	1.18E+01	4.63E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.04E-03	3.06E-03	0.00E+00	0.00E+00	3.06E-03
Xe-135	3.10E-01	9.99E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	3.99E-03	4.00E-03	0.00E+00	0.00E+00	4.00E-03
I-130	3.59E-01	5.01E-01	2.09E-01	1.33E-01	8.44E-01
I-131	2.40E+01	3.61E+01	3.10E+01	8.22E+01	1.49E+02
I-132	3.05E+01	3.47E+01	8.06E-01	6.55E-03	3.55E+01
I-133	4.34E+01	6.23E+01	3.53E+01	3.98E+01	1.37E+02
I-134	6.74E+00	6.91E+00	1.43E-03	4.54E-09	6.91E+00
I-135	2.60E+01	3.42E+01	7.54E+00	1.71E+00	4.34E+01
Cs-134	1.90E+01	1.92E+01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.82E+01	2.85E+01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.37E+01	1.38E+01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	1.01E+01	4.42E-07	0.00E+00	1.01E+01
Total	2.15E+02	2.96E+02	1.68E+02	3.56E+02	8.21E+02

Table 7.1-3
Activity Releases for Steam System Piping Failure with Accident-Initiated Iodine Spike

	Activity Release (Ci)				
Isotope	0–2 hour	0–8 hour	8–24 hour	24–72 hour	Total
Kr-85m	6.86E-02	1.83E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	2.82E-01	1.13E+00	2.25E+00	6.69E+00	1.01E+01
Kr-87	2.76E-02	4.10E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	1.12E-01	2.50E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	1.28E-01	5.07E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	1.59E-01	6.09E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	1.18E+01	4.63E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.04E-03	3.06E-03	0.00E+00	0.00E+00	3.06E-03
Xe-135	3.10E-01	9.99E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	3.99E-03	4.00E-03	0.00E+00	0.00E+00	4.00E-03
I-130	4.15E-01	1.42E+00	1.58E+00	1.01E+00	4.01E+00
I-131	2.57E+01	8.33E+01	1.56E+02	4.13E+02	6.53E+02
I-132	4.57E+01	1.44E+02	2.24E+01	1.82E-01	1.66E+02
I-133	4.85E+01	1.63E+02	2.27E+02	2.55E+02	6.45E+02
I-134	1.33E+01	3.20E+01	2.65E-01	8.42E-07	3.23E+01
I-135	3.20E+01	1.10E+02	7.83E+01	1.77E+01	2.06E+02
Cs-134	1.90E+01	1.92E+01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.82E+01	2.85E+01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.37E+01	1.38E+01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	1.01E+01	4.42E-07	0.00E+00	1.01E+01
Total	2.49E+02	6.54E+02	5.78E+02	9.20E+02	2.15E+03

Table 7.1-4
Activity Releases for Reactor Coolant Pump Shaft Seizure

	Activity Release (Ci)		
Isotope	Without Feedwater 0–1.5 hour	With Feedwater 6–8 hour	With Feedwater 0–8 hour
Kr-85m	8.16E+01	4.13E+01	2.79E+02
Kr-85	7.58E+00	1.01E+01	4.04E+01
Kr-87	1.20E+02	5.43E+00	2.13E+02
Kr-88	2.08E+02	6.05E+01	5.82E+02
Xe-131m	3.77E+00	4.95E+00	2.00E+01
Xe-133m	2.02E+01	2.48E+01	1.03E+02
Xe-133	6.66E+02	8.57E+02	3.49E+03
Xe-135m	3.24E+01	2.68E-06	3.30E+01
Xe-135	1.59E+02	1.32E+02	6.72E+02
Xe-138	1.29E+02	3.01E-06	1.31E+02
I-130	8.45E-01	5.65E-01	1.45E+00
I-131	3.77E+01	3.46E+01	8.05E+01
I-132	2.79E+01	3.95E+00	1.83E+01
I-133	4.86E+01	3.64E+01	8.98E+01
I-134	2.88E+01	2.09E-01	5.74E+00
I-135	4.19E+01	2.05E+01	5.79E+01
Cs-134	1.29E+00	1.11E+00	2.59E+00
Cs-136	5.63E-01	3.47E-01	8.63E-01
Cs-137	7.74E-01	6.51E-01	1.52E+00
Cs-138	6.08E+00	1.13E+00	4.08E+00
Rb-86	1.33E-02	1.27E-02	2.91E-02
Total	1.62E+03	1.23E+03	5.82E+03

Table 7.1-5
Activity Releases for Spectrum of Rod Cluster Control Assembly Ejection Accidents

	Activity Release (Ci)					
Isotope	0–2 hour	**0–8 hour**	**8–24 hour**	**24–96 hour**	**96–720 hour**	**Total**
Kr-85m	1.12E+02	1.77E+02	3.87E+01	1.77E+00	2.51E-05	2.18E+02
Kr-85	5.01E+00	1.06E+01	1.49E+01	3.35E+01	2.88E+02	3.47E+02
Kr-87	1.82E+02	2.08E+02	1.03E+00	8.37E-05	0.00E+00	2.09E+02
Kr-88	2.91E+02	4.10E+02	3.49E+01	3.59E-01	8.41E-09	4.45E+02
Xe-131m	4.94E+00	1.04E+01	1.42E+01	2.86E+01	1.16E+02	1.69E+02
Xe-133m	2.67E+01	5.48E+01	6.49E+01	8.45E+01	5.31E+01	2.57E+02
Xe-133	8.79E+02	1.84E+03	2.40E+03	4.27E+03	8.45E+03	1.70E+04
Xe-135m	7.34E+01	7.35E+01	4.33E-09	0.00E+00	0.00E+00	7.35E+01
Xe-135	2.15E+02	3.87E+02	2.09E+02	4.35E+01	1.79E-01	6.39E+02
Xe-138	2.99E+02	2.99E+02	3.19E-09	0.00E+00	0.00E+00	2.99E+02
I-130	4.90E+00	1.22E+01	4.32E+00	2.03E-01	2.95E-04	1.67E+01
I-131	1.36E+02	3.81E+02	2.31E+02	3.10E+01	1.68E+01	6.60E+02
I-132	1.53E+02	2.52E+02	9.85E+00	8.24E-03	0.00E+00	2.62E+02
I-133	2.72E+02	7.12E+02	3.18E+02	2.28E+01	2.41E-01	1.05E+03
I-134	1.66E+02	1.95E+02	1.37E-01	4.48E-08	0.00E+00	1.95E+02
I-135	2.39E+02	5.36E+02	1.19E+02	2.39E+00	7.32E-05	6.57E+02
Cs-134	3.08E+01	9.30E+01	6.03E+01	7.76E+00	5.16E+00	1.66E+02
Cs-136	8.79E+00	2.63E+01	1.67E+01	2.05E+00	6.58E-01	4.57E+01
Cs-137	1.79E+01	5.41E+01	3.51E+01	4.52E+00	3.05E+00	9.68E+01
Cs-138	1.09E+02	1.16E+02	1.68E-03	0.00E+00	0.00E+00	1.16E+02
Rb-86	3.62E-01	1.09E+00	6.96E-01	8.67E-02	3.42E-02	1.91E+00
Total	3.23E+03	5.84E+03	3.58E+03	4.53E+03	8.93E+03	2.29E+04

Table 7.1-6
Activity Releases for Failure of Small Lines Carrying Primary Coolant Outside Containment

Isotope	Activity Release (Ci) 0–2 hour
Kr-85m	1.24E+01
Kr-85	4.40E+01
Kr-87	7.05E+00
Kr-88	2.21E+01
Xe-131m	1.99E+01
Xe-133m	2.50E+01
Xe-133	1.84E+03
Xe-135m	2.59E+00
Xe-135	5.20E+01
Xe-138	3.65E+00
I-130	1.89E+00
I-131	9.26E+01
I-132	3.49E+02
I-133	2.01E+02
I-134	1.58E+02
I-135	1.68E+02
Cs-134	4.16E+00
Cs-136	6.16E+00
Cs-137	3.00E+00
Cs-138	2.21E+00
Total	3.02E+03

Table 7.1-7
Activity Releases for Steam Generator Tube Rupture with Preexisting Iodine Spike

	Activity Release (Ci)			
Isotope	0–2 hour	**0–8 hour**	**8–24 hour**	**Total**
Kr-85m	5.53E+01	7.46E+01	7.53E-03	7.46E+01
Kr-85	2.20E+02	3.29E+02	1.34E-01	3.29E+02
Kr-87	2.39E+01	2.75E+01	9.12E-05	2.75E+01
Kr-88	9.22E+01	1.19E+02	5.43E-03	1.19E+02
Xe-131m	9.96E+01	1.48E+02	5.91E-02	1.48E+02
Xe-133m	1.24E+02	1.83E+02	6.61E-02	1.83E+02
Xe-133	9.19E+03	1.37E+04	5.29E+00	1.37E+04
Xe-135m	3.44E+00	3.45E+00	0.00E+00	3.45E+00
Xe-135	2.46E+03	3.47E+02	7.10E-02	3.47E+02
Xe-138	4.56E+00	4.57E+00	0.00E+00	4.57E+00
I-130	1.79E+00	1.85E+00	2.68E-01	2.12E+00
I-131	1.21E+02	1.26E+02	3.06E+01	1.57E+02
I-132	1.42E+02	1.42E+02	1.92E+00	1.44E+02
I-133	2.16E+02	2.24E+02	4.06E+01	2.64E+02
I-134	2.74E+01	2.74E+01	4.23E-03	2.74E+01
I-135	1.27E+02	1.30E+02	1.17E+01	1.42E+02
Cs-134	1.63E+00	1.69E+00	2.16E-01	1.90E+00
Cs-136	2.42E+00	2.51E+00	3.14E-01	2.82E+00
Cs-137	1.17E+00	1.22E+00	1.56E-01	1.37E+00
Cs-138	5.64E-01	5.64E-01	5.73E-07	5.64E-01
Total	1.29E+04	1.56E+04	9.14E+01	1.56E+04

Table 7.1-8
Activity Releases for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

	Activity Release (Ci)			
Isotope	0–2 hour	**0–8 hour**	**8–24 hour**	**Total**
Kr-85m	5.53E+01	7.46E+01	7.53E-03	7.46E+01
Kr-85	2.20E+02	3.29E+02	1.34E-01	3.29E+02
Kr-87	2.39E+01	2.75E+01	9.12E-05	2.75E+01
Kr-88	9.22E+01	1.19E+02	5.43E-03	1.19E+02
Xe-131m	9.96E+01	1.48E+02	5.91E-02	1.48E+02
Xe-133m	1.24E+02	1.83E+02	6.61E-02	1.83E+02
Xe-133	9.19E+03	1.37E+04	5.29E+00	1.37E+04
Xe-135m	3.44E+00	3.45E+00	0.00E+00	3.45E+00
Xe-135	2.46E+03	3.47E+02	7.10E-02	3.47E+02
Xe-138	4.56E+00	4.57E+00	0.00E+00	4.57E+00
I-130	8.87E-01	1.05E+00	8.24E-01	1.87E+00
I-131	4.36E+01	5.51E+01	6.76E+01	1.23E+02
I-132	1.47E+02	1.52E+02	1.29E+01	1.65E+02
I-133	9.33E+01	1.13E+02	1.08E+02	2.22E+02
I-134	5.59E+01	5.59E+01	5.94E-02	5.60E+01
I-135	7.61E+01	8.60E+01	4.38E+01	1.30E+02
Cs-134	1.63E+00	1.69E+00	2.16E-01	1.90E+00
Cs-136	2.42E+00	2.51E+00	3.14E-01	2.82E+00
Cs-137	1.17E+00	1.22E+00	1.56E-01	1.37E+00
Cs-138	5.64E-01	5.64E-01	5.73E-07	5.64E-01
Total	1.27E+04	1.54E+04	2.40E+02	1.56E+04

Table 7.1-9 (Sheet 1 of 3)
Activity Releases for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

	Activity Release (Ci)							
Isotope	1.4–3.4 hr	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-72 hr**	**72-96 hr**	**96-720 hr**	**Total**
I-130	5.64E+01	3.24E+01	7.85E+01	6.21E+00	5.11E-01	1.17E-01	6.00E-03	1.18E+02
I-131	1.68E+03	9.19E+02	2.57E+03	2.56E+02	1.33E+02	5.84E+01	5.79E+02	4.52E+03
I-132	1.23E+03	8.79E+02	1.26E+03	1.62E+01	6.00E-03	0.00E+00	0.00E+00	2.16E+03
I-133	3.23E+03	1.82E+03	4.72E+03	3.71E+02	7.41E+01	9.90E+00	7.80E+00	7.00E+03
I-134	6.60E+02	7.09E+02	4.29E+02	3.07E-02	0.00E+00	0.00E+00	0.00E+00	1.14E+03
I-135	2.56E+03	1.54E+03	3.36E+03	1.56E+02	4.79E+00	1.00E-02	0.00E+00	5.06E+03
Kr-85m	1.42E+03	6.32E+02	3.14E+03	1.87E+03	8.60E+01	0.00E+00	0.00E+00	5.73E+03
Kr-85	8.31E+01	3.22E+01	2.65E+02	7.06E+02	1.06E+03	5.28E+02	1.36E+04	1.62E+04
Kr-87	1.10E+03	6.88E+02	1.26E+03	5.00E+01	0.00E+00	0.00E+00	0.00E+00	2.00E+03
Kr-88	3.11E+03	1.50E+03	5.76E+03	1.70E+03	1.70E+01	0.00E+00	0.00E+00	8.98E+03
Xe-131m	8.26E+01	3.21E+01	2.62E+02	6.79E+02	9.42E+02	4.31E+02	5.57E+03	7.92E+03
Xe-133m	4.43E+02	1.74E+02	1.37E+03	3.15E+03	3.14E+03	9.65E+02	2.58E+03	1.14E+04
Xe-133	1.47E+04	5.71E+03	4.62E+04	1.16E+05	1.46E+05	5.97E+04	4.07E+05	7.81E+05
Xe-135m	1.06E+01	3.33E+01	2.62E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.59E+01
Xe-135	3.15E+03	1.31E+03	8.33E+03	1.01E+04	2.06E+03	4.00E+01	1.00E+01	2.19E+04
Xe-138	3.11E+01	1.14E+02	6.90E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.21E+02
Rb-86	3.04E+00	1.72E+00	4.60E+00	2.80E-01	1.00E-03	0.00E+00	8.00E-03	6.61E+00
Cs-134	2.58E+02	1.46E+02	3.92E+02	2.40E+01	1.00E-01	0.00E+00	1.20E+00	5.63E+02
Cs-136	7.33E+01	4.14E+01	1.11E+02	6.70E+00	0.00E+00	0.00E+00	2.00E-01	1.59E+02
Cs-137	1.51E+02	8.49E+01	2.28E+02	1.41E+01	0.00E+00	0.00E+00	7.00E-01	3.28E+02
Cs-138	1.50E+02	2.60E+02	6.96E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.30E+02
Sb-127	2.42E+01	1.14E+01	3.67E+01	2.14E+00	1.00E-02	0.00E+00	1.00E-02	5.03E+01
Sb-129	5.10E+01	2.71E+01	6.23E+01	1.48E+00	0.00E+00	0.00E+00	0.00E+00	9.09E+01
Te-127m	3.15E+00	1.47E+00	4.83E+00	2.95E-01	2.00E-03	0.00E+00	1.30E-02	6.61E+00
Te-127	2.05E+01	1.02E+01	2.81E+01	1.11E+00	0.00E+00	0.00E+00	0.00E+00	3.94E+01

Table 7.1-9 (Sheet 2 of 3)
Activity Releases for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

	Activity Release (Ci)							
Isotope	1.4–3.4 hr	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-72 hr**	**72-96 hr**	**96-720 hr**	**Total**
Te-129m	1.07E+01	5.01E+00	1.64E+01	1.00E+00	1.00E-02	0.00E+00	3.00E-02	2.25E+01
Te-129	1.88E+01	1.39E+01	1.45E+01	3.00E-02	0.00E+00	0.00E+00	0.00E+00	2.84E+01
Te-131	3.17E+01	1.51E+01	4.69E+01	2.51E+00	0.00E+00	0.00E+00	1.00E-02	6.45E+01
Te-132	3.23E+02	1.52E+02	4.89E+02	2.84E+01	1.00E-01	0.00E+00	1.00E-01	6.70E+02
Sr-89	9.23E+01	4.31E+01	1.45E+02	5.40E+00	1.00E-01	0.00E+00	3.00E-01	1.94E+02
Sr-90	7.95E+00	3.71E+00	1.22E+01	7.50E-01	0.00E+00	0.00E+00	4.00E-02	1.67E+01
Sr-91	9.68E+01	4.79E+01	1.33E+02	5.30E+00	0.00E+00	0.00E+00	0.00E+00	1.86E+02
Sr-92	6.83E+01	3.91E+01	7.40E+01	1.00E+00	0.00E+00	0.00E+00	0.00E+00	1.14E+02
Ba-139	5.44E+01	3.74E+01	4.56E+01	1.50E-01	0.00E+00	0.00E+00	0.00E+00	8.32E+01
Ba-140	1.63E+02	7.61E+01	2.49E+02	1.51E+01	0.00E+00	0.00E+00	4.00E-01	3.41E+02
Mo-99	2.15E+01	1.01E+01	3.24E+01	1.86E+00	1.00E-02	0.00E+00	0.00E+00	4.44E+01
Tc-99m	1.47E+01	7.54E+00	1.91E+01	5.90E-01	0.00E+00	0.00E+00	0.00E+00	2.72E+01
Ru-103	1.73E+01	8.08E+00	2.65E+01	1.62E+00	0.00E+00	1.00E-02	6.00E-02	3.63E+01
Ru-105	8.18E+00	4.33E+00	1.00E+01	2.40E-01	0.00E+00	0.00E+00	0.00E+00	1.46E+01
Ru-106	5.70E+00	2.66E+00	8.75E+00	5.40E-01	0.00E+00	0.00E+00	3.00E-02	1.20E+01
Rh-105	1.03E+01	4.88E+00	1.53E+01	8.30E-01	0.00E+00	0.00E+00	0.00E+00	2.10E+01
Ce-141	3.89E+00	1.82E+00	5.96E+00	3.64E-01	1.00E-03	1.00E-03	1.20E-02	8.16E+00
Ce-143	3.46E+00	1.64E+00	5.14E+00	2.78E-01	1.00E-03	0.00E+00	0.00E+00	7.06E+00
Ce-144	2.94E+00	1.37E+00	4.51E+00	2.76E-01	1.00E-03	1.00E-03	1.30E-02	6.17E+00
Pu-238	9.16E-03	4.28E-03	1.41E-02	8.60E-04	0.00E+00	0.00E+00	4.00E-05	1.93E-02
Pu-239	8.06E-04	3.76E-04	1.24E-03	7.60E-05	0.00E+00	1.00E-06	3.00E-06	1.70E-03
Pu-240	1.18E-03	5.52E-04	1.81E-03	1.11E-04	1.00E-06	0.00E+00	5.00E-06	2.48E-03
Pu-241	2.65E-01	1.24E-01	4.08E-01	2.50E-02	1.00E-04	0.00E+00	1.20E-03	5.58E-01
Np-239	4.48E+01	2.12E+01	6.75E+01	3.84E+00	1.00E-02	1.00E-02	1.00E-02	9.26E+01
Y-90	8.08E-02	3.81E-02	1.22E-01	7.00E-03	0.00E+00	0.00E+00	0.00E+00	1.67E-01

Table 7.1-9 (Sheet 3 of 3)
Activity Releases for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

	Activity Release (Ci)							
Isotope	1.4–3.4 hr	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-72 hr**	**72-96 hr**	**96-720 hr**	**Total**
Y-91	1.19E+00	5.54E-01	1.82E+00	1.11E-01	1.00E-03	0.00E+00	4.00E-03	2.49E+00
Y-92	7.89E-01	4.32E-01	9.19E-01	1.80E-02	0.00E+00	0.00E+00	0.00E+00	1.37E+00
Y-93	1.21E+00	6.00E-01	1.68E+00	6.80E-02	0.00E+00	0.00E+00	0.00E+00	2.35E+00
Nb-95	1.59E+00	7.46E-01	2.44E+00	1.49E-01	1.00E-03	0.00E+00	5.00E-03	3.34E+00
Zr-95	1.59E+00	7.41E-01	2.43E+00	1.49E-01	0.00E+00	0.00E+00	6.00E-03	3.33E+00
Zr-97	1.43E+00	6.89E-01	2.05E+00	9.80E-02	0.00E+00	0.00E+00	0.00E+00	2.84E+00
La-140	1.67E+00	7.92E-01	2.50E+00	1.39E-01	0.00E+00	0.00E+00	0.00E+00	3.43E+00
La-141	1.03E+00	5.54E-01	1.23E+00	2.70E-02	0.00E+00	0.00E+00	0.00E+00	1.81E+00
La-142	5.38E-01	3.57E-01	4.74E-01	2.00E-03	0.00E+00	0.00E+00	0.00E+00	8.33E-01
Nd-147	6.16E-01	2.89E-01	9.42E-01	5.70E-02	0.00E+00	0.00E+00	1.00E-03	1.29E+00
Pr-143	1.39E+00	6.50E-01	2.13E+00	1.28E-01	1.00E-03	0.00E+00	3.00E-03	2.91E+00
Am-241	1.20E-04	5.59E-05	1.84E-04	1.13E-05	0.00E+00	0.00E+00	6.00E-07	2.52E-04
Cm-242	2.82E-02	1.32E-02	4.33E-02	2.65E-03	1.00E-05	1.00E-05	1.20E-04	5.93E-02
Cm-244	3.46E-03	1.62E-03	5.32E-03	3.26E-04	1.00E-06	0.00E+00	1.60E-05	7.28E-03
Total	3.53E+04	1.72E+04	8.14E+04	1.35E+05	1.54E+05	6.17E+04	4.29E+05	8.78E+05

Table 7.1-10
Activity Releases for Fuel-Handling Accident

Isotope	Activity Release (Ci) 0–2 hour
Kr-85m	8.40E+00
Kr-85	1.10E+03
Kr-88	3.00E-01
Xe-131m	5.52E+02
Xe-133m	2.30E+03
Xe-133	8.88E+04
Xe-135m	1.02E+03
Xe-135	5.68E+03
I-130	7.00E-01
I-131	3.47E+02
I-132	2.44E+02
I-133	1.08E+02
I-135	3.20E+00
Total	1.00E+05

Table 7.1-11
Atmospheric Dispersion Factors

Accident	Location	Time (hr)	DCD X/Q (sec/m^3)	Site X/Q (sec/m^3)	X/Q Ratio (Site/DCD)
LOCA	EAB	0 - 2	5.1E-04	9.46E-05	1.85E-01
	LPZ	0 - 8	2.2E-04	1.07E-05	4.86E-02
		8 - 24	1.6E-04	8.67E-06	5.42E-02
		24 - 96	1.0E-04	5.52E-06	5.52E-02
		96 - 720	8.0E-05	2.89E-06	3.61E-02
Other	EAB	0 - 2	1.0E-03	9.46E-05	9.46E-02
	LPZ	0 - 8	5.0E-04	1.07E-05	2.14E-02
		8 - 24	3.0E-04	8.67E-06	2.89E-02
		24 - 96	1.5E-04	5.52E-06	3.68E-02
		96 - 720	8.0E-05	2.89E-06	3.61E-02

Note: It is seen that the site X/Q values are bounded by the DCD X/Q values for all time steps.

Table 7.1-12
Summary of Design Basis Accident Doses

DCD/SRP Section	Accident	Site Dose (rem TEDE) EAB	LPZ	Limit[a]	Dose Table
15.1.5	Steam System Piping Failure				
	Preexisting Iodine Spike	0.095	0.018	25.0	7.1-13
	Accident-Initiated Iodine Spike	0.10	0.052	2.5	7.1-14
15.2.8	Feedwater System Pipe Break	(b)	(b)		
15.3.3	Reactor Coolant Pump Shaft Seizure				
	No Feedwater	0.076	0.0083	2.5	7.1-15
	Feedwater Available	0.057	0.017	2.5	7.1-16
15.3.4	Reactor Coolant Pump Shaft Break	(c)	(c)		
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	0.34	0.12	6.3	7.1-17
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	0.20	0.022	2.5	7.1-18
15.6.3	Steam Generator Tube Rupture				
	Preexisting Iodine Spike	0.21	0.027	25.0	7.1-19
	Accident-Initiated Iodine Spike	0.10	0.018	2.5	7.1-20
15.6.5	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	4.6	1.1	25.0	7.1-21
15.7.4	Fuel-Handling Accident	0.49	0.055	6.3	7.1-22

a) NUREG-1555 specifies a dose limit of 25 rem TEDE for all DBAs. The more restrictive limits shown in the table apply to safety analysis report doses, but are shown here to demonstrate that even these more restrictive limits are met.
b) Feedwater System Pipe Break is bounded by Steam System Piping Failure, as indicated in AP1000 DCD.
c) Reactor Coolant Pump Shaft Break is bounded by Reactor Coolant Pump Shaft Seizure, as indicated in AP1000 DCD.

Table 7.1-13
Doses for Steam Piping Failure with Preexisting Iodine Spike

	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0–2 hour	1.00E+00		9.46E-02	9.46E-02	
0–8 hour		5.81E-01	2.14E-02		1.24E-02
8–24 hour		7.18E-02	2.89E-02		2.08E-03
24–96 hour		1.08E-01	3.68E-02		3.97E-03
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	1.00E+00	7.61E-01		9.46E-02	1.85E-02
Limit				25	25

Table 7.1-14
Doses for Steam Piping Failure with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0–2 hour	1.10E+00		9.46E-02	1.04E-01	
0–8 hour		1.02E+00	2.14E-02		2.18E-02
8–24 hour		3.77E-01	2.89E-02		1.09E-02
24–96 hour		5.36E-01	3.68E-02		1.97E-02
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	1.10E+00	1.93E+00		1.04E-01	5.24E-02
Limit				2.5	2.5

Table 7.1-15
Doses for Reactor Coolant Pump Shaft Seizure with No Feedwater

	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0–2 hour	8.00E-01		9.46E-02	7.57E-02	
0–8 hour		3.89E-01	2.14E-02		8.32E-03
8–24 hour		0.00E+00	2.89E-02		0.00E+00
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	8.00E-01	3.89E-01		7.57E-02	8.32E-03
Limit				2.5	2.5

Table 7.1-16
Doses for Reactor Coolant Pump Shaft Seizure with Feedwater Available

	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
6–8 hour[a]	6.00E-01		9.46E-02	5.68E-02	
0–8 hour		7.94E-01	2.14E-02		1.70E-02
8–24 hour		0.00E+00	2.89E-02		0.00E+00
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	6.00E-01	7.94E-01		5.68E-02	1.70E-02
Limit				2.5	2.5

a) The six-to eight-hour time frame is the highest dose two-hour interval for this event.

Table 7.1-17
Doses for Spectrum of Rod Cluster Control Assembly Ejection Accidents

Time	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0–2 hour	3.60E+00		9.46E-02	3.41E-01	
0–8 hour		4.58E+00	2.14E-02		9.80E-02
8–24 hour		7.84E-01	2.89E-02		2.27E-02
24–96 hour		6.32E-02	3.68E-02		2.33E-03
96–720 hour		2.06E-02	3.61E-02		7.44E-04
Total	3.60E+00	5.45E+00		3.41E-01	1.24E-01
Limit				6.3	6.3

Table 7.1-18
Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment

Time	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0–2 hour	2.10E+00		9.46E-02	1.99E-01	
0–8 hour		1.02E+00	2.14E-02		2.18E-02
8–24 hour		0.00E+00	2.89E-02		0.00E+00
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	2.10E+00	1.02E+00		1.99E-01	2.18E-02
Limit				2.5	2.5

Table 7.1-19
Doses for Steam Generator Tube Rupture with Preexisting Iodine Spike

Time	DCD Dose (rem TEDE)		X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)	
	EAB	LPZ		EAB	LPZ
0–2 hour	2.20E+00		9.46E-02	2.08E-01	
0–8 hour		1.16E+00	2.14E-02		2.48E-02
8–24 hour		7.24E-02	2.89E-02		2.09E-03
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	2.20E+00	1.23E+00		2.08E-01	2.69E-02
Limit				25	25

Table 7.1-20
Doses for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

Time	DCD Dose (rem TEDE)		X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)	
	EAB	LPZ		EAB	LPZ
0–2 hour	1.10E+00		9.46E-02	1.04E-01	
0–8 hour		6.27E-01	2.14E-02		1.34E-02
8–24 hour		1.69E-01	2.89E-02		4.88E-03
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	1.10E+00	7.96E-01		1.04E-01	1.83E-02
Limit				2.5	2.5

Table 7.1-21
Doses for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

Time	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
1.4–3.4 hour[a]	2.46E+01		1.85E-01	4.56E+00	
0–8 hour		2.17E+01	4.86E-02		1.06E+00
8–24 hour		7.50E-01	5.42E-02		4.06E-02
24–96 hour		2.93E-01	5.52E-02		1.62E-02
96–720 hour		5.49E-01	3.61E-02		1.98E-02
Total	2.46E+01	2.33E+01		4.56E+00	1.13E+00
Limit				25	25

a) The 1.4 to 3.4 hour time frame is the highest dose two-hour interval for this event.

Table 7.1-22
Doses for Fuel Handling Accident

Time	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0–2 hour	5.20E+00		9.46E-02	4.92E-01	
0–8 hour		2.59E+00	2.14E-02		5.54E-02
8–24 hour		0.00E+00	2.89E-02		0.00E+00
24–96 hour		0.00E+00	3.68E-02		0.00E+00
96–720 hour		0.00E+00	3.61E-02		0.00E+00
Total	5.20E+00	2.59E+00		4.92E-01	5.54E-02
Limit				6.3	6.3

7.2 SEVERE ACCIDENTS

Severe accidents are defined as accidents with substantial damage to the reactor core and degradation of containment systems. Because the probability of a severe accident is very low for the AP1000, such accidents are not part of the design basis for the plant. However, the NRC requires, in its *Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants* (50 FR 32138), the completion of a probabilistic risk assessment for severe accidents for new reactor designs. This requirement is codified in regulation 10 CFR 52.47, *Contents of Applications*.

Westinghouse completed a probabilistic risk assessment for the AP1000 design (Westinghouse 2004) as part of their application for design certification. The AP1000 design was reviewed by NRC, and the review was documented in NUREG-1793, *Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design* (U.S. NRC 2004). Subsequently, NRC certified the design, concluding that, following resolution of open items, this advanced design meets NRC's safety goals and represents an improvement in safety over currently operating reactors in the United States

The Westinghouse analysis used generic, but conservative, meteorology and regional characteristics. SCE&G presents in this section an update of the generic probabilistic risk assessment analysis of severe accidents to include VCSNS site-specific characteristics and impacts over the entire life cycle of a severe accident. The purpose is to disclose the complete impacts of a severe accident, demonstrate that the impacts are comparable to those approved for the AP1000 certification, and support the severe accident mitigation alternatives analyses in Section 7.3.

7.2.1 WESTINGHOUSE METHODOLOGY

The Westinghouse probabilistic risk assessment for the AP1000 established an event tree which defined the possible end states of the containment following a severe accident. These end states can logically be grouped into three categories: (1) an intact containment with normal leakage or a larger leak with a containment isolation failure, (2) a containment breach, possibly due to high containment pressure or a hydrogen detonation, and (3) containment bypass such as a steam generator tube rupture. Using the EPRI code Modular Accident Analysis Program, Westinghouse determined that six source term categories would represent the entire suite of potential severe accidents. An accident frequency was assigned to each of the six categories (Table 7.2-1).

The six source term categories or accident categories are as follows:

- Intact Containment – Containment integrity is maintained throughout the accident. The release of radioactivity to the environment is due to nominal design leakage.

- Containment Bypass – Radioactivity is released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to the onset of core damage. This accident category contributes to the large, early release frequency.

- Containment Isolation Failure – Radioactivity is released through a failure of the valves that close the penetrations between containment and the environment. Containment failure occurs prior to the onset of core damage. This accident category contributes to the large, early release frequency.

- Early Containment Failure – Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after the onset of core damage but before core relocation. Such phenomena could include hydrogen detonation, hydrogen diffusion flame, steam explosions, or vessel failures. This accident category contributes to the large, early release frequency.

- Intermediate Containment Failure – Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after core relocation but before 24 hours have passed since initiation of the accident. Such phenomena could include hydrogen detonation and hydrogen deflagration. This accident category contributes to large releases but does not occur early in the accident life cycle.

- Late Containment Failure – Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon more than 24 hours after initiation of the accident. Such phenomena could include the failure of containment heat removal. This accident category contributes to large releases but does not occur early in the accident life cycle.

Westinghouse then used the NRC code MACCS2 (Chanin and Young 1997) to model the environmental consequences of the severe accidents. MACCS2 was developed specifically for NRC to evaluate severe accidents at nuclear power plants. The meteorology Westinghouse used to represent a generic AP1000 site is specified in the EPRI's Utility Requirements Document (EPRI 1999). This meteorology is from an actual site database selected because it is expected to provide impacts greater than those that would be expected at 80 to 90% of U.S. operating plants. The population considered also was selected to provide impacts greater than those that would be expected at 80% to 90% of the plants. The Westinghouse analysis focused on 24 hours following core damage as a measure of the consequences from a large release and, therefore, did not address the chronic pathways such as ingestion, inhalation of resuspended material, or groundshine subsequent to plume passage.

Additional details on the Westinghouse analysis are found in Westinghouse (2004) and reported in the AP1000 DCD (Westinghouse 2008).

7.2.2 SCE&G METHODOLOGY

SCE&G also used the MACCS2 computer code (Version 1.13.1) to evaluate consequences of severe accidents. The pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water. The MACCS2 code primarily addresses dose from the air pathway, but also calculates dose from surface runoff and deposition on surface water. The code also evaluates the extent of contamination. Differences between the Westinghouse generic analysis and the VCSNS site-specific analysis include: 1) SCE&G incorporated site-specific parameters describing such factors as meteorology, population, evacuation, agricultural production and property valuations; 2) SCE&G extended the analysis to include long-term exposure pathways, such as ingestion, over the life cycle of the accident (Ingestion exposure was determined using the COMIDA2 food model option of MACCS2); and 3) SCE&G incorporated more conservative model parameters in place of those from EPRI [i.e., MACCS' sample problem A parameters in place of those from EPRI (1999)].

To assess human health impacts, SCE&G determined health risks such as the collective dose to the 50-mile population, number of latent cancer fatalities, and number of early fatalities associated with a severe accident. Economic cost risks were also determined, including the costs associated with short-term relocation of people, decontamination of property and equipment, interdiction of food supplies, and indirect costs resulting from loss of use of the property and incomes derived as a result of the accident.

Five files provide input to a MACCS2 analysis. One provides data to calculate the amount of material released to the atmosphere that is dispersed and deposited. The calculation uses a Gaussian plume model. Important inputs in this file include the core inventory, nuclide release fractions, and geometry of the reactor and associated buildings. This input data is the same as those in the MACCS2 input files used by Westinghouse in the generic probabilistic risk assessment. A second file provides inputs to calculations regarding exposure in the time period immediately following the release. Important site-specific information includes emergency response parameters such as evacuation time. The third input file provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors. These files access a meteorological file, which uses actual VCSNS meteorological monitoring data and a site characteristics file which is built using SECPOP2000 (U.S. NRC 2003).

MACCS2 requires an entire calendar year of meteorological data. Year 2007 meteorological data measured at the Unit 2 and 3 site were analyzed. Sensitivity analyses considered meteorological data from 2008 at the Units 2 and 3 site along with three other years (July 2003 to June 2006) measured at the Unit 1 meteorological data tower.

SECPOP2000 incorporates 2000 census data for the 50-mile region around the VCSNS site. For this analysis, the census data was modified to include transient populations and projected to the year 2060. MACCS2 also requires the spatial distribution of certain agriculture and economic data (fraction of land devoted to farming, annual farm sales, fraction of farm sales resulting from dairy production, and property value of farm and non-farm land) in the same manner as the population. This was again done by applying the SECPOP2000 program, changing the regional economic data format to comply with MACCS2 input requirements. In this case, SECPOP2000 was used to access data from the 1997 National Census of Agriculture. (The County97.dat file provided by SECPOP2000 was modified to correct two errors (generally known as the missing notes parameter error and the missing county numbers error) in the issued version). The program's specification of crop production parameters for the 50-mile region (e.g, fraction of farmland devoted to grains, vegetables, etc.) was also applied.

SCE&G used the resulting MACCS2 calculations and accident frequency information to determine risk. The sum of the accident frequencies is known as the core damage frequency and includes only internally initiated events. Risk is the product of frequency of an accident times the consequences of the accident. Consequences include radiation dose and economic cost. Dose-risk is the product of the collective dose times the accident frequency. Because the AP1000's severe accident analysis addressed a suite of accidents, the individual risks were summed to provide a total risk. The same process was applied to estimating cost-risk. Therefore, these risks are reported as person-rem per reactor year or dollars per reactor year.

SCE&G assumed a ground-level release height and no release heat for each base case accident release hypothesized. Each of those assumptions was investigated with a sensitivity calculation. Release heights at the middle and top of containment and heat release rates of 1 and 10 megawatts per release segment were considered. The dose-risk varied by less than 9% for each of those. The previously discussed sensitivity of dose risk to variation in meteorology (using the 2008 Units 2 and 3 site data and three alternate years of data measured at the Unit 1 site) resulted in dose risk variations of less than 6% from the base case. The 2007 meteorological data resulted in the larger dose risk. The choice of MACCS sample problem A parameters (e.g., sheltering factors, deposition velocities) rather than EPRI (1999) parameters was shown to be conservative; the dose risk for the latter was 25% less than for the former.

7.2.3 CONSEQUENCES TO POPULATION GROUPS

7.2.3.1 Air Pathways

Each of the six accident categories was analyzed with MACCS2 to estimate population dose, number of early and latent fatalities, water ingestion dose, cost, and farm land requiring decontamination. The analysis assumed that 95% of the population was evacuated following declaration of a general emergency. For each accident category, SCE&G calculated the risk for each analytical endpoint (dose, fatalities, cost, and contaminated land) by multiplying it by the accident category frequency. The results are provided in Table 7.2-1.

7.2.3.2 Surface Water Pathways

People can be exposed to radiation when deposited airborne radioactivity runs off into or is deposited onto surface water. The exposure pathway can be from drinking the water, external radiation from submersion in the water, external radiation from activities near the shoreline, or ingestion of fish or shellfish. MACCS2 only calculates the dose from drinking the water. The MACCS2 severe accident dose-risk to the 50-mile population from drinking water is 6.4×10^{-3} person-rem per year of AP1000 operation. This value is included with the air pathways dose and is the sum of all six accident category risks.

Surface water pathways involving swimming, fishing, and boating are not modeled by MACCS2. Surface water bodies within the 50-mile region of VCSNS include rivers, reservoirs, creeks, and ponds. The NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet discharging to small rivers (including the Broad River) in NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (U.S. NRC 1996). The NRC evaluation estimated the uninterdicted aquatic food pathway dose risk as 0.4 person-rem per reactor year for sites on a small river. This analysis assumes that the Monticello Reservoir aquatic foods would be interdicted, but that less control might be established on water released to the Broad River by the Fairfield Pumped Storage Facility, resulting in contamination of aquatic foods that might be subject to less control.

The NRC concluded in NUREG-1437 that population doses from drinking water and aquatic food pathways are small relative to the atmospheric pathway for most sites (including VCSNS). Because the AP1000 atmospheric pathway doses are significantly lower than those of the current nuclear fleet, the doses from surface water sources would be consistently lower for the AP1000 as well.

7.2.3.3 Groundwater Pathways

People can also receive a dose from groundwater pathways. Radioactivity released during an accident can enter groundwater that serves as a source of drinking water or irrigation, or can move through an aquifer that eventually discharges to surface water. SCE&G evaluated the consequences of a spill of 22,400 gallons of radiologically contaminated water from an effluent holdup tank directly to groundwater. The evaluation determined that all isotopes would be small fractions of 10 CFR 20 effluent concentration limits before they reached the nearest potable water supply in an unrestricted area.

NUREG-1437 also evaluated the groundwater pathway dose, based on the analysis in NUREG-0440, *Liquid Pathway Generic Study* (LPGS). NUREG-0440 analyzed a core meltdown that contaminated groundwater that subsequently contaminated surface water. However, NUREG-0440 did not analyze direct drinking of groundwater because of the limited number of potable groundwater wells.

The LPGS results provide conservative, uninterdicted population dose estimates for six generic categories of plants. These dose estimates were one or more orders of magnitude less than those attributed to the atmospheric pathway. NUREG-1437 compares Unit 1 liquid pathway severe accident doses to the results of NUREG-0440 with results consistent with the LPGS conclusion that the atmospheric pathway dominates that from groundwater pathways. The proposed location for Units 2 and 3 has the same groundwater characteristics as the location of Unit 1, and the accident frequency for the AP1000 is lower than that of Unit 1. Therefore, the dose risk from the AP1000 groundwater pathway would be smaller than that from Unit 1.

7.2.4 COMPARISON TO NRC SAFETY GOALS

SCE&G compared the severe accident risks from Units 2 and 3 against two risk goals identified by the NRC (51 FR 30028) as described below. The results are presented in Table 7.2-2.

7.2.4.1 Individual Risk Goal

The risk of prompt fatalities that might result from reactor accidents to an average individual in the vicinity of a nuclear power plant should not exceed 0.1% of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. population are generally exposed. As noted in the Safety Goals Policy statement (51 FR 30028), "vicinity" is defined as the area within 1 mile of the plant site boundary. "Prompt Fatality Risks" are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. Such risks are the sum of risks which result in fatalities from such activities as driving, household chores, occupational activities, etc. For this evaluation, the sum of prompt fatality risks was taken as the U.S. accidental death risk value of 37.6 deaths per 100,000 people per year based upon National Center for Health Statistics and U.S. Census Bureau data average for 2002 to 2004 (USCB 2007, CDC 2006a, CDC 2006b, and CDC 2007).

7.2.4.2 Societal Risk Goal

The risk of cancer fatalities that might result from nuclear power plant operations to the population in the area near a nuclear power plant should not exceed 0.1% of the sum of the cancer fatality risks resulting from all other causes. As noted in the Safety Goal Policy Statement (51 FR 30028), "near" is defined as within 10 miles of the plant. The cancer fatality risk was taken as 191.2 deaths per 100,000 people per year based upon National Center for Health Statistics and U.S. Census Bureau data average for 2002 to 2004 (USCB 2007, CDC 2006a, CDC 2006b, and CDC 2007).

7.2.5 CONCLUSIONS

The total calculated dose-risk to the 50-mile population from airborne releases from an AP1000 reactor at VCSNS would be 0.103 person-rem per reactor year (Table 7.2-1); the dose risk from short-term exposure pathways only is 0.023

person-rem per reactor year. This latter value is less than the 0.043 person-rem per reactor year reported by Westinghouse in the DCD (Westinghouse 2008). Westinghouse did not include long-term (chronic) exposure pathways in their dose-risk, and the VCSNS value does. Therefore, SCE&G concludes that the site-specific VCSNS AP1000 severe accident dose-risk is less than that predicted for the generic AP1000 analysis.

The AP1000 dose-risk at the VCSNS site is less than the population risk for all current reactors that have undergone license renewal, and less than that for the five reactors analyzed in NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* (U.S. NRC 1989). As reported in NUREG-1811 (U.S. NRC 2006), the minimum dose-risk for reactors currently undergoing license renewal is 0.55 person-rem per reactor year. The airborne pathway dose-risk from severe accidents for Unit 1 is 0.95 person-rem per reactor year (SCE&G 2002).

SCE&G's comparative analysis indicates that risk from the surface water pathway is SMALL. The risks of groundwater contamination from an AP1000 accident would be much less than the risk from surface water contamination for currently licensed reactors. The risk of groundwater contamination from an AP1000 accident is smaller than the risk from currently licensed reactors. Additionally, interdiction could substantially reduce the groundwater pathway risks.

For comparison, as reported in Section 5.4, the total collective dose from Units 2 and 3 normal operations is expected to be 34.5 person-rem per year. As previously described, dose-risk is dose times frequency. Normal operations have a frequency of one. Therefore, the two-unit dose-risk for normal operations is 34.5 person-rem per reactor year. Comparing this value to the severe accident two-unit dose-risk of 0.206 (0.103 times 2) person-rem per reactor year indicates that the dose-risk from severe accidents is less than 1% of the dose-risk from normal operations.

The probability-weighted risk of cancer fatalities (early and late) from a severe accident for Unit 2 or 3 is reported in Table 7.2-1 as 6.4 × 10-5 fatalities per reactor year. The probability of an individual dying from any cancer from any cause is approximately 0.23 over a lifetime.

Section 7.2 References

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Table 7.2-1
Impacts to the Population and Land from Severe Accidents Analysis for the AP1000

		Environmental Risk					
			Number of Fatalities (per reactor year)				
Accident Category	Accident Frequency (per reactor year)[a]	Population Dose-Risk (person-rem/ reactor year)	Early	Late	Cost in Dollars (per reactor year)	Water Ingestion Dose (person-rem/reactor year)	Land Requiring Decontamination (acres/reactor year)
Intact Containment	2.2×10^{-7}	1.3×10^{-3}	0.0	7.8×10^{-7}	0.15	1.0×10^{-5}	2.5×10^{-6}
Containment Bypass	1.1×10^{-8}	0.081	2.2×10^{-8}	5.0×10^{-5}	220	5.4×10^{-3}	2.5×10^{-3}
Containment Isolation Failure	1.3×10^{-9}	3.0×10^{-3}	1.7×10^{-10}	1.8×10^{-6}	7.1	1.1×10^{-4}	1.1×10^{-4}
Early Containment Failure	7.5×10^{-9}	0.018	1.4×10^{-9}	1.1×10^{-5}	44	8.2×10^{-4}	6.1×10^{-4}
Intermediate Containment Failure	1.9×10^{-10}	6.1×10^{-4}	1.0×10^{-11}	3.7×10^{-7}	1.7	1.1×10^{-5}	3.1×10^{-5}
Late Containment Failure	3.5×10^{-13}	1.6×10^{-6}	0.0	9.7×10^{-10}	5.6×10^{-3}	2.3×10^{-9}	7.4×10^{-8}
Total	2.4×10^{-7}	0.10	2.4×10^{-8}	6.4×10^{-5}	270	6.4×10^{-3}	3.2×10^{-3}

a) Westinghouse (2004).

Table 7.2-2
Comparison to NRC Safety Goals

	Safety Risk	
	Early Fatality Risk (individual 0-1 mile) (deaths per reactor year)	**Late Fatalities (0-10 mile cancers) (deaths per year per reactor year)**
Safety Goal[a]	$<3.8 \times 10^{-7}$	$<1.9 \times 10^{-6}$
VCSNS Unit 2 or 3	1.4×10^{-10}	3.5×10^{-12}

a) USCB (2007), CDC (2006a), CDC (2006b), and CDC (2007)

7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES

Regulations of the Council on Environmental Quality regarding the National Environmental Policy Act require that a discussion on environmental consequences include mitigation measures (40 CFR 1502.16(h)). The Council on Environmental Quality has stated that mitigation measures should be considered even for impacts that, by themselves, would not be significant, if the overall proposed action could have significant impacts.

As described in Section 7.2, Westinghouse performed a generic severe accident analysis for the AP1000 as part of the design certification process (Westinghouse 2007). The Westinghouse analysis determined that severe accident impacts are small and that no potential mitigating design alternatives are cost-effective, that is, appropriate mitigating measures are already incorporated into the plant design. Section 7.2 extends the Westinghouse generic severe accident analysis to examine the SCE&G proposed new nuclear units at VCSNS and determined that the generic conclusions remain valid for the VCSNS site. The analysis in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented at SCE&G's site to mitigate these small impacts.

7.3.1 THE SAMA ANALYSIS PROCESS

Design or procedural modifications that could mitigate the consequences of a severe accident are known as severe accident mitigation alternatives (SAMAs). In the past, SAMAs were known as SAMDAs, severe accident mitigation design alternatives, which primarily focused on design changes and did not consider procedural modification SAMAs. The Westinghouse 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 four steps:

1. Define the base case — The base case is the dose-risk and cost-risk of a severe accident before implementation of any SAMAs. A plant's probabilistic risk assessment is a primary source of data in calculating the base case. The base case risks are converted to a monetary value to use for screening SAMAs. Section 7.2 presents the base case for a single AP1000 unit at the VCSNS site, without the monetization step.

2. Identify and screen potential SAMAs — Potential SAMAs can be identified from the plant's Individual Plant Examination, the plant's probabilistic risk assessment, and the results of other plants' SAMA analyses. This list of potential SAMAs is assigned a conservatively low implementation cost based on historical costs, similar design changes and/or engineering judgement, then compared to the base case screening value. SAMAs with higher implementation cost than the base case are not evaluated further.

3. Determine the cost and net value of each SAMA — Each SAMA remaining after Step 2 has a detailed engineering cost evaluation developed using

current plant engineering processes. If the SAMA continues to pass the screening value Step 4 is performed.

4. Determine the benefit associated with each screened SAMA — Each SAMA that passes the screening in Step 3 is evaluated using the probabilistic risk assessment model to determine the reduction in risk associated with implementation of the proposed SAMA. The reduction in risk benefit is then monetized and compared to the detailed cost estimate. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

In the absence of a completed plant with established procedural controls, the SCE&G analysis is limited to demonstrating that the VCSNS site is bounded by the Westinghouse Design Control Document analysis and determining what magnitude of plant-specific design or procedural modification would be cost-effective. Determining the magnitude of cost-effective design or procedural modifications is the same as “1. Define base case” for existing nuclear units. The base case benefit value is calculated by assuming the current dose-risk of the unit could be reduced to zero and assigning a defined dollar value for this change in risk. Any design or procedural change cost that exceeded the benefit value would not be considered cost-effective. The dose-risk and cost-risk results (Section 7.2 analyses) are monetized in accordance with methods established in NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*, (U.S. NRC 1997). NUREG/BR-0184 presents methods for determination of the value of decreases in risk using four types of attributes: public health, occupational health, offsite property, and onsite property. Any SAMAs in which the conservatively low implementation cost exceeds the base case monetization would not be expected to pass the screening in Step 2. If the SCE&G baseline analysis produces a value that is below that expected for implementation of any reasonable SAMA, no matter how inexpensive, then the remaining steps of the SAMA analysis are not necessary.

7.3.2 THE AP1000 SAMDA ANALYSIS

The Westinghouse SAMDA analysis is presented in Appendix 1B of the AP1000 Design Control Document (Westinghouse 2008). Westinghouse compiled a list of potential SAMDAs based on the AP600 analysis and other plant designs and suggestions from the AP600/AP1000 design staff. Some SAMDAs were then screened out based on their inapplicability to the AP1000 or the fact that they were already included in the AP1000 design. Rough implementation costs that far exceeded any reasonable benefit were also excluded. The 15 SAMDAs that passed the screening process are as follows and are described more fully in the AP1000 Design Control Document.

- Chemical volume and control system upgrade to mitigate small loss-of-coolant accidents
- Filtered containment vent
- Normal residual heat removal system inside containment
- Self-actuating containment isolation valves
- Passive containment spray
- Active high-pressure safety injection system
- Steam generator shell-side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank
- Increased steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse in-containment refueling water storage tank injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High-pressure containment design
- Improved reliability of diverse actuation system

These remaining SAMDAs were quantified by the probabilistic risk assessment model to determine the reduction in risk for implementing the SAMDA. Each SAMDA was assumed to reduce the risk of the accident sequences that they address to zero, a conservative assumption. Using the cost-benefit methodology of NUREG/BR-0184, the maximum averted cost risk was calculated for each SAMDA. The maximum averted cost risk calculation used the dose-risks and cost-risks calculated for the severe accidents described in Subsection 7.2.1. Westinghouse calculated the base case maximum averted cost risk to be $21,000 using a 7% discount rate.

Westinghouse next compared the implementation costs for each SAMDA to the $21,000 value and 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 $30,000, with the others having costs at least an order of magnitude greater. The one potential SAMDA was further evaluated but not found to be cost-effective.

In its Finding of No Significant Impact relating to the certification of the AP1000 design, (U.S. NRC 2005) concluded, “none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that any other design changes would be justified in the future on the basis of person-rem exposure because the estimated CDFs [core damage frequencies] are very low on an absolute scale.”

7.3.3 MONETIZATION OF THE VCSNS UNITS 2 AND 3 BASE CASE

The principal inputs to the calculations are the core damage frequency (reported in Section 7.2), dose-risk and cost-risk (reported in Table 7.2-1), dollars per person-rem ($2,000 as provided by NRC in NUREG/BR-0184), licensing period (40 years), and economic discount rate (7% and 3% are NRC precedents). Both the Westinghouse and SCE&G severe accident analyses described in Section 7.2

calculate risks from internal events. For this SAMDA analysis, the base-case core damage frequency, dose-risk, and cost-risk for internal events were escalated to account for external events, both at power and at shutdown. As explained in Westinghouse (2008), dose-risk and cost-risk were scaled up by the ratio of the total (internal and external events) frequency divided by the internal events frequency ($5.1 \times 10^{-7}/2.4 \times 10^{-7}$ per reactor year). With these inputs, the monetized value of reducing the base case core damage frequency to zero is presented in Table 7.3-1. The monetized value, known as the maximum averted cost-risk, is conservative because no SAMA can reduce the core damage frequency to zero.

The maximum averted cost risk of $31,681 for a single AP1000 at SCE&G's proposed site is so low that SCE&G does not believe there are any design changes, over those already incorporated into the advanced reactor designs, that could be determined to be cost-effective. With a conservative 3% discount rate, the valuation of the averted risk is $56,923. These values remain well below the cost of implementing the SAMDAs, with the exception of the self-actuating containment valves design alternative. As demonstrated in Westinghouse (2008), and confirmed for SCE&G, the benefit of the latter design change is much less than its implementation cost.

Accordingly, further evaluation of design-related SAMAs is not warranted. SCE&G does not believe that administrative SAMAs, such as those relating to procedures or training, are appropriate for evaluation. The purpose of this analysis is to demonstrate that design changes for an AP1000 at the VCSNS site are not cost beneficial. Evaluation of administrative SAMAs would not be appropriate until a plant design is finalized and plant administrative processes and procedures are being developed. COLA Part 2, Final Safety Analysis Report, Chapter 18, Human Factors Engineering, and the AP1000 Design Control Document (Westinghouse 2008) describe the human factors engineering process that would apply to development of procedures and training. The process addresses risk-important tasks, emergency response guidelines, and interactions with risk-significant systems, structures, and components. Although a SAMA analysis would not be performed at that time (SAMA is a component of National Environmental Policy Act documentation), risk-informed decision-making techniques would be used, as appropriate, during procedure and training development.

Section 7.3 References

1. U.S. NRC 1997, U.S. NRC, Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184. Office of Nuclear Reactor Regulation. Washington, D.C. January. 1997.

2. U.S. NRC 2005, U.S. NRC, *Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the AP1000 Standard Plant Design*. Docket No. 52-006, SECY 05-0227 (accession number ML053630176). Washington D.C. January 24, 2005.

3. Westinghouse (Westinghouse Electric Corporation) 2008. *Design Control Document,* Revision 17, Appendix 1B, “Severe Accident Mitigation Design Alternatives.” U.S. NRC, Washington, D.C., September 22, 2008.

Table 7.3-1
Monetization of the SCE&G AP1000 Base Case

	7% Discount Rate	3% Discount Rate
Offsite exposure cost	$5,843	$10,144
Offsite economic cost	$7,608	$13,208
Onsite exposure cost	$241	$487
Onsite cleanup cost	$7,353	$15,335
Replacement power cost	$10,637	$17,750
Total	$31,681	$56,923

7.4 TRANSPORTATION ACCIDENTS

Subsection 5.11.2 describes the methodology used by SCE&G to analyze the impacts of transportation of radioactive materials, including accidents.

NRC analyzed the transportation of radioactive materials in its assessments of environmental impacts for the proposed ESP sites at North Anna, Clinton, and Grand Gulf (U.S. NRC 2006a, 2006b, and 2006c). SCE&G reviewed the NRC analyses for guidance in assessing transportation impacts for the proposed AP1000 units at the VCSNS site.

7.4.1 RADIOLOGICAL IMPACTS OF TRANSPORTATION ACCIDENTS

7.4.1.1 Transportation of Unirradiated Fuel

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 CFR 51.52. Accident risks are calculated as frequency times consequence. Accident frequencies for transportation of fuel to future reactors are expected to be lower than those used in the analysis in AEC 1972, which forms the basis for Table S-4 of 10 CFR 51.52, because of improvements in highway safety and security. Traffic accident, injury, and fatality rates have fallen over the past 30 years. The consequences of accidents that are severe enough to result in a release of unirradiated particles to the environment from advanced light water reactors (LWR) fuels are not significantly different from those for current generation LWRs. The fuel form, cladding, and packaging are similar to those LWRs analyzed in AEC 1972. Consequently, as described in NUREG-1811 (U.S. NRC 2006a), NUREG-1815 (U.S. NRC 2006b), and NUREG-1817 (U.S. NRC 2006c), the risks of accidents during transportation of unirradiated fuel to the VCSNS site would be expected to be smaller than the reference LWR results listed in Table S-4.

7.4.1.2 Transportation of Spent Fuel

SCE&G used the RADTRAN 5 computer code to estimate impacts of transportation accidents involving spent fuel shipments. RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences (i.e., “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 radionuclide inventories of the advanced LWR spent fuel after five years decay were obtained from INEEL (2003) and a screening analysis performed to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations. This screening identified the radionuclides that would contribute more than 99.999% of the dose from inhalation of radionuclides released following a transportation accident (U.S. NRC 2006a, 2006b, and 2006c). The dominant radionuclides are similar regardless of the fuel type. The spent fuel inventory used in this analysis for the AP1000 is presented in Table 7.4-1.

The specific quantities and characteristics of the crud deposited on AP1000 spent fuel from corrosion products generated elsewhere in the reactor coolant system are unknown at this time because of insufficient operating experience. The spent fuel transportation accident risks were calculated using the dominant radionuclide inventory presented in Table 7.4-1. Westinghouse subsequently provided estimates for those radionuclides expected to be present in the form of crud. Assuming a minimum decay period of 5 years, the expected Co-60 activity as crud is approximately 4.09 Ci/MTU. Sb-125 is also expected to be present as crud. The estimated activity of Sb-125 as crud (0.111 Ci/MTU) is less than 0.003% of the total Sb-125 inventory in the fuel. These crud values were not included in the RADTRAN 5 calculations. However, the total activity of the crud components is roughly five orders of magnitude lower than the fission and activation products of the fuel. Therefore, from a radiological dose standpoint, the crud contribution would be negligible.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71, "Packaging and Transportation of Radioactive Material." 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. According to Sprung et al. (2000), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01% (i.e., more than 99.99% of all accidents would result in no release of radioactive material from the shipping cask). This analysis assumed that shipping casks for advanced LWR spent fuels would provide mechanical and thermal protection of the spent fuel cargo that is equivalent to that for current generation spent fuel.

SCE&G performed the RADTRAN 5 accident risk calculations using radionuclide inventories per shipment for the spent fuel from the AP1000 assuming 0.5 MTU per shipment. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (MTU per year) to derive estimates of the annual accident risks associated with spent fuel shipments from the AP1000. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 23 MTU per year for the AP1000. (This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2).

SCE&G used the release fractions for current generation LWR fuels to approximate the impacts from the advanced LWR spent fuel shipments. This assumes that the fuel materials and containment systems (i.e., cladding, fuel coatings) behave similarly to current LWR fuel under applied mechanical and thermal conditions.

Using RADTRAN 5, SCE&G calculated the population dose from the released radioactive material for four possible exposure pathways:

1. External dose from exposure to the passing cloud of radioactive material.

2. External dose from the radionuclides deposited on the ground by the passing plume (the radiation exposure from this pathway was included even though the area surrounding a potential accidental release would be evacuated and decontaminated, thus preventing long-term exposures from this pathway).

3. Internal dose from inhalation of airborne radioactive contaminants.

4. Internal dose from resuspension of radioactive materials that were deposited on the ground (the radiation exposures from this pathway were included even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures).

The analysis assumed interdiction of foodstuffs and evacuation after an accident so no internal dose due to ingestion of contaminated foods was calculated. External doses from increased radiation fields surrounding a shipping cask with damaged shielding, was considered but not included in the analysis. It is possible that shielding materials incorporated into the cask structures could become damaged as a result of an accident. However, SCE&G did not include loss of shielding events in its analysis because their contribution to spent fuel transportation risk is much smaller than the dispersal accident risks from the pathways listed above.

SCE&G calculated the environmental consequences of transportation accidents when shipping spent fuel from the VCSNS site to a spent fuel repository assumed to be at Yucca Mountain, Nevada. The shipping distances and population distribution information for the route were the same as those used for the "incident-free" transportation impacts analysis (described in Subsection 5.11.2).

Table 7.4-2 presents unit (per MTU) accident risks associated with transportation of spent fuel from the VCSNS site to the proposed Yucca Mountain repository. The accident risks are provided in the form of a collective population dose (i.e., person-rem over the shipping campaign). The table also presents estimates of accident risk per reactor year normalized to the reference reactor analyzed in AEC 1972. SCE&G also calculated the transportation accident impacts for the alternative sites (Savannah River Site, Cope Generating Station, Saluda County green field site) within the region of interest. The Fa-1 site is located 5 miles north-northwest of VCSNS and the population dose associated with accidents during transportation of spent fuel from that location would be the same as those estimated for VCSNS.

SCE&G estimated the risk to the public from radiation exposure using the nominal probability coefficient for total detrimental health effects (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 1 × 10^6 person-rem) per reference reactor year from ICRP Publication 60 (ICRP 1991). These values are presented in Table 7.4-2. These estimated risks are quite small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation. Therefore, no detectable increases in environmental risk effects are expected as a result of accidents that may result from shipping spent fuel from the VCSNS site to a spent fuel disposal repository.

7.4.2 NONRADIOLOGICAL IMPACTS OF TRANSPORTATION ACCIDENTS

Nonradiological impacts would include the projected number of accidents, injuries, and fatalities that could result from shipments of radioactive materials to or from the VCSNS site and return of empty containers. Nonradiological impacts were estimated using accident, injury, and fatality rates from Table 4 of *State-Level Accident Rates for Surface Freight Transportation: A Reexamination* (Saricks and Tompkins 1999). These data are representative of the traffic accident, injury, and fatality rates for heavy truck shipments similar to those that would be used to transport radioactive materials to and from the site. These rates (measured in impacts per vehicle-mile traveled) are multiplied by the annual numbers of shipments and estimated travel distances for the shipments to estimate annual impacts for the AP1000. These estimates include the human health impacts projected to result from traffic accidents involving shipments of radioactive materials; they do not consider the radiological or hazardous characteristics of the cargo.

7.4.2.1 Transportation of Unirradiated Fuel

The nonradiological accident impacts that could result from shipments of unirradiated fuel to VCSNS and return of empty containers from the site are presented in Table 7.4-3. The nonradiological impacts for the reference LWR analyzed in WASH-1238 are also shown for comparison. Nationwide median rates for interstate highway transportation from Saricks and Tompkins (1999) were used to estimate the annual impacts. Consistent with the incident-free transportation analysis described in Section 5.11.2, an average one-way shipping distance of 2000 miles was used to evaluate the unirradiated fuel shipments. The differences between the reference LWR and AP1000 results are due to the lower number of shipments per year (when normalized for electrical output) projected for the AP1000 units at VCSNS. The values presented in Table 7.4-3 would be doubled for a two-unit plant.

7.4.2.2 Transportation of Spent Fuel

The general approach to calculating the nonradiological impacts for spent fuel shipments is similar to that for other radioactive materials shipments. The main difference is the spent fuel shipping route characteristics are better defined allowing the state-specific accident statistics in Saricks and Tompkins (1999) to be used in the analysis. State-by-state shipping distances and road types were obtained from the TRAGIS output file (see Subsection 5.11.2.2 for a discussion of the TRAGIS routing model). The shipping distances were doubled to allow for return shipments of empty containers to VCSNS. This information, the annual number of shipments, and state-specific accident statistics were used to estimate the nonradiological impacts presented in Table 7.4-4. The values presented in Table 7.4-4 would be doubled for a two-unit plant.

7.4.2.3 Transportation of Radioactive Waste

Nonradiological impacts of radioactive waste shipments were calculated using the same general approach as the unirradiated fuel shipments. A shipping distance of 500 miles was assumed consistent with the analysis in WASH-1238. Because the destination of the waste shipments is not known, the national median accident, injury, and fatality rates from Saricks and Tompkins (1999) were used to calculate the values presented in Table 7.4-5. The nonradiological impacts for the reference LWR analyzed in WASH-1238 are also shown for comparison. The differences between the reference LWR and AP1000 are due to the lower number of radioactive waste shipments (when normalized for electrical output) projected for the AP1000. The values presented in Table 7.4-5 would be doubled for a two-unit plant.

7.4.3 CONCLUSION

The transportation accident risks results for the AP1000 for unirradiated and spent fuel and radioactive waste are less than the nonradiological effects of accidents in transportation (one fatal injury in 100 reactor years and one nonfatal injury per 10 reactor years) indicated in Table S-4. Based on this analysis, the overall transportation accident risks associated with unirradiated fuel, spent fuel, and radioactive waste shipments from the proposed AP1000 units at VCSNS are consistent with the risks associated with transportation of the radioactive materials from current generation reactors presented in WASH-1238 and Table S-4 of 10 CFR 51.52 (reproduced in Table 5.11-1) and thus will be SMALL.

Section 7.4 References

1. ICRP (International Commission on Radiological Protection) 1991. *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Pergammon Press, Oxford, United Kingdom, 1991.

2. INEEL (Idaho National Engineering and Environmental Laboratory) 2003. Early Site Permit Environmental Report Sections and Supporting Documentation, Engineering Design File Number 3747, Idaho Falls, Idaho, 2003.

3. U.S. AEC (U.S. Atomic Energy Commission) 1972, *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants*, WASH-1238, Washington D.C., December 1972.

4. U.S. NRC 2006a, *Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site*, NUREG-1811, Office of New Reactors, Washington D.C., December 2006.

5. U.S. NRC 2006b, *Environmental Impact Statement for an Early Site Permit (ESP) at the Exelon ESP Site*, NUREG-1815, Office of Nuclear Reactor Regulation, Washington D.C., July 2006.

6. U.S. NRC 2006c, *Environmental Impact Statement for an Early Site Permit (ESP) at the Grand Gulf ESP Site*, NUREG-1817, Office of Nuclear Reactor Regulation, Washington D.C., April 2006.

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8. Sprung, J. L., D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, 2000. *Reexamination of Spent Fuel Shipment Risk Estimates*, NUREG/CR-6672, Volume 1, Office of Nuclear Material Safety and Safeguards, U.S. NRC, Washington D.C., March 2000.

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

Radionuclide	AP1000 Inventory Ci/MTU
Am-241	727
Am-242m	13.1
Am-243	33.4
Ce-144	8,870
Cm-242	28.3
Cm-243	30.7
Cm-244	7,750
Cm-245	1.21
Cs-134	4.80×10^4
Cs-137	9.31×10^4
Eu-154	9,130
Eu-155	4,620
Pm-147	1.76×10^4
Pu-238	6,070
Pu-239	255
Pu-240	543
Pu-241	6.96×10^4
Pu-242	1.82
Ru-106	1.55×10^4
Sb-125	3,830
Sr-90	6.19×10^4
Y-90	6.19×10^4

Source: NRC (2006a, 2006b, 2006c)
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 per MTU)[a]	MTU per reference reactor year	Population Dose (person-rem per reference reactor year)[a]	Total detrimental Health effects per reference reactor year
VCSNS/Fa-1[b]	5.26×10^{-8}	19.5	1.03×10^{-6}	7.48×10^{-10}
SRS	1.01×10^{-7}	19.5	1.96×10^{-6}	1.43×10^{-9}
Cope	6.08×10^{-8}	19.5	1.19×10^{-6}	8.65×10^{-10}
Saluda County	5.18×10^{-8}	19.5	1.01×10^{-6}	7.37×10^{-10}

a) Value presented is the product of probability times collective dose.
b) The Fa-1 site is located 5 miles north-northwest of VCSNS. The accident risk for the Fa-1 site would be almost identical to that for VCSNS.

Table 7.4-3
Nonradiological Impacts of Transporting Unirradiated Fuel to VCSNS for the AP1000

Reactor	Total Shipments Normalized to Reference LWR	One-Way Shipping Distance (miles)	Total Round-Trip Shipping Distance (miles)	Annual Impacts		
				Fatalities per Year	Injuries per Year	Accidents per Year
Reference LWR	252	2000	1.01×10^6	3.7×10^{-4}	0.0078	0.011
AP1000	196	2000	7.84×10^5	2.9×10^{-4}	0.0061	0.0089

Table 7.4-4
Nonradiological Impacts of Transporting Spent Fuel from VCSNS for the AP1000

State	Highway Type	One-Way Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
Arizona	Interstate	357	4.2×10^{-4}	0.0053	0.0059
Arkansas	Interstate	283	2.2×10^{-4}	0.0035	0.0048
California	Interstate	265	2.3×10^{-4}	0.0041	0.0053
Nevada	Primary	33	6.9×10^{-5}	0.0011	0.0016
	Interstate	107	8.9×10^{-5}	0.0020	0.0030
New Mexico	Interstate	371	$5.5 x 10^{-4}$	0.0054	0.0053
North Carolina	Interstate	86	1.6×10^{-4}	0.0034	0.0037
Oklahoma	Interstate	332	5.6×10^{-4}	0.012	0.011
South Carolina	Primary	14	4.4×10^{-5}	5.6×10^{-4}	8.0×10^{-4}
	Interstate	86	2.8×10^{-4}	0.0036	0.0050
Tennessee	Interstate	459	$5.8 x 10^{-4}$	0.0053	0.0071
Texas	Interstate	176	2.9×10^{-4}	0.012	0.013
Totals		2568	0.0035	0.058	0.067

Table 7.4-5
Nonradiological Impacts of Transporting Radioactive Waste from VCSNS for the AP1000

Reactor	Shipments per Year Normalized to Reference LWR	One-Way Shipping Distance (miles)	Annual Impacts: Fatalities per Year	Annual Impacts: Injuries per Year	Annual Impacts: Accidents per Year
Reference LWR	46	500	6.8×10^{-4}	0.014	0.021
AP1000	21	500	3.1×10^{-4}	0.0065	0.0096