Chapter 7 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, 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 design bases accidents (DBAs) considered in this section are from the *AP1000 Design Control Document (DCD)* **(Westinghouse 2005)** and SSAR Chapter 15 in Part 2 of this ESP application. 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 VEGP site without undue risk to the health and safety of the public.

7.1.2 Evaluation Methodology

The AP1000 DCD presents the radiological consequences for the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with VEGP site data to demonstrate that the DCD analyses are bounding for the VEGP 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 (χ/Q values). Since the site-specific χ/Q values are bounded by the DCD χ/Q values, this approach demonstrates that the site-specific doses are within those calculated in the DCD.

SSAR Chapter 15, Accident Analysis, uses conservative assumptions to perform bounding safety analyses that substantially overstate the environmental impact of the identified accidents.

Among the conservative assumptions in SSAR Chapter 15 is the use of time-dependent χ/Q values corresponding to the top 5th 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 50th percentile site-specific χ/Q values during the first two hours of the accident, reflecting more realistic meteorological conditions. The χ/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 (RG 1.145) with site-specific meteorological data. As indicated in Section 2.7.5, the RG 1.145 methodology is implemented in the NRC-sponsored PAVAN computer program. This program computes χ/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 (non direction-specific) χ/Q values. For a given location, either the EAB or the LPZ, the 0 – 2 hour χ/Q value is the 50th percentile overall value calculated by PAVAN. For the LPZ, the χ/Q values for all subsequent times are calculated by logarithmic interpolation between the 50th percentile χ/Q value and the annual average χ/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 χ/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 **(EPA 1988)**, while the EDE is based on the dose conversion factors in Federal Guidance Report 12 **(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 (RG 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 RG 1.183, based on 102 percent 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 to 7.1-10.

7.1.4 Radiological Consequences

Environmental report design basis accident doses are evaluated based on more realistic meteorological conditions than in the site safety analysis report. 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 χ/Q value, developed in Section 2.7.5.2, to the

DCD χ/Q value as indicated in *AP1000 Accident Releases and Doses as Function of Time* **(Westinghouse 2006b)**. The time-dependent DCD χ/Q values and the time-dependent site χ/Q values and their ratios are shown in Table 7.1-11. As all site χ/Q values are bounded by DCD χ/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 to 7.1-22. For each accident, the EAB dose shown is for the two-hour period that yields the maximum dose, in accordance with RG 1.183.

The results of the VEGP 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 apply 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 RG 1.183. Where applied, the more restrictive dose limit is either 10 or 25 percent 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 to 7.1-22 are from RG 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 RG 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.

Table 7.1-1 Selection of Accidents

SRP/DCD Section	SRP 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	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 Pre-Existing Iodine Spike

	Activity Release (Ci)				
Isotope	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-72 hr**	**Total**
Kr-85m	6.86E-02	1.14E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	2.82E-01	8.46E-01	2.25E+00	6.69E+00	1.01E+01
Kr-87	2.76E-02	1.34E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	1.12E-01	1.37E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	1.28E-01	3.79E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	1.59E-01	4.51E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	1.18E+01	3.45E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.04E-03	1.33E-05	0.00E+00	0.00E+00	3.06E-03
Xe-135	3.10E-01	6.90E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	3.99E-03	1.14E-05	0.00E+00	0.00E+00	4.00E-03
I-130	3.59E-01	1.42E-01	2.09E-01	1.33E-01	8.44E-01
I-131	2.40E+01	1.21E+01	3.10E+01	8.22E+01	1.49E+02
I-132	3.05E+01	4.14E+00	8.06E-01	6.55E-03	3.55E+01
I-133	4.34E+01	1.90E+01	3.53E+01	3.98E+01	1.37E+02
I-134	6.74E+00	1.63E-01	1.43E-03	4.54E-09	6.91E+00
I-135	2.60E+01	8.16E+00	7.54E+00	1.71E+00	4.34E+01
Cs-134	1.90E+01	1.95E-01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.82E+01	2.86E-01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.37E+01	1.41E-01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	1.02E-03	4.42E-07	0.00E+00	1.01E+01
Total	2.15E+02	8.15E+01	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 hr**	**2-8 hr**	**8-24 hr**	**24-72 hr**	**Total**
Kr-85m	6.86E-02	1.14E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	2.82E-01	8.46E-01	2.25E+00	6.69E+00	1.01E+01
Kr-87	2.76E-02	1.34E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	1.12E-01	1.37E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	1.28E-01	3.79E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	1.59E-01	4.51E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	1.18E+01	3.45E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.04E-03	1.33E-05	0.00E+00	0.00E+00	3.06E-03
Xe-135	3.10E-01	6.90E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	3.99E-03	1.14E-05	0.00E+00	0.00E+00	4.00E-03
I-130	4.20E-01	9.95E-01	1.58E+00	1.01E+00	4.01E+00
I-131	2.60E+01	5.73E+01	1.56E+02	4.13E+02	6.53E+02
I-132	4.62E+01	9.74E+01	2.24E+01	1.82E-01	1.66E+02
I-133	4.91E+01	1.14E+02	2.27E+02	2.55E+02	6.45E+02
I-134	1.34E+01	1.86E+01	2.65E-01	8.42E-07	3.23E+01
I-135	3.24E+01	7.74E+01	7.83E+01	1.77E+01	2.06E+02
Cs-134	1.90E+01	1.95E-01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.82E+01	2.86E-01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.37E+01	1.41E-01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	1.02E-03	4.42E-07	0.00E+00	1.01E+01
Total	2.51E+02	4.03E+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)				
	No Feedwater	Feedwater Available			
Isotope	0-1.5 hr	0-2 hr	2-8 hr	6-8 hr	Total
Kr-85m	8.16E+01	1.05E+02	1.74E+02	4.13E+01	2.79E+02
Kr-85	7.58E+00	1.01E+01	3.03E+01	1.01E+01	4.04E+01
Kr-87	1.20E+02	1.43E+02	6.97E+01	5.43E+00	2.13E+02
Kr-88	2.08E+02	2.62E+02	3.20E+02	6.05E+01	5.82E+02
Xe-131m	3.77E+00	5.03E+00	1.49E+01	4.95E+00	1.99E+01
Xe-133m	2.02E+01	2.69E+01	7.64E+01	2.48E+01	1.03E+02
Xe-133	6.66E+02	8.87E+02	2.60E+03	8.57E+02	3.49E+03
Xe-135m	3.24E+01	3.28E+01	1.43E-01	2.68E-06	3.30E+01
Xe-135	1.59E+02	2.08E+02	4.64E+02	1.32E+02	6.72E+02
Xe-138	1.29E+02	1.30E+02	3.72E-01	3.01E-06	1.30E+02
I-130	8.45E-01	1.17E-01	1.33E+00	5.65E-01	1.45E+00
I-131	3.77E+01	5.39E+00	7.51E+01	3.46E+01	8.05E+01
I-132	2.79E+01	3.45E+00	1.48E+01	3.95E+00	1.83E+01
I-133	4.86E+01	6.86E+00	8.29E+01	3.64E+01	8.98E+01
I-134	2.88E+01	2.76E+00	2.98E+00	2.09E-01	5.74E+00
I-135	4.19E+01	5.68E+00	5.22E+01	2.05E+01	5.79E+01
Cs-134	1.29E+00	1.82E-01	2.40E+00	1.11E+00	2.59E+00
Cs-136	5.63E-01	8.45E-02	7.79E-01	3.47E-01	8.63E-01
Cs-137	7.74E-01	1.10E-01	1.41E+00	6.51E-01	1.52E+00
Cs-138	6.08E+00	7.29E-01	3.35E+00	1.13E+00	4.08E+00
Rb-86	1.33E-02	1.83E-03	2.73E-02	1.27E-02	2.91E-02
Total	1.62E+03	1.84E+03	3.99E+03	1.23E+03	5.82E+03

Note: The release period of 6–8 hr yields the maximum 2-hr EAB dose with feedwater available.

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

	Activity Release (Ci)					
Isotope	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-96 hr**	**96-720 hr**	**Total**
Kr-85m	1.12E+02	6.48E+01	3.87E+01	1.77E+00	2.51E-05	2.18E+02
Kr-85	5.01E+00	5.60E+00	1.49E+01	3.35E+01	2.88E+02	3.47E+02
Kr-87	1.82E+02	2.60E+01	1.03E+00	8.37E-05	0.00E+00	2.09E+02
Kr-88	2.91E+02	1.18E+02	3.49E+01	3.59E-01	8.41E-09	4.45E+02
Xe-131m	4.94E+00	5.46E+00	1.42E+01	2.86E+01	1.16E+02	1.69E+02
Xe-133m	2.67E+01	2.81E+01	6.49E+01	8.45E+01	5.31E+01	2.57E+02
Xe-133	8.79E+02	9.58E+02	2.40E+03	4.27E+03	8.45E+03	1.70E+04
Xe-135m	7.34E+01	5.30E-02	4.33E-09	0.00E+00	0.00E+00	7.35E+01
Xe-135	2.15E+02	1.72E+02	2.09E+02	4.35E+01	1.79E-01	6.39E+02
Xe-138	2.99E+02	1.38E-01	3.19E-09	0.00E+00	0.00E+00	2.99E+02
I-130	4.90E+00	7.28E+00	4.32E+00	2.03E-01	2.95E-04	1.67E+01
I-131	1.36E+02	2.45E+02	2.31E+02	3.10E+01	1.68E+01	6.60E+02
I-132	1.53E+02	9.94E+01	9.85E+00	8.24E-03	0.00E+00	2.62E+02
I-133	2.72E+02	4.40E+02	3.18E+02	2.28E+01	2.41E-01	1.05E+03
I-134	1.66E+02	2.85E+01	1.37E-01	4.48E-08	0.00E+00	1.95E+02
I-135	2.39E+02	2.97E+02	1.19E+02	2.39E+00	7.32E-05	6.57E+02
Cs-134	3.08E+01	6.22E+01	6.03E+01	7.76E+00	5.16E+00	1.66E+02
Cs-136	8.79E+00	1.75E+01	1.67E+01	2.05E+00	6.58E-01	4.57E+01
Cs-137	1.79E+01	3.62E+01	3.51E+01	4.52E+00	3.05E+00	9.68E+01
Cs-138	1.09E+02	7.05E+00	1.68E-03	0.00E+00	0.00E+00	1.16E+02
Rb-86	3.62E-01	7.27E-01	6.96E-01	8.67E-02	3.42E-02	1.91E+00
Total	3.23E+03	2.62E+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 hr
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 Pre-Existing Iodine Spike

	Activity Release (Ci)			
Isotope	**0-2 hr**	**2-8 hr**	**8-24 hr**	**Total**
Kr-85m	5.53E+01	1.93E+01	7.53E-03	7.46E+01
Kr-85	2.20E+02	1.09E+02	1.34E-01	3.29E+02
Kr-87	2.39E+01	3.61E+00	9.12E-05	2.75E+01
Kr-88	9.22E+01	2.65E+01	5.43E-03	1.19E+02
Xe-131m	9.96E+01	4.88E+01	5.91E-02	1.48E+02
Xe-133m	1.24E+02	5.91E+01	6.61E-02	1.83E+02
Xe-133	9.19E+03	4.47E+03	5.29E+00	1.37E+04
Xe-135m	3.44E+00	5.86E-03	0.00E+00	3.45E+00
Xe-135	2.46E+02	1.02E+02	7.10E-02	3.47E+02
Xe-138	4.56E+00	5.07E-03	0.00E+00	4.57E+00
I-130	1.79E+00	5.39E-02	2.68E-01	2.12E+00
I-131	1.21E+02	5.27E+00	3.06E+01	1.56E+02
I-132	1.42E+02	7.43E-01	1.92E+00	1.44E+02
I-133	2.16E+02	7.63E+00	4.06E+01	2.64E+02
I-134	2.74E+01	4.40E-03	4.23E-03	2.74E+01
I-135	1.27E+02	2.70E+00	1.17E+01	1.42E+02
Cs-134	1.63E+00	6.05E-02	2.16E-01	1.90E+00
Cs-136	2.42E+00	8.86E-02	3.14E-01	2.82E+00
Cs-137	1.17E+00	4.37E-02	1.56E-01	1.37E+00
Cs-138	5.64E-01	2.91E-06	5.73E-07	5.64E-01
Total	1.07E+04	4.85E+03	9.14E+01	1.56E+04

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

Isotope	Activity Release (Ci)			
	0-2 hr	2-8 hr	8-24 hr	Total
Kr-85m	5.53E+01	1.93E+01	7.53E-03	7.46E+01
Kr-85	2.20E+02	1.09E+02	1.34E-01	3.29E+02
Kr-87	2.39E+01	3.61E+00	9.12E-05	2.75E+01
Kr-88	9.22E+01	2.65E+01	5.43E-03	1.19E+02
Xe-131m	9.96E+01	4.88E+01	5.91E-02	1.48E+02
Xe-133m	1.24E+02	5.91E+01	6.61E-02	1.83E+02
Xe-133	9.19E+03	4.47E+03	5.29E+00	1.37E+04
Xe-135m	3.44E+00	5.86E-03	0.00E+00	3.45E+00
Xe-135	2.46E+02	1.02E+02	7.10E-02	3.47E+02
Xe-138	4.56E+00	5.07E-03	0.00E+00	4.57E+00
I-130	8.87E-01	1.62E-01	8.24E-01	1.87E+00
I-131	4.36E+01	1.14E+01	6.76E+01	1.23E+02
I-132	1.47E+02	4.86E+00	1.29E+01	1.65E+02
I-133	9.33E+01	2.00E+01	1.08E+02	2.22E+02
I-134	5.59E+01	6.04E-02	5.94E-02	5.60E+01
I-135	7.61E+01	9.88E+00	4.38E+01	1.30E+02
Cs-134	1.63E+00	6.05E-02	2.16E-01	1.90E+00
Cs-136	2.42E+00	8.86E-02	3.14E-01	2.82E+00
Cs-137	1.17E+00	4.37E-02	1.56E-01	1.37E+00
Cs-138	5.64E-01	2.91E-06	5.73E-07	5.64E-01
Total	1.05E+04	4.88E+03	2.40E+02	1.56E+04

Table 7.1-9 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**	**0-8 hr**	**8-24 hr**	**24-96 hr**	**96-720 hr**	**Total**
I-130	5.64E+01	1.12E+02	5.37E+00	7.10E-01	1.27E-02	1.18E+02
I-131	1.68E+03	3.49E+03	2.66E+02	2.39E+02	7.19E+02	4.71E+03
I-132	1.23E+03	2.14E+03	1.64E+01	1.46E-02	0.00E+00	2.15E+03
I-133	3.23E+03	6.54E+03	3.83E+02	1.04E+02	1.04E+01	7.04E+03
I-134	6.60E+02	1.14E+03	2.96E-01	6.79E-08	0.00E+00	1.14E+03
I-135	2.56E+03	4.89E+03	1.58E+02	6.09E+00	3.16E-03	5.06E+03
Kr-85m	1.42E+03	3.77E+03	1.87E+03	8.56E+01	1.22E-03	5.73E+03
Kr-85	8.31E+01	2.97E+02	7.06E+02	1.59E+03	1.36E+04	1.62E+04
Kr-87	1.10E+03	1.95E+03	4.97E+01	4.05E-03	0.00E+00	1.99E+03
Kr-88	3.11E+03	7.26E+03	1.70E+03	1.75E+01	4.09E-07	8.97E+03
Xe-131m	8.26E+01	2.94E+02	6.79E+02	1.37E+03	5.57E+03	7.91E+03
Xe-133m	4.43E+02	1.54E+03	3.15E+03	4.11E+03	2.58E+03	1.14E+04
Xe-133	1.47E+04	5.19E+04	1.16E+05	2.06E+05	4.07E+05	7.80E+05
Xe-135m	1.06E+01	3.59E+01	2.14E-07	0.00E+00	0.00E+00	3.59E+01
Xe-135	3.15E+03	9.64E+03	1.01E+04	2.11E+03	8.68E+00	2.19E+04
Xe-138	3.11E+01	1.20E+02	1.58E-07	0.00E+00	0.00E+00	1.20E+02
Rb-86	3.04E+00	6.32E+00	2.99E-01	9.83E-02	5.13E-01	7.23E+00
Cs-134	2.58E+02	5.38E+02	2.57E+01	9.11E+00	7.74E+01	6.50E+02
Cs-136	7.33E+01	1.52E+02	7.16E+00	2.28E+00	9.88E+00	1.72E+02
Cs-137	1.51E+02	3.13E+02	1.50E+01	5.32E+00	4.57E+01	3.79E+02
Cs-138	1.50E+02	3.30E+02	2.18E-03	0.00E+00	0.00E+00	3.30E+02
Sb-127	2.42E+01	4.80E+01	2.29E+00	5.67E-01	7.82E-01	5.16E+01
Sb-129	5.10E+01	8.94E+01	1.51E+00	4.95E-03	4.90E-08	9.09E+01
Te-127m	3.15E+00	6.30E+00	3.16E-01	1.11E-01	8.71E-01	7.60E+00
Te-127	2.05E+01	3.83E+01	1.15E+00	2.75E-02	1.33E-04	3.94E+01
Te-129m	1.07E+01	2.15E+01	1.07E+00	3.65E-01	2.36E+00	2.52E+01
Te-129	1.88E+01	2.83E+01	2.69E-02	3.54E-08	0.00E+00	2.84E+01
Te-131m	3.17E+01	6.20E+01	2.64E+00	3.35E-01	7.81E-02	6.50E+01

Table 7.1-9 (cont.) 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**	**0-8 hr**	**8-24 hr**	**24-96 hr**	**96-720 hr**	**Total**
Te-132	3.23E+02	6.40E+02	3.02E+01	7.04E+00	7.83E+00	6.85E+02
Sr-89	9.23E+01	1.85E+02	9.24E+00	3.19E+00	2.26E+01	2.20E+02
Sr-90	7.95E+00	1.59E+01	7.99E-01	2.84E-01	2.44E+00	1.94E+01
Sr-91	9.68E+01	1.81E+02	5.46E+00	1.35E-01	7.06E-04	1.87E+02
Sr-92	6.83E+01	1.13E+02	1.01E+00	5.15E-04	0.00E+00	1.14E+02
Ba-139	5.44E+01	8.30E+01	1.49E-01	9.91E-07	0.00E+00	8.32E+01
Ba-140	1.63E+02	3.25E+02	1.61E+01	5.11E+00	2.17E+01	3.68E+02
Mo-99	2.15E+01	4.25E+01	1.98E+00	4.29E-01	3.78E-01	4.53E+01
Tc-99m	1.47E+01	2.66E+01	6.05E-01	5.27E-03	1.33E-06	2.72E+01
Ru-103	1.73E+01	3.46E+01	1.73E+00	5.93E-01	3.99E+00	4.09E+01
Ru-105	8.18E+00	1.44E+01	2.48E-01	8.86E-04	1.17E-08	1.46E+01
Ru-106	5.70E+00	1.14E+01	5.73E-01	2.03E-01	1.70E+00	1.39E+01
Rh-105	1.03E+01	2.02E+01	8.81E-01	1.29E-01	4.14E-02	2.12E+01
Ce-141	3.89E+00	7.78E+00	3.88E-01	1.32E-01	8.45E-01	9.15E+00
Ce-143	3.46E+00	6.78E+00	2.93E-01	4.05E-02	1.14E-02	7.13E+00
Ce-144	2.94E+00	5.89E+00	2.96E-01	1.05E-01	8.68E-01	7.15E+00
Pu-238	9.16E-03	1.83E-02	9.21E-04	3.27E-04	2.82E-03	2.24E-02
Pu-239	8.06E-04	1.61E-03	8.10E-05	2.88E-05	2.48E-04	1.97E-03
Pu-240	1.18E-03	2.37E-03	1.19E-04	4.22E-05	3.63E-04	2.89E-03
Pu-241	2.66E-01	5.31E-01	2.67E-02	9.48E-03	8.14E-02	6.49E-01
Np-239	4.48E+01	8.87E+01	4.08E+00	8.15E-01	5.70E-01	9.41E+01
Y-90	8.08E-02	1.60E-01	7.44E-03	1.59E-03	1.35E-03	1.70E-01
Y-91	1.19E+00	2.37E+00	1.19E-01	4.12E-02	3.00E-01	2.83E+00
Y-92	7.89E-01	1.35E+00	1.80E-02	2.86E-05	0.00E+00	1.37E+00
Y-93	1.21E+00	2.28E+00	7.08E-02	1.98E-03	1.42E-05	2.35E+00
Nb-95	1.60E+00	3.19E+00	1.59E-01	5.44E-02	3.55E-01	3.76E+00
Zr-95	1.59E+00	3.18E+00	1.59E-01	5.52E-02	4.08E-01	3.80E+00
Zr-97	1.43E+00	2.74E+00	1.03E-01	6.73E-03	3.71E-04	2.85E+00
La-140	1.67E+00	3.29E+00	1.46E-01	2.36E-02	9.62E-03	3.47E+00

Table 7.1-9 (cont.) 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**	**0-8 hr**	**8-24 hr**	**24-96 hr**	**96-720 hr**	**Total**
La-141	1.03E+00	1.79E+00	2.71E-02	6.41E-05	2.01E-10	1.81E+00
La-142	5.38E-01	8.31E-01	2.09E-03	3.39E-08	0.00E+00	8.33E-01
Nd-147	6.16E-01	1.23E+00	6.06E-02	1.90E-02	7.29E-02	1.38E+00
Pr-143	1.39E+00	2.78E+00	1.37E-01	4.40E-02	1.94E-01	3.15E+00
Am-241	1.20E-04	2.39E-04	1.20E-05	4.27E-06	3.68E-05	2.92E-04
Cm-242	2.82E-02	5.65E-02	2.83E-03	9.98E-04	8.08E-03	6.84E-02
Cm-244	3.46E-03	6.93E-03	3.48E-04	1.24E-04	1.06E-03	8.47E-03
Total	3.53E+04	9.85E+04	1.35E+05	2.15E+05	4.30E+05	8.79E+05

Table 7.1-10 Activity Releases for Fuel Handling Accident

Isotope	Activity Release (Ci) 0-2 hr
Kr-85m	3.42E+02
Kr-85	1.11E+03
Kr-87	6.00E-02
Kr-88	1.07E+02
Xe-131m	5.54E+02
Xe-133m	2.80E+03
Xe-133	9.66E+04
Xe-135m	1.26E+03
Xe-135	2.49E+04
I-130	2.51E+00
I-131	3.76E+02
I-132	3.01E+02
I-133	2.40E+02
I-135	3.94E+01
Total	1.29E+05

Table 7.1-11 Atmospheric Dispersion Factors

Accident	Location	Time (hr)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
LOCA	EAB	0 – 2	5.10E-04	7.38E-05	1.45E-01
	LPZ	0 – 8	2.20E-04	1.40E-05	6.36E-02
		8 – 24	1.60E-04	1.22E-05	7.63E-02
		24 – 96	1.00E-04	9.15E-06	9.15E-02
		96 – 720	8.00E-05	6.04E-06	7.55E-02
Other Accidents	EAB	0 – 2	8.00E-04	7.38E-05	9.23E-02
	LPZ	0 – 8	5.00E-04	1.40E-05	2.80E-02
		8 – 24	3.00E-04	1.22E-05	4.07E-02
		24 – 96	1.50E-04	9.15E-06	6.10E-02
		96 – 720	8.00E-05	6.04E-06	7.55E-02

Note: The DCD χ/Q values for LOCA are consistent with AP1000 DCD Table 15A-5. Although not indicated as such in the DCD, a different set of χ/Q values was used by Westinghouse to calculate doses for accidents other than LOCA **(Westinghouse 2006b)**. It is seen that the site χ/Q values are bounded by the DCD χ/Q values for all time steps.

Table 7.1-12 Summary of Design Basis Accident Doses

DCD/SRP		Site Dose (rem TEDE)			
Section	**Accident**	**EAB**	**LPZ**	**Limit[1]**	**Dose Table**
15.1.5	Steam System Piping Failure				
	Pre-Existing Iodine Spike	0.07	0.03	25	7.1-13
	Accident-Initiated Iodine Spike	0.08	0.08	2.5	7.1-14
15.2.8	Feedwater System Pipe Break	[2]	[2]		
15.3.3	Reactor Coolant Pump Shaft Seizure				
	No Feedwater	0.06	0.01	2.5	7.1-15
	Feedwater Available	0.05	0.02	2.5	7.1-16
15.3.4	Reactor Coolant Pump Shaft Break	[3]	[3]		
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	0.27	0.17	6.3	7.1-17
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	0.16	0.03	2.5	7.1-18
15.6.3	Steam Generator Tube Rupture				
	Pre-Existing Iodine Spike	0.17	0.04	25	7.1-19
	Accident-Initiated Iodine Spike	0.08	0.02	2.5	7.1-20
15.6.5		3.5	1.5	25	7.1-21
15.7.4		0.52	0.10	6.3	7.1-22

1 NUREG-1555 specifies a dose limit of 25 rem TEDE for all design basis accidents. 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.

2 Feedwater System Pipe Break is bounded by Steam System Piping Failure, as indicated in AP1000 DCD.

3 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 System Piping Failure with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	8.0E-01		9.23E-02	7.38E-02	
0-8 hr		5.81E-01	2.80E-02		1.63E-02
8-24 hr		7.18E-02	4.07E-02		2.92E-03
24-96 hr		1.08E-01	6.10E-02		6.59E-03
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	8.0E-01	7.61E-01		7.38E-02	2.58E-02
Limit				25	25

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

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	9.00E-01		9.23E-02	8.30E-02	
0-8 hr		1.02E+00	2.80E-02		2.87E-02
8-24 hr		3.77E-01	4.07E-02		1.53E-02
24-96 hr		5.36E-01	6.10E-02		3.27E-02
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	9.00E-01	1.94E+00		8.30E-02	7.67E-02
Limit				2.5	2.5

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

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	**EAB**	**LPZ**	**(Site/DCD)**	**EAB**	**LPZ**
0-2 hr	7.00E-01		9.23E-02	6.46E-02	
0-8 hr		3.89E-01	2.80E-02		1.09E-02
8-24 hr		0.00E+00	4.07E-02		0.00E+00
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	7.00E-01	3.89E-01		6.46E-02	1.09E-02
Limit				2.5	2.5

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

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	**EAB**	**LPZ**	**(Site/DCD)**	**EAB**	**LPZ**
6-8 hr	5.00E-01		9.23E-02	4.61E-02	
0-8 hr		7.94E-01	2.80E-02		2.22E-02
8-24 hr		0.00E+00	4.07E-02		0.00E+00
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	5.00E-01	7.94E-01		4.61E-02	2.22E-02
Limit				2.5	2.5

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	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0-2 hr	2.90E+00		9.23E-02	2.68E-01	
0-8 hr		4.58E+00	2.80E-02		1.28E-01
8-24 hr		7.84E-01	4.07E-02		3.19E-02
24-96 hr		6.32E-02	6.10E-02		3.86E-03
96-720 hr		2.06E-02	7.55E-02		1.56E-03
Total	2.90E+00	5.45E+00		2.68E-01	1.66E-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	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0-2 hr	1.70E+00		9.23E-02	1.57E-01	
0-8 hr		1.02E+00	2.80E-02		2.86E-02
8-24 hr		0.00E+00	4.07E-02		0.00E+00
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	1.70E+00	1.02E+00		1.57E-01	2.86E-02
Limit				2.5	2.5

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

Time	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	1.80E+00		9.23E-02	1.66E-01	
0-8 hr		1.16E+00	2.80E-02		3.25E-02
8-24 hr		7.24E-02	4.07E-02		2.94E-03
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	1.80E+00	1.23E+00		1.66E-01	3.55E-02
Limit				25	25

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

Time	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	9.00E-01		9.23E-02	8.30E-02	
0-8 hr		6.27E-01	2.80E-02		1.76E-02
8-24 hr		1.69E-01	4.07E-02		6.87E-03
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	9.00E-01	7.96E-01		8.30E-02	2.44E-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

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
1.4-3.4 hr	2.43E+01		1.45E-01	3.52E+00	
0-8 hr		2.17E+01	6.36E-02		1.38E+00
8-24 hr		7.69E-01	7.63E-02		5.86E-02
24-96 hr		3.71E-01	9.15E-02		3.39E-02
96-720 hr		8.70E-01	7.55E-02		6.57E-02
Total	2.43E+01	2.37E+01		3.52E+00	1.54E+00
Limit				25	25

Table 7.1-22 Doses for Fuel Handling Accident

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	5.60E+00		9.23E-02	5.17E-01	
0-8 hr		3.44E+00	2.80E-02		9.63E-02
8-24 hr		0.00E+00	4.07E-02		0.00E+00
24-96 hr		0.00E+00	6.10E-02		0.00E+00
96-720 hr		0.00E+00	7.55E-02		0.00E+00
Total	5.60E+00	3.44E+00		5.17E-01	9.63E-02
Limit				6.3	6.3

Section 7.1 References

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

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

(Westinghouse 2005) AP1000 Document APP-GW-GL-700, *AP1000 Design Control Document,* Revision 15, Westinghouse Electric Company, 2005.

(Westinghouse 2006b) Westinghouse Document No. LTR-CRA-06-21, *AP1000 Accident Releases and Doses as Function of Time*, Westinghouse Electric Company, February 1, 2006.

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7.2 Severe Accidents

This section evaluates the potential environmental impacts of severe accidents on the VEGP site from the proposed Units 3 and 4 Westinghouse AP1000 reactors. Southern Nuclear Company (SNC) has updated the Westinghouse AP1000 DCD severe accident analysis with VEGP-specific data to demonstrate the Vogtle Electric Generating Plant (VEGP) site is bounded by the Nuclear Regulatory Commission (NRC)-approved analysis **(Westinghouse 2004; NRC 2005)**.

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*, 1985, the completion of a probabilistic risk assessment (PRA) 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*, 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 U.S.

The Westinghouse analysis used generic (but conservative) meteorology and regional characteristics. SNC presents in this section an update of the generic probabilistic risk assessment analysis of severe accidents to include site-specific characteristics of the VEGP site 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 less than 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 a containment 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 Electric Power Research Institute code Modular Accident Analysis Program (MAAP), Westinghouse determined that six source term categories would represent the entire suite of potential severe accidents. An accident frequency ("core damage frequency") was assigned to each of the six categories (Table 7.2-1).

The six source term categories or accident classes 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 class 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 class 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 prior to core relocation. Such phenomena could include hydrogen detonation, hydrogen diffusion flame, steam explosions, or vessel failures. This accident class 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 has passed since initiation of the accident. Such phenomena could include hydrogen detonation and hydrogen deflagration. This accident class 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 class 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 Electric Power Research Institute's Utility Requirements Document **(EPRI 1999)**. This meteorology is an actual site database selected because it is expected to provide impacts greater than those that would be expected at 80 to 90 percent 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 percent of the plants. The Westinghouse analysis focused on 24 hours following core damage and did not address the ingestion pathway.

Additional details on the Westinghouse analysis are found in Westinghouse **(2004)** and reported in the AP1000 Design Control Document **(Westinghouse 2005)**.

7.2.2 SNC Methodology

SNC also used the MACCS2 computer code to evaluate consequences of severe accidents. The 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 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. A significant difference between the Westinghouse generic analysis and the VEGP site-specific analysis is that SNC used site-specific meteorology and population data and included the ingestion pathway over the entire life cycle of the accident.

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

Five input files provide information 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 site-specific inputs in this file include the core inventory, release fractions, and geometry of the reactor and associated buildings. These input data are 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 information 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 (The Westinghouse analysis did not include this third file). These files access a meteorological file, which uses actual [VEGP] meteorological monitoring data from 1999 and a site characteristics file which is built using SECPOP2000 **(NRC 2003)**. SECPOP2000 incorporates 2000 census data for the 50-mile region around the VEGP site. For this analysis the census data were modified to include transient populations and projected to the year 2065. Population data for 2060 and 2070 are presented in Table 2.5.1-1. SNC prepared a calculation package supporting this analysis.

SNC used the results of the 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. The consequence can be either radiation dose or 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 are summed to provide a total risk. The same process was applied to estimating cost-risk. Therefore, risk can be reported as person-rem per reactor year or dollars per reactor year.

7.2.3 Consequences to Population Groups

This section evaluates impacts of severe accidents from air, surface water and groundwater pathways. The MACCS2 code was used to evaluate the doses from the air pathway and from drinking water with VEGP site-specific data. MACCS2 does not model other surface water and groundwater dose pathways. These were analyzed qualitatively based on a comparison of the AP1000 atmospheric doses to those of the existing nuclear fleet.

The current U.S. nuclear fleet has an exceptional safety record. The AP1000 is one of a new generation of reactors that incorporated passive safety features, making it inherently safer than existing reactors. The core damage frequency (CDF) is a measure of the impacts of potential accidents. CDF is estimated using PRA modeling which evaluates how changes to the reactor or auxiliary systems can change the severity of the accident. The CDF for the AP1000 is less than the CDFs for the current nuclear fleet.

7.2.3.1 Air Pathways

The potential severe accidents for the AP1000 were grouped into the six accident classes based on similarity of characteristics. Each class was assigned a set of characteristics representative of the elements of that class. Each accident class was analyzed with MACCS2 to estimate population dose, number of early and latent fatalities, cost, and farm land requiring decontamination. The analysis assumed that 95 percent of the population was evacuated following declaration of a general emergency.

For each accident class, SNC calculated the risk for each analytical endpoint (population dose, fatalities, cost, and contaminated land) by multiplying it by the accident class frequency. The results are provided in Table 7.2-1. The calculation considers other analytical endpoints 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.

7.2.3.2 Surface Water Pathways

People can be exposed to radiation when airborne radioactivity 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 2.1×10^{-3} person-rem per reactor year for the AP1000. This value is the sum of all six accident class risks.

Surface water pathways involving swimming, fishing, and boating are not modeled by MACCS2. Surface water bodies within the 50-mile region of VEGP include the Savannah River, other rivers, creeks, and ponds. The NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet discharging to small rivers (including the Savannah River) in NUREG-1437, the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (1996). The NRC evaluation estimated the aquatic food pathway dose risk to be 0.4 person-rem per reactor year.

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 VEGP). Because the AP1000 atmospheric pathway doses are significantly lower that 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. (SNC has evaluated the consequences of a radioactive spill not associated with an accident in the ESP SSAR Section 2.4.13 and determined that if radioactive liquids were released directly to groundwater, all isotopes would be below maximum permissible concentrations before they reached the Savannah River. NUREG-1437 also evaluated the groundwater pathway dose, based on the analysis in NUREG-0440 (1978), the *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 compared potential contamination at the existing VEGP site to the results of NUREG-0440 and found it to be 10^{-5} to 10^{-4} times the NUREG-0440 conclusions for a small river site. The proposed location for VEGP Units 3 and 4 has the same groundwater characteristics as the location of the existing units and the CDF for the AP1000 is lower that that of the existing units, therefore, the doses from the AP1000 groundwater pathway would be smaller than from the existing units.

7.2.4 Conclusions

The total calculated dose-risk to the 50-mile population from airborne releases from an AP1000 reactor at VEGP will not exceed 0.042 person-rem per reactor year (Table 7.2-1). This value is less than the 0.043 reported by Westinghouse in the Design Control Document **(Westinghouse 2005)**. The difference is more pronounced than it appears, because the Westinghouse analysis is based on a 24-hour dose but the SNC analysis is based on the entire life-cycle of the accidents considered.

The AP1000 dose-risk at the VEGP 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 RG 1.174, *An approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*. As reported in NUREG-1793 *Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design*, 2004, the minimum dose-risk reported for reactors currently undergoing license renewal is 0.55 person-rem per reactor year. The airborne pathway dose-risk from severe accidents for the existing VEGP reactors is 35 person-rem per reactor year (50 FR 32138).

Two population centers near the mouth of the Savannah River, Savannah, Georgia and Beaufort County, S.C., use the Savannah River as a source of drinking water. Also, shell-fishing near the mouth of the river provides foods to the population. SNC's qualitative analysis indicates that risk from the surface water pathway is small. The risks of groundwater contamination from an AP1000 accident are several orders of magnitude less than the risk from surface water contamination for currently licensed reactors. The risk of groundwater contamination from an AP14000 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 normal operations is expected to be 0.2 person-rem per year for the AP1000. As previously described, dose-risk is dose times frequency. Normal operation has a frequency of one. Therefore, the dose-risk for normal operation is 0.2 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.042 person-rem per reactor year indicates that the dose risk from severe accidents is approximately 20 percent of dose risk from normal operations.

The probability-weighted risk of an early cancer fatality from a severe accident for the AP1000 at VEGP is reported in Table 7.2-1 as 2.8×10^{-7} fatalities per reactor year. The lifetime probability of an individual dying from any cancer is 2.3×10^{-1}.

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

Accident Class	Core Damage Frequency (per reactor year)[1]	Environmental Risk: Population Dose-Risk (person-rem/ reactor year)	Number of Fatalities (per reactor year): Early	Number of Fatalities (per reactor year): Late	Cost in Dollars (per reactor year)	Land Requiring Decontamination (acres/reactor year)
1. Intact containment	2.2×10^{-7}	7.0×10^{-4}	0	9.7×10^{-8}	0.18	0
2. Containment bypass	1.1×10^{-8}	3.2×10^{-2}	1.5×10^{-7}	5.3×10^{-6}	40	5.3×10^{-4}
3. Containment isolation failure	1.3×10^{-9}	1.6×10^{-3}	9.5×10^{-9}	2.5×10^{-7}	1.4	0
4. Early containment failure	7.5×10^{-9}	8.0×10^{-3}	5.5×10^{-8}	1.2×10^{-6}	9.6	6.5×10^{-5}
5. Intermediate containment failure	1.9×10^{-10}	2.8×10^{-4}	6.9×10^{-8}	4.4×10^{-8}	.25	8.0×10^{-7}
6. Late containment failure	3.5×10^{-13}	6.7×10^{-7}	0	9.8×10^{-11}	0.0012	2.2×10^{-8}
Total	2.4×10^{-7}	4.2×10^{-2}	2.8×10^{-7}	6.9×10^{-6}	51	6.0×10^{-4}

[1] Source: **Westinghouse 2004**

Section 7.2 References

(Chanin and Young 1997) Chanin, D. I. and M. L. Young. Code Manual for MACCS2: Volume 1, User's Guide, SAND97-0594, Sandia National Laboratories, Albuquerque, New Mexico, March.

(EPRI 1999) Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Volume III, ALWR Passive Plant, Revision B, Palo Alto, California, March.

(NRC 2003) U.S. Nuclear Regulatory Commission, SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program, NUREG/CR-6525, Rev. 1, Washington, D.C. August.

(NRC 2005) U.S. Nuclear Regulatory Commission, 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.

(Westinghouse 2004) Westinghouse Electric Company, LLC, *Probabilistic Risk Assessment*, Revision 8, Pittsburgh, Pennsylvania.

(Westinghouse 2005) Westinghouse Electric Company, LLC, *Design Control Document, Revision 15*, Appendix 1B, "Severe Accident Mitigation Design Alternatives," NRC Accession Number ML053460409, U.S. Nuclear Regulatory Commission, Washington, D.C., November 11.

7.3 Severe Accident Mitigation Measures

This section updates the Westinghouse DCD Severe Accidents Mitigation Measures analysis **(Westinghouse 2005)** with VEGP site and regional data. The VEGP-site specific analysis demonstrates that the severe accident mitigation alternatives determined not to be cost beneficial by Westinghouse are also not cost beneficial when VEGP site-specific data are considered **(NRC 2005)**.

Regulations of the Council on Environmental Quality (CEQ) regarding the National Environmental Policy Act require that a discussion on environmental consequences include mitigation measures (40 CFR 1502.16(h). CEQ 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 Chapters 4 and 5, the construction and operation of a nuclear power plant has 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 2005)**. 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 SNC proposed new nuclear units at VEGP and determined that the generic conclusions remain valid for the VEGP site. The analysis in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented at SNC's site to mitigate these small impacts. SNC prepared a calculation package supporting this analysis.

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 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 the ESP project, 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 ESP analysis is limited to demonstrating that the VEGP site is bounded by the Westinghouse DCD 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 you could reduce the current dose risk of the unit to zero and assigning 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*, 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 SNC 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. SNC prepared a calculation package supporting this analysis.

7.3.2 The AP1000 SAMA Analysis

In the certification process, only design alternatives are of interest. The Westinghouse SAMDA analysis is presented in Appendix 1B of the AP1000 Design Control Document **(Westinghouse 2005)**.

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 DCD.

- 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 values
- 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 Section 7.2.1. Westinghouse calculated the base case maximum averted cost risk to be $21,000 using a 7 percent 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 NRC **(2006)** 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 frequency] are very low on an absolute scale."

7.3.3 Monetization of the VEGP Units 3 and 4 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). For this project, the base-case core damage frequency, dose-risk, and cost-risk were escalated in this analysis to account for not only internal events but also external events, both at power and at shutdown. 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 $18,000 for a single AP1000 at SNC's proposed site, is so low that SNC does not believe there are any design changes, over those already incorporated into the advanced reactor designs, which could be determined to be cost-effective. Even with a conservative three percent discount rate, the valuation of the averted risk is only $34,000. Conceivably, there could be administrative changes applicable to both AP1000 units that could be less than the combined project averted risk monetization.

These values compare to the Westinghouse generic analysis results of $21,000 for the seven percent discount rate and $43,000 for the three percent discount rate. The SNC analysis used actual population and meteorological characteristics that would result in lower impacts than did the conservative values used in the generic analysis.

Accordingly, further evaluation of design-related SAMAs is not warranted. 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 would be incorporated into the plants' management systems as part of its baseline.

Table 7.3-1 Monetization of the SNC AP1000 Base Case

	7% Discount Rate	3% Discount Rate
Offsite exposure cost	$693	$2,191
Offsite economic cost	$421	$1,331
Onsite exposure cost	$176	$380
Onsite cleanup cost	$6,093	$12,708
Replacement power cost	$10,578	$17,651
Total	$17,960	$34,261

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(NRC 2005) U.S. Nuclear Regulatory Commission, 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.

(Westinghouse 2005) Westinghouse Electric Corporation, *Design Control Document, Revision 15*, Appendix 1B, "Severe Accident Mitigation Design Alternatives," NRC Accession Number ML053460409, U.S. Nuclear Regulatory Commission, Washington, D.C., November 11, 2005.

7.4 Transportation Accidents

Section 5.11.2 described the methodology used by SNC to analyze the impacts of transportation, including accidents.

7.4.1 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 fuel for advanced LWRs 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, *Draft Environmental Impact Statement for an Early Site Permit at North Anna Power Station ESP Site*, 2004; NUREG-1815, *Draft Environmental Impact Statement for an Early Site Permit at Exelon ESP Site*, 2005; and NUREG-1817, *Environmental Impact Statement for an Early Site Permit at Grand Gulf ESP Site*, 2006, the risks of accidents during transport of unirradiated fuel to the VEGP site would be expected to be smaller than the reference LWR results listed in Table S-4.

7.4.2 Transportation of Spent Fuel

In its assessments of other proposed ESP sites, NRC 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).

NRC obtained the radionuclide inventories of the advanced LWR spent fuel after five years decay from INEEL (2003) and performed a screening analysis 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 percent of the dose from inhalation of radionuclides released following a transportation accident. NRC found that the dominant radionuclides are similar regardless of the fuel type. The spent fuel inventory used in the NRC 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. According to Sprung et al. **(2000)**, 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 NRC analysis assumed that shipping casks for advanced LWR spent fuels would provide equivalent mechanical and thermal protection of the spent fuel cargo.

NRC performed the RADTRAN 5 accident risk calculations using unit radionuclide inventories (curies/metric ton uranium [Ci/MTU]) for the spent fuel shipments from the advanced LWRs. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (MTU/yr) to derive estimates of the annual accident risks associated with spent fuel shipments from each potential advanced LWR. The amounts of spent fuel shipped per year were assumed to be equivalent to the annual discharge quantities: 23 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.)

NRC 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, NRC calculated the population dose from the released radioactive material for five 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 NRC analysis included the radiation exposure from this pathway 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 NRC analysis included the radiation exposures from this pathway even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures)
5. internal dose from ingestion of contaminated food (the NRC analysis assumed interdiction of foodstuffs and evacuation after an accident so no internal dose due to ingestion of contaminated foods was calculated).

A sixth pathway, 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, NRC 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.

NRC calculated the environmental consequences of transportation accidents when shipping spent fuel from other potential new reactor sites to a spent fuel repository assumed to be at Yucca Mountain, Nevada. The shipping distances and population distribution information for the routes were the same as those used for the "incident-free" transportation impacts analysis (described in Section 5.11.2).

SNC used the results of the NRC analysis for transportation of spent fuel from the Savannah River Site to Yucca Mountain to conservatively estimate the potential impacts for spent fuel transportation from VEGP, due to the proximity of the two sites (see Section 5.11.2.1 for further discussion). As discussed in Section 5.11.2.1, analysis of this transportation route is also bounding for the alternative sites (Farley, Hatch) or a green field site within the SNC region of interest. The NRC analysis included the AP1000 reactor design.

Table 7.4-2 presents unit (per MTU) accident risks associated with transportation of spent fuel from the VEGP 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).

7.4.3 Conclusion

Considering the uncertainties in the data and computational methods, NRC concluded that the overall transportation accident risks associated with advanced LWR spent fuel shipments are likely to be SMALL and are consistent with the risks associated with transportation of spent fuel 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 proposed new reactors at the VEGP site.

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	8870
Cm-242	28.3
Cm-243	30.7
Cm-244	7750
Cm-245	1.21
Cs-134	4.80E+4
Cs-137	9.31E+4
Eu-154	9.13E+3
Eu-155	4620
Pm-147	1.76E+4
Pu-238	6070
Pu-239	255
Pu-240	543
Pu-241	6.96E+4
Pu-242	1.82
Ru-106	1.55E+4
Sb-125	3830
Sr-90	6.19E+4
Y-90	6.19E+4

Source: NUREG-1811, NUREG-1815, NUREG-1817

Ci/MTU = curies per metric ton uranium

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

Unit Population Dose (person-rem per MTU)[1]	MTU per reference reactor year	Population Dose (person-rem per reference reactor year)[2]
2.4×10^{-6}	19.5	4.7E-5

1 Based on SRS information presented in Table G-13 of NUREG-1811 for AP1000. Value presented is the product of probability times collective dose.

2 Value presented is the product of probability times collective dose.

Section 7.4 References

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