CHAPTER 7

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

INVOLVING RADIOACTIVE MATERIALS

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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 at the Bellefonte Nuclear Plant, Units 3 and 4 (BLN) site. The chapter is divided into four sections that address design basis accidents, severe accidents, severe accident mitigation design alternatives, and transportation accidents.

- Design Basis Accidents (Section 7.1).
- Severe Accidents (Section 7.2).
- Severe Accident Mitigation Design Alternatives (Section 7.3).
- Transportation Accidents (Section 7.4).

7.1 DESIGN BASIS ACCIDENTS

This section reviews and analyzes the design basis accidents (DBAs), as identified in NUREG-1555, to demonstrate that reactors can be operated at the BLN without undue risk to the health and safety of the public.

7.1.1 SELECTION OF ACCIDENTS

The DBAs considered in this section come from the AP1000 design control document (DCD) Chapter 15. Table 7.1-1 lists the NUREG-1555 DBAs that have the potential to release radioactivity 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 DCD. The DBAs cover a spectrum of events, including those of relatively greater probability of occurrence and those that are less probable but have greater severity. The radiological consequences of the accidents listed in Table 7.1-1 are assessed to demonstrate that units can be sited and operated at the BLN without undue risk to the health and safety of the public.

7.1.2 EVALUATION METHODOLOGY

The DCD presents the radiological consequences for the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with BLN data to demonstrate that the DCD analyses are bounding for the BLN. The basis 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 postulated loss-of-coolant accidents (LOCA) are expected to more closely approach 10 CFR 50.34 limits than the other DBAs of greater probability of occurrence but lesser magnitude of activity releases. For these other accidents, the calculated doses are compared to the acceptance criteria in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000, and NUREG-0800 to demonstrate that the consequences of the postulated accidents are acceptable.

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 off-site location, the breathing rate of the individual at the off-site location, and activity-to-dose conversion factors. The breathing rate of the individual at the off-site location specified in Table 15.6.5-2 of the DCD is used for the analysis. 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). Because 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.

The accident analyses presented in DCD Chapter 15 uses conservative assumptions as specified in Regulatory Guide 1.183 to perform bounding safety analyses that substantially overstate the environmental effect of the identified accidents. Among the conservative assumptions in DCD Chapter 15 is the use of time-dependent χ/Q values corresponding to the top fifth percentile of meteorology, meaning that conditions would be more favorable for dispersion 95 percent of the time. In addition to the use of atmospheric dispersion factors corresponding to adverse conditions, the DCD Chapter 15 design basis analyses also used conservative assumptions on the core and coolant source terms, the types of radioactive materials released, and the release paths to the environment. These conservative assumptions are maintained for the dose assessments presented in this section except that realistic atmospheric dispersion factors are used. The doses in this environmental report (ER) are calculated based on the 50th percentile site-specific χ/Q values reflecting more realistic meteorological conditions consistent with NUREG-1555.

The χ/Q values are calculated using the guidance in NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Revision 1, with site-specific meteorological data. As indicated in Subsection 2.7.3, the NRC Regulatory Guide 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 it calculates atmospheric stability for each of 16 downwind direction sectors. It then calculates overall (nondirection-specific) χ/Q values. For a given location, either the EAB or the LPZ, the 0 – 2-hr. 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 off-site 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 (Reference 1), while the EDE is based on the dose conversion factors in Federal Guidance Report 12 (Reference 2). Appendix 15A of the DCD provides information on the methodologies used to calculate CEDE and EDE values for the various postulated accidents. As indicated in NRC Regulatory Guide 1.183, the dose conversion factors in Federal Guidance Reports 11 and 12 (References 1 and 2) used for the various postulated accidents are acceptable to the NRC staff.

7.1.3 SOURCE TERMS

The design basis accident source terms, methodology, and assumptions in the DCD are based on the alternative source term methods outlined in NRC Regulatory Guide 1.183. The activity releases and doses are based on 102 percent of the rated core thermal power of 3400 megawatts thermal (MWt). The core inventory for the AP1000 is provided in Appendix 15A of the DCD. The core fission product inventory was determined by the use of the ORIGEN code. The time-dependent isotopic activities released to the environment from each of the evaluated accidents are presented in Tables 7.1-2, 7.1-3, 7.1-4, 7.1-5, 7.1-6, 7.1-7, 7.1-8, 7.1-9, and 7.1-10.

7.1.4 RADIOLOGICAL CONSEQUENCES

The design basis accident doses are evaluated in the ER on the basis of more realistic meteorological conditions than those in the DCD Chapter 15. 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 DCD dose by the ratio of the site χ/Q value, developed in Table 2.7-118 of Subsection 2.7.3, to the DCD χ/Q value. The time-dependent DCD χ/Q values, 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, 7.1-14, 7.1-15, 7.1-16, 7.1-17, 7.1-18, 7.1-19, 7.1-20, 7.1-21, and 7.1-22. For each accident, the EAB dose shown is for the 2-hr. period that yields the maximum dose, in accordance with NRC Regulatory Guide 1.183.

The results of the BLN 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 with an exceedingly low probability of occurrence and a low risk of public exposure to radiation. For events with a higher probability of occurrence, the dose limits are taken from NRC Regulatory Guide 1.183. Although conformance to these dose limits is not required for an ER, the limits are shown in the tables for comparison purposes.

The TEDE dose limits shown in Tables 7.1-12, 7.1-13, 7.1-14, 7.1-15, 7.1-16, 7.1-17, 7.1-18, 7.1-19, 7.1-20, 7.1-21, and 7.1-22 are from NRC Regulatory Guide 1.183, Table 6, for all formally designated accidents except the Reactor Coolant Pump Shaft Break (SRP Subsection 15.3.4) and the Failure of Small Lines Carrying Primary Coolant Outside Containment (SRP Subsection 15.6.2). Although NRC Regulatory Guide 1.183 does not address these two accidents, NUREG-0800 indicates that the dose limit is a "small fraction" or 10 percent of the 10 CFR Part 100 guideline of 25 rem, meaning a limit of 2.5 rem for these accidents. All doses are within the acceptance criteria.

7.1.5 REFERENCES

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

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

TABLE 7.1-1 (Sheet 1 of 2)
SELECTION OF ACCIDENTS

SRP/DCD Subsection	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 Subsection 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 Subsection 15.1.5 Accident.
15.3.3[a]	Reactor Coolant Pump Rotor Seizure	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Yes	Addressed in DCD Subsection 15.3.3.
15.3.4[a]	Reactor Coolant Pump Shaft Break	Reactor Coolant Pump Shaft Break	Yes	In the DCD, this is bounded by Subsection 15.3.3 Accident.
15.4.8[b]	Spectrum of Rod Ejection Accidents (PWR)	Spectrum of Rod Cluster Control Assembly Ejection Accidents	No	Evaluated for completeness. Addressed in DCD Subsection 15.4.8.
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	Addressed in DCD Subsection 15.6.2.
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	Steam Generator Tube Rupture	Yes	Addressed in DCD Subsection 15.6.3.
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 Subsection 15.6.5.

TABLE 7.1-1 (Sheet 2 of 2)
SELECTION OF ACCIDENTS

SRP/DCD Subsection	SRP Description	DCD Description	Identified in NUREG-1555, Section 7.1 Appendix A	Comment
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 Subsection 15.6.5.
15.7.4	Radiological Consequences of Fuel Handling Accidents	Fuel Handling Accident	Yes	Addressed in DCD Subsection 15.7.4.

a) These sections for Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break are in a common SRP Section 15.3.3-15.3.4

b) The source of this accident is Section 15.4.8 of NUREG-0800. This event is not included in NUREG-1555, Section 7.1, Appendix A, “Design Basis Accidents Included in Section 15 of the Standard Review Plan.”

TABLE 7.1-2
ACTIVITY RELEASES FOR STEAM SYSTEM PIPING FAILURE WITH PRE-EXISTING IODINE SPIKE

Activity Release (Ci)

Isotope	0 – 8 hr.	8 – 24 hr.	24 – 72 hr.	Total
Kr-85m	1.83E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	1.13E+00	2.25E+00	6.69E+00	1.01E+01
Kr-87	4.10E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	2.50E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	5.07E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	6.09E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	4.63E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.06E-03	0.00E+00	0.00E+00	3.06E-03
Xe-135	9.99E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	4.00E-03	0.00E+00	0.00E+00	4.00E-03
I-130	5.01E-01	2.09E-01	1.33E-01	8.44E-01
I-131	3.61E+01	3.10E+01	8.22E+01	1.49E+02
I-132	3.47E+01	8.06E-01	6.55E-03	3.55E+01
I-133	6.23E+01	3.53E+01	3.98E+01	1.37E+02
I-134	6.91E+00	1.43E-03	4.54E-09	6.91E+00
I-135	3.42E+01	7.54E+00	1.71E+00	4.34E+01
Cs-134	1.92E+01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.85E+01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.38E+01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	4.42E-07	0.00E+00	1.01E+01
Total	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 – 8 hr.	8 – 24 hr.	24 – 72 hr.	Total
Kr-85m	1.83E-01	6.80E-02	6.18E-03	2.57E-01
Kr-85	1.13E+00	2.25E+00	6.69E+00	1.01E+01
Kr-87	4.10E-02	5.29E-04	8.60E-08	4.15E-02
Kr-88	2.50E-01	4.04E-02	8.27E-04	2.91E-01
Xe-131m	5.07E-01	9.81E-01	2.70E+00	4.19E+00
Xe-133m	6.09E-01	1.04E+00	2.05E+00	3.70E+00
Xe-133	4.63E+01	8.64E+01	2.16E+02	3.49E+02
Xe-135m	3.06E-03	0.00E+00	0.00E+00	3.06E-03
Xe-135	9.99E-01	8.35E-01	3.38E-01	2.17E+00
Xe-138	4.00E-03	0.00E+00	0.00E+00	4.00E-03
I-130	1.42E+00	1.58E+00	1.01E+00	4.01E+00
I-131	8.33E+01	1.56E+02	4.13E+02	6.53E+02
I-132	1.44E_02	2.24E+01	1.82E-01	1.66E+02
I-133	1.63E+02	2.27E+02	2.55E+02	6.45E+02
I-134	3.20E+01	2.65E-01	8.42E-07	3.23E+01
I-135	1.10E+02	7.83E+01	1.77E+01	2.06E+02
Cs-134	1.92E+01	5.19E-01	1.54E+00	2.12E+01
Cs-136	2.85E+01	7.43E-01	2.06E+00	3.13E+01
Cs-137	1.38E+01	3.74E-01	1.11E+00	1.53E+01
Cs-138	1.01E+01	4.42E-07	0.00E+00	1.01E+01
Total	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)

	No Feedwater	Feedwater Available
Isotope	0-1.5 hr	0-8 hr
Kr-85m	8.16E+01	2.79E+02
Kr-85	7.58E+00	4.04E+01
Kr-87	1.20E+02	2.13E+02
Kr-88	2.08E+02	5.82E+02
Xe-131m	3.77E+00	2.00E+01
Xe-133m	2.02E+01	1.03E+02
Xe-133	6.66E+02	3.49E+03
Xe-135m	3.24E+01	3.30E+01
Xe-135	1.59E+02	6.72E+02
Xe-138	1.29E+02	1.31E+02
I-130	8.45E-01	1.45E+00
I-131	3.77E+01	8.05E+01
I-132	2.79E+01	1.83E+01
I-133	4.86E+01	8.98E+01
I-134	2.88E+01	5.74E+00
I-135	4.19E+01	5.79E+01
Cs-134	1.29E+00	2.59E+00
Cs-136	5.63E-01	8.63E-01
Cs-137	7.74E-01	1.52E+00
Cs-138	6.08E+00	4.08E+00
Rb-86	1.33E-02	2.91E-02
Total	1.62E+03	5.82E+03

TABLE 7.1-5
ACTIVITY RELEASES FOR SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

Activity Release (Ci)

Isotope	0-8 hr	8-24 hr	24-96 hr	96-720 hr	Total
Kr-85m	1.77E+02	3.87E+01	1.77E+00	2.51E-05	2.18E+02
Kr-85	1.06E+01	1.49E+01	3.35E+01	2.88E+02	3.47E+02
Kr-87	2.08E+02	1.03E+00	8.37E-05	0.00E+00	2.09E+02
Kr-88	4.10E+02	3.49E+01	3.59E-01	8.41E-09	4.45E+02
Xe-131m	1.04E+01	1.42E+01	2.86E+01	1.16E+02	1.69E+02
Xe-133m	5.48E+01	6.49E+01	8.45E+01	5.31E+01	2.57E+02
Xe-133	1.84E+03	2.40E+03	4.27E+03	8.45E+03	1.70E+04
Xe-135m	7.35E+01	4.33E-09	0.00E+00	0.00E+00	7.35E+01
Xe-135	3.87E+02	2.09E+02	4.35E+01	1.79E-01	6.39E+02
Xe-138	2.99E+02	3.19E-09	0.00E+00	0.00E+00	2.99E+02
I-130	1.22E+01	4.32E+00	2.03E-01	2.95E-04	1.67E+01
I-131	3.81E+02	2.31E+02	3.10E+01	1.68E+01	6.60E+02
I-132	2.52E+02	9.85E+00	8.24E-03	0.00E+00	2.62E+02
I-133	7.12E+02	3.18E+02	2.28E+01	2.41E-01	1.05E+03
I-134	1.95E+02	1.37E-01	4.48E-08	0.00E+00	1.95E+02
I-135	5.36E+02	1.19E+02	2.39E+00	7.32E-05	6.57E+02
Cs-134	9.30E+01	6.03E+01	7.76E+00	5.16E+00	1.66E+02
Cs-136	2.63E+01	1.67E+01	2.05E+00	6.58E-01	4.57E+01
Cs-137	5.41E+01	3.51E+01	4.52E+00	3.05E+00	9.68E+01
Cs-138	1.16E+02	1.68E-03	0.00E+00	0.00E+00	1.16E+02
Rb-86	1.09E+00	6.96E-01	8.67E-02	3.42E-02	1.91E+00
Total	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

Activity Release (Ci)

Isotope	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	0-8 hr	8-24 hr	Total
Kr-85m	7.46E+01	7.53E-03	7.46E+01
Kr-85	3.29E+02	1.34E-01	3.29E+02
Kr-87	2.75E+01	9.12E-05	2.75E+01
Kr-88	1.19E+02	5.43E-03	1.19E+02
Xe-131m	1.48E+02	5.91E-02	1.48E+02
Xe-133m	1.83E+02	6.61E-02	1.83E+02
Xe-133	1.37E+04	5.29E+00	1.37E+04
Xe-135m	3.45E+00	0.00E+00	3.45E+00
Xe-135	3.47E+02	7.10E-02	3.47E+02
Xe-138	4.57E+00	0.00E+00	4.57E+00
I-130	1.85E+00	2.68E-01	2.12E+00
I-131	1.26E+02	3.06E+01	1.56E+02
I-132	1.42E+02	1.92E+00	1.44E+02
I-133	2.24E+02	4.06E+01	2.64E+02
I-134	2.74E+01	4.23E-03	2.74E+01
I-135	1.30E+02	1.17E+01	1.42E+02
Cs-134	1.69E+00	2.16E-01	1.90E+00
Cs-136	2.51E+00	3.14E-01	2.82E+00
Cs-137	1.22E+00	1.56E-01	1.37E+00
Cs-138	5.64E-01	5.73E-07	5.64E-01
Total	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	0-8 hr	8-24 hr	Total
Kr-85m	7.46E+01	7.53E-03	7.46E+01
Kr-85	3.29E+02	1.34E-01	3.29E+02
Kr-87	2.75E+01	9.12E-05	2.75E+01
Kr-88	1.19E+02	5.43E-03	1.19E+02
Xe-131m	1.48E+02	5.91E-02	1.48E+02
Xe-133m	1.83E+02	6.61E-02	1.83E+02
Xe-133	1.37E+04	5.29E+00	1.37E+04
Xe-135m	3.45E+00	0.00E+00	3.45E+00
Xe-135	3.47E+02	7.10E-02	3.47E+02
Xe-138	4.57E+00	0.00E+00	4.57E+00
I-130	1.05E+00	8.24E-01	1.87E+00
I-131	5.51E+01	6.76E+01	1.23E+02
I-132	1.52E+02	1.29E+01	1.65E+02
I-133	1.13E+02	1.08E+02	2.22E+02
I-134	5.59E+01	5.94E-02	5.60E+01
I-135	8.60E+01	4.38E+01	1.30E+02
Cs-134	1.69E+00	2.16E-01	1.90E+00
Cs-136	2.51E+00	3.14E-01	2.82E+00
Cs-137	1.22E+00	1.56E-01	1.37E+00
Cs-138	5.64E-01	5.73E-07	5.64E-01
Total	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	0-8 hr	8-24 hr	24-96 hr	96-720 hr	Total
I-130	2.52E+01	2.56E+00	5.97E-01	1.07E-02	2.83E+01
I-131	7.87E+02	1.39E+02	2.01E+02	6.05E+02	1.73E+03
I-132	4.78E+02	6.03E+00	1.23E-02	4.72E-12	4.84E+02
I-133	1.47E+03	1.90E+02	8.77E+01	8.75E+00	1.76E+03
I-134	2.55E+02	9.53E-02	5.72E-08	1.05E-32	2.55E+02
I-135	1.10E+03	6.99E+01	5.12E+00	2.66E-03	1.17E+03
Kr-85m	3.77E+03	1.87E+03	8.56E+01	1.22E-03	5.73E+03
Kr-85	2.97E+02	7.06E+02	1.59E+03	1.36E+04	1.62E+04
Kr-87	1.95E+03	4.97E+01	4.05E-03	3.68E-20	1.99E+03
Kr-88	7.26E+03	1.70E+03	1.75E+01	4.09E-07	8.97E+03
Xe-131m	2.94E+02	6.79E+02	1.37E+03	5.57E+03	7.92E+03
Xe-133m	1.54E+03	3.15E+03	4.11E+03	2.58E+03	1.14E+04
Xe-133	5.19E+04	1.16E+05	2.06E+05	4.07E+05	7.80E+05
Xe-135m	3.59E+01	2.14E-07	1.35E-26	0.00E+00	3.59E+01
Xe-135	9.64E+03	1.01E+04	2.11E+03	8.68E+00	2.19E+04
Xe-138	1.20E+02	1.58E-07	3.46E-28	0.00E+00	1.20E+02
Rb-86	1.26E+00	5.97E-02	1.97E-02	1.03E-01	1.45E+00
Cs-134	1.08E+02	5.14E+00	1.82E+00	1.55E+01	1.30E+02
Cs-136	3.05E+01	1.43E+00	4.57E-01	1.98E+00	3.43E+01
Cs-137	6.27E+01	3.00E+00	1.06E+00	9.15E+00	7.59E+01
Cs-138	6.59E+01	4.37E-04	6.68E-15	3.07E-55	6.59E+01
Sb-127	9.60E+00	4.57E-01	1.14E-01	1.57E-01	1.03E+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	0-8 hr	8-24 hr	24-96 hr	96-720 hr	Total
Sb-129	1.79E+01	3.01E-01	9.89E-04	9.81E-09	1.82E+01
Te-127m	1.26E+00	6.32E-02	2.22E-02	1.74E-01	1.52E+00
Te-127	7.66E+00	2.29E-01	5.49E-03	2.66E-05	7.89E+00
Te-129m	4.29E+00	2.14E-01	7.30E-02	4.71E-01	5.05E+00
Te-129	5.67E+00	5.38E-03	7.09E-09	1.42E-27	5.67E+00
Te-131m	1.24E+01	5.28E-01	6.69E-02	1.56E-02	1.30E+01
Te-132	1.28E+02	6.04E+00	1.41E+00	1.57E+00	1.37E+02
Sr-89	3.69E+01	1.85E+00	6.38E-01	4.53E+00	4.39E+01
Sr-90	3.18E+00	1.60E-01	5.68E-02	4.88E-01	3.89E+00
Sr-91	3.62E+01	1.09E+00	2.70E-02	1.41E-04	3.73E+01
Sr-92	2.26E+01	2.03E-01	1.03E-04	1.02E-12	2.28E+01
Ba-139	1.66E+01	2.97E-02	1.98E-07	3.97E-23	1.66E+01
Ba-140	6.50E+01	3.21E+00	1.02E+00	4.33E+00	7.36E+01
Mo-99	8.50E+00	3.96E-01	8.58E-02	7.56E-02	9.06E+00
Tc-99m	5.32E+00	1.21E-01	1.05E-03	2.67E-07	5.45E+00
Ru-103	6.92E+00	3.46E-01	1.19E-01	7.97E-01	8.19E+00
Ru-105	2.87E+00	4.97E-02	1.77E-04	2.34E-09	2.92E+00
Ru-106	2.28E+00	1.15E-01	4.06E-02	3.40E-01	2.78E+00
Rh-105	4.04E+00	1.76E-01	2.58E-02	8.29E-03	4.25E+00
Ce-141	1.56E+00	7.77E-02	2.64E-02	1.69E-01	1.83E+00
Ce-143	1.36E+00	5.87E-02	8.09E-03	2.29E-03	1.43E+00
Ce-144	1.18E+00	5.91E-02	2.09E-02	1.74E-01	1.43E+00

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	0-8 hr	8-24 hr	24-96 hr	96-720 hr	Total
Pu-238	3.67E-03	1.84E-04	6.55E-05	5.63E-04	4.48E-03
Pu-239	3.23E-04	1.62E-05	5.76E-06	4.96E-05	3.94E-04
Pu-240	4.73E-04	2.38E-05	8.44E-06	7.26E-05	5.78E-04
Pu-241	1.06E-01	5.34E-03	1.90E-03	1.63E-02	1.30E-01
Np-239	1.77E+01	8.16E-01	1.63E-01	1.14E-01	1.88E+01
Y-90	3.20E-02	1.49E-03	3.18E-04	2.69E-04	3.41E-02
Y-91	4.74E-01	2.37E-02	8.23E-03	5.99E-02	5.66E-01
Y-92	2.70E-01	3.59E-03	5.72E-06	4.25E-12	2.74E-01
Y-93	4.56E-01	1.42E-02	3.95E-04	2.85E-06	4.71E-01
Nb-95	6.38E-01	3.19E-02	1.09E-02	7.11E-02	7.52E-01
Zr-95	6.35E-01	3.18E-02	1.11E-02	8.15E-02	7.59E-01
Zr-97	5.49E-01	2.06E-02	1.35E-03	7.41E-05	5.71E-01
La-140	6.58E-01	2.92E-02	4.72E-03	1.92E-03	6.94E-01
La-141	3.57E-01	5.41E-03	1.28E-05	4.02E-11	3.62E-01
La-142	1.66E-01	4.18E-04	6.79E-09	5.76E-23	1.67E-01
Nd-147	2.46E-01	1.21E-02	3.79E-03	1.46E-02	2.77E-01
Pr-143	5.55E-01	2.75E-02	8.79E-03	3.88E-02	6.30E-01
Am-241	4.79E-05	2.41E-06	8.55E-07	7.35E-06	5.85E-05
Cm-242	1.13E-02	5.67E-04	2.00E-04	1.62E-03	1.37E-02
Cm-244	1.39E-03	6.97E-05	2.47E-05	2.13E-04	1.69E-03
Total	8.16E+04	1.35E+05	2.15E+05	4.29E+05	8.61E+05

TABLE 7.1-10
ACTIVITY RELEASES FOR FUEL HANDLING ACCIDENT

Activity Release (Ci)

Isotope	0 – 2 hr.
Kr-85m	8.40E+00
Kr-85	1.10E+03
Kr-88	3.0E-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	7.0E-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 χ/Q (s/m^3)	Site χ/Q (s/m^3)	χ/Q Ratio (Site/DCD)
All Accidents	EAB	0 – 2	1.00E-03	1.04E-04	1.04E-01
	LPZ	0 – 8	5.00E-04	9.65E-06	1.93E-02
		8 – 24	3.00E-04	8.35E-06	2.78E-02
		24 – 96	1.50E-04	6.09E-06	4.06E-02
		96 – 720	8.00E-05	3.88E-06	4.85E-02

Note: The χ/Q values used for the various postulated accident dose analyses are consistent with DCD Table 15A-5. It is seen that the siteχ/Q values are bounded by the DCD χ/Q values for all time intervals. The site χ/Q values were obtained from Table 2.7-118 of Subsection 2.7.3.

TABLE 7.1-12
SUMMARY OF DESIGN BASIS ACCIDENT DOSES

DCD/SRP Subsection	DCD Description Accident	Site Dose (rem TEDE) EAB	LPZ	Limit[a]	Reference Dose Table
15.1.5	Steam System Piping Failure				
	Pre-Existing Iodine Spike	0.10	0.02	25	7.1-13
	Accident-Initiated Iodine Spike	0.11	0.05	2.5	7.1-14
15.2.8	Feedwater System Pipe Break	(b)	(b)		
15.3.3	Reactor Coolant Pump Shaft Seizure				
	No Feedwater	0.08	0.01	2.5	7.1-15
	Feedwater Available	0.06	0.02	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[d]	0.37	0.11	6.3	7.1-17
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	0.22	0.02	2.5	7.1-18
15.6.3	Steam Generator Tube Rupture				
	Pre-Existing Iodine Spike	0.23	0.02	25	7.1-19
	Accident-Initiated Iodine Spike	0.11	0.02	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	1.20	0.31	25	7.1-21
15.7.4	Fuel Handling Accident	0.54	0.05	6.3	7.1-22

a) 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 doses, but they 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 Subsection 15.2.8.3 of the DCD.

c) Reactor Coolant Pump Shaft Break is bounded by Reactor Coolant Pump Shaft Seizure, as indicated in Subsection 15.3.4.2 of the DCD.

d) The source of this accident is Section 15.4.8 of NUREG- 0800. This event is not included in NUREG-1555, Section 7.1, Appendix A, “Design Basis Accidents Included in Section 15 of the Standard Review Plan.”

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[(a)]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	1.0E+00		1.04E-01	1.04E-01	
0 – 8 hr.		5.81E-01	1.93E-02		1.12E-02
8 – 24 hr.		7.18E-02	2.78E-02		2.00E-03
24 – 96 hr.		1.08E-01	4.06E-02		4.38E-03
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	1.0E+00	7.61E-01		1.04E-01	1.76E-02
Limit				25	25

a) EAB dose obtained from Subsection 15.1.5.4.6 of the DCD.

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[(a)]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	1.10E+00		1.04E-01	1.14E-01	
0 – 8 hr.		1.02E+00	1.93E-02		1.97E-02
8 – 24 hr.		3.77E-01	2.78E-02		1.05E-03
24 – 96 hr.		5.36E-01	4.06E-02		2.18E-02
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	1.10E+00	1.93E+00		1.14E-01	5.19E-02
Limit				2.5	2.5

a) EAB dose obtained from Subsection 15.1.5.4.6 of the DCD.

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[a]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	8.00E-01		1.04E-01	8.32E-01	
0 – 8 hr.		3.89E-01	1.93E-02		7.51E-03
8 – 24 hr.		0.00E+00	2.78E-02		0.00E+00
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	8.00E-01	3.89E-01		8.32E-01	7.51E-03
Limit				2.5	2.5

a) EAB dose obtained from Subsection 15.3.3.3.6 of the DCD.

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[a]	LPZ	(Site/DCD)	EAB	LPZ
6 – 8 hr.[b]	6.00E-01		1.04E-01	6.24E-02	
0 – 8 hr.		7.94E-01	1.93E-02		1.53E-02
8 – 24 hr.		0.00E+00	2.78E-02		0.00E+00
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	6.00E-01	7.94E-01		6.24E-02	1.53E-02
Limit				2.5	2.5

a) EAB dose obtained from Subsection 15.3.3.3.6 of the DCD.

b) The releases for the 6 – 8 hr. period are identified because this is the limiting time period for the 2-hr. EAB dose.

TABLE 7.1-17
DOSES FOR SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB[a]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	3.60E+00		1.04E-01	3.74E-01	
0 – 8 hr.		4.58E+00	1.93E-02		8.84E-02
8 – 24 hr.		7.84E-01	2.78E-02		2.18E-02
24 – 96 hr.		6.32E-02	4.06E-02		2.57E-03
96 – 720 hr.		2.06E-02	4.85E-02		9.99E-04
Total	3.60E+00	5.45E+00		3.74E-01	1.14E-01
Limit				6.3	6.3

a) EAB dose obtained from Subsection 15.4.8.3.6 of the DCD.

TABLE 7.1-18
DOSES FOR FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB[a]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	2.10E+00		1.04E-01	2.18E-01	
0 – 8 hr.		1.02E+00	1.93E-02		1.97E-02
8 – 24 hr.		0.00E+00	2.78E-02		0.00E+00
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	2.10E+00	1.02E+00		2.18E-01	1.97E-02
Limit				2.5	2.5

a) EAB dose obtained from Subsection 15.6.2.6 of the DCD.

TABLE 7.1-19
DOSES FOR STEAM GENERATOR TUBE RUPTURE WITH PRE-EXISTING IODINE SPIKE

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB[a]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	2.20E+00		1.04E-01	2.29E-01	
0 – 8 hr.		1.16E+00	1.93E-02		2.24E-02
8 – 24 hr.		7.24E-02	2.783E-02		2.02E-03
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	2.20E+00	1.23E+00		2.29E-01	2.44E-02
Limit				25	25

a) EAB dose obtained from Subsection 15.6.3.3.6 of the DCD.

TABLE 7.1-20
DOSES FOR STEAM GENERATOR TUBE RUPTURE WITH ACCIDENT-INITIATED IODINE SPIKE

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB[(a)]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	1.10E+00		1.04E-01	1.14E-01	
0 – 8 hr.		6.27E-01	1.93E-02		1.21E-02
8 – 24 hr.		1.69E-01	2.78E-02		4.70E-03
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	1.10E+00	7.96E-01		1.14E-01	1.68E-02
Limit				2.5	2.5

a) EAB dose obtained from Subsection 15.6.3.3.6 of the DCD.

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[(a)]	LPZ	(Site/DCD)	EAB	LPZ
1.2 – 3.2 hr.[(b)]	1.15E+01		1.04E-01	1.20E+00	
0 – 8 hr.		1.22E+01	1.93E-02		2.35E-01
8 – 24 hr.		9.31E-01	2.78E-02		2.59E-02
24 – 96 hr.		4.58E-01	4.06E-02		1.86E-02
96 – 720 hr.		6.09E-01	4.85E-02		2.95E-02
Total	1.15E+01	1.42E+01		1.20E+00	3.10E-01
Limit				25	25

a) EAB dose obtained from Subsection 15.6.5.3.8.1 and Table 15.6.5-3 of the DCD.

b) Per DCD Section 15.6.5.3.8.1, the reported exclusion area boundary doses are for the time period of 1.2 – 3.2 hr. This is the 2-hr. interval that has the highest calculated doses. The dose that would be incurred over the first 2 hr. of the accident is well below the reported dose.

TABLE 7.1-22
DOSES FOR FUEL HANDLING ACCIDENT

	DCD Dose (rem TEDE)		χ/Q Ratio	Site Dose (rem TEDE)	
Time	EAB[(a)]	LPZ	(Site/DCD)	EAB	LPZ
0 – 2 hr.	5.20E+00		1.04E-01	5.41E-01	
0 – 8 hr.		2.59E+00	1.93E-02		5.00E-02
8 – 24 hr.		0.00E+00	2.78E-02		0.00E+00
24 – 96 hr.		0.00E+00	4.06E-02		0.00E+00
96 – 720 hr.		0.00E+00	4.85E-02		0.00E+00
Total	5.20E+00	2.59E+00		5.41E-01	5.00E-02
Limit				6.3	6.3

a) EAB dose obtained from Subsection 15.7.4.5 of the DCD.

7.2 SEVERE ACCIDENTS

7.2.1 INTRODUCTION

This section discusses the probabilities and consequences of accidents of greater severity than the design basis accidents. As a class, they are considered less likely to occur, but because their consequences could be more severe, they are considered important both in terms of impact to the environment and off-site costs. These severe accidents can be distinguished from design basis accidents in two primary respects: (1) they involve substantial physical deterioration of the fuel in the reactor core, including overheating to the point of melting, and (2) they involve deterioration of the capability of the containment system to perform its intended function of limiting the release of radioactive materials to the environment.

7.2.2 APPLICABILITY OF EXISTING GENERIC SEVERE ACCIDENT STUDIES

Section 5.3.3 of NUREG-1437, Volume 1, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), presents a thorough U.S. Nuclear Regulatory Commission (NRC) staff assessment of the impacts of severe accidents during the license renewal period. Methodologies described therein were developed to evaluate the environmental impacts and off-site costs of severe accidents (Reference 7). The resulting conclusions are considered, for reasons discussed below, broadly applicable beyond the license renewal context. Thus, the conclusions contained in the GEIS provide a basis for comparison for the evaluation of the BLN site described in this section.

As described in the GEIS, the purpose of the GEIS evaluation of severe accidents was to use, to the extent practical, the available severe accident results in conjunction with those factors that are important to risk and that change with time, to estimate the consequences of nuclear plant accidents for all plants for a time period that exceeds the time frame of existing analyses. This estimation process was completed by predicting increases or decreases in consequences as the plant lifetime was extended past the normal license period by considering the projected changes in the risk factors. The primary assumption in this analysis was that regulatory controls ensure that the physical plant condition (i.e., the predicted probability of and radioactive releases from an accident) is maintained at a constant level during the operating period. Therefore, the frequency and magnitude of a release remains relatively constant. In other words, significant changes in consequences would result only from changes in the plant's external environment. The logical approach, then, would be to incorporate the most significant environmental factors into calculations of consequences for subsequent correlation with existing analyses (which use the consequence computer codes).

The NRC staff concluded in NUREG-1437 that the primary factors affecting risk are the site population, which reflects the number of people potentially at risk of severe accident exposure, and wind direction, which reflects the likelihood of exposure (Reference 7).

While the NUREG-1437 discussions dealt with the environmental impacts of accidents during operation after license renewal, the primary assumption for this evaluation was that the frequency, or likelihood of occurrence, and magnitude of an accident at a given plant would not increase during the plant lifetime because the plant's licensing basis is maintained in accordance with the regulatory controls. This assumption is equally applicable for new units. Therefore, the thorough generic analysis of severe accident impacts presented in the GEIS provides an appropriate basis for comparison when evaluating severe accident impacts of new units.

However, it was recognized that the changing environment around the plant is not subjected to regulatory controls, introducing the potential for changing risk (Reference 7). Thus, the site-specific environmental considerations, i.e., population and meteorology, were evaluated in the GEIS and are considered in the following sections.

7.2.3 EVALUATION OF POTENTIAL SEVERE ACCIDENT RELEASES

7.2.3.1 Methodology

The severe accident consequence analysis was performed using the Melcor Accident Consequence Code System (MACCS2) risk analysis code. An AP1000 was evaluated for the BLN site.

The analysis was performed with the MACCS2 version designated as Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2 V.1.13.1, CCC-652 Code Package (Reference 4). MACCS2, Version 1.13.1, released in January 2004, simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS2 are atmospheric transport, mitigating actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. The MACCS2 program was chosen for this analysis because it is NRC-endorsed, as stated in the MACCS2 User’s Guide, and has been used in other severe accident consequence evaluations. The BLN model had no important deviations from the default code input values, except for site-specific values and reactor design information. The code values modified for the AP1000 were primarily the source term data from the AP1000 Probabilistic Risk Assessment (PRA). These data include the radionuclide inventory, power level, release fractions and corresponding frequencies, plume release start time, plume release height, delay, and duration. Values for the ATMOS input data file (one of the five input files used by MACCS2) were modified, as necessary, to use data appropriate for the AP1000 source terms and probability frequencies. The remaining four MACCS2 input files were reviewed and modified as necessary.

One year of site-specific hourly meteorological data, which represents current conditions, was used in the analysis. Stability class was calculated using the BLN site meteorological data and the methodology of Regulatory Guide 1.23, Table 1. In accordance with EPA recommendations, short periods of missing data were replaced by interpolating from the values immediately before the data gap to the values immediately after the data gap, while longer periods of missing data were replaced with data from nearby days which had similar meteorological conditions as before and after the data gaps (Reference 6). Meteorology is further discussed in FSAR Section 2.3.

Morning and afternoon mixing height values used for this analysis are the same as those presented in Table 2.3-303 of the BLN FSAR. The treatment of rain/precipitation events follows the default recommended parameter values given in the ATMOS file supplied with the MACCS2 code.

The population distribution and land-use information for the region surrounding the BLN site are specified in the SITE input data file. Contained in the SITE input data file are the geometry data used for the site (spatial intervals and wind directions), population distribution, fraction of the area that is land, watershed data for the liquid pathways model, information on agricultural land use and growing seasons, and regional economic information.

A 50-mi. radius area around the site was divided into 16 directions that are equivalent to a standard navigational compass rosette. This rosette was further divided into inner radial rings as shown in BLN FSAR Figures 2.1-205 and 2.1-206.

The results presented in this section are based on 2007 population data. These data are used because they provide an accurate model of the actual population near the BLN site. In the MACCS2 evaluation, however, the model is projected through the year 2017, and the results remain acceptable.

The land fractions are estimated from FSAR Figures 2.1-205 and 2.1-206.

Regional indices are identified as either Alabama, Georgia, or Tennessee for region indexing. The default economic values supplied by the code were increased by the Consumer Price Index (CPI) ratio of the November 1988 value of 118.3 (when the NUREG-1150 data above were generated) to the November 2006 value of 194.3, where CPI data were taken from the U.S. Department of Labor Bureau of Labor Statistics web page (Reference 2). Details regarding farm acreage for the counties within a 50-mi. radius of the plant were taken from the Agricultural Marketing Services branch of the U.S. Department of Agriculture (USDA) agricultural statistics state summary web pages (Reference 1). The fraction of farmland, based on farm acreage and total area for each county, and updated economic values, based on the CPI ratio, are shown in Table 7.2-1.

The crop information required by MACCS2 was collected from county statistics on the same USDA web pages described above. These were combined and weighted by the total farmland area within the 50-mi. radius to produce a single composite measure, as shown in Table 7.2-2.

The growing season was conservatively assumed to be all year long in the MACCS2 analysis.

The EARLY module of the MACCS2 code models the time period immediately following a radioactive release. This period, which may extend up to one week after the arrival of the first plume at any downwind spatial interval, is commonly referred to as the emergency phase. The subsequent intermediate and long-term periods are treated by the CHRONC module of the code. In the EARLY module, the user may specify emergency response scenarios that include evacuation, sheltering, and dose-dependent relocation. The EARLY module has the capability of combining results from up to three different emergency response scenarios. This is accomplished by appending change records to the EARLY input data file. The first emergency response scenario is defined in the main body of the EARLY input data file. Up to two additional response scenarios can be defined through change record sets positioned at the end of the file.

The emergency evacuation model has been modeled as a single evacuation zone extending out 10 mi. from the site. For the purposes of this analysis, an average evacuation speed of 4.0 mph is used with a 7200-second delay between the alarm and start of evacuation, with no sheltering for the base case. Once evacuees are more than 25 mi. from the site, they no longer factor into the analysis. The evacuation scenario is modeled so that 95 percent of the population is evacuated, compared to no evacuation, for the purpose of composite results.

The ATMOS input data file calculates the dispersion and deposition of material released “source terms” to the atmosphere as a function of downwind distance. Source term release fractions (RELFRC) are shown in Table 7.2-3, and plume characterizations are shown in Table 7.2-4. These data include the source term release fractions, plume start time, plume release height, delay, and duration. The leading contributors to dose consequences are the release categories defined in Notes a) through g) of Table 7.2-3. The source terms relate to the leading contributors to core damage frequency, listed in Table 33-4 of the PRA, and to large-release frequency, listed in Table 43-6 of the PRA. The PRA results can be found in the summary in FSAR Section 19.59.

The data in Table 7.2-3 and 7.2-4 are from Table 49-2 of the AP1000 PRA. The four plumes in Table 49-2 of the AP1000 PRA were collapsed into two plumes, in order to maintain consistency with previous analyses such as that performed for North Anna, using the following steps:

1. The release fractions for the first two plumes in AP1000 PRA Table 49-2 were added together to produce a release fraction for the first plume in Table 7.2-3. Similarly, the third and fourth plumes in AP1000 PRA Table 49-2 were combined for the second plume in Table 7.2-3. This assures that the total release is the same.

2. The alarm time in Table 7.2-4 is equal to the first plume release start time in AP1000 PRA Table 49-2.

3. The first plume duration in Table 7.2-4 is the maximum of the first two plume durations in AP1000 PRA Table 49-2. Similarly, the second plume duration in Table 7.2-4 is the maximum of the third and fourth plume durations in AP1000 PRA Table 49-2.

4. The plume delays in Table 7.2-4 were taken as the first and second plume start times in AP1000 PRA Table 49-2. Thus the inventory is released faster in this approach than in the four plume approach.

5. The Ref Time term in Table 7.2-4, which calculates the plume position according to its leading edge (0) or midpoint (0.5), is equal to the plume position in AP1000 PRA Table 49-2 for the first and second plumes, respectively, to be consistent with the plume delay approach.

The plume release height is the plant vent elevation from Table 15A-7 of the DCD. The vent elevation is used rather than a ground level release because the vent is the location of plant releases in an actual severe accident event. Furthermore, when considering off-site dose risks, a slightly elevated release is more conservative than a ground level release because fewer radionuclides are deposited in the low population areas immediately surrounding the BLN site. Parameters are assigned to each source term according to release category. Each released plume is assumed to have two segments.

The results of the dose and dollar risk assessments are provided in Table 7.2-5. Risk is defined in these results as the product of release category frequency and the dose or cost associated with the release category. The total risk is assumed to be the sum of all scenarios.

The sum of the mean values for affected farmland areas for all release scenarios are also shown in Table 7.2-5. Each of these mean values has also been multiplied by their release category frequency. The values for total early and latent fatalities per reactor year were conservatively calculated as the sum of all release scenarios. Tables 7.2-6 and 7.2-7 support the calculated dose per reactor year and dollars per reactor year risks presented in Table 7.2-5. The release frequency data come from Table 1B-1 of the DCD. External events were considered in developing the release frequencies, but as stated in Chapter 58 of the AP1000 PRA, “as per the site selection criteria defined in Chapter 2 of the DCD, a frequency of occurrence of 10^{-6} per year, for an accident external to AP1000 that has a potential consequence serious enough to affect the safety of the plant according to 10 CFR Part 100 guidelines, is not exceeded.” (Reference 9) Therefore, external events were determined to be negligible compared to internal events and were not incorporated into the release frequencies.

7.2.3.2 Consequences

Due to the extremely low frequency of severe accidents in the AP1000 PRA results, the severe accident individual dose for the BLN site is also low. The weighted total dose risk is 2.88×10^{-2} person-rem/reactor year (RY). This is much lower than the background radiation. NUREG-1437, Volume 1, Section 5.5.2.1 indicates that the average individual dose caused by all other sources is 3×10^{-1} rem/year. Since the weighted total dose risk from severe accidents is much lower than the background radiation, it can also be concluded that the impact on the local biota is considered negligible. Radiological dose consequences and health effects associated with normal and anticipated operational releases are discussed in Subsection 5.4.3.

Surface water dose pathways are an extension of the air pathway. These pathways cover the effects of radioactive material deposited on open bodies of water. The surface water pathways of interest include external radiation from submersion in water and activities near the water, ingestion of water, and ingestion of fish and other aquatic creatures. Of these pathways, the MACCS2 code evaluates only the ingestion of contaminated water. The risks associated with this surface water pathway calculated for the BLN site are included in the last column of Table 7.2-9. NUREG-1437 Table 5.13 indicates that Bellefonte is classified as a "small river" site and Table 5.17 indicates that for small river sites, drinking water is the dominant liquid pathway compared to seafood ingestion and shoreline exposure. Furthermore, the water ingestion dose risk of 4.51×10^{-3} person-rem/RY is small compared to the total dose risk of 2.88×10^{-2} person-rem/RY. Finally, should a severe accident occur at the BLN site, it is likely that federal, state, and local officials would restrict access to the river below the site and in contaminated areas above the site. These actions would further reduce surface water pathway exposures.

In addition to surface water, groundwater must be considered in the liquid pathways dose. Subsection 2.3.2.3.1 states that local groundwater use in the vicinity appears limited to mainly individual residences. Also, as discussed in Subsection 2.3.1.5.4, there is no use or anticipated use of groundwater at the BLN site. Potable, sanitary, and fire suppression water is supplied by the Scottsboro Municipal Water System, which draws water from the Tennessee River.

Subsection 2.3.1.5.4 further states that the majority of groundwater discharge from the BLN site is to the Town Creek embayment. As shown in Subsection 2.3.1.5.6, the average groundwater travel time from Unit 3 to the Town Creek embayment is 655 days, which allows ample time for interdiction and other prevention activities. Therefore, the groundwater pathway is expected to be negligible.

The results of severe accidents for current generation reactors are compared to the severe accident risk calculated in the MACCS2 analysis in Table 7.2-8, where the data for the current generation reactors were taken from the Systems Energy Resources, Inc. (SERI) Response to Requests for Additional Information dated August 10, 2004 (Reference 5). The conclusion is that the low frequency of releases associated with the AP1000 design makes the severe accident risk of a future unit at this site extremely low. Additional severe accident analysis results are reported in Table 7.2-9.

Because the AP1000 CDF and frequency of releases are extremely low and the dose risk is significantly lower than background radiation, the BLN site risk is considered acceptable. Section 5.5.2.1 of NUREG-1437, Volume 1, indicated these predicted effects of a severe accident “are not expected to exceed a small fraction of that risk to which the population is already exposed.” Section 5.5.2.1 goes on to state that “the probability-weighted consequences from atmospheric releases associated with severe accidents is judged to be of small significance for all plants.”

7.2.3.3 Evaluation of Economic Impacts of Severe Accidents

This section discusses the potential economic impacts as a result of postulated severe accidents at a nuclear reactor on the BLN site. MACCS2 calculated severe accident costs based on the following:

- Evacuation costs.
- Value of crops contaminated and condemned.
- Value of milk contaminated and condemned.
- Costs of decontamination of property.
- Indirect costs resulting from the loss of use of property and incomes derived as a result of the accident.

The MACCS2 code calculated the cost of severe accidents at the BLN site to be 76.80 dollars/reactor year (RY). This low cost is mostly due to the extremely low frequencies expected for accidents of this magnitude. Socioeconomic impacts from a severe accident are further discussed in ER Section 7.3.

7.2.4 CONCLUSION

The environmental impacts of a postulated severe accident at the BLN site could be severe, but due to the inherent safety of the AP1000 design, as reflected by the extremely low likelihood of such an accident as compared to the generation plants, the risk-weighted impacts are determined to be acceptable. Also, the total dose risk value of 2.88×10^{-2} person-rem/RY calculated in the MACCS2 analysis is bounded by the dose risk of 4.32×10^{-2} person-rem/RY given in Table 1B-1 of the DCD. This is especially significant because the DCD value is calculated using only the first 24 hrs. after an accident, while the MACCS2 analysis calculates consequences for the entire event.

Additionally, the Commission's Safety Goal Policy Statement, issued in 1986, states that "the risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1 percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed" and that "the risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes." (Reference 8) From the Centers for Disease Control and Prevention website, there were 2,398,343 deaths in the United States in the year 2004. Of these deaths, 550,270 resulted from cancer, while 108,694 resulted from accidents (Reference 3). This means that the cancer fatality risk from "all other causes" is 0.229 and the prompt fatality risk from "other accidents" is 0.045. One-tenth of one percent of each of these risks results in a value of 2.29×10^{-4} for cancer fatalities and 4.50×10^{-5} for prompt fatalities. As shown in Table 7.2-5, the risk for latent (cancer) fatalities is 1.83×10^{-5}, and the risk for early (prompt) fatalities is 0, both of which are bounded by their respective criteria. Therefore, the early and latent fatality risks from a severe accident at the BLN site are considered acceptable.

7.2.5 REFERENCES

1. Agricultural Marketing Services, Website, http://www.ams.usda.gov/statesummaries/AL/District.htm, http://www.ams.usda.gov/statesummaries/GA/District.htm, and http://www.ams.usda.gov/statesummaries/TN/District.htm, accessed June 5, 2007.

2. Bureau of Labor Statistics, Website, http://www.bls.gov/cpi/, accessed May 9, 2007.

3. Centers for Disease Control and Prevention, Website, http://www.cdc.gov/nchs/fastats/deaths.htm, accessed August 2, 2007.

4. Chanin, D.I. and M.L. Young, *Code Manual for MACCS2: Volume 1, User's Guide*, SAND97-0594, Sandia National Laboratories, Albuquerque, New Mexico, May 1998.

5. Systems Energy Resources, Inc. letter to USNRC, "Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 4)," CNRO-2004-00050, Docket No. 52-009, August 10, 2004 [ADAMS Accession No. ML042290395].

6. U.S. Environmental Protection Agency, *Procedures for Substituting Values for Missing NWS Meteorological Data for Use in Regulatory Air Quality Models*, July 7, 1992.

7. U.S. Nuclear Regulatory Commission, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, NUREG-1437.

8. U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants: Policy Statement," 51FR28044, August 4, 1986.

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

TABLE 7.2-1
STATE ECONOMIC STATISTICS CORRECTED FOR INFLATION AND FARM FRACTION

Region[(a)]	State	Fraction farm	Fraction dairy	Farm sales ($/hectare)	Property value ($/hectare)	Non-farm property value ($/person)
1	ALA	0.383	0.040	754	2,995	101,804
9	GA	0.274	0.060	1,007	3,095	119,866
40	TENN	0.386	0.153	591	3,038	108,372

a) The region values are the numbers recorded in the MACCS2 site input file to designate a particular state.

TABLE 7.2-2
DISTRICT FARM STATISTICS AND WEIGHTED COMPOSITES

	AL-4	AL-5	GA-10	GA-11	TN-3	TN-4	Composite
Pasture	0.490	0.397	0.582	0.644	0.498	0.391	0.459
Stored Forage	0.140	0.115	0.190	0.132	0.172	0.169	0.145
Grains	0.045	0.092	0.009	0.029	0.047	0.124	0.071
Green Leafy	0.002	0.002	0.000	0.002	0.001	0.001	0.002
Other	0.001	0.001	0.000	0.001	0.000	0.000	0.000
Legumes/seeds	0.040	0.101	0.011	0.016	0.056	0.095	0.066
Roots/tubers	0.001	0.001	0.000	0.000	0.000	0.000	0.001

a) AL-4 and AL-5 are Alabama electoral districts 4 and 5.

b) GA-10 and GA-11 are Georgia electoral districts 10 and 11.

c) TN-3 and TN-4 are Tennessee electoral districts 3 and 4.

TABLE 7.2-3
AP1000 SOURCE TERM RELEASE FRACTIONS

Release Category	Xe/Kr	I	Cs	Te	Sr	Ru	La	Ce	Ba
CFI[a]	8.0E-1	3.3E-3	3.3E-3	4.4E-4	2.2E-2	9.3E-3	8.1E-3	4.3E-5	1.7E-2
CFI	1.2E-1	0.0E+0	0.0E+0	6.0E-6	0.0E+0	0.0E+0	1.1E-2	4.1E-5	0.0E+0
CFE[b]	8.2E-1	5.7E-2	5.5E-2	1.4E-3	3.5E-3	1.4E-2	6.5E-5	1.0E-6	5.3E-3
CFE	1.4E-1	0.0E+0	0.0E+0	6.0E-7	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
DIRECT[c]	4.4E-3	3.6E-5	3.5E-5	2.4E-6	3.2E-5	3.9E-5	4.1E-6	1.8E-8	3.6E-5
DIRECT	3.5E-3	0.0E+0	0.0E+0	5.4E-9	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
IC[d]	1.5E-3	1.2E-5	1.2E-5	8.1E-7	1.1E-5	1.3E-5	1.4E-6	5.9E-9	1.2E-5
IC	1.2E-3	0.0E+0	0.0E+0	1.8E-9	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
BP[e]	1.E+0	2.2E-1	2.0E-1	9.4E-3	3.6E-3	4.5E-2	1.3E-4	3.2E-6	8.9E-3
BP	0.0E+0	2.3E-1	7.6E-2	6.9E-3	0.0E+0	0.0E+0	0.0E+0	0.0E+0	1.0E-6
CI[f]	6.9E-1	4.6E-2	2.1E-2	1.7E-3	2.0E-2	4.0E-2	2.4E-4	3.0E-6	3.2E-2
CI	8.4E-2	0.0E+0	0.0E+0	9.4E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
CFL[g]	1.5E-3	1.2E-5	1.2E-5	1.0E-6	1.7E-5	1.7E-5	1.2E-5	4.8E-8	1.7E-5
CFL	9.8E-1	2.1E-5	1.2E-5	3.7E-5	2.8E-3	1.4E-3	1.4E-1	5.3E-4	2.6E-3

a) Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after core relocation but before 24 hrs. Such phenomena include: hydrogen detonation and hydrogen deflagration.

b) Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after the onset of core damage but prior to core relocation. Such phenomena include: hydrogen detonation, hydrogen diffusion flame, steam explosions, and vessel failures.

c) A modification of the IC release category in which no credit is assumed for aerosol nuclide deposition in the middle annulus. This is a conservative case used only as a sensitivity evaluation.

d) Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

e) Fission products are released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

f) Fission product release occurs through a failure of the system or valves that close the penetrations between containment and the environment. Containment failure occurs prior to onset of core damage.

g) Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after 24 hrs. Such phenomena include the failure of containment heat removal (failure of passive containment cooling). (Reference 9)

TABLE 7.2-4
AP1000 PLUME CHARACTERIZATION DATA

Release Category	Alarm (s)	Number of Plume Releases	Risk-Dominant Plume	Ref Time	Plume Heat (W)	Plume Release Height (ft)	Plume Duration (s)	Plume Delay (s)
CFI	2924	2	1	0.0	0	182.7	53,830	2,924
CFI	2924	2	1	0.5	0	182.7	86,400	32,590
CFE	3004	2	1	0.0	0	182.7	70,160	3,004
CFE	3004	2	1	0.0	0	182.7	86,400	19,810
DIRECT	4378	2	1	0.5	0	182.7	80,432	4,378
DIRECT	4378	2	1	0.0	0	182.7	86,400	84,810
IC	4378	2	1	0.5	0	182.7	80,432	4,378
IC	4378	2	1	0.0	0	182.7	86,400	84,810
BP	31890	2	1	0.5	0	182.7	40,050	31,890
BP	31890	2	1	0.0	0	182.7	86,400	46,440
CI	100.8	2	1	0.5	0	182.7	86,380	100.8
CI	100.8	2	1	0.5	0	182.7	75,300	50,020
CFL	2922	2	1	0.5	0	182.7	81,640	2,922
CFL	2922	2	1	0.5	0	182.7	86,400	26,360

TABLE 7.2-5
RESULTS SUMMARY WITHIN 50 MI. OF THE BLN SITE

Dose Risk (person-rem/yr)	Dollar Risk ($/yr)	Affected Farmland (acres)[a]	Early Fatalities (per RY)	Latent Fatalities (per RY)
2.88E-2	7.68E+1	5.41E-3	0.00E+00	1.83E-5

a) This value reflects the sum of affected farmland areas that have been multiplied by their release category frequency, whereas the affected farmland areas shown in the MACCS2 analysis are neither multiplied by release category frequency or summed. However, the same MACCS2 data are used as the basis for both values.

TABLE 7.2-6
MEAN VALUE FOR TOTAL DOSE RISK ASSESSMENT IN PERSON-REM/RY

Release Category	Frequency (per RY)	Dose Risk
CFI	1.89E-10	1.29E-04
CFE	7.47E-09	6.43E-03
IC	2.21E-07	9.08E-04
BP	1.05E-08	2.04E-02
CI	1.33E-09	9.22E-04
CFL	3.45E-13	3.42E-09
Total		2.88E-02

TABLE 7.2-7
DOLLAR RISK ASSESSMENT IN DOLLARS/RY

Release Category	Frequency (per RY)	Dollar Risk
CFI	1.89E-10	3.74E-01
CFE	7.47E-09	1.48E+01
IC	2.21E-07	5.59E-01
BP	1.05E-08	5.89E+01
CI	1.33E-09	2.21E+00
CFL	3.45E-13	1.26E-03
Total		7.68E+01

TABLE 7.2-8
POPULATION DOSE COMPARISON AMONG PLANTS

Plant	Population Dose within 50 mi (person-rem/yr)[a]
Zion	5.47E+1
Grand Gulf	5.2E-1
Surry	6.E+0
North Anna	2.51E+1
BLN AP1000	2.88E-2

a) Data for the current generation reactors were taken from Reference 5.

TABLE 7.2-9

SEVERE ACCIDENT IMPACTS TO THE POPULATION AND FARMLAND[a]

Release Category	Core Damage Frequency (per RY)	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Cost-Risk (dollars per RY)	Affected Farmland Area (acres/RY)[b] per year	Water Ingestion Pathway (person-rem/RY)
CFI	1.89E-10	1.29E-4	0	1.03E-7	3.74E-1	3.00E-5	7.90E-6
CFE	7.47E-9	6.43E-3	0	3.57E-6	1.48E+1	1.48E-3	5.80E-4
IC	2.21E-7	9.08E-4	0	4.60E-7	5.59E-1	4.71E-5	7.71E-6
BP	1.05E-8	2.04E-2	0	1.34E-5	5.89E+1	3.63E-3	3.82E-3
CI	1.33E-9	9.22E-4	0	7.55E-7	2.21E+0	2.17E-4	8.79E-5
CFL	3.45E-13	3.42E-7	0	3.93E-10	1.26E-3	7.22E-8	1.63E-9
Total	2.40E-7	2.88E-2	0	1.83E-5	7.68E+1	5.41E-3	4.51E-3

a) This table provides impacts within a 50-mi. radius of the BLN Site.

b) These values reflect affected farmland areas that have been multiplied by their release category frequency, whereas the affected farmland areas shown in the MACCS2 analysis are not multiplied by release category frequency. However, the same MACCS2 data are used as the basis for both values.

7.3 SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

This section updates the Westinghouse Severe Accidents Mitigation Design Alternatives discussion provided in Appendix 1B of the AP1000 Design Control Document (DCD) with BLN site and regional data. The BLN site-specific analysis demonstrates that the severe accident mitigation design alternatives determined not to be cost beneficial by Westinghouse are also not cost beneficial when BLN site-specific data are considered.

As described in Section 7.2, Westinghouse performed a generic severe accident analysis for the AP1000 in Appendix 1B of the AP1000 DCD as part of the design certification process. 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. As stated in 10 CFR Part 52 VI(B)(7), the environmental issues concerning SAMDAs for the AP1000 design and Appendix 1B of the generic DCD are closed for plants referencing this appendix whose site parameters are within those specified in the SAMDA evaluation. Since the site parameters used in Appendix 1B could not be shown to be bounding for BLN, addition evaluations were performed, as described in Sections 7.2 and 7.3. Section 7.2 extends the Westinghouse generic severe accident analysis to examine the Units 3 and 4 at the BLN site and determined that the generic conclusions remain valid for the BLN site. The analysis in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented at the BLN site to mitigate these small severe accident impacts. A site-specific calculation package supporting this analysis was prepared.

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). SAMAs are somewhat broader than SAMDAs, severe accident mitigation design alternatives, which primarily focus on design changes and do not consider procedural modifications. 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 and analyzing potential SAMAs includes four steps:

1. Define the base case - The base case is the dose-risk and cost-risk of severe accidents 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 BLN 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 judgment, and is 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 of each SAMA - A detailed engineering cost evaluation is developed using current plant engineering processes for each SAMA remaining after Step 2. 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 current analysis is limited to demonstrating that the BLN site is bounded by the Westinghouse DCD analysis and determining what magnitude of plant-specific design or procedural modification would be cost-effective. 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*, 1997 (Reference 2). NUREG/BR-0184 presents methods for determination of the value of decreases in risk, using the following attributes: public health, occupational health, off-site property, replacement power costs, and on-site 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 BLN 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. This AP1000 SAMDA analyses evaluated the leading contributors to (1) core damage frequency, (2) large release frequency, and (3) dose consequences to determine if there are any design alternatives which provide a cost beneficial risk reduction. 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 (CVS) upgraded to mitigate small loss-of-coolant accidents (LOCAs).
- Filtered containment vent.
- Normal residual heat removal system (RNS) located 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 (IRWST).
- Increase of steam generator secondary side pressure capacity.
- Secondary containment filtered ventilation.
- Diverse IRWST injection valves.
- Diverse containment recirculation valves.
- Ex-vessel core catcher.
- High-pressure containment design.
- Diverse actuation system improved reliability.

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. 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. Another calculation of the maximum attainable benefit is made with the conservative discount rate of 3 percent previously used by the NRC. The resulting value is $43,000, which is still very small to justify any appreciable investment.

In its Finding of No Significant Impact relating to the certification of the AP1000 design, NRC (Reference 1) concluded, “none of the potential design alternatives are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that other design changes would be identified and justified in the future on the basis of cost-benefit considerations, because the estimated CDFs [core damage frequency] for the AP1000 are very low on an absolute scale.” Pursuant to 10 CFR 51.55(b), it was confirmed that the design changes that are incorporated into the referenced DCD, as defined in Section 1.1, did not change the SAMDA screening or evaluation results or conclusions. Specifically, the SAMDAs assessed as being rejected for the certified AP1000 design, as documented in DCD Revision 15 Appendix 1B, have not become cost-beneficial for the BLN site. Nor have any new SAMDAs been identified for the BLN site.

7.3.3 MONETIZATION OF THE BLN BASE CASE

The principal inputs to the site-specific calculations are the core damage frequency (reported in Section 7.2), dose-risk and dollar-risk (reported in Tables 7.2-6 and 7.2-7), dollars per person-rem ($2,000 as provided by NRC in NUREG/BR-0184), licensing period (40 years), and economic discount rate (7 percent and 3 percent are NRC precedents). 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 less than $10,000 for a single AP1000 at the BLN site is so low that there are no design changes, over those already incorporated into the advanced reactor designs, that could be determined to be cost-effective. Even with a conservative 3-percent discount rate, the valuation of the averted risk is less than $22,000.

These values compare to the Westinghouse generic analysis results of $21,000 for the 7-percent discount rate and $43,000 for the 3-percent discount rate. The plant specific analysis used actual population and meteorological characteristics that result in lower impacts than did the conservative values used in the generic DCD analysis.

Accordingly, further evaluation of design-related SAMAs is not warranted. Due to the costs associated with processing administrative changes (including procedures and training costs), administrative changes are likely to cost more than the maximum averted cost-risk. Furthermore, because administrative changes would likely have a small impact on risk, the reduction in risk benefit of administrative changes would likely be substantially less than the cost of the administrative changes. Evaluation of administrative controls would not be appropriate until a plant design is finalized and plant administrative processes and procedures are being developed. At that time, risk insights will be considered in the development of plant procedures and training.

7.3.4 REFERENCES

1. 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, OMB-3150-0151(accession number ML060260400), Washington D.C., 71 FR 4478, January 27, 2006.

2. U.S. Nuclear Regulatory Commission, *Regulatory Analysis Technical Evaluation Handbook*, NUREG/BR-0184, January 1, 1997.

TABLE 7.3-1
MONETIZATION OF THE BLN AP1000 BASE CASE

	7% Discount Rate	3% Discount Rate
Off-site exposure cost	$412	$1,024
Off-site economic cost	$549	$1,366
On-site exposure cost	$122	$341
On-site cleanup cost	$3,579	$7,984
Replacement power cost	$5,152	$11,253
Total	$9,814	$21,967

7.4 TRANSPORTATION 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 Reference 1, 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 WASH-1238. Consequently, as described in References 2, 3, and 4, the overall transportation accident risks associated with advanced reactor spent fuel shipments are likely to be SMALL and are consistent with the risks associated with transportation of spent fuel from current-generation reactors.

7.4.2 TRANSPORTATION OF SPENT FUEL

In its assessments of 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).

A screening analysis was conducted on the inventories reported in Reference 5 by the NRC to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations. The screening identified the radionuclides that would contribute more than 99.999 percent of the dose from inhalation and the results are reported in the References 2, 3, and 4.

Radionuclide inventories are important parameters in the calculation of accident risks. The radionuclide inventories used in this analysis were taken directly from References 2, 3, 4, and 5, with the exception of Co-60.

Co-60 inventories for this analysis were taken directly from Reference 6. The following discussion is from Section 7.2.3.5 of Reference 6 and provides a discussion regarding the importance of including Co-60 in the overall source term.

> “During reactor operation, corrosion products formed in the reactor’s primary cooling system deposit on fuel assembly surfaces where elements in these deposits are activated by neutron bombardment. The resulting radioactive deposits are called CRUD. Due to vibratory loads during incident free transportation, impact loads during collision accidents, and thermal loads during accidents that lead to fires, portions of these radioactive deposits may spall from the rods. Then, if some of these spalled materials become airborne during an accident, their release to the atmosphere could contribute to the radiation exposures caused by the accident. Although CRUD contains a number of radionuclides, only Co-60 would contribute significantly to these radiation exposures.

> Since the CRUD deposits on typical PWR spent fuel rods typically contain 0.2 Ci of Co-60 per rod and the generic PWR assemblies for which ORIGEN inventories were calculated contain respectively 289 spent fuel rods, the amounts of Co-60 produced by activation of deposits on assembly surfaces is 57.8 Ci for the generic PWR assembly (115.6 Ci/MTU based on 0.5 MTU/assembly)."

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

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR Part 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 Reference 7, 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.

The 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 amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantities: 24 MTU/yr for the AP1000. This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2.

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. The same assumptions regarding release fractions were used in this analysis.

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

Table 7.4-2 presents unit (per MTU) accident risks associated with transportation of spent fuel from the proposed advanced reactor sites to the proposed Yucca Mountain repository. The accident risks are provided in the form of a unit collective population dose (i.e., person-rem per MTU). The table also presents estimates of accident risk per reactor year normalized to the reference reactor analyzed in Reference 1.

7.4.3 NONRADIOLOGICAL IMPACTS

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

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

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

The general approach used, in this document, to calculate nonradiological impacts of unirradiated and spent fuel shipments is based on the approach used in the Yucca Mountain SEIS which used adjusted state-level accident, injury, and fatality statistics from References 7 and 8 (Table 7.4-3). The round trip distances between the proposed advanced reactor sites and the fuel fabrication facility (assumed to be located in Lynchburg, Virginia) and Yucca Mountain, Nevada (Table 7.4-4) provided the data for the last part of the equation. State-by-state shipping distances were obtained from the Web-TRAGIS output file and combined with the annual number of shipments and accident, injury, and fatality rates by state from References 7 and 8, to calculate nonradiological impacts. The results are shown in Table 7.4-4 and are compared to those reported in Table S-4. It should be noted that because of the larger round trip distances and greater number of shipments, 95 percent of the total nonradiological impacts (fresh fuel and spent nuclear fuel), are from the shipment of spent nuclear fuel. It should also be noted that the fatalities/RRY calculated for the shipment of fresh and spent nuclear fuel are slightly larger than those reported in Table S-4. This is primarily due to the longer shipping distances and adjusted accident, injury, and fatality rate data that were used for the shipment of fresh fuel to and spent fuel from Bellefonte and the alternative sites versus what was used for the basis to support Table S-4.

7.4.4 CONCLUSION

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

7.4.5 REFERENCES

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

2. U.S. Nuclear Regulatory Commission, Draft Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site, NUREG-1811, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C., November 2004.

3. U.S. Nuclear Regulatory Commission, Draft Environmental Impact Statement for an Early Site Permit (ESP) at the Exelon ESP Site, NUREG-1815, Office or Nuclear Reactor Regulation, Washington, D.C., July 2006.

4. U.S. Nuclear Regulatory Commission, Environmental Impact Statement for an Early Site Permit (ESP) at the Grand Gulf ESP Site, NUREG-1817, Office or Nuclear Reactor Regulation, Washington, D.C., April 2006.

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

6. U.S. Nuclear Regulatory Commission. Reexamination of SpentFuel Shipment Risk Estimates. NUREG/CR-6672. Office or Nuclear Reactor Regulation. Washington, D.C. March 2000.

7. Saricks, C.L., and M.M. Tompkins, State-Level Accident Rates of Surface Freight Transportation: A Re-examination, Argonne National Laboratory, Energy Systems Division, Center for Transportation Research, April 1999.

8. Blower, D and A. Matteson, Center for National Truck Statistics, Evaluation of the Motor Carrier Management Information System Crash File, Phase 1. University of Michigan. March 2003.

TABLE 7.4-1
RADIONUCLIDE INVENTORY USED IN TRANSPORTATION ACCIDENT RISK CALCULATIONS FOR THE AP1000

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

a) Ci/MTU = curies per metric ton uranium

b) Cobalt-60 is the key radionuclide constituent of fuel assembly Crud (rounded up). Source: Reference 6.

Source: References 2,3,4, and 5

TABLE 7.4-2
SPENT FUEL TRANSPORTATION ACCIDENT RISKS FOR THE AP1000

Site	Unit Population Dose (Person-rem/MTU)	MTU/RRY[a]	Population Dose (person-rem/RRY)[b]
BLN	1.68E-06	19.5	3.27E-05
PBN	1.78E-06	19.5	3.47E-05
YCN	1.67E-06	19.5	3.25E-05
HVN	1.64E-06	19.5	3.19E-05
Murphy Hill	1.83E-06	19.5	3.57E-05
Table S-4			SMALL

a) Based on 39 normalized shipments/yr and 0.5 MTU/shipment

b) Value presented is the product of unit population dose times the product of MTU/shipment and normalized shipments/year.

TABLE 7.4-3 (Sheet 1 of 2)
ADJUSTED ACCIDENT, INJURY AND FATALITY RATES FOR THE UNITED STATES[(a)]

	Accidents/Trucks-mi.		Fatalities/Trucks-mi.		Injuries/Trucks-mi.	
State/Parameter	Interstate	Total	Interstate	Total	Interstate	Total
Alabama	7.45E-07	9.96E-07	2.17E-08	5.55E-08	2.86E-07	4.12E-07
Arizona	3.49E-07	2.83E-07	2.38E-08	2.38E-08	2.25E-07	1.77E-07
Arkansas	3.54E-07	3.91E-07	1.57E-08	5.63E-08	1.90E-07	2.40E-07
California	4.23E-07	2.19E-07	1.77E-08	9.12E-09	2.40E-07	1.24E-07
Colorado	1.18E-06	1.15E-06	2.90E-08	4.44E-08	6.08E-07	5.86E-07
Connecticut	2.38E-06	2.33E-06	3.67E-08	4.84E-08	1.18E-06	1.19E-06
Delaware	1.37E-06	1.92E-06	1.42E-08	5.95E-08	6.60E-07	9.87E-07
Florida	1.82E-07	2.35E-07	1.95E-08	2.72E-08	1.06E-07	1.37E-07
Georgia	*	1.77E-06	*	4.94E-08	*	8.87E-07
Idaho	7.79E-07	1.04E-06	9.62E-09	6.31E-08	5.92E-07	7.61E-07
Illinois	5.86E-07	7.82E-07	2.11E-08	2.78E-08	2.90E-07	3.17E-07
Indiana	5.94E-07	4.46E-07	1.71E-08	2.17E-08	2.70E-07	2.22E-07
Iowa	2.96E-07	3.91E-07	2.38E-08	3.40E-08	1.66E-07	2.19E-07
Kansas	7.50E-07	1.01E-06	1.32E-08	5.81E-08	4.91E-07	6.66E-07
Kentucky	8.19E-07	1.37E-06	3.25E-08	5.81E-08	4.26E-07	6.97E-07
Louisiana	*	5.84E-07	*	2.33E-08	*	3.56E-07
Maine	1.16E-06	1.09E-06	2.30E-08	1.98E-08	6.02E-07	6.44E-07
Maryland	1.43E-06	1.96E-06	1.64E-08	5.04E-08	8.87E-07	1.17E-06
Massachusetts	2.27E-07	4.09E-07	2.03E-09	9.62E-09	9.85E-08	2.01E-07
Michigan	7.47E-07	5.68E-07	2.72E-08	2.72E-08	5.04E-07	4.25E-07
Minnesota	4.52E-07	4.65E-07	7.60E-09	3.04E-08	1.63E-07	2.33E-07
Mississippi	1.27E-07	1.66E-07	6.34E-09	8.61E-09	7.53E-08	1.10E-07
Missouri	1.23E-06	1.42E-06	3.14E-08	4.99E-08	6.07E-07	7.05E-07
Montana	1.64E-06	1.54E-06	3.44E-08	5.15E-08	4.94E-07	4.99E-07
Nebraska	8.43E-07	1.15E-06	3.48E-08	4.75E-08	3.80E-07	5.01E-07
Nevada	5.94E-07	6.47E-07	1.67E-08	2.25E-08	2.86E-07	3.12E-07
New Hampshire	6.95E-07	1.01E-06	0.00E+00	2.99E-08	3.15E-07	4.52E-07
New Jersey	1.49E-06	1.30E-06	3.07E-08	1.80E-08	7.55E-07	7.32E-07
New Mexico	2.98E-07	2.85E-07	2.99E-08	2.78E-08	2.22E-07	2.09E-07
New York	*	9.11E-07	*	3.14E-08	*	3.57E-07
North Carolina	9.14E-07	8.82E-07	3.78E-08	4.10E-08	6.12E-07	6.10E-07
North Dakota	7.98E-07	9.03E-07	2.59E-08	2.82E-08	3.65E-07	4.89E-07
Ohio	4.33E-07	3.06E-07	9.88E-09	9.88E-09	2.70E-07	2.06E-07
Oklahoma	7.08E-07	7.29E-07	3.36E-08	3.73E-08	5.58E-07	5.50E-07
Oregon	*	5.70E-07	*	5.17E-08	*	2.62E-07

TABLE 7.4-3 (Sheet 2 of 2)
ADJUSTED ACCIDENT, INJURY AND FATALITY RATES FOR THE UNITED STATES[(a)]

	Accidents/Trucks-mi.		Fatalities/Trucks-mi.		Injuries/Trucks-mi.	
State/Parameter	Interstate	Total	Interstate	Total	Interstate	Total
Pennsylvania	1.36E-06	1.79E-06	3.43E-08	6.16E-08	7.40E-07	1.03E-06
Rhode Island	*	*	*	*	*	*
South Carolina	*	1.24E-06	*	6.58E-08	*	6.37E-07
South Dakota	6.15E-07	6.05E-07	1.55E-08	3.22E-08	3.32E-07	3.07E-07
Tennessee	3.25E-07	4.20E-07	2.53E-08	3.30E-08	1.77E-07	2.45E-07
Texas	1.59E-06	1.74E-06	3.30E-08	6.84E-08	1.06E-06	1.04E-06
Utah	7.66E-07	8.98E-07	3.01E-08	3.52E-08	4.89E-07	5.49E-07
Vermont	4.97E-07	7.87E-07	*	2.46E-08	2.93E-07	4.25E-07
Virginia	1.04E-06	7.00E-07	4.09E-08	2.95E-08	5.99E-07	4.17E-07
Washington	7.00E-07	5.41E-07	4.55E-09	1.34E-08	3.48E-07	2.70E-07
West Virginia	4.54E-07	5.68E-07	4.26E-08	7.05E-08	2.16E-07	2.70E-07
Wisconsin	1.19E-06	1.45E-06	2.30E-08	5.63E-08	6.44E-07	7.92E-07
Wyoming	1.79E-06	1.79E-06	2.74E-08	3.14E-08	6.24E-07	6.24E-07

a) Values from Table 4 presented in Reference 7 were adjusted by the values obtained in Tables 1 and 2 of Reference 8 to form the basis for this table

TABLE 7.4-4
NONRADIOLOGICAL IMPACTS RESULTING FROM THE SHIPMENT OF UNIRRADIATED AND SPENT NUCLEAR FUEL, NORMALIZED TO REFERENCE LWR

Site	Round Trip Distance One Shipment Unirradiated Fuel - mi	Round Trip Distance One Shipment Spent Fuel – mi	Round-trip Distance, mi/RRY[(a)]	Accidents/ RRY	Injuries/ RRY	Fatalities/ RRY
BLN	1,038	4,499	180,547	9.32E-02	6.04E-02	4.57E-03
PBN	511	4,798	189,626	9.49E-02	6.15E-02	4.82E-03
YCN	1,260	4,544	183,390	9.60E-02	6.11E-02	4.63E-03
HVN	1,001	4,293	172,332	8.95E-02	5.84E-02	4.37E-03
Murphy Hill	1,099	4,688	188,217	1.00E-01	6.33E-02	4.77E-03
Table S-4	----	----	144,357	NC[(b)]	1.00E-01	1.00E-03

a) Based on 4.9 shipments/yr of unirradiated waste, 39 shipments/yr of spent fuel, and round trip shipping distances for both Unirradiated and spent fuel.

b) Not calculated.