ENVIRONMENTAL REPORT

CHAPTER 7

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

7.1 DESIGN BASIS ACCIDENTS

Design basis accidents (DBAs) are events that are not expected to occur, but are evaluated to demonstrate the adequacy of the plant design since the consequences of their occurrence have the potential for radioactive material to be released. DBAs having a potential for radiological releases to the environment are identified in Section 7.1, Appendix A of NUREG-1555 (NRC, 1999) and are listed in Table 7.1-1 along with DBAs applicable for the U.S. EPR. The DBAs are based on Chapter 15 of NUREG-0800 (NRC, 2007) and Regulatory Guide 1.183 (NRC, 2000).

Sources of radioactivity are generated within the reactor core. Radioactivity releases are dependent on the specific accident and may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. Design input used in the DBA radiological consequences evaluations for the U.S. EPR follows the Alternative Source Term Methodology outlined in Regulatory Guide 1.183 (NRC, 2000). The design basis primary and secondary coolant source term activity concentrations for the U.S. EPR are provided in Table 7.1-2 and Table 7.1-3, respectively. Table 7.1-4 lists the design basis source term inventories for the core.

Primary and secondary coolant concentrations are based on the U.S. EPR Technical Specification limits for halogens and noble gases, the American National Standards Institute/ American Nuclear Society (ANSI/ANS)-18.1 Standard (ANS, 1999) for activation products and tritium, and 0.25% fuel defects for remaining radionuclides. For certain accidents (i.e., Steam System Piping Failures, and Steam Generator Tube Rupture), the radiological consequences analyses account for iodine spiking which causes the concentration of various radioactive iodines in the primary coolant to significantly increase to levels described in Table 7.1-2. The iodine appearance rates (i.e., rates at which iodine isotopes are transferred from the core to the primary coolant via assumed fuel cladding defects) used in DBA analyses for the U.S. EPR were based on a conservative Reactor Coolant System letdown purification flow rate. Referring to Table 7.1-3, no secondary coolant noble gas source term is applicable since noble gas leakage from the Reactor Coolant System is assumed to enter the steam phase directly. Design basis core source terms were determined for a power level of 4,612 MWt, which is equivalent to the rated core thermal power of 4,590 MWt plus 0.48% calorimetric uncertainty. Core inventories are bounding for U-235 fuel enrichments ranging between two and five percent and burnups up to 62,000 MWd/MTU. As an exception, the radionuclide inventory for the Fuel Handling Accident (FHA) and Rod Ejection Accident (REA) is derived using a fuel enrichment of 5 wt% in U-235 and burnup steps ranging between approximately 5 and 41 GWD/MTU and is referred to as the "maximized" inventory.

For each of the accident scenarios listed in Table 7.1-1, it is postulated that some quantity of radioactivity is released at the accident location inside a plant building and eventually released into the environment. Radiological consequences of these accidents depend on the type and amount of radioactivity released and meteorological conditions. Potential consequences are assessed to demonstrate that environmental impacts, quantified in doses to individuals at the exclusion area boundary (EAB) distance of 0.33 mi (0.53 km) (measured from the centerline of the Reactor Building and encompassing the analytical distance of 0.31 mi (0.50 km) from all release points) and the low population zone (LPZ) distance of 1.5 mi (2.4km), meet regulatory dose acceptance criteria.

The accident doses are expressed as total effective dose equivalent (TEDE). For each applicable DBA, TEDE/accident doses are calculated based on time-dependent activities released to the environment. Dose receptor variables include the exposure interval, the atmospheric dispersion of the activity during transport from the release point to the EAB and LPZ, the breathing rate of an individual at the EAB and LPZ, and dose conversion factors for the inhalation and external exposure pathways. In accordance with Section C.4.1.5 of Regulatory Guide 1.183 (NRC, 2000), the period of most adverse release of radioactive materials to the environment was assumed to occur coincident with the period of most unfavorable atmospheric dispersion. Except for atmospheric dispersion and departures from the U.S. EPR design methodology for the following DBAs: Small Line Break (SLB), Locked Rotor Accident (LRA), REA and FHA, the other variables are independent of the Bell Bend Nuclear Power Plant (BBNPP) site and specific to the U.S. EPR design.

BBNPP site-specific atmospheric dispersion characteristics are provided in Section 2.7. Table 7.1-5 contains the 50th percentile BBNPP site-specific X/Q values for the EAB and LPZ. For the EAB, the postulated DBA doses and X/Q values are calculated for a 2 hour interval. For the LPZ, doses and X/Q values are calculated for the accident duration (up to 30 days).

For the DBAs applicable to the U.S. EPR, the time-dependent releases to the atmosphere are presented in Table 7.1-14 to Table 7.1-25. The time-dependent postulated TEDE doses at the BBNPP EAB and LPZ are provided in Table 7.1-6 through Table 7.1-12 and an overall summary is presented in Table 7.1-13. All doses are below the dose acceptance criteria.

7.1.1 References

ANS, 1999. Radioactive Source Term for Normal Operation for Light Water Reactors, ANSI/ ANS-18.1, American National Standards Institute/American Nuclear Society, 1999.

NRC, 1999. Environmental Standard Review Plan, NUREG-1555, Nuclear Regulatory Commission, October 1999.

NRC, 2000. Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183, Nuclear Regulatory Commission, July 2000.

NRC, 2007. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Nuclear Regulatory Commission, March 2007.

Table 7.1-1— Design Basis Accidents

(Page 1 of 2)

NUREG-1555 DBA Description	U.S. EPR DBA Description	Remarks
Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	Steam System Piping Failures	Following the guidance provided in Section 15.0.3 of NUREG-0800 and Regulatory Guide 1.183, the limiting accident for the U.S. EPR Steam System Piping Failures was determined to be a double-ended guillotine break of a main steam line in one of the Safeguards Buildings.
Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	Not Applicable	For the U.S. EPR, the consequences of the steam system piping failures were determined to bound those for the feedwater system line breaks.
Reactor Coolant Pump Rotor Seizure	Reactor Coolant Pump Locked Rotor Accident / Broken Shaft	For the U.S. EPR, this postulated accident scenario is based on Section 15.0.3 of NUREG-0800 and the Alternate Source Term Methodology in Appendix G of Regulatory Guide 1.183.
Reactor Coolant Pump Shaft Break		
Radiological Consequences of Control Rod Drop Accident (BWR)	Not Applicable	The U.S. EPR is a pressurized water reactor. Refer to Section 3.2.
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Failure of Small Lines Carrying Primary Coolant Outside Containment	Based on the guidance in Section 15.6.2 of NUREG-0800, the limiting accident scenario for the U.S. EPR Failure of Small Lines Carrying Primary Coolant Outside Containment was determined to be a double-ended guillotine break in the Fuel Building of a sampling line connected to the RCS cross-over leg #3.
Radiological Consequences of Steam Generator Tube Failures (PWR)	Steam Generator Tube Rupture	The analysis was based on guidance in Section 15.0.3 of NUREG-0800 and in Regulatory Guide 1.183 and incorporated the clarifications provided in NRC Regulatory Issue Summary 2006-04, Section 9, namely, the inclusion of the alkalis (in addition to the halogens and noble gases). Two alternative accident scenarios were postulated: A Steam Generator Tube Rupture (SGTR) with a pre-accident iodine spike and a SGTR with a concurrent iodine spike.
Radiological Consequences of a Design Basis Loss of Coolant Accident Including Containment Leakage Contribution	Loss of Coolant Accidents resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	For the U.S. EPR, a radiological analysis was performed to determine the EAB and LPZ cloud immersion and inhalation doses at the EAB and LPZ for Loss of Coolant Accidents using the Alternate Source Term Methodology.
Radiological Consequences of a Design Basis Loss of Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment	Loss of Coolant Accidents resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	For the U.S. EPR, a radiological analysis was performed to determine the EAB and LPZ cloud immersion and inhalation doses at the EAB and LPZ for Loss of Coolant Accidents using the Alternate Source Term Methodology.

Table 7.1-1— Design Basis Accidents
(Page 2 of 2)

NUREG-1555 DBA Description	U.S. EPR DBA Description	Remarks
Radiological Consequences of a Design Basis Loss of Coolant Accident: Leakage from Main Steam Isolation Valve Leakage Control System (BWR)	Not Applicable	The U.S. EPR is a pressurized water reactor. Refer to Section 3.2.
Radiological Consequences of Fuel Handling Accidents	Fuel Handling Accident	For the U.S. EPR, the postulated accident scenario followed the guidance in Section 15.0.3 of NUREG-0800 and in Regulatory Guide 1.183, and was postulated to occur in either an open Containment or in the Fuel Building following a 34-hour decay.
Not Applicable	Rod Ejection Accident	For the U.S. EPR, the analysis was based on the guidance in Section 15.0.3 of NUREG-0800 and in Regulatory Guide 1.183. The recent NRC concern regarding the fission-product gap inventory for reactivity-induced accidents and the interim acceptance criteria and guidance, were also considered.

Table 7.1-2— U.S. EPR Design Basis Primary Coolant Activity
Notes (a)(b)
(Page 1 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)
Noble Gases	
Kr-83m	1.28E-01 (4.74E+03)
Kr-85m	5.71E-01 (2.11E+04)
Kr-85	5.31E+00 (1.96E+05)
Kr-87	3.26E-01 (1.21E+04)
Kr-88	1.03 E+00 (3.81E+04)
Kr-89	2.42E-02 (8.95E+02)
Xe-131m	1.08E+00 (4.00E+04)
Xe-133m	1.35E+00 (5.00E+04)
Xe-133	9.47E+01 (3.50E+06)
Xe-135m	1.95E-01 (7.22E+03)
Xe-135	3.40E+00 (1.26E+05)
Xe-137	4.57E-02 (1.69E+03)
Xe-138	1.64E-01 (6.07E+03)
Halogens	
Br-83	3.16E-02 (1.17E+03)
Br-84	1.67E-02 (6.18E+02)
Br-85	2.01E-03 (7.44E+01)
I-129	4.59E-08 (1.70E-03)
I-130	4.97E-02 (1.84E+03)
I-131	7.43E-01 (2.75E+04)
I-132	3.71E-01 (1.37E+04)
I-133	1.25E+00 (4.63E+04)
I-134	2.40E-01 (8.88E+03)
I-135	7.90E-01 (2.92E+04)
Alkalis	
Rb-86m	5.32E-07 (1.97E-02)
Rb-86	3.66E-03 (1.35E+02)
Rb-88	1.02E+00 (3.77E+04)
Rb-89	4.72E-02 (1.75E+03)
Cs-134	4.18E-01 (1.55E+04)
Cs-136	1.00E-01 (3.70E+03)
Cs-137	1.60E-01 (5.92E+03)
Cs-138	2.35E-01 (8.70E+03)
Tellurium Group	
Sb-125	1.56E-06 (5.77E-02)
Sb-127	6.99E-06 (2.59E-01)
Sb-129	8.53E-06 (3.16E-01)
Te-127m	6.19E-04 (2.29E+01)
Te-127	3.05E-03 (1.13E+02)
Te-129m	1.79E-03 (6.62E+01)
Te-129	3.00E-03 (1.11E+02)
Te-131m	4.36E-03 (1.61E+02)
Te-131	3.01E-03 (1.11E+02)
Te-132	4.70E-02 (1.74E+03)

Table 7.1-2— U.S. EPR Design Basis Primary Coolant Activity
Notes (a)(b)
(Page 2 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)
Te-134	6.80E-03 (2.52E+02)
Barium/Strontium Group	
Sr-89	6.35E-04 (2.35E+01)
Sr-90	4.32E-05 (1.60E+00)
Sr-91	1.02E-03 (3.77E+01)
Sr-92	1.73E-04 (6.40E+00)
Ba-137m	1.50E-01 (5.55E+03)
Ba-139	2.30E-02 (8.51E+02)
Ba-140	6.74E-04 (2.49E+01)
Noble Metals	
Mo-99	1.21E-01 (4.48E+03)
Tc-99m	5.24E-02 (1.94E+03)
Ru-103	1.00E-04 (3.70E+00)
Ru-105	1.47E-04 (5.44E+00)
Ru-106	5.83E-05 (2.16E+00)
Rh-103m	8.85E-05 (3.27E+00)
Rh-105	6.62E-05 (2.45E+00)
Rh-106	5.84E-05 (2.16E+00)
Cerium Group	
Ce-141	9.12E-05 (3.37E+00)
Ce-143	7.96E-05 (2.95E+00)
Ce-144	6.93E-05 (2.56E+00)
Pu-238	5.97E-07 (2.21E-02)
Pu-239	2.51E-08 (9.29E-04)
Pu-240	5.72E-08 (2.12E-03)
Pu-241	1.03E-05 (3.81E-01)
Np-239	1.41E-03 (5.22E+01)
Lanthanides	
Y-90	1.03E-05 (3.81E-01)
Y-91m	5.23E-04 (1.94E+01)
Y-91	8.10E-05 (3.00E+00)
Y-92	1.41E-04 (5.22E+00)
Y-93	6.50E-05 (2.41E+00)
Zr-95	9.31E-05 (3.44E+00)
Zr-97	7.37E-05 (2.73E+00)
Nb-95	9.35E-05 (3.46E+00)
Ag-110m	9.87E-07 (3.65E-02)
Ag-110	4.72E-08 (1.75E-03)
La-140	1.76E-04 (6.51E+00)
La-141	5.77E-05 (2.13E+00)
La-142	3.38E-05 (1.25E+00)
Pr-143	9.20E-05 (3.40E+00)
Pr-144	6.94E-05 (2.57E+00)
Nd-147	3.77E-05 (1.39E+00)
Am-241	1.18E-08 (4.37E-04)

Table 7.1-2— U.S. EPR Design Basis Primary Coolant Activity
Notes (a)(b)
(Page 3 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)
Cm--242	5.35E-06 (1.98E-01)
Cm--244	2.83E-06 (1.05E-01)
Activation Products	
Na-24	3.7E-02 (1.37E+03)
Cr-51	2.0E-03 (7.40E+01)
Mn-54	1.0E-03 (3.70E+01)
Fe-55	7.6E-04 (2.81E+01)
Fe-59	1.9E-04 (7.03E+00)
Co-58	2.9E-03 (1.07E+02)
Co-60	3.4E-04 (1.26E+01)
Zn-65	3.2E-04 (1.18E+01)
W-187	1.8E-03 (6.66E+01)
Tritium	
H-3	1.0E+00 (3.70E+04)

Key:
µCi/gm - microcuries per gram
Bq/gm - Becquerels per gram
Notes:

a. This table lists the design basis source term activity and the magnitude of source terms for offsite releases for the U.S. EPR primary coolant.

b. Following an accident, iodine spiking causes the concentration of radioactive iodines I-131 through I-135 to significantly increase.

Table 7.1-3— U.S. EPR Design Basis Secondary Coolant Activity

Notes (a)(b)
(Page 1 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)
Halogens	
Br-83	1.61E-03 (5.96E+01)
Br-84	3.05E-04 (1.13E+01)
Br-85	3.93E-06 (1.45E-01)
I-129	4.81E-09 (1.78E-04)
I-130	4.33E-03 (1.60E+02)
I-131	7.67E-02 (2.84E+03)
I-132	2.27E-02 (8.40E+02)
I-133	1.17E-01 (4.33E+03)
I-134	6.68E-03 (2.47E+02)
I-135	5.99E-02 (2.22E+03)
Alkalis	
Rb-86m	3.99E-12 (1.48E-07)
Rb-86	7.27E-06 (2.69E-01)
Rb-88	1.26E-04 (4.66E+00)
Rb-89	5.02E-06 (1.86E-01)
Cs-134	8.38E-04 (3.10E+01)
Cs-136	1.98E-04 (7.33E+00)
Cs-137	3.21E-04 (1.19E+01)
Cs-138	5.00E-05 (1.85E+00)
Tellurium Group	
Sb-125	1.74E-09 (6.44E-05)
Sb-127	7.60E-09 (2.81E-04)
Sb-129	6.01E-09 (2.22E-04)
Te-127m	6.89E-07 (2.55E-02)
Te-127	2.82E-06 (1.04E-01)
Te-129m	1.99E-06 (7.36E-02)
Te-129	1.94E-06 (7.18E-02)
Te-131m	4.48E-06 (1.66E-01)
Te-131	1.33E-06 (4.92E-02)
Te-132	5.07E-05 (1.88E+00)
Te-134	1.64E-06 (6.07E-02)
Barium/Strontium Group	
Sr-89	7.16E-07 (2.65E-02)
Sr-90	4.81E-08 (1.78E-03)
Sr-91	9.01E-07 (3.33E-02)
Sr-92	1.00E-07 (3.70E-03)
Ba-137m	3.01E-04 (1.11E+01)
Ba-139	1.03E-05 (3.81E-01)
Ba-140	7.45E-07 (2.76E-02)
Noble Metals	
Mo-99	1.30E-04 (4.81E+00)
Tc-99m	7.47E-05 (2.76E+00)
Ru-103	1.11E-07 (4.11E-03)
Ru-105	1.09E-07 (4.03E-03)

Table 7.1-3— U.S. EPR Design Basis Secondary Coolant Activity

Notes (a)(b)
(Page 2 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)
Ru-106	6.49E-08 (2.40E-03)
Rh-103m	9.97E-08 (3.69E-03)
Rh-105	7.58E-08 (2.80E-03)
Rh-106	6.49E-08 (2.40E-03)
Cerium Group	
Ce-141	1.01E-07 (3.74E-03)
Ce-143	8.24E-08 (3.05E-03)
Ce-144	7.72E-08 (2.86E-03)
Pu-238	6.65E-10 (2.46E-05)
Pu-239	2.80E-11 (1.04E-06)
Pu-240	6.37E-11 (2.36E-06)
Pu-241	1.15E-08 (4.26E-04)
Np-239	1.50E-06 (5.55E-02)
Lanthanides	
Y-90	1.29E-08 (4.77E-04)
Y-91m	5.38E-07 (1.99E-02)
Y-91	9.17E-08 (3.39E-03)
Y-92	1.33E-07 (4.92E-03)
Y-93	5.81E-08 (2.15E-03)
Zr-95	1.04E-07 (3.85E-03)
Zr-97	7.15E-08 (2.65E-03)
Nb-95	1.04E-07 (3.85E-03)
Ag-110m	1.10E-09 (4.07E-05)
Ag-110	1.47E-11 (5.44E-07)
La-140	2.28E-07 (8.44E-03)
La-141	4.06E-08 (1.50E-03)
La-142	1.51E-08 (5.59E-04)
Pr-143	1.02E-07 (3.77E-03)
Pr-144	7.72E-08 (2.86E-03)
Nd-147	4.16E-08 (1.54E-03)
Am-241	1.32E-11 (4.88E-07)
Cm-242	5.96E-09 (2.21E-04)
Cm-244	3.15E-09 (1.17E-04)
Activation Products	
Na-24	3.53E-05 (1.31E+00)
Cr-51	2.22E-06 (8.21E-02)
Mn-54	1.11E-06 (4.11E-02)
Fe-55	8.47E-07 (3.13E-02)
Fe-59	2.11E-07 (7.81E-03)
Co-58	3.23E-06 (1.20E-01)
Co-60	3.79E-07 (1.40E-02)
Zn-65	3.56E-07 (1.32E-02)
W-187	1.81E-06 (6.70E-02)
Tritium	
H-3	1.0E-03 (3.70E+01)

Table 7.1-3— U.S. EPR Design Basis Secondary Coolant Activity

Notes (a)(b)

(Page 3 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)

Key:
µCi/gm - microcuries per gram
Bq/gm - Becquerels per gram

Notes:

a. This table lists the design basis source term activity and the magnitude of source terms for offsite releases for the U.S. EPR secondary coolant.

b. Noble gases are not applicable since they are assumed to enter the steam phase.

Table 7.1-4— U.S. EPR Core Inventory[1,2,3]

(Page 1 of 3)

Radionuclide	Inventory Ci (Bq)	
	Bounding	Maximized for FHA and REA
Noble Gases		
Kr-83m	1.96E+07 (7.25E+17)	1.96E+07 (7.25E+17)
Kr-85m	4.50E+07 (1.67E+18)	4.50E+07 (1.67E+18)
Kr-85	2.10E+06 (7.77E+16)	1.60E+06 (5.92E+16)
Kr-87	9.02E+07 (3.34E+18)	9.02E+07 (3.34E+18)
Kr-88	1.28E+08 (4.74E+18)	1.28E+08 (4.74E+18)
Kr-89	1.61E+08 (5.96E+18)	1.61E+08 (5.96E+18)
Xe-131m	1.54E+06 (5.70E+16)	1.35E+06 (5.00E+16)
Xe-133m	8.92E+06 (3.30E+17)	7.72E+06 (2.86E+17)
Xe-133	2.89E+08 (1.07E+19)	2.55E+08 (9.44E+18)
Xe-135m	5.49E+07 (2.03E+18)	4.84E+07 (1.79E+18)
Xe-135	9.26E+07 (3.43E+18)	9.26E+07 (3.43E+18)
Xe-137	2.52E+08 (9.32E+18)	2.25E+08 (8.33E+18)
Xe-138	2.45E+08 (9.07E+18)	2.30E+08 (8.51E+18)
Halogens		
Br-83	1.96E+07 (7.25E+17)	1.96E+07 (7.25E+17)
Br-84	3.62E+07 (1.34E+18)	3.62E+07 (1.34E+18)
Br-85	4.45E+07 (1.65E+18)	4.45E+07 (1.65E+18)
I-129	8.33E+00 (3.08E+11)	4.75E+00 (1.76E+11)
I-130	1.32E+07 (4.88E+17)	4.64E+06 (1.72E+17)
I-131	1.39E+08 (5.14E+18)	1.21E+08 (4.48E+18)
I-132	2.01E+08 (7.44E+18)	1.75E+08 (6.48E+18)
I-133	2.90E+08 (1.07E+19)	2.55E+08 (9.44E+18)
I-134	3.18E+08 (1.18E+19)	2.86E+08 (1.06E+19)
I-135	2.69E+08 (9.95E+18)	2.38E+08 (8.81E+18)
Alkalis		
Rb-86m	5.53E+04 (2.05E+15)	3.11E+04 (1.15E+15)
Rb-86	5.80E+05 (2.15E+16)	3.07E+05 (1.14E+16)
Rb-88	1.29E+08 (4.77E+18)	1.29E+08 (4.77E+18)
Rb-89	1.67E+08 (6.18E+18)	1.67E+08 (6.18E+18)
Cs-134	6.48E+07 (2.40E+18)	2.65E+07 (9.81E+17)
Cs-136	1.61E+07 (5.96E+17)	8.50E+06 (3.15E+17)
Cs-137	2.47E+07 (9.14E+17)	1.67E+07 (6.18E+17)
Cs-138	2.69E+08 (9.95E+18)	2.50E+08 (9.25E+18)
Tellurium Group		
Sb-125	3.83E+06 (1.42E+17)	NA
Sb-127	1.80E+07 (6.66E+17)	NA
Sb-129	4.85E+07 (1.79+18)	NA
Te-127m	2.43E+06 (8.99E+16)	NA
Te-127	1.79E+07 (6.62E+17)	NA
Te-129m	7.08E+06 (2.62E+17)	NA
Te-129	4.78E+07 (1.77E+18)	NA
Te-131m	2.04E+07 (7.55E+17)	NA
Te-131	1.24E+08 (4.59E+18)	NA
Te-132	1.98E+08 (7.33E+18)	NA
Te-134	2.50E+08 (9.25E+18)	NA

Table 7.1-4— U.S. EPR Core Inventory[1,2,3]
(Page 2 of 3)

Radionuclide	Inventory Ci (Bq)	
	Bounding	Maximized for FHA and REA
Barium/Strontium Group		
Sr-89	1.61E+08 (5.96E+18)	NA
Sr-90	1.69E+07 (6.25E+17)	NA
Sr-91	2.07E+08 (7.66E+18)	NA
Sr-92	2.14E+08 (7.92E+18)	NA
Ba-137m	2.34E+07 (8.66E+17)	NA
Ba-139	2.62E+08 (9.69E+18)	NA
Ba-140	2.52E+08 (9.32E+18)	NA
Noble Metals		
Mo-99	2.59E+08 (9.58E+18)	NA
Tc-99m	2.27E+08 (8.40E+18)	NA
Ru-103	2.42E+08 (8.95E+18)	NA
Ru-105	1.96E+08 (7.25E+18)	NA
Ru-106	1.43E+08 (5.29E+18)	NA
Rh-103m	2.18E+08 (8.07E+18)	NA
Rh-105	1.75E+08 (6.48E+18)	NA
Rh-106	1.58E+08 (5.85E+18)	NA
Cerium Group		
Ce-141	2.24E+08 (8.29E+18)	NA
Ce-143	2.28E+08 (8.44E+18)	NA
Ce-144	1.70E+08 (6.29E+18)	NA
Pu-238	1.46E+06 (5.40E+16)	NA
Pu-239	6.14E+04 (2.27E+15)	NA
Pu-240	1.40E+05 (5.18E+15)	NA
Pu-241	2.53E+07 (9.36E+17)	NA
Np-239	3.82E+09 (1.41E+20)	NA
Lanthanides		
Y-90	1.79E+07 (6.62E+17)	NA
Y-91m	1.20E+08 (4.44E+18)	NA
Y-91	1.96E+08 (7.25E+18)	NA
Y-92	2.14E+08 (7.92E+18)	NA
Y-93	2.34E+08 (8.66E+18)	NA
Zr-95	2.29E+08 (8.47E+18)	NA
Zr-97	2.43E+08 (8.99E+18)	NA
Nb-95	2.29E+08 (8.47E+18)	NA
Ag-110m	2.42E+06 (8.95E+16)	NA
Ag-110	7.15E+07(2.65E+18)	NA
La-140	2.54E+08 (9.40E+18)	NA
La-141	2.41E+08 (8.92E+18)	NA
La-142	2.35E+08 (8.70E+18)	NA
Pr-143	2.26E+08 (8.36E+18)	NA
Pr-144	1.72E+08 (6.35E+18)	NA
Nd-147	9.44E+07 (3.49E+18)	NA
Am-241	2.88E+04 (1.07E+15)	NA
Cm-242	1.31E+07 (4.85E+17)	NA
Cm-244	6.94E+06 (2.57E+17)	NA

Table 7.1-4— U.S. EPR Core Inventory[1,2,3]
(Page 3 of 3)

Radionuclide	Inventory Ci (Bq)	
	Bounding	**Maximized for FHA and REA**

Key:
Ci - curies
Bq - Bequerels
NA - not applicable
Notes:

1. Core inventories are bounding for U-235 fuel enrichments ranging between two and five percent and burnups up to 62,000 megawatt days per metric ton uranium (MWd/MTU) [Ref. 7.1.9 (Section 2.0 (a))]
2. As an exception, the radionuclide inventory for the REA and FHA is derived using a fuel enrichment of 5 wt% in U-235 and burnup steps ranging between approximately 5 and 41 GWD/MTU, referred to as the "maximized" inventory.[Ref. 7.1.11]
3. The design basis power level is 4,612 MWt [Ref. 7.1.9 (Section 2.0 (a))]

Table 7.1-5— 50th Percentile BBNPP Site Atmospheric Dispersion Factors

Time Interval (hrs)	Atmospheric Disperson Factor (sec/m^3) (Nominal, 50% Meteorology)	
	EAB (Worst 2-hr)	**LPZ (0 to 30 days)**
Loss of Coolant Accident (LOCA)		
0 to 1.5	n/a	1.932E-05
1.5 to 3.5[(a)]	1.437E-04	2.347E-05
3.5 to 8	n/a	1.932E-05
8 to 24		1.624E-05
24 to 96		1.244E-05
96 to 720		8.485E-06
All Other Accidents		
0 to 2	1.437E-04	2.347E-05
2 to 8	n/a	1.932E-05
8 to 24		1.624E-05
24 to 96		1.244E-05
96 to 720		8.485E-06

Note:
(a) In accordance with Regulatory Guide 1.183 (Section 4.1.5), the period of most adverse release of radioactive materials to the environment was assumed to occur coincident with the period of most unfavorable atmospheric dispersion. The critical receptor is in the W sector, at 0.33 miles from the reactor building center line.

Table 7.1-6— Steam System Piping Failure

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
Pre-Accident Iodine Spike		
0-2 hr	3.497E-02/ 3.497E-04	5.712E-03/5.712E-05
2-8 hr		2.586E-03/2.586E-05
8-24 hr		1.573E-04/1.573E-06
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	3.497E-02/ 3.497E-04	8.455E-03/8.455E-05
Limit	25/2.5E-01	25/2.5E-01
Accident-Initiated Iodine Spike		
0-2 hr	3.870E-02/ 3.870E-04	6.320E-03/6.320E-05
2-8 hr		2.053E-02/2.053E-04
8-24 hr		2.247E-03/2.247E-05
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	3.870E-02/ 3.870E-04	2.910E-02/2.910E-04
Limit	2.5/2.5E-02	2.5/2.5E-02
Accident-Induced 3.3% Fuel Rod Clad Failure		
0-2 hr	7.580E-01/ 7.580E-03	1.238E-01/1.238E-03
2-8 hr		2.303E-01/2.303E-03
8-24 hr		1.490E-02/1.490E-04
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	7.580E-01/ 7.580E-03	3.690E-01/3.690E-03
Limit	25/2.5E-01	25/2.5E-01
Accident-Induced 0.58% Fuel Overheat		
0-2 hr	8.395E-01/ 8.395E-03	1.371E-01/1.371E-03
2-8 hr		2.359E-0l/2.359E-03
8-24 hr		1.464E-02/1.464E-04
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	8.395E-01/ 8.395E-03	3.876E-01/3.876E-03
Limit	25/2.5E-01	25/2.5E-01
Accident-Induced 1.24% Fuel Rod Clad Failure		
0-2 hr	3.021E-01 / 3.021E-03	4.934E-02 / 4.934E-04
2-8 hr		8.670E-02 / 8.670E-04
8-24 hr		5.603E-03 / 5.603E-05
24-96 hr		0.00E+00
96-720 hr		0.00E+00
Total	3.021E-01 / 3.021E-03	1.416E-01 / 1.416E-01
Limit	25 / 2.5E-01	25 / 2.5E-01

Table 7.1-7— Reactor Coolant Pump Locked Rotor Accident / Broken Shaft with 8% Fuel Rod Clad Failure

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
0-2 hr	2.307E-01/ 2.307E-03	3.768E-02 / 3.768E-04
2-8 hr		4.739E-02 / 4.739E-04
8-24 hr		0.00E+00/0.00E+00
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	2.307E-01/ 2.307E-03	8.507E-02 / 8.507E-04
Limit	2.5/0.025	2.5/0.025

Table 7.1-8— Failure of Small Lines Carrying Primary Coolant Outside Containment[1]

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
0-2 hr	8.633E-02 / 8.633E-04	1.410E-02/1.410E-04
2-8 hr		0.00E+00/0.00E+00
8-24 hr		0.00E+00/0.00E+00
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	8.633E-02 / 8.633E-04	1.410E-02/1.410E-04
Limit	2.5/0.025	2.5/0.025

[1]The Nuclear Sampling System Line Break (1/4-inch line) bounds the Chemical and Volume Control System Line Break (6-inch line) for the EAB and LPZ

Table 7.1-9— Steam Generator Tube Rupture

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
Pre-Accident Iodine Spike		
0-2 hr	1.588E-01 / 1.588E-03	2.593E-02/2.593E-04
2-8 hr		5.297E-03/5.297E-05
8-24 hr		4.274E-03/4.274E-05
24-96 hr		2.191E-03/2.191E-05
96-720 hr		1.078E-02/ 1.078E-04
Total	1.588E-01 / 1.588E-03	4.847E-02/4.847E-04
Limit	25/0.25	25/0.25
Concurrent Iodine Spike		
0-2 hr	1.049E-01/ 1.049E-03	1.714E-02/1.714E-04
2-8 hr		3.837E-03/3.837E-05
8-24 hr		4.895E-03/4.895E-05
24-96 hr		l.341E-02/1.341E-04
96-720 hr		9.811E-02/9.811E-04
Total	1.049E-01/ 1.049E-03	l.374E-01/1.374E-03
Limit	2.5/0.025	2.5/0.025

Table 7.1-10— Loss of Coolant Accident

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
0-1.5 hr		7.388E-02/7.388E-04
1.5-3.5	1.794E+00 / 1.794E-02	2.930E-01/2.930E-03
3.5-8 hr		3.901E-01/3.901E-03
8-24 hr		5.413E-01 /5413E-03
24-96 hr		3.765E-01 /3.765E-03
96-720 hr		3.342E-01/ 3.342E-03
Total	1.794E+00 / 1.794E-02	2.009E+00/2.009E-02
Limit	25/0.25	25/0.25

Table 7.1-11— Fuel Handling Accident
[Maximized source term and 72-hour decay]

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
0-2 hr	5.136E-01 / 5.136E-03	8.388E-02 /8.388E-04
2-8 hr		5.469E-04/ 5.469E-06
8-24 hr		8.662E-05 / 8.662E-07
24-96 hr		5.041E-05 /5.041E-07
96-720 hr		5.844E-06/ 5.844E-08
Total	5.136E-01 / 5.136E-03	8.457E-02 / 8.457E-04
Limit	6.3/0.063	6.3/0.063

Table 7.1-12— Rod Ejection Accident

Time	Site TEDE Dose (rem/Sieverts)	
	EAB	LPZ
0-2 hr	5.360E-01 / 5.360E-03	8.755E-02 / 8.755E-04
2-8 hr		1.946E-01 / 1.946E-03
8-24 hr		0.00E+00/0.00E+00
24-96 hr		0.00E+00/0.00E+00
96-720 hr		0.00E+00/0.00E+00
Total	5.360E-01 / 5.360E-03	2.822E-01 / 2.822E-03
Limit	6.3/6.30E-02	6.3/6.30E-02

Table 7.1-13— Summary of the DBA Radiological Consequences at Offsite Receptors from BBNPP

Design Basis Accident		EAB TEDE Dose (rem / Sieverts)	LPZ TEDE Dose (rem / Sieverts)	Regulatory TEDE Dose Acceptance Criteria (rem / Sieverts)
Steam System Piping Failures	Pre-accident iodine spike	3.5E-02/3.5E-04	8.5E-03/8.5E-05	25 / 0.25
	Concurrent iodine spike	3.9E-02/3.9E-04	2.9E-02/2.9E-04	2.5 / 0.025
	3.3% clad failure	7.6E-01/7.6E-03	3.7E-01 / 3.7E-03	25 / 0.25
	0.58% fuel overheat	8.4E-01/8.4E-03	3.9E-01/3.9E-03	25 / 0.25
	1.24% clad failure	3.0E-01/3.0E-03	1.4E-01/1.4E-03	25 / 0.25
Reactor Coolant Pump Locked Rotor Accident / Broken Shaft (8% clad failure)		2.3E-01/2.3E-03	8.5E-02/8.5E-04	2.5 / 0.025
Failure of Small Lines Carrying Primary Coolant Outside Containment		8.6E-02/8.6E-04	1.4E-02/1.4E-04	2.5 / 0.025
Steam Generator Tube Rupture	Pre-accident Iodine Spike	1.6E-01/1.6E-03	4.8E-02/4.8E-04	25 / 0.25
	Concurrent Iodine Spike	1.0E-01/1.0E-03	1.4E-01/1.4E-03	2.5 / 0.025
Loss Of Coolant Accident		1.8E+00/1.8E-02	2.0E+00/2.0E-02	25 / 0.25
Fuel Handling Accident (72-hr decay)		5.1E-01/5.1E-03	8.5E-02/8.5E-04	6.3 / 0.063
Rod Ejection Accident (26% clad failure)		5.4E-01/5.4E-03	2.8E-01/2.8E-03	6.3 / 0.063

Key:
EAB – Exclusion area boundary
LPZ – Low population zone
TEDE – Total effective dose equivalent

Table 7.1-14— Radionuclide Releases to Atmosphere for SteamSystem Piping Failure with Pre-Accident Iodine Spike

(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	2.167E-02	8.018E+08	2.145E-02	7.937E+08	3.182E-04	1.177E+07	4.344E-02	1.607E+09
Kr-85m	1.115E-01	4.126E+09	1.858E-01	6.875E+09	4.350E-03	1.610E+08	3.016E-01	1.116E+10
Kr-85	1.205E+00	4.459E+10	3.613E+00	1.337E+11	1.505E-01	5.569E+09	4.969E+00	1.839E+11
Kr-87	4.505E-02	1.667E+09	2.194E-02	8.118E+08	9.099E-05	3.367E+06	6.709E-02	2.482E+09
Kr-88	1.849E-01	6.841E+09	2.258E-01	8.355E+09	3.674E-03	1.359E+08	4.144E-01	1.533E+10
Kr-89	2.093E-04	7.744E+06	8.419E-16	3.115E-05	1.370E-50	5.069E-40	2.093E-04	7.744E+06
Xe-131m	2.446E-01	9.050E+09	7.271E-01	2.690E+10	3.027E-02	1.120E+09	1.002E+00	3.707E+10
Xe-133m	3.042E-01	1.126E+10	8.850E-01	3.275E+10	3.985E-02	1.474E+09	1.229E+00	4.547E+10
Xe-133	2.140E+01	7.918E+11	6.307E+01	2.334E+12	2.646E+00	9.790E+10	8.711E+01	3.223E+12
Xe-135m	3.843E-01	1.422E+10	8.821E-01	3.264E+10	8.834E-02	3.269E+09	1.355E+00	5.014E+10
Xe-135	9.137E-01	3.381E+10	3.733E+00	1.381E+11	4.540E-01	1.680E+10	5.100E+00	1.887E+11
Xe-137	4.777E-04	1.767E+07	1.767E-13	6.538E-03	2.237E-42	8.277E-32	4.777E-04	1.767E+07
Xe-138	6.324E-03	2.340E+08	1.790E-05	6.623E+05	9.525E-14	3.524E-03	6.341E-03	2.346E+08
Br-83	2.522E-01	9.331E+09	4.130E-03	1.528E+08	7.641E-05	2.827E+06	2.564E-01	9.487E+09
Br-84	4.771E-02	1.765E+09	4.524E-05	1.674E+06	7.550E-09	2.794E+02	4.775E-02	1.767E+09
Br-85	6.133E-04	2.269E+07	1.092E-18	4.040E-08	1.546E-56	5.720E-46	6.133E-04	2.269E+07
I-129	7.539E-07	2.789E+04	3.757E-08	1.390E+03	1.301E-09	4.814E+01	7.928E-07	2.933E+04
I-130	6.787E-01	2.511E+10	2.685E-02	9.935E+08	8.749E-04	3.237E+07	7.064E-01	2.614E+10
I-131	1.516E+01	5.609E+11	8.621E+00	3.190E+11	1.226E+00	4.536E+10	2.501E+01	9.254E+11
I-132	4.788E+00	1.772E+11	1.069E+00	3.955E+10	4.889E-02	1.809E+09	5.906E+00	2.185E+11
I-133	2.350E+01	8.695E+11	1.244E+01	4.603E+11	1.602E+00	5.927E+10	3.754E+01	1.389E+12
I-134	1.620E+00	5.994E+10	1.135E-01	4.200E+09	5.052E-04	1.869E+07	1.734E+00	6.416E+10
I-135	1.246E+01	4.610E+11	5.510E+00	2.039E+11	5.515E-01	2.041E+10	1.852E+01	6.852E+11
Rb-86m	1.353E-09	5.006E+01	1.255E-45	4.644E-35	0.000E+00	0.000E+00	1.353E-09	5.006E+01
Rb-86	1.398E-03	5.173E+07	7.207E-04	2.667E+07	1.024E-04	3.789E+06	2.221E-03	8.218E+07
Rb-88	1.915E-01	7.086E+09	2.517E-01	9.313E+09	4.103E-03	1.518E+08	4.474E-01	1.655E+10
Rb-89	1.838E-03	6.801E+07	3.266E-06	1.208E+05	1.619E-13	5.990E-03	1.841E-03	6.812E+07
Cs-134	1.609E-01	5.953E+09	8.300E-02	3.071E+09	1.185E-02	4.385E+08	2.557E-01	9.461E+09
Cs-136	3.808E-02	1.409E+09	1.963E-02	7.263E+08	2.782E-03	1.029E+08	6.048E-02	2.238E+09
Cs-137	6.160E-02	2.279E+09	3.177E-02	1.175E+09	4.536E-03	1.678E+08	9.791E-02	3.623E+09

Table 7.1-14— Radionuclide Releases to Atmosphere for SteamSystem Piping Failure with Pre-Accident Iodine Spike

(Page 2 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Cs-138	2.051E-02	7.589E+08	1.254E-03	4.640E+07	1.886E-07	6.978E+03	2.177E-02	8.055E+08
Sr-89	7.189E-07	2.660E+04	2.557E-06	9.461E+04	3.082E-07	1.140E+04	3.584E-06	1.326E+05
Ba-137m	5.786E-02	2.141E+09	3.006E-02	1.112E+09	4.291E-03	1.588E+08	9.220E-02	3.411E+09
Total	8.386E+01	3.103E+12	1.016E+02	3.759E+12	6.875E+00	2.544E+11	1.923E+02	7.115E+12

Table 7.1-15— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced (Coincident) Iodine Spike
(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	2.167E-02	8.018E+08	2.145E-02	7.937E+08	3.182E-04	1.177E+07	4.344E-02	1.607E+09
Kr-85m	1.115E-01	4.126E+09	1.858E-01	6.875E+09	4.350E-03	1.610E+08	3.016E-01	1.116E+10
Kr-85	1.205E+00	4.459E+10	3.613E+00	1.337E+11	1.505E-01	5.569E+09	4.969E+00	1.839E+11
Kr-87	4.505E-02	1.667E+09	2.194E-02	8.118E+08	9.099E-05	3.367E+06	6.709E-02	2.482E+09
Kr-88	1.849E-01	6.841E+09	2.258E-01	8.355E+09	3.674E-03	1.359E+08	4.144E-01	1.533E+10
Kr-89	2.093E-04	7.744E+06	8.419E-16	3.115E-05	1.370E-50	5.069E-40	2.093E-04	7.744E+06
Xe-131m	2.446E-01	9.050E+09	7.308E-01	2.704E+10	3.188E-02	1.180E+09	1.007E+00	3.726E+10
Xe-133m	3.045E-01	1.127E+10	9.837E-01	3.640E+10	8.092E-02	2.994E+09	1.369E+00	5.065E+10
Xe-133	2.140E+01	7.918E+11	6.448E+01	2.386E+12	3.237E+00	1.198E+11	8.912E+01	3.297E+12
Xe-135m	7.205E-01	2.666E+10	1.136E+01	4.203E+11	2.616E+00	9.679E+10	1.470E+01	5.439E+11
Xe-135	1.023E+00	3.785E+10	1.721E+01	6.368E+11	5.434E+00	2.011E+11	2.367E+01	8.758E+11
Xe-137	4.777E-04	1.767E+07	1.767E-13	6.538E-03	2.237E-42	8.277E-32	4.777E-04	1.767E+07
Xe-138	6.324E-03	2.340E+08	1.790E-05	6.623E+05	9.525E-14	3.524E-03	6.341E-03	2.346E+08
Br-83	2.522E-01	9.331E+09	4.130E-03	1.528E+08	7.641E-05	2.827E+06	2.564E-01	9.487E+09
Br-84	4.771E-02	1.765E+09	4.524E-05	1.674E+06	7.550E-09	2.794E+02	4.775E-02	1.767E+09
Br-85	6.133E-04	2.269E+07	1.092E-18	4.040E-08	1.546E-56	5.720E-46	6.133E-04	2.269E+07
I-129	7.539E-07	2.789E+04	3.757E-08	1.390E+03	1.301E-09	4.814E+01	7.928E-07	2.933E+04
I-130	6.787E-01	2.511E+10	2.685E-02	9.935E+08	8.749E-04	3.237E+07	7.064E-01	2.614E+10
I-131	1.627E+01	6.020E+11	6.254E+01	2.314E+12	1.557E+01	5.761E+11	9.438E+01	3.492E+12
I-132	8.145E+00	3.014E+11	3.962E+01	1.466E+12	6.683E+00	2.473E+11	5.445E+01	2.015E+12
I-133	2.653E+01	9.816E+11	1.129E+02	4.177E+12	2.685E+01	9.935E+11	1.663E+02	6.153E+12
I-134	5.642E+00	2.088E+11	2.468E+01	9.132E+11	2.899E+00	1.073E+11	3.322E+01	1.229E+12
I-135	1.595E+01	5.902E+11	7.814E+01	2.891E+12	1.675E+01	6.198E+11	1.108E+02	4.100E+12
Rb-86m	1.353E-09	5.006E+01	1.255E-45	4.644E-35	0.000E+00	0.000E+00	1.353E-09	5.006E+01
Rb-86	1.398E-03	5.173E+07	7.207E-04	2.667E+07	1.024E-04	3.789E+06	2.221E-03	8.218E+07
Rb-88	1.915E-01	7.086E+09	2.517E-01	9.313E+09	4.103E-03	1.518E+08	4.474E-01	1.655E+10
Rb-89	1.838E-03	6.801E+07	3.266E-06	1.208E+05	1.619E-13	5.990E-03	1.841E-03	6.812E+07
Cs-134	1.609E-01	5.953E+09	8.300E-02	3.071E+09	1.185E-02	4.385E+08	2.557E-01	9.461E+09
Cs-136	3.808E-02	1.409E+09	1.963E-02	7.263E+08	2.782E-03	1.029E+08	6.048E-02	2.238E+09

Table 7.1-15— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced (Coincident) Iodine Spike
(Page 2 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Cs-137	6.160E-02	2.279E+09	3.177E-02	1.175E+09	4.536E-03	1.678E+08	9.791E-02	3.623E+09
Cs-138	2.051E-02	7.589E+08	1.254E-03	4.640E+07	1.886E-07	6.978E+03	2.177E-02	8.055E+08
Sr-89	7.189E-07	2.660E+04	2.557E-06	9.461E+04	3.082E-07	1.140E+04	3.584E-06	1.326E+05
Ba-137m	5.786E-02	2.141E+09	3.006E-02	1.112E+09	4.291E-03	1.588E+08	9.220E-02	3.411E+09
Total	9.932E+01	3.675E+12	4.172E+02	1.544E+13	8.034E+01	2.973E+12	5.968E+02	2.208E+13

Table 7.1-16— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 3.3% Clad Failure

(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs) with Accident-Induced 3.3 % Clad Failure							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	3.280E+01	1.214E+12	3.559E+01	1.317E+12	1.238E+00	4.581E+10	6.963E+01	2.576E+12
Kr-85m	8.444E+01	3.124E+12	1.407E+02	5.206E+12	3.320E+00	1.228E+11	2.285E+02	8.455E+12
Kr-85	1.031E+01	3.815E+11	3.093E+01	1.144E+12	1.288E+00	4.766E+10	4.253E+01	1.574E+12
Kr-87	1.192E+02	4.410E+12	5.806E+01	2.148E+12	2.408E-01	8.910E+09	1.775E+02	6.568E+12
Kr-88	2.202E+02	8.147E+12	2.688E+02	9.946E+12	4.376E+00	1.619E+11	4.934E+02	1.826E+13
Kr-89	1.332E+01	4.928E+11	5.359E-11	1.983E+00	8.719E-46	3.226E-35	1.332E+01	4.928E+11
Xe-131m	3.583E+00	1.326E+11	1.068E+01	3.952E+11	4.523E-01	1.674E+10	1.472E+01	5.446E+11
Xe-133m	1.946E+01	7.200E+11	5.604E+01	2.073E+12	2.403E+00	8.891E+10	7.790E+01	2.882E+12
Xe-133	6.466E+02	2.392E+13	1.908E+03	7.060E+13	8.055E+01	2.980E+12	2.635E+03	9.750E+13
Xe-135m	4.150E+01	1.536E+12	4.615E+01	1.708E+12	4.800E+00	1.776E+11	9.245E+01	3.421E+12
Xe-135	1.998E+02	7.393E+12	5.351E+02	1.980E+13	3.532E+01	1.307E+12	7.702E+02	2.850E+13
Xe-137	2.515E+01	9.306E+11	9.302E-09	3.442E+02	1.178E-37	4.359E-27	2.515E+01	9.306E+11
Xe-138	9.017E+01	3.336E+12	2.552E-01	9.442E+09	1.358E-09	5.025E+01	9.042E+01	3.346E+12
Br-83	1.094E+01	4.048E+11	9.155E+00	3.387E+11	4.542E-01	1.681E+10	2.055E+01	7.604E+11
Br-84	1.069E+01	3.955E+11	5.777E-01	2.137E+10	1.566E-04	5.794E+06	1.126E+01	4.166E+11
Br-85	1.663E+00	6.153E+10	2.161E-13	7.996E-03	3.269E-51	1.210E-40	1.663E+00	6.153E+10
I-129	6.476E-06	2.396E+05	1.488E-05	5.506E+05	2.258E-06	8.355E+04	2.362E-05	8.739E+05
I-130	9.312E+00	3.445E+11	1.780E+01	6.586E+11	2.217E+00	8.203E+10	2.933E+01	1.085E+12
I-131	1.643E+02	6.079E+12	3.897E+02	1.442E+13	5.846E+01	2.163E+12	6.125E+02	2.266E+13
I-132	1.121E+02	4.148E+12	8.941E+01	3.308E+12	4.225E+00	1.563E+11	2.057E+02	7.611E+12
I-133	2.124E+02	7.859E+12	4.391E+02	1.625E+13	5.933E+01	2.195E+12	7.109E+02	2.630E+13
I-134	1.242E+02	4.595E+12	2.356E+01	8.717E+11	1.065E-01	3.941E+09	1.479E+02	5.472E+12
I-135	1.789E+02	6.619E+12	2.877E+02	1.064E+13	2.996E+01	1.109E+12	4.966E+02	1.837E+13
Rb-86m	1.764E-03	6.527E+07	2.996E-39	1.109E-28	0.000E+00	0.000E+00	1.764E-03	6.527E+07
Rb-86	9.539E-01	3.529E+10	2.456E+00	9.087E+10	3.714E-01	1.374E+10	3.781E+00	1.399E+11
Rb-88	2.406E+02	8.902E+12	2.999E+02	1.110E+13	4.885E+00	1.807E+11	5.454E+02	2.018E+13
Rb-89	8.269E+01	3.060E+12	2.451E-01	9.069E+09	1.281E-08	4.740E+02	8.293E+01	3.068E+12
Cs-134	1.069E+02	3.955E+12	2.768E+02	1.024E+13	4.209E+01	1.557E+12	4.258E+02	1.575E+13
Cs-136	2.650E+01	9.805E+11	6.805E+01	2.518E+12	1.026E+01	3.796E+11	1.048E+02	3.878E+12
Cs-137	4.081E+01	1.510E+12	1.057E+02	3.911E+12	1.607E+01	5.946E+11	1.626E+02	6.016E+12

Table 7.1-16— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 3.3% Clad Failure

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Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs) with Accident-Induced 3.3 % Clad Failure							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Cs-138	2.696E+02	9.975E+12	2.276E+01	8.421E+11	4.151E-03	1.536E+08	2.923E+02	1.082E+13
Sr-89	5.497E-02	2.034E+09	1.946E-01	7.200E+09	2.451E-02	9.069E+08	2.741E-01	1.014E+10
Ba-137m	3.860E+01	1.428E+12	1.000E+02	3.700E+12	1.520E+01	5.624E+11	1.538E+02	5.691E+12
Total	3.138E+03	1.161E+14	5.224E+03	1.933E+14	3.776E+02	1.397E+13	8.739E+03	3.233E+14

Table 7.1-17— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 0.58% Fuel Overheat

(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs) with Accident-Induced 0.58% Fuel Overheat							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	1.098E+02	4.063E+12	1.038E+02	3.841E+12	2.549E+00	9.431E+10	2.162E+02	7.999E+12
Kr-85m	2.957E+02	1.094E+13	4.928E+02	1.823E+13	1.158E+01	4.285E+11	8.001E+02	2.960E+13
Kr-85	1.721E+01	6.368E+11	5.163E+01	1.910E+12	2.150E+00	7.955E+10	7.099E+01	2.627E+12
Kr-87	4.179E+02	1.546E+13	2.035E+02	7.530E+12	8.440E-01	3.123E+10	6.223E+02	2.303E+13
Kr-88	7.737E+02	2.863E+13	9.445E+02	3.495E+13	1.537E+01	5.687E+11	1.733E+03	6.412E+13
Kr-89	4.684E+01	1.733E+12	1.884E-10	6.971E+00	3.065E-45	1.134E-34	4.684E+01	1.733E+12
Xe-131m	1.197E+01	4.429E+11	3.560E+01	1.317E+12	1.483E+00	5.487E+10	4.905E+01	1.815E+12
Xe-133m	6.769E+01	2.505E+12	1.938E+02	7.171E+12	8.011E+00	2.964E+11	2.695E+02	9.972E+12
Xe-133	2.213E+03	8.188E+13	6.514E+03	2.410E+14	2.708E+02	1.002E+13	8.997E+03	3.329E+14
Xe-135m	1.112E+02	4.114E+12	8.124E+01	3.006E+12	8.435E+00	3.121E+11	2.008E+02	7.430E+12
Xe-135	6.807E+02	2.519E+13	1.677E+03	6.205E+13	8.537E+01	3.159E+12	2.443E+03	9.039E+13
Xe-137	8.839E+01	3.270E+12	3.271E-08	1.210E+03	4.140E-37	1.532E-26	8.839E+01	3.270E+12
Xe-138	3.178E+02	1.176E+13	8.992E-01	3.327E+10	4.786E-09	1.771E+02	3.187E+02	1.179E+13
Br-83	1.904E+01	7.045E+11	1.609E+01	5.953E+11	7.982E-01	2.953E+10	3.592E+01	1.329E+12
Br-84	1.875E+01	6.938E+11	1.015E+00	3.756E+10	2.752E-04	1.018E+07	1.976E+01	7.311E+11
Br-85	2.922E+00	1.081E+11	3.798E-13	1.405E-02	5.745E-51	2.126E-40	2.922E+00	1.081E+11
I-129	1.081E-05	4.000E+05	2.613E-05	9.668E+05	3.967E-06	1.468E+05	4.091E-05	1.514E+06
I-130	1.585E+01	5.865E+11	3.127E+01	1.157E+12	3.897E+00	1.442E+11	5.102E+01	1.888E+12
I-131	1.792E+02	6.630E+12	4.277E+02	1.582E+13	6.411E+01	2.372E+12	6.709E+02	2.482E+13
I-132	1.943E+02	7.189E+12	1.571E+02	5.813E+12	7.425E+00	2.747E+11	3.588E+02	1.328E+13
I-133	3.595E+02	1.330E+13	7.712E+02	2.853E+13	1.043E+02	3.859E+12	1.235E+03	4.570E+13
I-134	2.175E+02	8.048E+12	4.141E+01	1.532E+12	1.872E-01	6.926E+09	2.591E+02	9.587E+12
I-135	3.073E+02	1.137E+13	5.054E+02	1.870E+13	5.265E+01	1.948E+12	8.654E+02	3.202E+13
Rb-86m	1.290E-03	4.773E+07	2.191E-39	8.107E-29	0.000E+00	0.000E+00	1.290E-03	4.773E+07
Rb-86	7.010E-01	2.594E+10	1.804E+00	6.675E+10	2.727E-01	1.009E+10	2.777E+00	1.027E+11
Rb-88	6.770E+02	2.505E+13	1.053E+03	3.896E+13	1.716E+01	6.349E+11	1.747E+03	6.464E+13
Rb-89	9.740E+01	3.604E+12	3.763E-01	1.392E+10	1.278E-08	4.729E+02	9.778E+01	3.618E+12
Cs-134	7.845E+01	2.903E+12	2.031E+02	7.515E+12	3.087E+01	1.142E+12	3.124E+02	1.156E+13
Cs-136	1.947E+01	7.204E+11	4.995E+01	1.848E+12	7.537E+00	2.789E+11	7.696E+01	2.848E+12
Cs-137	2.990E+01	1.106E+12	7.740E+01	2.864E+12	1.177E+01	4.355E+11	1.191E+02	4.407E+12

Table 7.1-17— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 0.58% Fuel Overheat
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Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs) with Accident-Induced 0.58% Fuel Overheat							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Cs-138	4.164E+02	1.541E+13	5.014E+01	1.855E+12	5.701E-03	2.109E+08	4.666E+02	1.726E+13
Sr-89	7.331E-02	2.712E+09	2.692E-01	9.960E+09	2.321E-02	8.588E+08	3.657E-01	1.353E+10
Ba-137m	2.829E+01	1.047E+12	7.327E+01	2.711E+12	1.113E+01	4.118E+11	1.127E+02	4.170E+12
Total	7.814E+03	2.891E+14	1.376E+04	5.091E+14	7.187E+02	2.659E+13	2.229E+04	8.247E+14

Table 7.1-18— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 1.24% Clad Failure
(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	1.234E+01	4.566E+11	1.339E+01	4.954E+11	4.654E-01	1.722E+10	2.619E+01	9.690E+11
Kr-85m	3.180E+01	1.177E+12	5.299E+01	1.961E+12	1.250E+00	4.625E+10	8.604E+01	3.183E+12
Kr-85	4.628E+00	1.712E+11	1.388E+01	5.136E+11	5.781E-01	2.139E+10	1.908E+01	7.060E+11
Kr-87	4.483E+01	1.659E+12	2.183E+01	8.077E+11	9.054E-02	3.350E+09	6.675E+01	2.470E+12
Kr-88	8.286E+01	3.066E+12	1.011E+02	3.741E+12	1.647E+00	6.094E+10	1.857E+02	6.871E+12
Kr-89	5.007E+00	1.853E+11	2.014E-11	7.452E-01	3.276E-46	1.212E-35	5.007E+00	1.853E+11
Xe-131m	1.499E+00	5.546E+10	4.467E+00	1.653E+11	1.887E-01	6.982E+09	6.154E+00	2.277E+11
Xe-133m	7.499E+00	2.775E+11	2.160E+01	7.992E+11	9.244E-01	3.420E+10	3.002E+01	1.111E+12
Xe-133	2.563E+02	9.483E+12	7.562E+02	2.798E+13	3.187E+01	1.179E+12	1.044E+03	3.863E+13
Xe-135m	1.561E+01	5.776E+11	1.737E+01	6.427E+11	1.804E+00	6.675E+10	3.479E+01	1.287E+12
Xe-135	7.553E+01	2.795E+12	2.021E+02	7.478E+12	1.331E+01	4.925E+11	2.910E+02	1.077E+13
Xe-137	9.450E+00	3.497E+11	3.495E-09	1.293E+02	4.426E-38	1.638E-27	9.450E+00	3.497E+11
Xe-138	3.388E+01	1.254E+12	9.589E-02	3.548E+09	5.105E-10	1.889E+01	3.398E+01	1.257E+12
Br-83	4.268E+00	1.579E+11	3.443E+00	1.274E+11	1.707E-01	6.316E+09	7.881E+00	2.916E+11
Br-84	4.045E+00	1.497E+11	2.171E-01	8.033E+09	5.883E-05	2.177E+06	4.262E+00	1.577E+11
Br-85	6.252E-01	2.313E+10	8.120E-14	3.004E-03	1.228E-51	4.544E-41	6.252E-01	2.313E+10
I-129	2.904E-06	1.074E+05	5.616E-06	2.078E+05	8.492E-07	3.142E+04	9.369E-06	3.467E+05
I-130	3.923E+00	1.452E+11	6.706E+00	2.481E+11	8.338E-01	3.085E+10	1.146E+01	4.240E+11
I-131	6.926E+01	2.563E+12	1.468E+02	5.432E+12	2.198E+01	8.133E+11	2.380E+02	8.806E+12
I-132	4.433E+01	1.640E+12	3.363E+01	1.244E+12	1.588E+00	5.876E+10	7.955E+01	2.943E+12
I-133	9.127E+01	3.377E+12	1.655E+02	6.124E+12	2.231E+01	8.255E+11	2.791E+02	1.033E+13
I-134	4.733E+01	1.751E+12	8.856E+00	3.277E+11	4.002E-02	1.481E+09	5.623E+01	2.081E+12
I-135	7.309E+01	2.704E+12	1.083E+02	4.007E+12	1.126E+01	4.166E+11	1.926E+02	7.126E+12
Rb-86m	6.630E-04	2.453E+07	1.126E-39	4.166E-29	0.000E+00	0.000E+00	6.630E-04	2.453E+07
Rb-86	3.593E-01	1.329E+10	9.232E-01	3.416E+10	1.396E-01	5.165E+09	1.422E+00	5.261E+10
Rb-88	9.053E+01	3.350E+12	1.129E+02	4.177E+12	1.838E+00	6.801E+10	2.052E+02	7.592E+12
Rb-89	3.107E+01	1.150E+12	9.210E-02	3.408E+09	4.812E-09	1.780E+02	3.116E+01	1.153E+12
Cs-134	4.026E+01	1.490E+12	1.041E+02	3.852E+12	1.582E+01	5.853E+11	1.602E+02	5.927E+12
Cs-136	9.981E+00	3.693E+11	2.558E+01	9.465E+11	3.858E+00	1.427E+11	3.942E+01	1.459E+12
Cs-137	1.537E+01	5.687E+11	3.975E+01	1.471E+12	6.041E+00	2.235E+11	6.116E+01	2.263E+12

Table 7.1-18— Radionuclide Releases to Atmosphere for Steam System Piping Failure with Accident-Induced 1.24% Clad Failure

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	Releases to Atmosphere During Specified Time Intervals (hrs)							
	0 to 2		2 to 8		8 to 24		Total	
Nuclide	**Ci**	**Bq**	**Ci**	**Bq**	**Ci**	**Bq**	**Ci**	**Bq**
Cs-138	1.013E+02	3.748E+12	8.553E+00	3.165E+11	1.560E-03	5.772E+07	1.099E+02	4.066E+12
Sr-89	2.066E-02	7.644E+08	7.312E-02	2.705E+09	9.210E-03	3.408E+08	1.030E-01	3.811E+09
Ba-137m	1.454E+01	5.380E+11	3.760E+01	1.391E+12	5.714E+00	2.114E+11	5.785E+01	2.140E+12
Total	1.223E+03	4.525E+13	2.008E+03	7.430E+13	1.437E+02	5.317E+12	3.375E+03	1.249E+14

Table 7.1-19— Radionuclide Releases to Atmosphere for Pump Locked Rotor Accident (LRA) with Accident-Induced 8.0% Clad Failure

	Releases to Atmosphere During Specified Time Intervals (hrs)					
	0 to 2		2 to 8		8 to 24	
Nuclide	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	4.810E+01	1.78E+12	3.789E+01	1.40E+12	8.599E+01	3.18E+12
Kr-85m	1.335E+02	4.94E+12	2.123E+02	7.86E+12	3.459E+02	1.28E+13
Kr-85	1.516E+01	5.61E+11	4.362E+01	1.61E+12	5.878E+01	2.17E+12
Kr-87	1.921E+02	7.11E+12	8.793E+01	3.25E+12	2.801E+02	1.04E+13
Kr-88	3.505E+02	1.30E+13	4.069E+02	1.51E+13	7.574E+02	2.80E+13
Kr-89	2.665E+01	9.86E+11	8.120E-11	3.00E+00	2.665E+01	9.86E+11
Xe-131m	5.422E+00	2.01E+11	1.546E+01	5.72E+11	2.088E+01	7.73E+11
Xe-133m	3.034E+01	1.12E+12	8.296E+01	3.07E+12	1.133E+02	4.19E+12
Xe-133	9.995E+02	3.70E+13	2.817E+03	1.04E+14	3.816E+03	1.41E+14
Xe-135m	4.093E+01	1.51E+12	8.209E+00	3.04E+11	4.914E+01	1.82E+12
Xe-135	2.999E+02	1.11E+13	6.620E+02	2.45E+13	9.619E+02	3.56E+13
Xe-137	4.994E+01	1.85E+12	1.410E-08	5.22E+02	4.994E+01	1.85E+12
Xe-138	1.603E+02	5.93E+12	3.866E-01	1.43E+10	1.607E+02	5.95E+12
Br-83	3.026E+00	1.12E+11	1.433E+00	5.30E+10	4.459E+00	1.65E+11
Br-84	4.425E+00	1.64E+11	6.152E-02	2.28E+09	4.486E+00	1.66E+11
Br-85	1.634E+00	6.05E+10	1.750E-14	6.48E-04	1.634E+00	6.05E+10
I-129	1.724E-06	6.38E+04	2.801E-06	1.04E+05	4.525E-06	1.67E+05
I-130	2.423E+00	8.97E+10	3.217E+00	1.19E+11	5.640E+00	2.09E+11
I-131	4.169E+01	1.54E+12	7.242E+01	2.68E+12	1.141E+02	4.22E+12
I-132	3.140E+01	1.16E+12	1.392E+01	5.15E+11	4.532E+01	1.68E+12
I-133	5.550E+01	2.05E+12	8.061E+01	2.98E+12	1.361E+02	5.04E+12
I-134	4.262E+01	1.58E+12	2.891E+00	1.07E+11	4.551E+01	1.68E+12
I-135	4.658E+01	1.72E+12	5.036E+01	1.86E+12	9.694E+01	3.59E+12
Rb-86m	1.779E-03	6.58E+07	2.376E-40	8.79E-30	1.779E-03	6.58E+07
Rb-86	2.210E-01	8.18E+09	4.492E-01	1.66E+10	6.702E-01	2.48E+10
Rb-88	3.094E+02	1.14E+13	4.536E+02	1.68E+13	7.630E+02	2.82E+13
Rb-89	6.288E+01	2.33E+12	1.232E-01	4.56E+09	6.300E+01	2.33E+12
Cs-134	2.473E+01	9.15E+11	5.069E+01	1.88E+12	7.542E+01	2.79E+12
Cs-136	6.139E+00	2.27E+11	1.244E+01	4.60E+11	1.858E+01	6.87E+11
Cs-137	9.440E+00	3.49E+11	1.936E+01	7.16E+11	2.880E+01	1.07E+12
Cs-138	2.012E+02	7.44E+12	1.931E+01	7.14E+11	2.206E+02	8.16E+12
Sr-89	2.305E-02	8.53E+08	9.627E-02	3.56E+09	1.193E-01	4.41E+09
Ba-137m	7.067E+00	2.61E+11	1.831E+01	6.77E+11	2.538E+01	9.39E+11
Total	3.203E+03	1.19E+14	5.174E+03	1.91E+14	8.377E+03	3.10E+14

Table 7.1-20— Radionuclide Releases to Atmosphere for Design-Basis Small Line Break

[Rupture of ¼" NSS Sampling line outside primary containment]

Nuclide	Total Release to Atmosphere [0 – 2 hr]	
	Ci	Bq
Kr-83m	2.464E-01	9.117E+09
Kr-85m	1.115E+00	4.126E+10
Kr-85	1.077E+01	3.985E+11
Kr-87	5.791E-01	2.143E+10
Kr-88	1.968E+00	7.282E+10
Kr-89	7.493E-03	2.772E+08
Xe-131m	2.190E+00	8.103E+10
Xe-133m	2.742E+00	1.015E+11
Xe-133	1.920E+02	7.104E+12
Xe-135m	8.559E+00	3.167E+11
Xe-135	8.257E+00	3.055E+11
Xe-137	1.704E-02	6.305E+08
Xe-138	1.746E-01	6.460E+09
Br-83	5.134E-02	1.900E+09
Br-84	2.140E-02	7.918E+08
Br-85	4.844E-04	1.792E+07
I-129	8.010E-08	2.964E+03
I-130	8.553E-02	3.165E+09
I-131	3.171E+01	1.173E+12
I-132	3.815E+01	1.412E+12
I-133	6.164E+01	2.281E+12
I-134	4.539E+01	1.679E+12
I-135	5.062E+01	1.873E+12
Rb-86	6.384E-03	2.362E+08
Rb-88	1.755E+00	6.494E+10
Rb-89	4.928E-02	1.823E+09
Cs-134	7.294E-01	2.699E+10
Cs-136	1.744E-01	6.453E+09
Cs-137	2.792E-01	1.033E+10
Cs-138	3.499E-01	1.295E+10
Sr-89	8.475E-06	3.136E+05
Ba-137m	2.638E-01	9.761E+09
Total	4.599E+02	1.702E+13

Table 7.1-21— Radionuclide Releases to Atmosphere for SGTR with a Pre-Accident Iodine Spike

(Page 1 of 2)

Nuclide	Releases to Atmosphere (Ci) During Specified Time Intervals (hrs)											
	0 to 2		2 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	5.579E+01	2.064E+12	5.208E+01	1.927E+12	1.113E+01	4.118E+11	1.110E-01	4.107E+09	1.024E-10	3.789E+00	1.191E+02	4.407E+12
Kr-85m	2.745E+01	1.016E+12	9.737E-02	3.603E+09	5.647E-02	2.089E+09	5.168E-03	1.912E+08	7.391E-08	2.735E+03	2.761E+01	1.022E+12
Kr-85	2.693E+02	9.964E+12	1.875E+00	6.938E+10	4.878E+00	1.805E+11	2.172E+01	8.036E+11	1.734E+02	6.416E+12	4.711E+02	1.743E+13
Kr-87	1.365E+01	5.051E+11	1.170E-02	4.329E+08	4.390E-04	1.624E+07	7.132E-08	2.639E+03	6.326E-25	2.341E-14	1.366E+01	5.054E+11
Kr-88	4.786E+01	1.771E+12	1.186E-01	4.388E+09	3.368E-02	1.246E+09	6.881E-04	2.546E+07	1.565E-11	5.791E-01	4.801E+01	1.776E+12
Kr-89	1.260E-01	4.662E+09	4.744E-16	1.755E-05	2.768E-50	1.024E-39	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.260E-01	4.662E+09
Xe-131m	5.483E+01	2.029E+12	7.018E-01	2.597E+10	1.810E+00	6.697E+10	7.458E+00	2.759E+11	3.116E+01	1.153E+12	9.596E+01	3.551E+12
Xe-133m	7.072E+01	2.617E+12	7.102E+00	2.628E+11	1.379E+01	5.102E+11	2.108E+01	7.800E+11	4.983E+00	1.844E+11	1.177E+02	4.355E+12
Xe-133	4.829E+03	1.787E+14	1.262E+02	4.669E+12	2.600E+02	9.620E+12	5.499E+02	2.035E+13	6.459E+02	2.390E+13	6.411E+03	2.372E+14
Xe-135m	1.530E+03	5.661E+13	3.263E+03	1.207E+14	3.062E+03	1.133E+14	7.187E+02	2.659E+13	4.064E-01	1.504E+10	8.574E+03	3.172E+14
Xe-135	4.299E+02	1.591E+13	5.069E+02	1.876E+13	4.845E+02	1.793E+13	1.206E+02	4.462E+12	1.232E-01	4.558E+09	1.542E+03	5.705E+13
Xe-137	2.887E-01	1.068E+10	9.932E-14	3.675E-03	4.492E-42	1.662E-31	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.887E-01	1.068E+10
Xe-138	3.434E+00	1.271E+11	9.959E-06	3.685E+05	2.041E-13	7.552E-03	8.199E-34	3.034E-23	0.000E+00	0.000E+00	3.434E+00	1.271E+11
Br-83	2.004E+00	7.415E+10	2.840E-03	1.051E+08	7.849E-04	2.904E+07	1.620E-05	5.994E+05	4.395E-14	1.626E-03	2.008E+00	7.430E+10
Br-84	5.904E-01	2.184E+10	4.270E-05	1.580E+06	1.939E-08	7.174E+02	4.027E-17	1.490E-06	1.788E-57	6.616E-47	5.904E-01	2.184E+10
Br-85	6.852E-04	2.535E+07	1.190E-18	4.403E-08	2.448E-56	9.058E-46	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.852E-04	2.535E+07
I-129	3.454E-06	1.278E+05	1.964E-08	7.267E+02	8.140E-08	3.012E+03	1.077E-06	3.985E+04	4.192E-05	1.551E+06	4.655E-05	1.722E+06
I-130	3.616E+00	1.338E+11	1.503E-02	5.561E+08	3.374E-02	1.248E+09	5.191E-02	1.921E+09	2.304E-03	8.525E+07	3.719E+00	1.376E+11
I-131	5.578E+01	2.064E+12	3.103E-01	1.148E+10	1.236E+00	4.573E+10	1.376E+01	5.091E+11	1.542E+02	5.705E+12	2.253E+02	8.336E+12
I-132	2.339E+01	8.654E+11	3.417E-02	1.264E+09	8.312E-03	3.075E+08	1.407E-04	5.206E+06	1.667E-13	6.168E-03	2.343E+01	8.669E+11
I-133	9.220E+01	3.411E+12	4.337E-01	1.605E+10	1.242E+00	4.595E+10	3.997E+00	1.479E+11	9.448E-01	3.496E+10	9.882E+01	3.656E+12
I-134	1.140E+01	4.218E+11	3.079E-03	1.139E+08	3.155E-05	1.167E+06	2.442E-10	9.035E+00	1.584E-34	5.861E-24	1.140E+01	4.218E+11
I-135	5.584E+01	2.066E+12	1.805E-01	6.679E+09	2.463E-01	9.113E+09	1.167E-01	4.318E+09	1.685E-04	6.235E+06	5.639E+01	2.086E+12
Rb-86	4.589E-03	1.698E+08	2.766E-05	1.023E+06	1.086E-04	4.018E+06	1.305E-03	4.829E+07	2.814E-02	1.041E+09	3.417E-02	1.264E+09
Rb-88	1.105E+00	4.089E+10	1.286E-03	4.758E+07	6.410E-04	2.372E+07	2.976E-05	1.101E+06	2.261E-12	8.366E-02	1.107E+00	4.096E+10
Rb-89	1.257E-02	4.651E+08	4.677E-08	1.730E+03	4.331E-15	1.602E-04	1.140E-33	4.218E-23	0.000E+00	0.000E+00	1.257E-02	4.651E+08
Cs-134	5.246E-01	1.941E+10	3.196E-03	1.183E+08	1.275E-02	4.718E+08	1.648E-01	6.098E+09	6.259E+00	2.316E+11	6.964E+00	2.577E+11
Cs-136	1.253E-01	4.636E+09	7.520E-04	2.782E+07	2.931E-03	1.084E+08	3.415E-02	1.264E+09	5.875E-01	2.174E+10	7.507E-01	2.778E+10
Cs-137	2.008E-01	7.430E+09	1.224E-03	4.529E+07	4.884E-03	1.807E+08	6.322E-02	2.339E+09	2.436E+00	9.013E+10	2.706E+00	1.001E+11
Cs-138	1.397E-01	5.169E+09	9.813E-06	3.631E+05	5.129E-09	1.898E+02	1.405E-17	5.199E-07	2.046E-57	7.570E-47	1.397E-01	5.169E+09

Table 7.1-21— Radionuclide Releases to Atmosphere for SGTR with a Pre-Accident Iodine Spike
(Page 2 of 2)

Nuclide	Releases to Atmosphere (Ci) During Specified Time Intervals (hrs)									
	0 to 2		2 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Ba-137m	1.883E-01	6.967E+09	1.148E-03	4.248E+07	4.579E-03	1.694E+08	5.927E-02	2.193E+09	2.284E+00	8.451E+10	2.537E+00	9.387E+10
Total	7.580E+03	2.805E+14	3.959E+03	1.465E+14	3.841E+03	1.421E+14	1.458E+03	5.395E+13	1.023E+03	3.785E+13	1.786E+04	6.608E+14

Table 7.1-22— Radionuclide Releases to Atmosphere for SGTR with Accident Induced (Coincident) Iodine Spike
(Page 1 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)											
	0 to 2		2 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	5.286E+01	1.956E+12	6.506E+01	2.407E+12	2.614E+01	9.672E+11	5.395E-01	1.996E+10	1.229E-09	4.547E+01	1.446E+02	5.350E+12
Kr-85m	2.938E+01	1.087E+12	2.475E-01	9.158E+09	2.560E-01	9.472E+09	2.342E-02	8.665E+08	3.350E-07	1.240E+04	2.990E+01	1.106E+12
Kr-85	2.693E+02	9.964E+12	1.875E+00	6.938E+10	4.878E+00	1.805E+11	2.172E+01	8.036E+11	1.734E+02	6.416E+12	4.711E+02	1.743E+13
Kr-87	1.365E+01	5.051E+11	1.170E-02	4.329E+08	4.390E-04	1.624E+07	7.132E-08	2.639E+03	6.326E-25	2.341E-14	1.366E+01	5.054E+11
Kr-88	4.786E+01	1.771E+12	1.186E-01	4.388E+09	3.368E-02	1.246E+09	6.881E-04	2.546E+07	1.565E-11	5.791E-01	4.801E+01	1.776E+12
Kr-89	1.260E-01	4.662E+09	4.744E-16	1.755E-05	2.768E-50	1.024E-39	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.260E-01	4.662E+09
Xe-131m	5.476E+01	2.026E+12	5.269E-01	1.950E+10	1.550E+00	5.735E+10	9.473E+00	3.505E+11	8.667E+01	3.207E+12	1.530E+02	5.661E+12
Xe-133m	6.924E+01	2.562E+12	4.025E+00	1.489E+11	1.188E+01	4.396E+11	4.107E+01	1.520E+12	2.417E+01	8.943E+11	1.504E+02	5.565E+12
Xe-133	4.808E+03	1.779E+14	8.294E+01	3.069E+12	2.349E+02	8.691E+12	9.134E+02	3.380E+13	1.558E+03	5.765E+13	7.597E+03	2.811E+14
Xe-135m	9.009E+02	3.333E+13	2.273E+03	8.410E+13	2.859E+03	1.058E+14	1.054E+03	3.900E+13	1.262E+00	4.669E+10	7.088E+03	2.623E+14
Xe-135	3.154E+02	1.167E+13	3.712E+02	1.373E+13	6.204E+02	2.295E+13	3.471E+02	1.284E+13	1.427E+00	5.280E+10	1.655E+03	6.124E+13
Xe-137	2.887E-01	1.068E+10	9.932E-14	3.675E-03	4.492E-42	1.662E-31	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.887E-01	1.068E+10
Xe-138	3.434E+00	1.271E+11	9.959E-06	3.685E+05	2.041E-13	7.552E-03	8.199E-34	3.034E-23	0.000E+00	0.000E+00	3.434E+00	1.271E+11
Br-83	3.105E+00	1.149E+11	2.064E-02	7.637E+08	3.304E-02	1.222E+09	1.187E-03	4.392E+07	4.062E-12	1.503E-01	3.159E+00	1.169E+11
Br-84	3.844E+00	1.422E+11	4.306E-03	1.593E+08	7.921E-04	2.931E+07	7.298E-12	2.700E-01	4.404E-52	1.629E-41	3.849E+00	1.424E+11
Br-85	7.119E-01	2.634E+10	4.381E-05	1.621E+06	6.904E-07	2.554E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.120E-01	2.634E+10
I-129	1.942E-06	7.185E+04	3.838E-08	1.420E+03	4.662E-07	1.725E+04	9.049E-06	3.348E+05	3.973E-04	1.470E+07	4.088E-04	1.513E+07
I-130	2.679E+00	9.912E+10	3.998E-02	1.479E+09	3.041E-01	1.125E+10	6.765E-01	2.503E+10	3.436E-02	1.271E+09	3.734E+00	1.382E+11
I-131	3.199E+01	1.184E+12	6.194E-01	2.292E+10	7.305E+00	2.703E+11	1.192E+02	4.410E+12	1.500E+03	5.550E+13	1.659E+03	6.138E+13
I-132	3.721E+01	1.377E+12	2.421E-01	8.958E+09	3.626E-01	1.342E+10	1.103E-02	4.081E+08	1.671E-11	6.183E-01	3.782E+01	1.399E+12
I-133	6.155E+01	2.277E+12	1.022E+00	3.781E+10	9.383E+00	3.472E+11	4.389E+01	1.624E+12	1.163E+01	4.303E+11	1.275E+02	4.718E+12
I-134	4.170E+01	1.543E+12	9.438E-02	3.492E+09	3.336E-02	1.234E+09	7.756E-07	2.870E+04	6.711E-31	2.483E-20	4.183E+01	1.548E+12
I-135	5.032E+01	1.862E+12	6.126E-01	2.267E+10	3.161E+00	1.170E+11	2.185E+00	8.085E+10	3.747E-03	1.386E+08	5.629E+01	2.083E+12
Rb-86	4.589E-03	1.698E+08	2.766E-05	1.023E+06	1.086E-04	4.018E+06	1.305E-03	4.829E+07	2.814E-02	1.041E+09	3.417E-02	1.264E+09
Rb-88	1.105E+00	4.089E+10	1.286E-03	4.758E+07	6.410E-04	2.372E+07	2.976E-05	1.101E+06	2.261E-12	8.366E-02	1.107E+00	4.096E+10
Rb-89	1.257E-02	4.651E+08	4.677E-08	1.730E+03	4.331E-15	1.602E-04	1.140E-33	4.218E-23	0.000E+00	0.000E+00	1.257E-02	4.651E+08
Cs-134	5.246E-01	1.941E+10	3.196E-03	1.183E+08	1.275E-02	4.718E+08	1.648E-01	6.098E+09	6.259E+00	2.316E+11	6.964E+00	2.577E+11
Cs-136	1.253E-01	4.636E+09	7.520E-04	2.782E+07	2.931E-03	1.084E+08	3.415E-02	1.264E+09	5.875E-01	2.174E+10	7.507E-01	2.778E+10
Cs-137	2.008E-01	7.430E+09	1.224E-03	4.529E+07	4.884E-03	1.807E+08	6.322E-02	2.339E+09	2.436E+00	9.013E+10	2.706E+00	1.001E+11
Cs-138	1.397E-01	5.169E+09	9.813E-06	3.631E+05	5.129E-09	1.898E+02	1.405E-17	5.199E-07	2.046E-57	7.570E-47	1.397E-01	5.169E+09

Table 7.1-22— Radionuclide Releases to Atmosphere for SGTR with Accident Induced (Coincident) Iodine Spike

(Page 2 of 2)

Nuclide	Releases to Atmosphere During Specified Time Intervals (hrs)											
	0 to 2		2 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Ba-137m	1.883E-01	6.967E+09	1.148E-03	4.248E+07	4.579E-03	1.694E+08	5.927E-02	2.193E+09	2.284E+00	8.451E+10	2.537E+00	9.387E+10
Total	6.801E+03	2.516E+14	2.802E+03	1.037E+14	3.780E+03	1.399E+14	2.554E+03	9.450E+13	3.368E+03	1.246E+14	1.930E+04	7.141E+14

Table 7.1-23— Radionuclide Releases to Atmosphere for Design Basis LOCA

(Page 1 of 3)

Nuclide	Releases to Atmosphere (Ci) During Specified Time Intervals (hrs)													
	0 to 1.5		1.5 to 3.5		3.5 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	7.297E+02	2.700E+13	2.751E+03	1.018E+14	4.641E+03	1.717E+14	4.187E+03	1.549E+14	1.072E+02	3.966E+12	3.150E-07	1.166E+04	1.242E+04	4.595E+14
Kr-85m	1.709E+03	6.323E+13	6.303E+03	2.332E+14	8.876E+03	3.284E+14	8.074E+03	2.987E+14	3.703E+02	1.370E+13	5.366E-03	1.985E+08	2.533E+04	9.372E+14
Kr-85	1.126E+02	4.166E+12	4.307E+02	1.594E+13	9.847E+02	3.643E+13	3.497E+03	1.294E+14	7.845E+03	2.903E+14	6.661E+04	2.465E+15	7.948E+04	2.941E+15
Kr-87	2.224E+03	8.229E+13	4.925E+03	1.822E+14	2.337E+03	8.647E+13	2.199E+02	8.136E+12	1.791E-02	6.627E+08	1.613E-19	5.968E-09	9.706E+03	3.591E+14
Kr-88	4.382E+03	1.621E+14	1.434E+04	5.306E+14	1.548E+04	5.728E+14	7.580E+03	2.805E+14	7.766E+01	2.873E+12	1.794E-06	6.638E+04	4.186E+04	1.549E+15
Kr-89	9.523E+00	3.524E+11	3.044E-06	1.126E+05	1.461E-17	5.406E-07	3.346E-43	1.238E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.523E+00	3.524E+11
Xe-131m	7.277E+01	2.692E+12	3.151E+02	1.166E+13	7.225E+02	2.673E+13	2.650E+03	9.805E+13	8.448E+03	3.126E+14	8.304E+04	3.072E+15	9.525E+04	3.524E+15
Xe-133m	4.023E+02	1.489E+13	1.806E+03	6.682E+13	4.148E+03	1.535E+14	1.551E+04	5.739E+14	3.840E+04	1.421E+15	2.689E+04	9.949E+14	8.716E+04	3.225E+15
Xe-133	1.326E+04	4.906E+14	5.898E+04	2.182E+15	1.353E+05	5.006E+15	4.923E+05	1.822E+16	1.172E+06	4.336E+16	2.331E+06	8.625E+16	4.202E+06	1.555E+17
Xe-135m	1.676E+03	6.201E+13	1.283E+04	4.747E+14	5.187E+04	1.919E+15	1.495E+05	5.532E+15	6.371E+04	2.357E+15	8.257E+01	3.055E+12	2.797E+05	1.035E+16
Xe-135	4.390E+03	1.624E+14	2.130E+04	7.881E+14	5.958E+04	2.204E+15	2.402E+05	8.887E+15	1.708E+05	6.320E+15	9.095E+02	3.365E+13	4.971E+05	1.839E+16
Xe-137	2.238E+01	8.281E+11	1.730E-04	6.401E+06	7.545E-14	2.792E-03	4.529E-35	1.676E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.238E+01	8.281E+11
Xe-138	6.229E+02	2.305E+13	9.854E+01	3.646E+12	3.005E-01	1.112E+10	5.518E-07	2.042E+04	1.111E-27	4.111E-17	0.000E+00	0.000E+00	7.217E+02	2.670E+13
Br-83	3.714E+00	1.374E+11	7.476E+00	2.766E+11	5.922E+00	2.191E+11	1.578E+00	5.839E+10	9.943E-03	3.679E+08	7.939E-12	2.937E-01	1.870E+01	6.919E+11
Br-84	3.206E+00	1.186E+11	1.399E+00	5.176E+10	1.010E-01	3.737E+09	2.106E-04	7.792E+06	1.010E-13	3.737E-03	1.200E-54	4.440E-44	4.706E+00	1.741E+11
Br-85	7.005E-01	2.592E+10	3.783E-10	1.400E+01	1.011E-22	3.741E-12	3.330E-51	1.232E-40	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.005E-01	2.592E+10
I-129	2.143E-06	7.929E+04	6.460E-06	2.390E+05	1.204E-05	4.455E+05	2.778E-05	1.028E+06	8.971E-05	3.319E+06	6.739E-04	2.493E+07	8.120E-04	3.004E+07
I-130	3.160E+00	1.169E+11	8.910E+00	3.297E+11	1.395E+01	5.162E+11	1.919E+01	7.100E+11	9.181E+00	3.397E+11	1.557E-01	5.761E+09	5.455E+01	2.018E+12
I-131	3.558E+01	1.316E+12	1.070E+02	3.959E+12	1.971E+02	7.293E+12	4.395E+02	1.626E+13	1.216E+03	4.499E+13	3.310E+03	1.225E+14	5.305E+03	1.963E+14
I-132	3.928E+01	1.453E+12	8.453E+01	3.128E+12	8.515E+01	3.151E+12	8.672E+01	3.209E+12	1.646E+02	6.090E+12	1.700E+02	6.290E+12	6.303E+02	2.332E+13
I-133	7.134E+01	2.640E+12	2.071E+02	7.663E+12	3.479E+02	1.287E+13	5.859E+02	2.168E+13	5.389E+02	1.994E+13	5.089E+01	1.883E+12	1.802E+03	6.667E+13
I-134	4.192E+01	1.551E+12	4.308E+01	1.594E+12	1.043E+01	3.859E+11	2.466E-01	9.124E+09	4.949E-07	1.831E+04	8.736E-32	3.232E-21	9.568E+01	3.540E+12
I-135	6.120E+01	2.264E+12	1.615E+02	5.976E+12	2.183E+02	8.077E+12	2.005E+02	7.419E+12	3.195E+01	1.182E+12	1.584E-02	5.861E+08	6.735E+02	2.492E+13
Rb-86m	2.457E-04	9.091E+06	8.331E-31	3.082E-20	2.805E-66	1.038E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.457E-04	9.091E+06
Rb-86	1.268E-01	4.692E+09	3.249E-01	1.202E+10	5.175E-01	1.915E+10	6.158E-01	2.278E+10	1.784E-01	6.601E+09	3.473E-02	1.285E+09	1.798E+00	6.653E+10
Rb-88	6.288E+01	2.327E+12	1.545E+02	5.717E+12	1.636E+02	6.053E+12	8.009E+01	2.963E+12	8.460E-01	3.130E+10	1.980E-08	7.326E+02	4.619E+02	1.709E+13
Rb-89	1.126E+01	4.166E+11	5.235E-01	1.937E+10	1.960E-03	7.252E+07	4.966E-09	1.837E+02	5.198E-29	1.923E-18	0.000E+00	0.000E+00	1.178E+01	4.359E+11
Cs-134	1.418E+01	5.247E+11	3.636E+01	1.345E+12	5.818E+01	2.153E+12	7.012E+01	2.594E+12	2.128E+01	7.874E+11	5.202E+00	1.925E+11	2.053E+02	7.596E+12
Cs-136	3.511E+00	1.299E+11	9.004E+00	3.331E+11	1.431E+01	5.295E+11	1.694E+01	6.268E+11	4.810E+00	1.780E+11	8.548E-01	3.163E+10	4.943E+01	1.829E+12
Cs-137	5.419E+00	2.005E+11	1.389E+01	5.139E+11	2.223E+01	8.225E+11	2.679E+01	9.912E+11	8.142E+00	3.013E+11	2.002E+00	7.407E+10	7.848E+01	2.904E+12

Table 7.1-23— Radionuclide Releases to Atmosphere for Design Basis LOCA
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Nuclide	Releases to Atmosphere (Ci) During Specified Time Intervals (hrs)													
	0 to 1.5		1.5 to 3.5		3.5 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Cs-138	4.511E+01	1.669E+12	2.603E+01	9.631E+11	1.839E+00	6.804E+10	3.106E-03	1.149E+08	3.645E-13	1.349E-02	3.186E-55	1.179E-44	7.298E+01	2.700E+12
Sb-125	7.674E-02	2.839E+09	3.605E-01	1.334E+10	5.787E-01	2.141E+10	6.973E-01	2.580E+10	2.117E-01	7.833E+09	5.185E-02	1.918E+09	1.977E+00	7.315E+10
Sb-127	3.566E-01	1.319E+10	1.658E+00	6.135E+10	2.602E+00	9.627E+10	2.947E+00	1.090E+11	7.152E-01	2.646E+10	6.814E-02	2.521E+09	8.347E+00	3.088E+11
Sb-129	8.062E-01	2.983E+10	3.074E+00	1.137E+11	3.076E+00	1.138E+11	1.172E+00	4.336E+10	1.262E-02	4.669E+08	9.903E-09	3.664E+02	8.142E+00	3.013E+11
Te-127m	5.087E-02	1.882E+09	2.290E-01	8.473E+09	3.677E-01	1.360E+10	4.432E-01	1.640E+10	1.345E-01	4.977E+09	3.221E-02	1.192E+09	1.257E+00	4.651E+10
Te-127	3.679E-01	1.361E+10	1.678E+00	6.209E+10	2.678E+00	9.909E+10	3.139E+00	1.161E+11	8.103E-01	2.998E+10	9.679E-02	3.581E+09	8.769E+00	3.245E+11
Te-129m	1.475E-01	5.458E+09	6.643E-01	2.458E+10	1.065E+00	3.941E+10	1.276E+00	4.721E+10	3.779E-01	1.398E+10	8.132E-02	3.009E+09	3.612E+00	1.336E+11
Te-129	9.137E-01	3.381E+10	3.758E+00	1.390E+11	4.244E+00	1.570E+11	2.219E+00	8.210E+10	2.610E-01	9.657E+09	5.294E-02	1.959E+09	1.145E+01	4.237E+11
Te-131m	4.117E-01	1.523E+10	1.808E+00	6.690E+10	2.706E+00	1.001E+11	2.700E+00	9.990E+10	4.296E-01	1.590E+10	9.844E-03	3.642E+08	8.066E+00	2.984E+11
Te-131	4.731E-01	1.750E+10	6.764E-01	2.503E+10	6.180E-01	2.287E+10	6.079E-01	2.249E+10	9.670E-02	3.578E+09	2.216E-03	8.199E+07	2.474E+00	9.154E+10
Te-132	4.076E+00	1.508E+11	1.819E+01	6.730E+11	2.841E+01	1.051E+12	3.181E+01	1.177E+12	7.423E+00	2.747E+11	6.147E-01	2.274E+10	9.053E+01	3.350E+12
Te-134	1.637E+00	6.057E+10	2.306E+00	8.532E+10	2.992E-01	1.107E+10	1.926E-03	7.126E+07	2.642E-11	9.775E-01	1.491E-43	5.517E-33	4.244E+00	1.570E+11
Sr-89	1.295E+00	4.792E+10	6.070E+00	2.246E+11	9.727E+00	3.599E+11	1.167E+01	4.318E+11	3.484E+00	1.289E+11	7.831E-01	2.897E+10	3.303E+01	1.222E+12
Sr-90	1.352E-01	5.002E+09	6.346E-01	2.348E+10	1.019E+00	3.770E+10	1.228E+00	4.544E+10	3.731E-01	1.380E+10	9.176E-02	3.395E+09	3.481E+00	1.288E+11
Sr-91	1.523E+00	5.635E+10	6.489E+00	2.401E+11	8.369E+00	3.097E+11	5.720E+00	2.116E+11	3.029E-01	1.121E+10	1.462E-04	5.409E+06	2.240E+01	8.288E+11
Sr-92	1.273E+00	4.710E+10	4.299E+00	1.591E+11	3.300E+00	1.221E+11	7.207E-01	2.667E+10	1.556E-03	5.757E+07	1.220E-12	4.514E-02	9.594E+00	3.550E+11
Ba-137m	4.246E+00	1.571E+11	1.310E+01	4.847E+11	2.103E+01	7.781E+11	2.535E+01	9.380E+11	7.702E+00	2.850E+11	1.894E+00	7.008E+10	7.332E+01	2.713E+12
Ba-139	1.252E+00	4.632E+10	2.933E+00	1.085E+11	1.185E+00	4.385E+10	7.377E-02	2.729E+09	2.809E-06	1.039E+05	3.953E-23	1.463E-12	5.444E+00	2.014E+11
Ba-140	2.011E+00	7.441E+10	9.409E+00	3.481E+11	1.500E+01	5.550E+11	1.775E+01	6.568E+11	5.031E+00	1.861E+11	8.876E-01	3.284E+10	5.008E+01	1.853E+12
Mo-99	6.680E-01	2.472E+10	1.185E+00	4.385E+10	1.843E+00	6.819E+10	2.036E+00	7.533E+10	4.535E-01	1.678E+10	3.193E-02	1.181E+09	6.218E+00	2.301E+11
Tc-99m	4.054E-01	1.500E+10	1.062E+00	3.929E+10	1.685E+00	6.235E+10	1.916E+00	7.089E+10	4.358E-01	1.612E+10	3.075E-02	1.138E+09	5.535E+00	2.048E+11
Ru-103	2.419E-01	8.950E+09	1.134E+00	4.196E+10	1.816E+00	6.719E+10	2.175E+00	8.048E+10	6.463E-01	2.391E+10	1.417E-01	5.243E+09	6.155E+00	2.277E+11
Ru-105	1.639E-01	6.064E+09	6.263E-01	2.317E+10	6.347E-01	2.348E+10	2.485E-01	9.195E+09	2.881E-03	1.066E+08	3.096E-09	1.146E+02	1.676E+00	6.201E+10
Ru-106	1.433E-01	5.302E+09	6.720E-01	2.486E+10	1.079E+00	3.992E+10	1.299E+00	4.806E+10	3.939E-01	1.457E+10	9.568E-02	3.540E+09	3.683E+00	1.363E+11
Rh-103m	2.180E-01	8.066E+09	1.022E+00	3.781E+10	1.637E+00	6.057E+10	1.961E+00	7.256E+10	5.827E-01	2.156E+10	1.277E-01	4.725E+09	5.549E+00	2.053E+11
Rh-105	1.753E-01	6.486E+09	8.191E-01	3.031E+10	1.284E+00	4.751E+10	1.375E+00	5.088E+10	2.453E-01	9.076E+09	7.574E-03	2.802E+08	3.907E+00	1.446E+11
Rh-106	1.433E-01	5.302E+09	6.720E-01	2.486E+10	1.079E+00	3.992E+10	1.299E+00	4.806E+10	3.939E-01	1.457E+10	9.568E-02	3.540E+09	3.683E+00	1.363E+11
Ce-141	4.504E-02	1.666E+09	2.100E-01	7.770E+09	3.363E-01	1.244E+10	4.027E-01	1.490E+10	1.191E-01	4.407E+09	2.551E-02	9.439E+08	1.139E+00	4.214E+10
Ce-143	4.473E-02	1.655E+09	2.032E-01	7.518E+09	3.060E-01	1.132E+10	3.105E-01	1.149E+10	5.212E-02	1.928E+09	1.426E-03	5.276E+07	9.179E-01	3.396E+10
Ce-144	3.421E-02	1.266E+09	1.595E-01	5.902E+09	2.560E-01	9.472E+09	3.085E-01	1.141E+10	9.342E-02	3.457E+09	2.261E-02	8.366E+08	8.743E-01	3.235E+10

Table 7.1-23— Radionuclide Releases to Atmosphere for Design Basis LOCA
(Page 3 of 3)

Nuclide	Releases to Atmosphere (Ci) During Specified Time Intervals (hrs)													
	0 to 1.5		1.5 to 3.5		3.5 to 8		8 to 24		24 to 96		96 to 720		Total	
	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Np-239	7.573E-01	2.802E+10	3.479E+00	1.287E+11	5.379E+00	1.990E+11	5.860E+00	2.168E+11	1.242E+00	4.595E+10	7.389E-02	2.734E+09	1.679E+01	6.212E+11
Pu-238	2.937E-04	1.087E+07	1.371E-03	5.073E+07	2.200E-03	8.140E+07	2.652E-03	9.812E+07	8.060E-04	2.982E+07	1.984E-04	7.341E+06	7.522E-03	2.783E+08
Pu-239	1.236E-05	4.573E+05	5.767E-05	2.134E+06	9.263E-05	3.427E+06	1.118E-04	4.137E+06	3.413E-05	1.263E+06	8.458E-06	3.129E+05	3.171E-04	1.173E+07
Pu-240	2.817E-05	1.042E+06	1.315E-04	4.866E+06	2.110E-04	7.807E+06	2.543E-04	9.409E+06	7.729E-05	2.860E+06	1.901E-05	7.034E+05	7.212E-04	2.668E+07
Pu-241	5.110E-03	1.891E+08	2.385E-02	8.825E+08	3.828E-02	1.416E+09	4.613E-02	1.707E+09	1.402E-02	5.187E+08	3.446E-03	1.275E+08	1.308E-01	4.840E+09
Y-90	3.140E-03	1.162E+08	2.339E-02	8.654E+08	6.936E-02	2.566E+09	1.818E-01	6.727E+09	1.423E-01	5.265E+09	7.603E-02	2.813E+09	4.961E-01	1.836E+10
Y-91m	5.663E-01	2.095E+10	3.441E+00	1.273E+11	5.191E+00	1.921E+11	3.634E+00	1.345E+11	1.924E-01	7.119E+09	9.288E-05	3.437E+06	1.302E+01	4.817E+11
Y-91	1.652E-02	6.112E+08	8.019E-02	2.967E+09	1.426E-01	5.276E+09	2.021E-01	7.478E+09	7.064E-02	2.614E+09	1.656E-02	6.127E+08	5.286E-01	1.956E+10
Y-92	3.112E-01	1.151E+10	2.236E+00	8.273E+10	3.968E+00	1.468E+11	2.181E+00	8.070E+10	2.160E-02	7.992E+08	1.599E-09	5.916E+01	8.719E+00	3.226E+11
Y-93	1.749E-02	6.471E+08	7.414E-02	2.743E+09	9.685E-02	3.583E+09	6.832E-02	2.528E+09	3.943E-03	1.459E+08	2.631E-06	9.735E+04	2.607E-01	9.646E+09
Zr-95	1.861E-02	6.886E+08	8.589E-02	3.178E+09	1.377E-01	5.095E+09	1.654E-01	6.120E+09	4.955E-02	1.833E+09	1.135E-02	4.200E+08	4.685E-01	1.733E+10
Zr-97	1.877E-02	6.945E+08	8.243E-02	3.050E+09	1.169E-01	4.325E+09	1.014E-01	3.752E+09	1.051E-02	3.889E+08	5.726E-05	2.119E+06	3.300E-01	1.221E+10
Nb-95	1.862E-02	6.889E+08	8.599E-02	3.182E+09	1.380E-01	5.106E+09	1.664E-01	6.157E+09	5.053E-02	1.870E+09	1.232E-02	4.558E+08	4.719E-01	1.746E+10
La-140	6.044E-02	2.236E+09	4.868E-01	1.801E+10	1.509E+00	5.583E+10	3.941E+00	1.458E+11	2.736E+00	1.012E+11	9.057E-01	3.351E+10	9.639E+00	3.566E+11
La-141	1.590E-02	5.883E+08	5.866E-02	2.170E+09	5.613E-02	2.077E+09	1.940E-02	7.178E+08	1.590E-04	5.883E+06	3.935E-11	1.456E+00	1.502E-01	5.557E+09
La-142	1.132E-02	4.188E+08	2.986E-02	1.105E+09	1.382E-02	5.113E+08	1.118E-03	4.137E+07	1.026E-07	3.796E+03	7.220E-23	2.671E-12	5.612E-02	2.076E+09
Pr-143	1.844E-02	6.823E+08	8.551E-02	3.164E+09	1.384E-01	5.121E+09	1.698E-01	6.283E+09	5.241E-02	1.939E+09	1.030E-02	3.811E+08	4.748E-01	1.757E+10
Pr-144	3.272E-02	1.211E+09	1.590E-01	5.883E+09	2.560E-01	9.472E+09	3.085E-01	1.141E+10	9.343E-02	3.457E+09	2.261E-02	8.366E+08	8.722E-01	3.227E+10
Nd-147	7.658E-03	2.833E+08	3.525E-02	1.304E+09	5.615E-02	2.078E+09	6.621E-02	2.450E+09	1.857E-02	6.871E+08	3.127E-03	1.157E+08	1.870E-01	6.919E+09
Am-241	2.343E-06	8.669E+04	1.083E-05	4.007E+05	1.740E-05	6.438E+05	2.105E-05	7.789E+05	6.475E-06	2.396E+05	1.695E-06	6.272E+04	5.978E-05	2.212E+06
Cm-242	1.065E-03	3.941E+07	4.917E-03	1.819E+08	7.889E-03	2.919E+08	9.495E-03	3.513E+08	2.870E-03	1.062E+08	6.862E-04	2.539E+07	2.692E-02	9.960E+08
Cm-244	5.651E-04	2.091E+07	2.610E-03	9.657E+07	4.190E-03	1.550E+08	5.049E-03	1.868E+08	1.534E-03	5.676E+07	3.772E-04	1.396E+07	1.432E-02	5.298E+08
Total	3.005E+04	1.112E+15	1.250E+05	4.625E+15	2.852E+05	1.055E+16	9.254E+05	3.424E+16	1.463E+06	5.413E+16	2.512E+06	9.294E+16	5.341E+06	1.976E+17

Table 7.1-24— Radionuclide Releases to Atmosphere for Fuel Handling Accident
[Maximized source term and 72-hr decay]

	Releases to Atmosphere During Specified Time Intervals (hrs)											
	0 to 2		2 to 8		8 to 24		24 to 96		96 to 720		Total	
Nuclide	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	2.429E-05	8.987E+05	3.484E-06	1.289E+05	7.205E-07	2.666E+04	7.024E-09	2.599E+02	0.000E+00	0.000E+00	2.850E-05	1.055E+06
Kr-85m	2.186E-01	8.088E+09	1.086E-03	4.018E+07	1.313E-10	4.858E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.197E-01	8.129E+09
Kr-85	1.121E+03	4.148E+13	7.604E+00	2.813E+11	2.326E-06	8.606E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.129E+03	4.177E+13
Kr-88	9.499E-04	3.515E+07	3.944E-06	1.459E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.538E-04	3.529E+07
Xe-131m	5.048E+02	1.868E+13	1.172E+01	4.336E+11	2.136E+01	7.903E+11	8.230E+01	3.045E+12	2.492E+02	9.220E+12	8.694E+02	3.217E+13
Xe-133m	1.543E+03	5.709E+13	2.599E+01	9.616E+11	2.957E+01	1.094E+12	3.817E+01	1.412E+12	3.811E+00	1.410E+11	1.641E+03	6.072E+13
Xe-133	7.072E+04	2.617E+15	6.975E+02	2.581E+13	4.174E+02	1.544E+13	5.388E+02	1.994E+13	5.379E+01	1.990E+12	7.243E+04	2.680E+15
Xe-135m	1.793E+01	6.634E+11	3.252E+01	1.203E+12	3.019E+01	1.117E+12	6.933E+00	2.565E+11	3.653E-03	1.352E+08	8.758E+01	3.240E+12
Xe-135	9.170E+02	3.393E+13	1.667E+01	6.168E+11	1.056E+01	3.907E+11	2.424E+00	8.969E+10	1.277E-03	4.725E+07	9.467E+02	3.503E+13
Br-83	2.634E-08	9.746E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.634E-08	9.746E+02
I-129	8.319E-06	3.078E+05	5.643E-08	2.088E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.375E-06	3.099E+05
I-130	1.404E-01	5.195E+09	8.507E-04	3.148E+07	1.859E-10	6.878E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.413E-01	5.228E+09
I-131	2.615E+02	9.676E+12	1.761E+00	6.516E+10	5.272E-07	1.951E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.633E+02	9.742E+12
I-132	1.035E-07	3.830E+03	3.830E-10	1.417E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.039E-07	3.844E+03
I-133	4.003E+01	1.481E+12	2.539E-01	9.394E+09	6.360E-08	2.353E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.028E+01	1.490E+12
I-135	2.110E-01	7.807E+09	1.159E-03	4.288E+07	1.890E-10	6.993E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.122E-01	7.851E+09
Rb-88	4.564E-04	1.689E+07	4.366E-06	1.615E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.608E-04	1.705E+07
Total	7.512E+04	2.779E+15	7.940E+02	2.938E+13	5.090E+02	1.883E+13	6.686E+02	2.474E+13	3.068E+02	1.135E+13	7.740E+04	2.864E+15

Table 7.1-25— Radionuclide Releases to Atmosphere for Rod Ejection Accident (REA) with Accident-Induced 26% Clad Failure

[Secondary-side releases without SG tube uncovery]

	Releases to Atmosphere During Specified Time Intervals (hrs)					
	0 to 2		2 to 8		8 to 24	
Nuclide	Ci	Bq	Ci	Bq	Ci	Bq
Kr-83m	4.713E+02	1.744E+13	3.879E+02	1.435E+13	8.592E+02	3.179E+13
Kr-85m	1.326E+03	4.906E+13	2.209E+03	8.173E+13	3.535E+03	1.308E+14
Kr-85	7.302E+01	2.702E+12	2.188E+02	8.096E+12	2.918E+02	1.080E+13
Kr-87	1.878E+03	6.949E+13	9.137E+02	3.381E+13	2.791E+03	1.033E+14
Kr-88	3.466E+03	1.282E+14	4.228E+03	1.564E+14	7.695E+03	2.847E+14
Kr-89	2.102E+02	7.777E+12	8.446E-10	3.125E+01	2.102E+02	7.777E+12
Xe-131m	5.278E+01	1.953E+12	1.566E+02	5.794E+12	2.094E+02	7.748E+12
Xe-133m	3.008E+02	1.113E+13	8.563E+02	3.168E+13	1.157E+03	4.281E+13
Xe-133	9.847E+03	3.643E+14	2.890E+04	1.069E+15	3.875E+04	1.434E+15
Xe-135m	3.493E+02	1.292E+13	6.359E+01	2.353E+12	4.129E+02	1.528E+13
Xe-135	2.976E+03	1.101E+14	6.805E+03	2.518E+14	9.781E+03	3.619E+14
Xe-137	3.971E+02	1.469E+13	1.468E-07	5.432E+03	3.971E+02	1.469E+13
Xe-138	1.423E+03	5.265E+13	4.026E+00	1.490E+11	1.427E+03	5.280E+13
Br-83	2.318E+00	8.577E+10	1.109E+01	4.103E+11	1.341E+01	4.962E+11
Br-84	1.340E+00	4.958E+10	4.784E-01	1.770E+10	1.818E+00	6.727E+10
Br-85	1.816E-02	6.719E+08	1.358E-13	5.025E-03	1.816E-02	6.719E+08
I-129	1.451E-06	5.369E+04	2.134E-05	7.896E+05	2.279E-05	8.432E+05
I-130	2.119E+00	7.840E+10	2.472E+01	9.146E+11	2.684E+01	9.931E+11
I-131	2.406E+01	8.902E+11	3.485E+02	1.289E+13	3.726E+02	1.379E+13
I-132	2.343E+01	8.669E+11	1.077E+02	3.985E+12	1.311E+02	4.851E+12
I-133	4.811E+01	1.780E+12	6.162E+02	2.280E+13	6.643E+02	2.458E+13
I-134	2.052E+01	7.592E+11	2.249E+01	8.321E+11	4.300E+01	1.591E+12
I-135	4.045E+01	1.497E+12	3.877E+02	1.434E+13	4.282E+02	1.584E+13
Rb-86m	3.435E-06	1.271E+05	9.253E-40	3.424E-29	3.435E-06	1.271E+05
Rb-86	1.192E-01	4.410E+09	1.756E+00	6.497E+10	1.876E+00	6.941E+10
Rb-88	2.836E+03	1.049E+14	4.711E+03	1.743E+14	7.548E+03	2.793E+14
Rb-89	2.112E+02	7.814E+12	1.177E+00	4.355E+10	2.124E+02	7.859E+12
Cs-134	1.337E+01	4.947E+11	1.981E+02	7.330E+12	2.114E+02	7.822E+12
Cs-136	3.309E+00	1.224E+11	4.861E+01	1.799E+12	5.192E+01	1.921E+12
Cs-137	5.096E+00	1.886E+11	7.554E+01	2.795E+12	8.064E+01	2.984E+12
Cs-138	1.250E+03	4.625E+13	1.935E+02	7.160E+12	1.443E+03	5.339E+13
Sr-89	1.940E-01	7.178E+09	8.238E-01	3.048E+10	1.018E+00	3.767E+10
Ba-137m	4.812E+00	1.780E+11	7.146E+01	2.644E+12	7.627E+01	2.822E+12
Total	2.726E+04	1.009E+15	5.156E+04	1.908E+15	7.882E+04	2.916E+15

7.2 SEVERE ACCIDENTS

This section evaluates the potential environmental impacts of severe accidents on the Bell Bend Nuclear Power Plant (BBNPP) site from the proposed U.S. EPR plant. The environmental impacts from a postulated severe accident have been estimated using BBNPP site-specific data to demonstrate acceptability for a Combined License (COL) Application.

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 U.S. EPR, such accidents are not part of the design basis for the plant. However, the Nuclear Regulatory Commission (NRC) requires, in its Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (FR, 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.

A PRA was completed for the U.S. EPR as part of the application for design certification. This section presents the applicable results of the probabilistic risk assessment and includes site-specific characteristics of the BBNPP site and impacts of a severe accident over the entire life cycle. The purpose of this report is to identify the severe accident offsite radiological impacts, demonstrate that the impacts are acceptable, and support the severe accident mitigation alternatives analyses in Section 7.3.

7.2.1 Methodology

7.2.1.1 Offsite Consequences

The probabilistic risk assessment for the U.S. EPR established containment event trees that define the possible end states of the containment following an accident sequence. The end states are grouped into five broad categories as follows:

1. Containment intact, isolated and not bypassed (RC 101)
2. Containment bypassed (RC701, 702, 802)
3. Containment not isolated (isolation failure) (RC 201-206)
4. Early failures (excluding not isolated and bypassed) (RC 301-304, 401-404)
5. Late containment failures (RC 501-504, 602)

Using the Electric Power Research Institute code Modular Accident Analysis Program (MAAP), 23 release categories (RC) are assigned to represent all potential severe accident scenarios. It should be noted that there are a total of 25 RCs, however two of them have zero frequency and are not included in the Level 3 PRA or in the results of this analysis. The release categories are described in Table 7.2-1. An accident frequency (release category frequency) is assigned to each of the 23 categories, and these are shown in Table 7.2-3.

The NRC code MACCS2 (Sandia, 1997) was used to model the environmental consequences of the severe accidents. MACCS2 was developed specifically for NRC to evaluate severe accidents at nuclear power plants. The exposure pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water.

The MACCS2 code primarily addresses dose from the air pathway, but also calculates dose from surface runoff and deposition on surface water. The code also evaluates the extent of contamination. The meteorology data used in the analysis was hourly data for one year that includes wind velocity (speed and direction), stability class, and rainfall.

To assess human health impacts, the analysis determined the expected number of early fatalities, expected number of latent cancer fatalities, and collective whole body dose from a severe accident to the year 2050 population within a 50-mile radius of the plant. 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.

MACCS2 requires five input files: MET, SITE, ATMOS, EARLY, and CHRONC. ATMOS 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. EARLY 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. CHRONC provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors. These files access a meteorological file, which uses actual SSES Units 1 and 2 meteorological monitoring data from the years 2001 through 2007 and a site characteristics file, which uses site-specific population data, land usage, watershed index, and regions.

For the Level 3 PRA, meteorological data for the BBNPP site for the years 2001 through 2007 were reviewed to determine the years with the least unusable data points. The year 2002 satisfied this criterion and was used for the base case.

7.2.1.2 Population Data

The population used in this analysis was generated using SECPOP2000 for the year 2000. Since the BBNPP site does not have an existing reactor, the coordinates of the site were used as input into SECPOP2000. Consistent with the BBNPP Level 3 PRA, population data from the 2000 U.S. Census and SECPOP2000 were compared with Susquehanna Steam Electric Station, proximate to the BBNPP site. This comparison was used to represent the population in the 50 mi (80 km) region surrounding the BBNPP site.

The 2000 population data were escalated to 2050 by a growth rate factor of 1.0549 per decade, which was estimated based on the population growth in the region from 2000 to 2005 by county and further compared to the growth rate estimated from the SECPOP2000 1990 data. The growth rate factors were comparable and the census-based escalation factor was used as it was slightly greater and therefore would be more conservative.

BBNPP has an expected start-up date of 2018, an operating life of 40 years and a 20-year license extension, bringing anticipated end-of-life to the year 2078. Recognizing that consequences increase with increasing time (i.e., increasing population), a time-averaged consequence can be estimated by looking at the midpoint of the BBNPP operational life, thus 2050 approximates to the base year used while calculating the consequences of a severe accident. Considering just the 40-year initial license period, selecting 2050 as the base year is conservative. As a sensitivity case, the approximate endpoint of the BBNPP operational life, 2080 is also evaluated.

7.2.1.3 Risk Calculation

Release heights vary, depending on the event sequence, ranging from ground level to the top of the containment annulus. The BBNNP Level 2 PRA provides the inputs for the MACCS2 analysis, and the spectrum of accident sequences analyzed includes containment releases with durations of up to 140 hours after the start of the accident in the cases of late containment failure. The MACCS2 analysis extends out to 5 years for the assessment of post accident interdiction measures and out to 30 years for the assessment of the long term dose to individuals..

The results of the MACCS2 calculations and accident frequency information were used to determine risk. The sum of all release category frequencies is the core damage frequency and includes internal and external initiating events. External events include internal fire events and internal flood events. Risk is the set of accident sequences, their respective frequencies and their respective consequences. Risk is often more simply quantified as the sum of the products of accident sequence frequencies and consequences. The consequence can be radiation dose or economic cost. Therefore, risk can be reported as a combination of person-rem per year and dollars per year.

7.2.2 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 water ingestion with BBNPP site-specific data. MACCS2 does not model other surface water and groundwater dose pathways. These were analyzed qualitatively based on a comparison of the U.S. EPR atmospheric doses to those of the existing U.S. nuclear fleet.

The current U.S. nuclear fleet has an exceptional safety record. Through evolutionary and innovative design, the U.S. EPR has enhanced the ability to both prevent potential core damage events and to mitigate them should they occur. A list of example U.S. EPR design features which reduce plant risk is provided below.

- Increased redundancy and separation
- Four safety trains including four EFW divisions
- Separate power divisions for each safety train, each with dedicated battery division and EDG
- Two divisions each have a backup alternate AC diesel generator for SBO-type scenario
- State-of-the-art digital I&C
- Stand-still Seal System for backup to RCP seals
- Main Feedwater System with Startup and Shutdown System
- In-containment refueling water storage tank to eliminate transfer to long term recirculation
- Two, dedicated severe accident battery divisions
- Dedicated severe accident depressurization valves to prevent high pressure melt scenarios which can challenge containment due to postulated direct containment heating

- Containment combustible gas control system, including passive autocatalytic recombiners and gas mixing system
- Core stabilization system
- Passive cooling of molten core debris
- Active spray for environmental control of the containment atmosphere
- Active recirculation cooling of the molten core debris and containment atmosphere

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 U.S. EPR is less than the CDFs for the current U.S. nuclear fleet.

7.2.2.1 Air Pathways

The potential severe accidents for the U.S. EPR were grouped into 23 release categories based on their similarity of characteristics. Each release category was assigned a set of characteristics representative of the elements of that class. Each release category 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 release category, risk was calculated by multiplying each consequence (population dose, fatalities, cost, and contaminated land) with its corresponding frequency. A summary of the results are provided in Table 7.2-3. The calculation considers other consequences, 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.

7.2.2.2 Surface Water Pathways

Population 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 water. The MACCS2 severe accident dose-risk to the 50-mile population from drinking water is 9.98E-04 person-rem per year for the U.S. EPR. This value is the sum of all 23 release categories.

Surface water pathways involving swimming, fishing, and boating are not modeled by MACCS2. Surface water bodies within the 50 mi (80 km) region of BBNPP include the Susquehanna River, Lehigh River, Beltzville Lake, and other smaller bodies of water. The NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet discharging to various bodies of water in NUREG 1437, the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NRC, 1996). The NRC evaluation concluded that with interdiction, the risk associated with the aquatic food pathway is found to be small relative to the atmospheric pathway for most sites and essentially the same as the atmospheric pathway for the few sites with large annual aquatic food harvests (which does not include BBNPP). Because the U.S. EPR atmospheric pathway doses are significantly lower that those of the current U.S. nuclear fleet, the doses from surface water sources would be consistently lower for the U.S. EPR as well.

7.2.2.3 Groundwater Pathways

Population 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. The consequences of a radioactive spill not associated with an accident in COL application FSAR Section 2.4.13 have been evaluated and it has been determined that if radioactive liquids were released directly to groundwater, all isotopes would be below maximum permissible concentrations before they reached the local groundwater sources.

NUREG-1437 also evaluated the groundwater pathway dose, based on the analysis in NUREG 0440 (NRC, 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 representative sites, including the existing Susquehanna Steam Electric Station (SSES). The conclusion for those sites is that the uninterdicted population doses are significantly less than the NUREG 0440 generic site. The proposed location for BBNPP has the same groundwater characteristics as the location of the existing SSES units and the CDF for the U.S. EPR is lower than that of the existing SSES units. Therefore, the doses from the BBNPP groundwater pathway would be smaller than from the existing SSES units.

7.2.3 Conclusions

The total calculated dose-risk to the 50 mi (80 km), year 2050 estimated population from airborne releases from a U.S. EPR reactor at BBNPP is expected to be approximately 0.22 person-rem per year (Table 7.2-3). The fraction of core inventory assumed to be released in each of the release categories is also included in Table 7.2-2. The number of persons exposed to doses greater than 200 rem (2 Sv) and 25 rem (0.25 Sv) are 1.92E-05 and 2.55E-04, respectively. It must be noted that these populations exceeding a dose are only calculated by MACCS2 for the early phase of an accident, the long-term dose that could be accumulated is not included in this result. Long-term doses are mitigated by emergency response and remedial measures.

The U.S. EPR dose-risk at the BBNPP site is less than the population risk for current reactors that have undergone license renewal, and less than that for the five reactors analyzed in NUREG-1150 (NRC, 1990). As reported in NUREG-1811 (NRC, 2006), the lowest dose-risk reported for reactors currently undergoing license renewal is 0.55 person-rem per year.

The analysis indicates that risk from the water ingestion dose is small at 9.98E-04 person-rem per year. As discussed in Section 7.2.2, risks from aquatic food pathway is small compared with the atmospheric pathway of the current U.S. nuclear fleet. As discussed in Section 7.2.3, the risk of groundwater contamination from a BBNPP severe accident is one or more orders of magnitude less than the risk from the atmospheric pathway for currently licensed reactors. Additionally, interdiction could substantially reduce the groundwater pathway risks.

The probability-weighted number of cancer fatalities from a severe accident for the U.S. EPR at BBNPP is reported in Table 7.2-3 as 1.30 E-04 per year, at 50 miles from the plant. The lifetime probability of an individual dying from any cancer is 2.3 E-01 (NCHS, 2007).

7.2.4 References

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NCHS, 2007. Table C, Percentage of total deaths, death rates, age-adjusted death rates for 2004, percentage change in age-adjusted death rates from 2003 to 2004 and ratio of age-adjusted death rates by race and sex for the 15 leading causes of death for the total population in 2004: United States, National Vital Statistics Report, Volume 55, Number 19, dated August 21, 2007, National Center for Health Statistics, Website: http://www.cdc.gov/nchs/data/nvsr/nvsr55/nvsr55_19.pdf, Date accessed: December 8, 2007.

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Table 7.2-1— Release Category Descriptions

Release Category	Description
RC101	No containment failure
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with molten core-concrete interaction (MCCI), melt not flooded ex-vessel, with containment spray
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex-vessel, without containment spray
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex-vessel with containment spray
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex-vessel without containment spray
RC206	Small containment failure due to failure to isolate 2 inch or smaller lines
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex-vessel, with containment spray
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex-vessel, without containment spray
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex-vessel, with containment spray
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex-vessel, without containment spray
RC401	Containment failures after breach and up to melt transfer to the spreading area, with MCCI, without debris flooding, with containment spray
RC402	Containment failures after breach and up to melt transfer to the spreading area, with MCCI, without debris flooding, without containment spray
RC403	Containment failures after breach and up to melt transfer to the spreading area, without MCCI, with debris flooding, with containment spray
RC404	Containment failures after breach and up to melt transfer to the spreading area, without MCCI, with debris flooding, without containment spray
RC501	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, with containment spray
RC502	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, without containment spray
RC503	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, with containment spray
RC504	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, without containment spray
RC602	Long term containment failure due to basemat failure, without debris flooding, without containment spray
RC701	Steam Generator Tube Rupture with Fission Product Scrubbing
RC702	Steam Generator Tube Rupture without Fission Product Scrubbing
RC802	Interfacing System LOCA without Fission Product Scrubbing

Table 7.2-2— Source Term Input to MACCS2

	XE/KR	I	Cs	Te	Sr	Ru	La	Ce	Ba
RC101	8.8E-3	2.4E-5	2.0E-5	5.3E-5	8.5E-6	4.4E-5	2.8E-7	7.3E-7	2.4E-5
RC201	3.6E-1	1.0E-1	9.5E-2	9.6E-3	7.8E-5	1.1E-3	3.4E-6	1.7E-5	4.1E-4
RC202	7.9E-1	2.3E-2	1.5E-2	2.0E-2	2.4E-4	3.4E-3	1.9E-5	6.8E-5	2.4E-3
RC203	8.9E-1	5.3E-2	2.8E-2	1.6E-1	1.4E-4	6.8E-3	1.5E-5	2.4E-4	2.2E-3
RC204	9.5E-1	2.8E-2	1.6E-2	3.6E-2	1.7E-4	5.3E-3	1.4E-5	6.2E-5	3.2E-3
RC205	9.8E-1	5.7E-2	3.6E-2	9.3E-2	4.0E-3	9.8E-3	3.0E-4	5.3E-4	6.1E-3
RC206	1.9E-1	5.6E-3	5.0E-3	9.0E-3	1.2E-3	7.3E-3	5.5E-5	1.8E-4	4.2E-3
RC301	7.9E-1	2.3E-2	1.5E-2	2.0E-2	2.4E-4	3.4E-3	1.9E-5	6.8E-5	2.4E-3
RC302	8.9E-1	5.3E-2	2.8E-2	1.6E-1	1.4E-4	6.8E-3	1.5E-5	2.4E-4	2.2E-3
RC303	9.5E-1	2.8E-2	1.6E-2	3.6E-2	1.7E-4	5.3E-3	1.4E-5	6.2E-5	3.2E-3
RC304	9.8E-1	5.7E-2	3.6E-2	9.3E-2	4.0E-3	9.8E-3	3.0E-4	5.3E-4	6.1E-3
RC401	8.0E-1	4.6E-3	2.3E-3	3.4E-3	2.7E-3	1.5E-3	8.0E-5	3.4E-4	5.2E-3
RC402	9.7E-1	2.0E-2	1.0E-2	1.2E-2	3.8E-3	2.1E-3	1.1E-4	4.9E-4	7.3E-3
RC403	8.0E-1	4.6E-3	2.3E-3	3.4E-3	2.7E-3	1.5E-3	8.0E-5	3.4E-4	5.2E-3
RC404	9.7E-1	2.0E-2	1.0E-2	1.2E-2	3.8E-3	2.1E-3	1.1E-4	4.9E-4	7.3E-3
RC501	9.9E-1	7.7E-4	4.0E-4	1.7E-2	7.4E-6	4.4E-5	2.2E-7	7.0E-7	2.4E-5
RC502	9.9E-1	7.7E-4	4.0E-4	1.7E-2	7.4E-6	4.4E-5	2.2E-7	7.0E-7	2.4E-5
RC503	1.0E+0	4.1E-4	6.9E-5	6.1E-4	8.5E-6	4.4E-5	2.8E-7	7.3E-7	2.4E-5
RC504	1.0E+0	4.1E-4	6.9E-5	6.1E-4	8.5E-6	4.4E-5	2.8E-7	7.3E-7	2.4E-5
RC602	9.9E-1	7.7E-4	4.0E-4	1.7E-2	7.4E-6	4.4E-5	2.2E-7	7.0E-7	2.4E-5
RC701	1.1E-1	4.2E-3	4.4E-3	6.9E-3	6.0E-4	4.8E-3	2.2E-5	1.1E-4	2.7E-3
RC702	1.1E-1	8.4E-2	8.7E-2	1.4E-1	1.2E-2	9.6E-2	4.5E-4	2.2E-3	5.4E-2
RC802	9.8E-1	7.1E-1	6.9E-1	6.4E-1	1.3E-1	5.7E-1	3.9E-3	2.2E-2	3.8E-1

Table 7.2-3— U.S. EPR Severe Accidents Analysis Impacts - 50-Mile Radius and 2050 Population

		Number of Fatalities (per year) at 50 mi (80km)		Environmental Risk (per year) at 50 mi (80km)			
Release Category	**Release Category Frequency (per year)**	**Early Fatalities**	**Latent Cancers**	**Population Dose-Risk (person-rem)**	**Water Ingestion Dose-Risk (person-rem)**	**Cost (dollars)**	**Land Requiring Decontamination (acres)**
RC101	3.43E-07	0.00E+00	4.36E-06	8.99E-03	9.50E-06	1.03E+00	1.58E-04
RC201	4.98E-10	8.67E-12	6.37E-07	1.40E-03	1.74E-05	1.78E+00	7.17E-05
RC202	3.97E-14	3.86E-17	4.84E-11	1.03E-07	2.53E-10	1.04E-04	6.75E-09
RC203	1.92E-12	3.19E-14	3.61E-09	7.35E-06	2.23E-08	7.89E-03	4.22E-07
RC204	2.78E-11	7.87E-14	3.67E-08	7.65E-05	1.85E-07	7.95E-02	4.98E-06
RC205	4.08E-10	3.61E-12	8.45E-07	1.67E-03	7.14E-06	1.94E+00	9.26E-05
RC206	1.65E-08	1.32E-09	1.31E-05	2.51E-02	9.24E-05	2.11E+01	1.50E-03
RC301	1.67E-12	1.62E-15	2.04E-09	4.34E-06	1.07E-08	4.36E-03	2.84E-07
RC302	2.18E-11	3.62E-13	4.10E-08	8.35E-05	2.53E-07	8.96E-02	4.80E-06
RC303	2.30E-09	6.51E-12	3.04E-06	6.33E-03	1.53E-05	6.58E+00	4.12E-04
RC304	1.75E-08	1.55E-10	3.62E-05	7.16E-02	3.06E-04	8.31E+01	3.97E-03
RC401	1.38E-11	0.00E+00	8.03E-09	1.74E-05	3.91E-08	1.25E-02	9.72E-07
RC402	2.75E-10	0.00E+00	3.05E-07	6.63E-04	1.87E-06	6.66E-01	4.46E-05
RC403	6.82E-10	0.00E+00	3.97E-07	8.59E-04	1.93E-06	6.20E-01	4.80E-05
RC404	1.34E-08	0.00E+00	1.49E-05	3.23E-02	9.11E-05	3.24E+01	2.17E-03
RC501	5.92E-13	0.00E+00	1.02E-10	2.24E-07	1.01E-10	3.10E-05	1.73E-08
RC502	2.87E-10	0.00E+00	4.94E-08	1.08E-04	4.91E-08	1.50E-02	8.38E-06
RC503	6.01E-10	0.00E+00	1.85E-08	4.12E-05	2.05E-08	1.34E-03	1.05E-06
RC504	1.19E-07	0.00E+00	3.67E-06	8.15E-03	4.06E-06	2.65E-01	2.07E-04
RC602	6.50E-10	0.00E+00	1.12E-07	2.46E-04	1.11E-07	3.40E-02	1.90E-05
RC701	1.02E-08	1.18E-13	8.06E-06	1.55E-02	3.07E-05	1.22E+01	8.57E-04
RC702	5.38E-09	1.85E-09	3.54E-05	4.14E-02	3.21E-04	3.49E+01	1.32E-03
RC802	2.64E-10	1.22E-09	8.58E-06	7.66E-03	9.87E-05	2.67E+00	5.97E-05
Total	5.31E-07	4.56E-09	1.30E-04	2.22E-01	9.98E-04	2.00E+02	1.10E-02

7.3 SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

The purpose of the severe accident mitigation design alternatives (SAMDA) analysis is to review and evaluate both design and non-hardware (i.e., operation and maintenance programs) alternatives that could significantly reduce the radiological risk from a postulated severe accident by preventing core damage and significant releases from the containment. The U.S. EPR Design Certification Environmental Report (U.S. EPR DC ER) (AREVA, 2009) for the U.S. EPR design submitted by AREVA NP evaluated both design and non-hardware alternatives.

The primary focus of the U.S. EPR DC ER was the severe accident mitigation design alternatives (SAMDA). However, non-hardware alternatives were identified in the analysis and will be addressed when the plant design is finalized and processes and procedures are being developed for the U.S. EPR. The conclusions drawn in the U.S. EPR DC ER are applicable to BBNPP.

7.3.1 SAMDA Analysis Methodology

The methodology used to develop a comprehensive list of U.S. EPR SAMDA candidates, define the screening criteria used to categorize the SAMDA candidates, and the cost-benefit evaluation is summarized in this section based on the U.S. EPR DC ER (AREVA, 2009) for the U.S. EPR.

The comprehensive list of SAMDA candidates was developed for the U.S. EPR design by reviewing industry documents for generic PWR enhancements and considering plant-specific enhancements. The SAMDA candidates were defined as enhancements to the U.S. EPR plant that have the potential to prevent core damage and significant releases from the containment. The primary industry document supporting the development of U.S. EPR generic PWR SAMDA candidates was NEI 05-01 (NEI, 2005).

In addition to the generic SAMDA candidates, the results of the Level 1 and Level 2 PRA were reviewed to identify plant-specific modifications for inclusion in the comprehensive list of SAMDA Candidates.

The U.S. EPR top 100 core damage frequency (CDF) cutsets were evaluated to identify those modifications that would reduce the likelihood of occurrence of the significant core damage sequences. As stated in the U.S. EPR FSAR Section 19.1.4.1.2.3 (Significant Cutsets and Sequences), ninety-five percent of the total CDF is represented by over 12,000 cutsets for the U.S. EPR design; however, the top 100 cutsets include all cutsets contributing >1 percent to the total CDF. For the U.S. EPR design, this equates to approximately 50 percent of the total CDF. In fact the selection of the top 100 cutsets conservatively includes cutsets of low importance. For example, the percentage of the individual contribution to the total CDF for the 101st cutset was 0.10 percent.

The U.S. EPR top 100 large release frequency (LRF) cutsets were evaluated to identify those modifications that would reduce the likelihood of occurrence of the significant containment challenges. This population of cutsets specifically excluded the contribution to LRF of core damage sequences due to Main Steam Line Break (MSLB) inside containment with main feedwater unisolated, as this sequence of events was determined not to lead to core damage or LRF. This exclusion ensures that the conservative treatment of an event does not artificially reduce the importance of other containment failure mechanisms. The top 100 LRF cutsets include all cutsets contributing greater than 1 percent to the total LRF. For the U.S. EPR design this equates to approximately 50 percent of the total LRF, and includes many low importance cutsets that individually contribute only 0.10 percent to the total LRF.

Consistent with current regulatory guidance and industry practice, the risk significant design alternatives for the U.S. EPR design have been addressed by detailed evaluations of the top 100 CDF and LRF cutsets to identify plant-specific modifications for inclusion in the comprehensive list of U.S. EPR SAMDA candidates. Through evaluation of the top 100 Level 1 PRA cutsets, numerous U.S. EPR specific operator actions and hardware-based SAMDA candidates were developed. When evaluating the top 100 LRF cutsets no additional SAMDA candidates were identified. The U.S. EPR DC ER (AREVA, 2009) provides a detailed list of the SAMDA candidates for the U.S. EPR design. The SAMDA candidates identified in the U.S. EPR DC ER are applicable to BBNPP.

The SAMDA candidates developed for the U.S. EPR design were qualitatively screened using seven categories. The intent of the screening is to identify the candidates for further risk-benefit calculation. For each SAMDA candidate, a screening criteria and basis for screening was identified to justify the implementation or exclusion of the SAMDA candidate in the U.S. EPR design. The seven categories used during the screening process included:

- Not applicable. The SAMDA candidates were identified to determine which are definitely not applicable to the U.S. EPR design. Potential enhancements that are not considered applicable to the U.S. EPR design are those developed for systems specifically associated with boiling water reactors (BWR) or with specific PWR equipment that is not in the U.S. EPR design.

- Already implemented. The SAMDA candidates were reviewed to ensure that the U.S. EPR design does not already include features recommended by a particular SAMDA candidate. Also, the intent of a particular SAMDA candidate may have been fulfilled by another design feature or modification. In these cases the SAMDA candidates are already implemented in the U.S. EPR plant design. If a SAMDA candidate has already been implemented at the plant, it is not retained.

- Combined. If one SAMDA candidate is similar to another SAMDA candidate, and can be combined with that candidate to develop a more comprehensive or plant-specific SAMDA candidate, only the combined SAMDA candidate is retained for screening.

- Excessive implementation cost. If a SAMDA candidate requires extensive changes that will obviously exceed the maximum benefit, even without an implementation cost estimate and therefore incurs an excessive implementation cost, it is not retained.

- Very low benefit. If a SAMDA candidate is related to a non-risk significant system for which change in reliability is known to have negligible impact on the risk profile, it is deemed to have a very low benefit and is not retained.

- Not required for design certification. Evaluation of any potential procedural or surveillance action SAMDA candidates are not appropriate until the plant design is finalized and the plant procedures are being developed. Therefore, if a SAMDA candidate is related to any of these enhancements, it is not retained for this analysis.

- Considered for further evaluation. If a particular SAMDA candidate was not categorized by any of the preceding categories, then the SAMDA candidate is considered for further evaluation and subject to a cost-benefit analysis.

The screening categories were chosen based on guidance from NEI 05-01. The U.S. EPR DC ER contains a detailed description of each of the categories. The screening categories are applicable to BBNPP.

The SAMDA candidates categorized as "Not required for design certification" in the AREVA NP Environmental Report Standard Design Certification were re-evaluated for BBNPP. These SAMDA candidates were re-evaluated using the screening methdology in AREVA NP Environmental Report Standard Design Certification. An additional screening category called "Not a design alternative" was used to capture any SAMDA candidate not related to plant design. This category included SAMDA candidates related to procedure modifications, training, or surveillance. If a SAMDA candidate is related to any of these enhancements, it is not retained for this analysis.

After the screening process was completed, the SAMDA candidates that were placed in the Considered for Further Evaluation category would require a cost-benefit evaluation. The cost-benefit evaluation of each SAMDA candidate would determine the cost of implementing the specific SAMDA candidate with the maximum averted cost risk from the implementation of the specific SAMDA candidate. The maximum averted cost risk, typically referred to as the maximum benefit, equates to the cost obtained by the elimination of all severe accident risk.

7.3.2 Severe Accident Cost Impact and Maximum Benefit for BBNPP

The severe accident impact is determined by summing the occupational exposure cost, on-site cost, public exposure, and off-site property damage. The methodologies provided in NEI 05-01 (NEI, 2005) and NUREG/BR-0184 (NRC, 1997) were used as guidance. The principal inputs to the calculations were the CDF, 2,000 dollars per person-rem (NRC, 1997), licensing period of 60 years, 7% best estimate discount rate (NEI, 2005), and 3% upper bound discount rate (NEI, 2005). The maximum benefit calculation performed in the U.S. EPR DC ER used the whole body dose and economic impact from U.S. EPR Level 3 PRA analysis, which was based on population data from 2000. The maximum benefit calculation for BBNPP uses the economic impact and whole body dose for a 2050 population (Table 7.3-1). The point estimate and mean value CDF with 2008 replacement power costs severe accident impact cost for BBNPP is also shown in Table 7.3-1.

The total cost impact of a severe accident (maximum benefit) must account for the risk contribution from internal initiating events, internal flooding, fire, and seismic. The total core damage frequency (CDF) at power for the U.S. EPR design includes the contribution from internal initiating events (55%), internal flooding (12%), and fire (33%) (AREVA, 2007b). A seismic margin assessment instead of a seismic PRA was completed for the U.S. EPR design. The seismic margin analysis yields valuable information regarding the ruggedness of the seismic design with respect to the potential severe accident (AREVA, 2007b). However, it does not result in the estimation of seismic CDF which is used to determine the cost impact of a severe accident in the SAMDA analysis. In order to account for the seismic contribution, it was assumed that the seismic risk is equivalent to the fire risk since the fire risk in the U.S. EPR PRA analysis was evaluated to be the highest external event risk at 33% of the total CDF.

Increasing the severe accident impact by 33 percent includes the contribution from seismic risk and is the maximum benefit for BBNPP. The maximum benefit for BBNPP based on the point estimate CDF with 2008 replacement power costs is $72,388.

The percentage contributions of each hazards group are slightly different for the mean value CDF. Therefore, seismic risk based on the mean value CDF is assumed to be 28 percent of total mean value CDF. The resulting maximum benefit on the mean value CDF would be $92,677.

7.3.3 Sensitivity Studies

Sensitivity cases were performed to investigate the sensitivity of certain parameters in the Bell Bend SAMDA analysis. A total of five sensitivity benefit calculations were performed for both the point estimate and mean value CDF with 2008 replacement power costs. Below is a brief description of the sensitivity cases.

- The first case investigated the sensitivity of the base case to the discount rate by assuming a lower discount rate of three percent. The method to calculate the present value of replacement power for a single event is discussed in U.S. EPR DC ER (AREVA 2009).
- The second case investigated the sensitivity of the base case to the discount rate by assuming a lower discount rate of five percent.
- The third case investigated the sensitivity of the base case to the on-site dose estimates. For the base case analysis, an immediate and long-term on-site dose to plant personnel following a severe accident is 3,300 rem and 20,000 rem, respectively. Therefore, this sensitivity case used the recommended high estimate dose values of 14,000 rem and 30,000 rem for immediate and long term dose on-site respectively, as suggested in (NRC, 1997).
- The fourth case investigated the sensitivity of the base case to the total on-site cleanup cost. For the base case analysis, the total on-site cleanup cost following a severe accident is taken to be $1,500,000. Therefore, this analysis assumed a high estimated on-site cleanup cost of $2,000,000 as suggested in (NRC, 1997).
- The fifth case also investigated the sensitivity of the increase in the replacement power cost for the U.S. EPR design. This sensitivity case projected that the cost of replacement power would double between 2008 and 2015. This would result in electricity cost of 24 cents/kw-h in 2015 based upon the assumption that the cost of electricity in 2008 is 12 cents/kw-h. The inflation rate for this sensitivity case was calculated using the the method outlined in (AREVA, 2009).

Table 7.3-2 and Table 7.3-3 provide the calculated benefit for the point estimate and mean value CDF with 2008 replacement power cost sensitivity cases discussed above.

7.3.4 Results and Summary

A total of 167 SAMDA candidates developed from industry and U.S. EPR documents were evaluated in the U.S. EPR DC ER completed by AREVA NP. The basis for screening is provided in detail for each SAMDA candidate in the U.S. EPR DC ER. Below is a summary of the results of the SAMDA analysis performed for the U.S. EPR and is applicable to BBNPP.

- Twenty-five SAMDA candidates were not applicable to the U.S. EPR design.
- Sixty-nine SAMDA candidates were already implemented into the U.S. EPR design either as suggested in the SAMDA or an equivalent replacement that fulfilled the intent of the SAMDA. These SAMDA candidates are summarized in Table 7.3-4.

- Four SAMDA candidates were combined with another SAMDA because they had the same intent.
- Fourty-three SAMDA candidates were categorized as not a design alternative because they were related to procedure modifications, training, or surveillance.
- One SAMDA candidate was categorized as very low benefit.
- Twenty-five SAMDA candidates were categorized as excessive implementation cost.
- None of the SAMDA candidates were categorized as consider for further evaluation.

The low probability of core damage events in the U.S. EPR coupled with reliable severe accident mitigation features provide significant protection to the public and the environment. Specific severe accident mitigation design alternatives from previous industry studies, and from U.S. EPR probabilistic risk assessment (PRA) insights, were measured against broad acceptance criteria in the U.S. EPR DC ER (AREVA, 2009). Since none of the SAMDA candidates were categorized as considered for further evaluation, a cost-benefit analysis (i.e., risk reduction, value impact ratios) was not required for the U.S. EPR SAMDA analysis. The overall conclusion of the U.S. EPR SAMDA analysis is that no additional plant modifications are cost beneficial to implement due to the robust design of the U.S. EPR with respect to prevention and mitigation of severe accidents. The maximum benefit from the U.S. EPR DC ER was reevaluated for BBNPP. The detailed analysis and conclusions in the U.S. EPR DC ER remain applicable for BBNPP.

7.3.5 References

AREVA, 2009. AREVA NP Environmental Report Standard Design Certification, ANP-10290, Revision 1, AREVA NP, September 2009.

AREVA, 2007b. AREVA NP U.S. EPR Final Safety Analysis Report, Revision 0, AREVA NP, December 2007.

NEI, 2005. Severe Accident Mitigation Alternatives (SAMA) Analysis, Guidance Document, NEI 05-01, Revision A, Nuclear Energy Institute November 2005.

NRC, 1997. Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184, Nuclear Regulatory Commission, January 1997.

Table 7.3-1— Severe Accident Cost Impact

	Point Estimate CDF (7% Discount Rate and 2008 Replacement Power Costs)	Mean Value CDF (7% Discount Rate and 2008 Replacement Power Costs)
Averted Occupational Exposure (AREVA, 2007a)	$264	$369
Averted Onsite Costs (AREVA, 2007a)	$45,102	$62,974
Averted Public Exposure	$6,247	$6,247
Averted Offsite Property Damage Costs	$2,814	$2,814
Severe Accident Cost Impact[(a)] Internal Events, Internal Flooding, Fire	$54,427	$72,404
Maximum Benefit[(b)] Internal Events, Internal Flooding, Fire, Seismic	$72,388	$92,677

Notes:
(a) Severe Accident Cost Impact is the sum of the Averted Occupational Exposure, Averted Onsite Cost, Averted Public Exposure and Averted Offsite Property Damage Cost.
(b) Maximum Benefit is calculated by increasing the Severe Accident Cost Impact by 33%.

Table 7.3-2— Maximum Benefit for Sensitivity Cases (Point Estimate CDF with 2008 Replacement Power Costs)

Case	Sensitivity Case 1: Discount Rate 3%	Sensitivity Case 2: Discount Rate - 5%	Sensitivity Case 3: High Estimated Dose (On-Site)	Sensitivity Case 4: High On-site Cleanup Costs	Sensitivity Case 5: Increase Replacement Power Cost via Inflation for 2015 Dollars
Immediate Dose Savings (On-site)	$97	$66	$209	$49	$49
Long Term Dose Savings (On-site)	$510	$317	$322	$215	$215
Total Accident Related Occupational Exposure **(AOE)**	$607	$384	$531	$264	$264
Cleanup/ Decontamination Savings (On-site)	$19,110	$13,053	$8,045	$10,727	$8,045
Replacement Power Savings (On-site)	$129,243	$62,524	$36,835	$36,835	$73,675
Averted Costs of On-site Property Damage **(AOSC)**	$148,353	$75,577	$44,880	$47,562	$81,720
Total On-site Benefit	**$148,960**	**$75,960**	**$45,411**	**$47,826**	**$81,984**
Averted Public Exposure **(APE)**	$12,354	$8,438	$6,248	$6,248	$6,248
Averted Offsite Damage Savings **(AOC)**	$5,565	$3,801	$2,814	$2,814	$2,814
Total Offsite Benefit	**$17,918**	**$12,239**	**$9,062**	**$9,062**	**$9,062**
Total Benefit (On-site + Offsite)	**$166,878**	**$88,199**	**$54,473**	**$56,888**	**$91,046**
Total Benefit (On-site + Offsite + External Events)	**$221,947**	**$117,305**	**$72,449**	**$75,611**	**$121,091**

Table 7.3-3— Maximum Benefit for Sensitivity Cases (Mean Value CDF with 2008 Replacement Power Costs)

Case	Sensitivity Case 1: Discount Rate 3%	Sensitivity Case 2: Discount Rate - 5%	Sensitivity Case 3: High Estimated Dose (On-Site)	Sensitivity Case 4: High On-site Cleanup Costs	Sensitivity Case 5: Increase Replacement Power Cost via Inflation for 2015 Dollars
Immediate Dose Savings (On-site)	$136	$93	$292	$69	$69
Long Term Dose Savings (On-site)	$712	$443	$449	$300	$300
Total Accident Related Occupational Exposure **(AOE)**	$847	$535	$741	$368	$368
Cleanup/ Decontamination Savings (On-site)	$26,682	$18,225	$11,233	$14,977	$11,233
Replacement Power Savings (On-site)	$180,452	$87,298	$51,430	$51,430	$102,867
Averted Costs of On-site Property Damage **(AOSC)**	$207,134	$105,522	$62,663	$66,407	$114,100
Total On-site Benefit	**$207,981**	**$106,058**	**$63,404**	**$66,775**	**$114,468**
Averted Public Exposure **(APE)**	$12,354	$8,438	$6,248	$6,248	$6,248
Averted Offsite Damage Savings **(AOC)**	$5,565	$3,801	$2,814	$2,814	$2,814
Total Offsite Benefit	$17,918	$12,239	$9,062	$9,062	$9,062
Total Benefit (On-site + Offsite)	**$225,900**	**$118,297**	**$72,466**	**$75,837**	**$123,530**
Total Benefit (On-site + Offsite + External Events)	**$289,151**	**$151,420**	**$92,756**	**$97,072**	**$158,118**

Table 7.3-4— SAMDA Candidates - Already Implemented

(Page 1 of 2)

SAMDA ID	Potential Enhancement
AC/DC-01	Provide additional DC battery capacity.
AC/DC-03	Add additional battery charger or portable, diesel-driven battery charger to existing DC system.
AC/DC-04	Improve DC bus load shedding.
AC/DC-05	Provide DC bus crossties
AC/DC-06	Provide additional DC power to the 120/240V vital AC system.
AC/DC-07	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.
AC/DC-09	Provide an additional diesel generator.
AC/DC-11	Improve 4.16 kV bus cross-tie ability.
AC/DC-14	Install a gas turbine generator.
AC/DC-16	Improve uninterruptible power supplies.
AC/DC-24	Bury off-site power lines.
AT-01	Add an independent boron injection system.
AT-02	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.
AT-07	Install motor generator set trip breakers in control room.
AT-08	Provide capability to remove power from the bus powering the control rods.
CB-01	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.
CB-04	Install self-actuating containment isolation valves.
CB-10	Replace SGs with a new design.
CB-12	Install a redundant spray system to depressurize the primary system during an SGTR.
CB-14	Provide improved instrumentation to detect SGTR, such as Nitrogen-16 monitors.
CB-16	Install a highly reliable (closed loop) SG shell-side heat removal system that relies on natural circulation and stored water sources.
CB-20	Install relief valves in the CCWS.
CC-01	Install an independent active or passive high pressure injection system.
CC-04	Add a diverse low pressure injection system.
CC-05	Provide capability for alternate injection via diesel-driven fire pump.
CC-06	Improve ECCS suction strainers.
CC-07	Add the ability to manually align ECCS recirculation.
CC-10	Provide an in-containment reactor water storage tank.
CC-15	Replace two of the four electric safety injection pumps with diesel-powered pumps.
CC-17	Create a reactor coolant depressurization system.
CC-21	Modify the containment sump strainers to prevent plugging.
CP-01	Create a reactor cavity flooding system.
CP-03	Use the fire water system as a backup source for the containment spray system.
CP-07	Provide post-accident containment inerting capability.
CP-08	Create a large concrete crucible with heat removal potential to contain molten core debris.
CP-11	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.
CP-13	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.
CP-17	Install automatic containment spray pump header throttle valves.
CP-20	Install a passive hydrogen control system.
CP-21	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.
CP-22	Install a secondary containment filtered ventilation.
CW-01	Add redundant DC control power for SW pumps.
CW-02	Replace ECCS pump motors with air-cooled motors.
CW-04	Add a SW pump.
CW-05	Enhance the screen wash system.

Table 7.3-4— SAMDA Candidates - Already Implemented

(Page 2 of 2)

SAMDA ID	Potential Enhancement
CW-06	Cap downstream piping of normally closed component cooling water drain and vent valves.
CW-10	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.
CW-15	Use existing hydro test pump for RCP seal injection.
CW-16	Install improved RCP seals.
CW-17	Install an additional component cooling water pump.
EPR-01	Provide an additional SCWS train.
EPR-05	Add redundant pressure sensors to the pressurizer and SG.
FR-03	Install additional transfer and isolation switches.
FR-05	Enhance control of combustibles and ignition.
FW-01	Install a digital feed water upgrade.
FW-02	Create ability for emergency connection of existing or new water sources to feedwater and condensate systems.
FW-04	Add a motor-driven feedwater pump.
FW-07	Install a new condensate storage tank (auxiliary feedwater storage tank).
FW-11	Use fire water system as a backup for SG inventory.
FW-15	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.
HV-01	Provide a redundant train or means of ventilation to the switch gear rooms.
HV-02	Add a diesel building high temperature alarm or redundant louver and thermostat.
HV-03	Stage backup fans in switchgear rooms.
HV-04	Add a switchgear room high temperature alarm.
HV-05	Create ability to switch EFW room fan power supply to station batteries in an SBO.
SR-01	Increase seismic ruggedness of plant components.
SR-02	Provide additional restraints for CO2 tanks.
OT-01	Install digital large break LOCA protection system.
OT-03	Install computer aided instrumentation system to assist the operator in assessing post-accident plant status.

7.4 TRANSPORTATION ACCIDENTS

The NRC evaluated the environmental effects of transportation of fuel and waste for light water reactors in WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Plants" (AEC, 1972) and NUREG-75/038, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, Supplement 1" (NRC, 1975) and found the impacts to be small. These NRC analyses provided the basis for Table S-4 in 10 CFR 51.52 (CFR, 2007) which summarizes the environmental impacts of transportation of fuel and radioactive wastes to and from a reference reactor.

10 CFR 51.52 requires that:

> Every environmental report prepared for … a light-water-cooled nuclear power reactor… contain a statement concerning transportation of fuel and radioactive wastes to and from the reactor. That statement shall indicate that the reactor and this transportation either meet all of the conditions in paragraph (a) of this section or all of the conditions in paragraph (b) of this section.

Table S-4 of 10 CFR 51.52 addresses two categories of environmental considerations: (1) normal conditions of transport and (2) accidents in transport.

The U.S. EPR design varies from the conditions of 10 CFR 51.52(a). Specifically,

- The reactor has a core thermal power level exceeding 3,800 MWth,
- The reactor fuel has a uranium-235 enrichment that may exceed 4% by weight,
- The uranium dioxide pellets are not encapsulated in Zircaloy rods,
- The average level of burnup of the irradiated fuel removed from the reactor will exceed 33,000 MWd/MTU.

Because the U.S. EPR varies from the conditions of 10 CFR51.52(a), a full description and analysis of transportation environmental impacts is required in accordance with 10 CFR 51.52(b). This section describes the environmental impact of postulated transportation accidents involving the shipment of radioactive materials including unirradiated (new) fuel, irradiated fuel, and radioactive waste as required by 10 CFR 51.52. The environmental impacts from the incident-free transportation of fuel and wastes to and from the new reactor is summarized in Section 5.11.

These evaluated impacts are compared to the respective impacts in 10 CFR 51.52 as shown in Table 7.4-1.

Radiological and non-radiological types of accident effects are analyzed. Two computer programs were used to perform this analysis. The TRAGIS (ORNL, 2003) computer code was used to determine the distance traveled by truck, the roads taken, and the population density along the routes. The RADTRAN 5.6 computer code was used to calculate population doses from the shipment (direct and effluent sources, not ingestion) given the routes defined by TRAGIS. The inputs to these codes are listed in Table 7.4-2 through Table 7.4-6 and Table 7.4-8 through Table 7.4-10.

7.4.1 Radiological Impacts

The radiological impact population dose was calculated using the RADTRAN computer code. The population dose impact from postulated accidents associated with the transportation of unirradiated fuel, irradiated fuel, and radioactive waste are provided in Table 7.4-11. The dose impact from all postulated transportation accident sources is 1.5E-4 person-rem/year (1.5E-6 person-Sv/year).

7.4.1.1 Unirradiated (New) Fuel

The WASH-1238 analysis (AEC, 1972) of postulated accidents during the transportation of unirradiated fuel found accident impacts to be negligible. The analysis states "the impact on the environment from radiation in transportation accidents involving unirradiated (current) fuel is considered to be negligible."

Additionally, as noted in NUREG-1815 (NRC, 2006), accident frequencies are likely to be lower in the future than those used in the analysis in WASH-1238 (AEC, 1972) because traffic accident, injury, and fatality rates have fallen since the initial analyses were performed.

Finally, advanced fuel behaves like fuel evaluated in the analyses provided in WASH-1238 (AEC, 1972). Again as noted in NUREG-1815 (NRC, 2006), there is no significant difference in the consequences of accidents severe enough to result in a release of unirradiated fuel particles to the environment between advanced LWRs and previous-generation LWRs because the fuel form, cladding, and packaging are similar to those analyzed in WASH-1238 (AEC, 1972).

Based on this information, the dose impact from nuclides released from postulated accidents involving new fuel is assumed to be negligible when compared to dose impact from postulated irradiated fuel and radiation waste transportation accidents. Therefore, quantitative analysis of dose from new fuel accidents was not performed.

7.4.1.2 Irradiated Fuel

The dose impact from postulated accidents during the shipment of irradiated fuel was evaluated using the TRAGIS code (ORNL, 2003) to define appropriate routing and population density along the route. This information was used as input to the RADTRAN code with U.S. EPR-specific design information to calculate a postulated annual dose from irradiated fuel transportation accidents.

The evaluation model assumed that irradiated fuel will be shipped to the site of the proposed Yucca Mountain repository. The distance from the Bell Bend Nuclear Plant (BBNPP) site to the proposed repository is 2,540.8 mi (4,089.5 km) based on a TRAGIS Highway Route Controlled Quantity (HRCQ) distance.

The model accident rate is the probability that an accident will occur during the trip along each route through each state. The route's average accident rate is the sum of the distance weighted accident rate through each state.

State-specific accident data from Table 4 of ANL/ESD/TM-150 (ANL, 1999) are shown in Table 7.4-4. Only the interstate data are used because the HRCQ route is mainly on Interstate roads.

The distance and demographic data for input to RADTRAN are listed in Table 7.4-2. The U.S. EPR average annual quantity of irradiated fuel shipped is assumed, consistent with

NUREG-1815 (NRC, 2006), to equal the average annual reload quantity. For the U.S. EPR this is 37.5 MTU of irradiated fuel per year (as provided in Section 5.11) to be shipped.

The source term in Table 7.4-3 is based on an equilibrium burnup of 52 GWd/MTU. The activity was decayed 5 years to account for the minimum decay period prior to shipment of irradiated fuel to the proposed geologic repository at Yucca Mountain, NV. The nuclides evaluated are those dominant nuclides described and listed in Appendix G of NUREG-1815 (NRC, 2006).

In addition to the source term assumed above, Cobalt-60 was used to represent fuel surface contamination and added at a level of 0.2 Ci/rod. This use of Cobalt-60 in the model was consistent with previously performed studies (SNL, 1991) (NRC, 2000) (DOE, 2002) that quantified fuel rod contamination levels and that concluded the maximum contribution from contamination is Cobalt-60. NUREG/CR-6672 estimated the maximum contamination from Cobalt-60 for PWR fuel at zero year decay is 0.168 Ci/rod (6.22E9 Bq/rod) (or approximately 0.2 Ci/rod (7.4E9 Bq/rod)). A U.S. EPR-specific calculation of Ci/rod was carried out that confirmed the 0.2 Ci/rod (7.4E9 Bq/rod) value was conservative.

The accident severity categories and related releases from Appendix G of NUREG-1815 (NRC, 2006) were used and are presented in Table 7.4-5. The model deposition velocities were consistent with Appendix E in DOE/EIS-0250 (DOE, 2002). The model severity fractions, release fractions, aerosol and respirable fractions are the conditional probabilities, given an accident occurs, for specific severity categories. The model severity and release fractions are for the 19 severity categories and the 5 chemical groups identified in NUREG-1815 (NRC, 2006), and are presented in Table 7.4-5. Gases are not deposited and have a 0.0 m/s deposition velocity. All other chemical groups are defined consistent with DOE/EIS-0250 (DOE, 2002) at 0.03 ft/s (0.01 m/s). Other RADTRAN parameters used were the default values from the RADCAT 2.3 User Guide (SNL, 2006), and from Appendix G of NUREG-1815 (NRC, 2006).

The evaluation determined that the dose impact from postulated transportation accidents involving irradiated fuel was 3.59E-06 person-rem/MTU (3.59E-08 person-Sv/MTU). Using the average annual reload requirements for a U.S. EPR of 37.5 MTU, the annual population dose impact is 1.4E-04 person-rem/year (1.4E-06 person-Sv/year) from postulated transportation accidents involving irradiated fuel.

7.4.1.3 Radioactive Waste

The population risk from radwaste transportation accidents is 5.3E-06 person-rem/yr (5.3E-08 person-Sv/year). (Table 7.1-11) This is the population dose for an accident divided by the mean number of years between accidents.

The TRAGIS computer code was used to calculate the routes, distances, and demographics along the route. It was conservatively assumed that all radwaste would be shipped to the farthest disposal repository in commercial mode. The route was from the plant to the Hanford site located in Washington State. It was along roads which allowed trucks and avoided ferry crossings. TRAGIS calculated the total one-way distance to be 2,639.7 mi (4,248.0 km). The distances through each state are listed in Table 7.4-10. The distances and population densities through the rural, suburban and urban settings are listed in Table 7.4-8 as well as the time spent stopped. These were all used as inputs to RADTRAN.

The RADTRAN computer code was used to calculate accident probability and population risk for the route. In an average year 1.99E+03 Ci (7.37E+13 Bq), is forecast to be shipped. This is described in Table 7.4-7 and will involve 15 shipments per year (as described in Section 5.11).

The fraction of various nuclides released, by accident category, are listed in Table 7.4-5. These release fractions are a function of 19 accident severity categories and 5 chemical groups. The values are from NUREG-1815 (NRC, 2006). The model release fractions, aerosol and respirable fractions are the conditional probabilities, given an accident occurs, for specific severity categories.

The model deposition velocities are consistent with Appendix E in DOE/EIS-0250 (DOE, 2002). All chemical groups are defined at 0.01 m/s.

Other RADTRAN parameters were the default values from the RADCAT 2.3 User Guide (SNL, 2006), and from Appendix G of NUREG-1815 (NRC, 2006).

The source term in Table 7.4-8 is based on the sum of all waste type expected (average) annual activities. The radionuclides chosen are >1% of the total activity (with the exception of Ag-110m, which is not in the RADTRAN 5.6 Library), and those in Table G-9 of NUREG-1815 (NRC, 2006) plus isotopes in the same family (such as Co-58 and Ru-103). On page G-23 of that report the NRC performed a screening analysis that showed that these were the dominant nuclides.

The model accident rate is the probability that an accident will occur during the trip along each road through each state. The route's average accident rate is the sum of the distance weighted accident rate through each state. Table 7.4-10 presents the individual state accident rate data compiled from ANL/ESD/TM-150 (ANL, 1999) and the associated average rate. Since the commercial route is mainly on Interstate roads, only the interstate rate data was used in the model.

The result from RADTRAN is the annual population dose per year of 5.3E-06 person-rem/yr (5.3E-08 person-Sv/yr).

7.4.2 Non-Radiological Impacts

Two non-radiological impacts associated with the postulated accidents during transportation of new fuel, irradiated fuel, and radioactive waste were calculated, the fatal injury rate per 100 reactor years and the nonfatal injury rate per 10 reactor years.

7.4.2.1 New Fuel

TRAGIS (ORNL, 2003) was used to calculate the commercial routing through each state. Interstate travel is the dominant road designation and was used for all route types. It was assumed that all shipments came from the fuel fabrication facility furthest from BBNPP located in Richland, WA.

As described in Section 5.11.3.1, the average number of new fuel shipments was assumed to be 7.5 per year, each covering the 2,628.4 mi (4,229.9 km) distance, including the return of the empty truck the same distance. This is based on the distances and road types from the calculation of radiological impacts above and the fatal injury rates from Table 4 of ANL/ESD/TM 150 (ANL, 1999).

Based on the above and the average fatality rate from Table 7.4-6 of 1.67E-08 fatalities/truck-mi (1.04E-08 fatalities/truck-km), the non-radiological fatal injury rate impact associated with postulated accidents as a result of new fuel shipments is 6.6E-02 per 100 reactor years.

Based on the same routes, distances, and assumptions above and the average nonfatal injury rate from Table 7.4-6 of 3.52E-07 nonfatal injuries/truck-mi (2.19E-07 nonfatal injuries/truck-km), the non-radiological nonfatal injury rate impact associated with postulated accidents as a result of new fuel shipments is 1.5E-01 (Table 7.4-12) nonfatal injuries per 10 reactor years.

7.4.2.2 Irradiated Fuel

The methodology for evaluating the fatal and nonfatal injury rates as a result of postulated accidents during the transportation of irradiated fuel is the same as that described in Section 7.4.2.1 above with the exceptions of the number of trips and the routing assumed in the TRAGIS evaluation. Twenty-one irradiated fuel shipments from the BBNPP site to the proposed Yucca Mountain repository per year were evaluated (as discussed in Section 5.11) and the TRAGIS Highway Route Controlled Quantity was utilized as the basis to calculate the shipping distance.

Based on the above and the accident rates from Table 7.4-4, the non-radiological fatal injury rate impact associated with postulated accidents as a result of irradiated fuel shipments is 1.8E-01 (Table 7.4-12) per 100 reactor years.

Based on the above and the accident rates from Table 7.4-4, the non-radiological nonfatal injury rate impact associated with postulated accidents as a result of irradiated fuel shipments is 4.1E-01 (Table 7.4-12) nonfatal injuries per 10 reactor years.

7.4.2.3 Radioactive Waste

The fatal injury rate for accidents associated with radwaste shipments is 1.1E-01 (Table 7.4-12) fatal injuries per 100 reactor years. This is based on the fatality rates from Table 4 of ANL/ESD/TM-150 (ANL, 1999). TRAGIS was used to calculate the commercial routing through each state. Interstate travel is the dominant road designation and was used for all route types.

It is assumed that all shipments go from the BBNPP site to the farthest potential disposal repository located in Hanford, WA 2,639.7 mi (4,248.0 km) and that the truck conservatively returns to the plant empty (doubling the traveled distance.) The state-specific fatality rates are in Table 7.4-10. The number of radwaste shipments from the site to Hanford per year is 15 as described in Section 5.11.3.3. The distance weighted fatality rate from Table 7.4-10 is 1.32E-08 fatalities/truck-mi (8.23E-09 fatalities/truck-km). The Radwaste Fatality (SFF) rate was calculated to be 1.1E-01 fatal injuries/100 reactor years (Table 7.4-12).

The nonfatal injury rate associated with radwaste shipments is 2.8E-01 nonfatal injuries per 10 reactor years. This is based on the distances and road types from the radiological impact calculations and the injury rates from Table 4 of ANL/ESD/TM-150 (ANL, 1999). TRAGIS was used to calculate the commercial routing through each state. Interstate travel is the dominant road designation and was used for all route types.

It is assumed that all shipments go from the site to the farthest potential disposal repository located in Hanford, WA 2,639.7 mi (4,248.0 km) and that the truck conservatively returns to plant empty (doubling the traveled distance.) The state-specific fatality rates are in Table 7.4-10. The number of radwaste shipments from the site to Hanford per year is 15 as described in Section 5.11.3.3. The average injury rate from Table 7.4-10 is 3.57E-07 injuries/truck-mi (2.22E-07 injuries/truck-km). The nonfatal Radwaste Injury rate was calculated to be 2.8E-01 nonfatal injuries/10 reactor years.

7.4.3 Summary and Conclusion

A detailed accident analysis of the environmental impacts for the transportation of unirradiated fuel, irradiated fuel, and radioactive waste (DOE, 1981) transported to and from the BBNPP site has been performed in accordance with 10 CFR 51.52(b) (CFR, 2007).

Table 7.4-11 summarizes the radiological impact, and Table 7.4-12 summarizes the non-radiological impact. These environmental impact results are bounded by 10 CFR 51.52(c) (CFR, 2007), Table S-4. These impacts represent the contribution of postulated transportation accidents to the environmental costs of operating the proposed facility.

As shown in Table 7.4-12, the calculated impacts from transportation accidents are less than those corresponding impacts listed in Table S-4 of 10 CFR 51.52 (CFR, 2007). Therefore the corresponding impacts from transportation accidents for the transportation of fuel and waste to and from the proposed facility are SMALL and will be less than those accepted by 10 CFR 51.52 (CFR, 2007).

7.4.4 References

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Table 7.4-1— 10 CFR 51.52 Summary Table S-4 Excerpt Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor Accidents in Transport

Types of Effects	Environmental Risk
Radiological Effects	**Small**
Common (nonradiological) causes	1 fatal injury in 100 reactor years 1 nonfatal injury in 10 reactor years $475 property damage per reactor year

Table 7.4-2— RADTRAN/TRAGIS Model Irradiated Fuel Input Parameters

Parameter	BBNPP Model U.S. EPR (English Units)	BBNPP Model U.S. EPR (SI Units)
TRAGIS Input:		
Route Mode	HRCQ	
Route Origin	BBNPP	
Route Destination	Yucca Mt, NV	
RADTRAN Input TRAGIS:		
Total Shipping Distance	2,541.2 mi	4,089.5 km
Travel Distance - Rural	2, 017.4 mi	3,246.7 km
Travel Distance - Suburban	469.8 mi	756.0 km
Travel Distance - Urban	54.0 mi	87.0 km
Population Density - Rural	28.7 person/mi^2	11.1 person/km^2
Population Density - Rural	765.9 person/mi^2	295.7 person/km^2
Population Density - Rural	6,082.0 person/mi^2	2,348.3 person/km^2
Stop Time, hr/trip	4.5 (1)	
RADTRAN Input from NRC Models (2)		
Vehicle Speed	55.0 mph	88.49 km/hr
Traffic Count - Rural, vehicles/hr	530	
Traffic Count - Suburban, vehicles/hr	760	
Traffic Count - Urban, vehicles/hr	2,400	
Dose Rate at 3.3 ft (1 m) from Vehicle	14 mrem/hr	0.14 mSv/hr
Packaging Length	17.1 ft	5.2 m
Packaging Diameter	3.3 ft	1.0 m
Number of Truck Crew	2	
Population Density at Stops (radii: 3.3 to 33 ft (1 to 10 m))	77,699.6 person/mi^2	30,000 person/km^2
Population Density at Stops (radii: 33 to 2,625 ft (10 to 800 m))	880.6 person/mi^2	340 person/km^2
Shielding Factor at Stops (radii: 3.3 to 33 ft (1 to 10 m))	1	
Shielding Factor at Stops (radii: 33 to 2,625 ft (10 to 800 m))	0.2	

Notes:
(1) Based on TRAGIS output: 9 stops at 30 minutes each.
(2) From NUREG-1815.

Table 7.4-3— Irradiated Fuel Source Term

Radionuclide	BBNPP Model U.S. EPR 5 Year Decay Ci/MTU	BBNPP Model U.S. EPR 5 Year Decay Bq/MTU
Am-241	1.25E+03	4.62E+13
Am-242m	2.38E+01	8.82E+11
Am-243	3.22E+01	1.19E+12
Cm-144	1.52E+04	5.62E+14
Cm-242	4.35E+01	1.61E+12
Cm-243	3.19E+01	1.18E+12
Cm-244	4.84E+03	1.79E+14
Cm-245	6.19E-01	2.29E+10
Co-60	7.59E+01	2.81E+12
Cs-134	5.84E+04	2.16E+15
Cs-137	1.42E+05	5.25E+15
Eu-154	1.16E+04	4.31E+14
Eu-155	5.73E+03	2.12E+14
I-129	4.65E-02	1.72E+09
Kr-85	1.05E+04	3.88E+14
Pm-147	3.54E+04	1.31E+15
Pu-238	6.95E+03	2.57E+14
Pu-239	4.24E+02	1.57E+13
Pu-240	7.24E+02	2.68E+13
Pu-241	1.17E+05	4.34E+15
Pu-242	2.28E+00	8.44E+10
Ru-106	2.05E+04	7.59E+14
Sb-125	5.35E+03	1.98E+14
Sr-90	1.03E+05	3.81E+15
Y-90	1.03E+05	3.82E+15

Table 7.4-4— Irradiated Fuel BBNPP Model Accident, Fatality & Injury Rates

	Accident Rate	Fatality Rate	Injury Rate	Distance	Distance Weighted Fraction		
State	Accidents / trk-km (Accidents / trk-mi)	Fatalities / trk-km (Fatalities / trk-mi)	Injuries / trk-km (Injuries / trk-mi)	km (mi)	Accident Rate	Fatality Rate	Injury Rate
AZ	1.32E-07	9.40E-09	1.17E-07	47	1.52E-09	1.08E-10	1.34E-09
	(2.12E-07)	(1.51E-08)	(1.88E-07)	(29.2)			
IA	1.12E-07	9.40E-09	8.60E-08	494.2	1.35E-08	1.14E-09	1.04E-08
	(1.80E-07)	(1.51E-08)	(1.38E-07)	(307.1)			
IL	2.22E-07	8.30E-09	1.50E-07	261.7	1.42E-08	5.31E-10	9.60E-09
	(3.57E-07)	(1.34E-08)	(2.41E-07)	(162.6)			
IN	2.25E-07	6.70E-09	1.40E-07	243.3	1.34E-08	3.99E-10	8.33E-09
	(3.62E-07)	(1.08E-08)	(2.25E-07)	(151.2)			
NE	3.19E-07	1.37E-08	1.97E-07	734.8	5.73E-08	2.46E-09	3.54E-08
	(5.13E-07)	(2.20E-08)	(3.17E-07)	(456.6)			
NV	2.25E-07	6.60E-09	1.48E-07	269.6	1.48E-08	4.35E-10	9.76E-09
	(3.62E-07)	(1.06E-08)	(2.38E-07)	(167.5)			
OH	1.64E-07	3.90E-09	1.40E-07	379.3	1.52E-08	3.62E-10	1.30E-08
	(2.64E-07)	(6.28E-09)	(2.25E-07)	(235.7)			
PA	5.14E-07	1.35E-08	3.83E-07	404.8	5.09E-08	1.34E-09	3.79E-08
	(8.27E-07)	(2.17E-08)	(6.16E-07)	(251.5)			
UT	2.90E-07	1.19E-08	2.53E-07	610.3	4.33E-08	1.78E-09	3.78E-08
	(4.67E-07)	(1.92E-08)	(4.07E-07)	(379.2)			
WY	6.74E-07	1.08E-08	3.23E-07	644.5	1.06E-07	1.70E-09	5.09E-08
	(1.08E-06)	(1.74E-08)	(5.20E-07)	(400.5)			
Sum:				4089.5	3.30E-07	1.02E-08	2.14E-07
				(2541.2)			
Fatalities per Accident [(a)] :						3.10E-02	
Note: (a) Fatalities per accident = Fatality Rate / Accident Rate.							

Table 7.4-5— Irradiated Fuel and Radwaste Models Severity & Release Fractions

		Release Fractions[a]				
Severity Category	**Severity Fraction**	**Gas**	**Cesium**	**Ruthenium**	**Particulate**	**Corrosion Product**
0	1.53E-08	0.8	2.40E-08	6.00E-07	6.00E-07	2.00E-03
1	5.88E-05	0.14	4.10E-09	1.00E-07	1.00E-07	1.40E-03
2	1.81E-06	0.18	5.40E-09	1.30E-07	1.30E-06	1.80E-03
3	7.49E-08	0.84	3.60E-05	3.80E-06	3.80E-06	3.20E-03
4	4.65E-07	0.43	1.30E-08	3.20E-07	3.20E-07	1.80E-03
5	3.31E-09	0.49	1.50E-08	3.70E-07	3.70E-07	2.10E-03
6	0	0.85	2.70E-05	2.10E-06	2.10E-06	3.10E-03
7	1.13E-08	0.82	2.40E-08	6.10E-07	6.10E-07	2.00E-02
8	8.03E-11	0.89	2.70E-08	6.70E-07	6.70E-07	2.20E-03
9	0	0.91	5.90E-06	6.80E-07	6.80E-07	2.50E-03
10	1.44E-10	0.82	2.40E-08	6.10E-07	6.10E-07	2.00E-03
11	1.02E-12	0.89	2.70E-08	6.70E-07	6.70E-07	2.20E-03
12	0	0.91	5.90E-06	6.80E-07	6.80E-07	2.50E-03
13	7.49E-11	0.84	9.60E-05	8.40E-05	1.80E-05	6.40E-03
14	0	0.85	5.50E-05	5.00E-05	9.00E-06	5.90E-03
15	0	0.91	5.90E-06	6.40E-06	6.80E-07	3.30E-03
16	0	0.91	5.90E-06	6.40E-06	6.80E-07	3.30E-03
17	5.86E-06	0.84	1.70E-05	6.70E-08	6.70E-08	2.50E-03
18	0.99993	0	0	0	0	0

Notes:
(a) Aerosol and Respirable Fractions set to 1.0.

Table 7.4-6— New Fuel BBNPP Transportation Fatality and Injury Rates

	Fatality Rate	Injury Rate	Distance	Distance Weighted Fraction	
State	Fatalities / trk-km (Fatalities / trk-mi)	Injuries / trk-km (Injuries / trk-mi)	km (mi)	Fatality Rate	Injury Rate
IA	9.40E-09	8.60E-08	491.3	1.09E-09	9.99E-09
	(1.51E-08)	(1.38E-07)	(305.3)		
ID	3.80E-09	3.07E-07	443.5	3.98E-10	3.22E-08
	(6.12E-09)	(4.94E-07)	(275.6)		
IL	8.30E-09	1.50E-07	261.7	5.13E-10	9.28E-09
	(1.34E-08)	(2.41E-07)	(162.6)		
IN	6.70E-09	1.40E-07	243.3	3.85E-10	8.05E-09
	(1.08E-08)	(2.25E-07)	(151.2)		
NE	1.37E-08	1.97E-07	728.6	2.36E-09	3.39E-08
	(2.20E-08)	(3.17E-07)	(452.7)		
OH	3.90E-09	1.40E-07	379.3	3.50E-10	1.26E-08
	(6.28E-09)	(2.25E-07)	(235.7)		
OR	2.04E-08	1.36E-07	335.5	1.62E-09	1.08E-08
	(3.28E-08)	(2.19E-07)	(208.5)		
PA	1.35E-08	3.83E-07	404.8	1.29E-09	3.66E-08
	(2.17E-08)	(6.16E-07)	(251.5)		
UT	1.19E-08	2.53E-07	240	6.75E-10	1.44E-08
	(1.92E-08)	(4.07E-07)	(149.1)		
WA	1.80E-09	1.80E-07	57.3	2.44E-11	2.44E-09
	(2.90E-09)	(2.90E-07)	(35.6)		
WY	1.08E-08	3.23E-07	644.5	1.65E-09	4.92E-08
	(1.74E-08)	(5.20E-07)	(400.5)		
Sum:			4229.9	1.04E-08	2.19E-07
			(2628.3)		

Table 7.4-7— EPR Radwaste Annual Generation

	Annual Activity [a]	
Waste Type	**Bq**	**Ci**
Evaporator Concentrates	5.55E+12	1.50E+02
Spent Resins (other)	3.96E+13	1.07E+03
Spent Resins (Rad Waste Demineralizer System)	1.10E+12	2.96E+01
Wet Waste from Demineralizers	6.25E+10	1.69E+00
Waste Drum for Solids Collection from Centrifuge System of KPF	6.25E+10	1.69E+00
Filters	2.54E+13	6.86E+02
Sludge	5.48E+11	1.48E+01
Mixed Waste	1.48E+09	4.00E-02
Non-Compressible DAW	1.10E+10	2.97E-01
Compressible DAW	2.22E+11	6.01E+00
Combustible DAW	1.18E+12	3.19E+01
Total	7.37E+13	1.99E+03
Note: [a] From Section 3.5.		
Conversion: 1 Ci = 3.7E+10 Bq.		

Table 7.4-8— RADTRAN/TRAGIS Model Radwaste Input Parameters

	BBNPP Model	
Parameter	**EPR**	
TRAGIS Input:		
Route Mode	Commercial	
Route Origin	BBNPP	
Route Destination	Hanford, WA	
RADTRAN Input TRAGIS:		
Total Shipping Distance, km (mi)	4248.0	(2639.7)
Travel Distance - Rural, km (mi)	3343.4	(2077.5)
Travel Distance - Suburban, km (mi)	815.7	(506.9)
Travel Distance - Urban, km (mi)	89.2	(55.5)
Population Density - Rural, person/km^2 (person/mi^2)	11.4	(29.5)
Population Density - Suburban, person/km^2 (person/mi^2)	303.4	(785.8)
Population Density - Urban, person/km^2 (person/mi^2)	2307.0	(5975.1)
Stop Time, hr/trip	5.0 [(a)]	
RADTRAN Input from NRC Models [(b)]		
Vehicle Speed, km/hr (mi/hr)	88.49	(55.0)
Traffic Count - Rural, vehicles/hr	530	
Traffic Count - Suburban, vehicles/hr	760	
Traffic Count - Urban, vehicles/hr	2400	
Dose Rate at 1-m from Vehicle, mSv/hr (mR/hr)	0.14	(14)
Packaging Length, m (ft)	5.2	(17.1)
Packaging Diameter, m (ft)	1.0	(3.3)
Number of Truck Crew	2	
Population Density at Stops (radii: 1-10m (3.3-32.8ft)), person/km^2 (person/mi^2)	30000	(77699.6)
Population Density at Stops (radii: 10-800m (32.8-2624ft)), person/km^2 (person/mi^2)	340	(880.6)
Shielding Factor at Stops (radii: 1-10m (3.3--32.8ft))	1	
Shielding Factor at Stops (radii: 10-800m (32.8-2624ft))	0.2	

Note:
(a) Based on TRAGIS output: 10 stops at 30 minutes each.
(b) From NUREG-1815

Table 7.4-9— Radwaste Annual Source Term

	RADTRAN Input	
	Annual Activity	
Radionuclide	**Bq**	**Ci**
CE144	2.87E+10	7.75E-01
CO 58	4.21E+12	1.14E+02
CO 60	8.75E+12	2.37E+02
CS134	6.79E+12	1.84E+02
CS137	1.29E+13	3.49E+02
FE 55	1.75E+13	4.73E+02
I129	3.35E+07	9.06E-04
I131	3.39E+08	9.16E-03
MN 54	1.33E+13	3.60E+02
PU241	1.26E+10	3.39E-01
RU103	4.62E+11	1.25E+01
RU106	7.71E+11	2.08E+01
SB124	4.22E+08	1.14E-02
SB125	1.38E+09	3.74E-02
SR89	4.92E+08	1.33E-02
SR 90	9.75E+10	2.64E+00
Y 90	9.43E+10	2.55E+00
ZN 65	3.46E+12	9.34E+01

Table 7.4-10— Radwaste BBNPP Transportation Accident, Fatality and Injury Rates

	Accident Rate	Fatality Rate	Injury Rate	Distance	Distance Weighted Fraction		
State	Accidents / trk-km (Accidents / trk-mi)	Fatalities / trk-km (Fatalities / trk-mi)	Injuries / trk-km (Injuries / trk-mi)	km (mi)	Accident Rate	Fatality Rate	Injury Rate
ID	2.95E-07	3.80E-09	3.07E-07	116.5	8.09E-09	1.04E-10	8.42E-09
	(4.75E-07)	(6.12E-09)	(4.94E-07)	(72.4)			
IL	2.22E-07	8.30E-09	1.50E-07	190.7	9.97E-09	3.73E-10	6.73E-09
	(3.57E-07)	(1.34E-08)	(2.41E-07)	(118.5)			
IN	2.25E-07	6.70E-09	1.40E-07	243.3	1.29E-08	3.84E-10	8.02E-09
	(3.62E-07)	(1.08E-08)	(2.25E-07)	(151.2)			
MN	1.71E-07	3.00E-09	8.40E-08	442.4	1.78E-08	3.12E-10	8.75E-09
	(2.75E-07)	(4.83E-09)	(1.35E-07)	(274.9)			
MT	6.20E-07	1.36E-08	2.56E-07	888.5	1.30E-07	2.84E-09	5.35E-08
	(9.98E-07)	(2.19E-08)	(4.12E-07)	(552.1)			
OH	1.64E-07	3.90E-09	1.40E-07	379.3	1.46E-08	3.48E-10	1.25E-08
	(2.64E-07)	(6.28E-09)	(2.25E-07)	(235.7)			
PA	5.14E-07	1.35E-08	3.83E-07	404.8	4.90E-08	1.29E-09	3.65E-08
	(8.27E-07)	(2.17E-08)	(6.16E-07)	(251.5)			
SD	2.33E-07	6.10E-09	1.72E-07	662.4	3.63E-08	9.51E-10	2.68E-08
	(3.75E-07)	(9.82E-09)	(2.77E-07)	(411.6)			
WA	2.65E-07	1.80E-09	1.80E-07	281.5	1.76E-08	1.19E-10	1.19E-08
	(4.26E-07)	(2.90E-09)	(2.90E-07)	(174.9)			
WI	4.49E-07	9.10E-09	3.33E-07	301.9	3.19E-08	6.47E-10	2.37E-08
	(7.23E-07)	(1.46E-08)	(5.36E-07)	(187.6)			
WY	6.74E-07	1.08E-08	3.23E-07	336.7	5.34E-08	8.56E-10	2.56E-08
	(1.08E-06)	(1.74E-08)	(5.20E-07)	(209.2)			
				4248.0	3.81E-07	8.23E-09	2.22E-07
Sum:				(2639.7)			
Fatalities per Accident [a] :						2.16E-02	
Note: [a] Fatalities per accident = Fatality Rate / Accident Rate.							

Table 7.4-11— Population Dose from Transportation Accidents

Environmental Impact	New Fuel	Irradiated Fuel	Radwaste	Total
EPR Dose, person-Sv/EPR-reactor-yr (person-rem/EPR-reactor-yr)	See Note	1.4E-06 (1.4E-04)	7.3E-08 (7.3E-06)	1.5E-06 (1.5E-04)
Note: The dose from new fuel accidents is assumed to be negligible compared to the doses from Irradiated Fuel and Radioactive Waste (see Section 7.4.2.).				

Table 7.4-12— EPR Summary of Annual Transportation Accident Non-Radiological Impact

Environmental Impact	New Fuel	Irradiated Fuel	Radwaste	Total	Table S-4
Fatal Injury per 100 reactor years	6.6E-02	1.8E-01	1.1E-01	3.6E-01	1
Non-Fatal Injury per 10 reactor years	1.5E-01	4.1E-01	2.8E-01	8.7E-01	1