TABLE OF CONTENTS

Page

7.1 DESIGN BASIS ACCIDENTS 7.1-1
7.2 SEVERE ACCIDENTS 7.2-1
7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES 7.3-1
7.4 TRANSPORTATION ACCIDENTS 7.4-1

LIST OF TABLES

Table 7.1-1 Design Basis Accidents

Table 7.1-2 U.S. EPR Design Basis Primary Coolant Activity

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

Table 7.1-4 U.S. EPR Bounding Core Inventory

Table 7.1-5 {CCNPP} Site and U.S. EPR Atmospheric Dispersion Factors

Table 7.1-6 Steam System Piping Failure

Table 7.1-7 Feedwater System Line Break

Table 7.1-8 Reactor Coolant Pump Locked Rotor Accident/Broken Shaft

Table 7.1-9 Failure of Small Lines Carrying Primary Coolant Outside Containment

Table 7.1-10 Steam Generator Tube Rupture

Table 7.1-11 Loss of Coolant Accident

Table 7.1-12 Fuel Handling Accident

Table 7.1-13 Rod Ejection Accident

Table 7.1-14 Summary of DBA {CCNPP} Site-Specific Doses

Table 7.2-1 Release Category Descriptions

Table 7.2-2 Source Term Release Fractions R 1

Table 7.2-3 {U.S. EPR Severe Accidents Analysis Impacts – 50-mile Radius and 2050 Population }

Table 7.3-1 Severe Accident Cost Impact R 1

Table 7.3-2 SAMDA Candidates – Already Implemented R 1

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

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

Table 7.4-3 Irradiated Fuel Source Term

Table 7.4-4 {Irradiated CCNPP Model Accident and Fatality Rates}

Table 7.4-5 Irradiated Fuel and Radioactive Waste Models Severity and Release Fractions

Table 7.4-6 {Irradiated Fuel CCNPP Transportation Injury Rates}

Table 7.4-7 {New Fuel CCNPP Transportation Fatality and Injury Rates}

Table 7.4-8 U.S. EPR Radwaste Annual Generation

Table 7.4-9 RADTRAN/TRAGIS Radwaste Model Input Parameters

Table 7.4-10 Radwaste Annual Source Term

LIST OF TABLES (Cont.)

Table 7.4-11 {Radwaste CCNPP Transportation Accident, Fatality and Injury Rates}

Table 7.4-12 Population Dose from Transportation Accidents

Table 7.4-13 {CCNPP Unit 3} Summary of Annual Transportation Accident Non-Radiological Impact

LIST OF FIGURES

None

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 to the environment. 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 proposed 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, Feedwater Pipe Break 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 22 MWt (approximately 1/2% of rated thermal power) to account for heat balance measurement uncertainty. Core inventories are bounding for U-235 fuel enrichments ranging between two and five percent and burnups up to 62,000 MWd/MTU.

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 0.5 mi (0.8 km) 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, the other variables are independent of the **{**Calvert Cliffs Nuclear Power Plant (CCNPP)**}** site and specific to the U.S. EPR design.

{CCNPP**}** site-specific atmospheric dispersion characteristics are provided in Section 2.7. To determine **{**CCNPP**}** site-specific TEDE doses, TEDE doses for the U.S. EPR were multiplied by the ratio of **{**CCNPP**}** site atmospheric dispersion factors to the U.S. EPR atmospheric dispersion factors. Atmospheric dispersion factors are referred to as 'χ/Q'. The accident χ/Q values for the subject site are based on site-specific meteorological data. U.S. EPR accident χ/Q values were derived based on five years of meteorological data. The accident χ/Q values for use in the U.S. EPR DCD are the highest values determined using both Calvert Cliffs and Nine Mile meteorological data. Two runs using different meteorological data were made and the largest χ/Q values for each sector/distance combination were used. The site-specific values for CCNPP Unit 3 used Calvert Cliffs meteorological data. For the EAB and LPZ accident χ/Q values, all compass headings/wind direction sectors were calculated and the maximum χ/Q values were used in accordance with Regulatory Guide 1.145. Therefore, for the dose comparison to determine whether the CCNPP Unit 3 doses are less than the DCD values, it does not matter that different meteorological data were used. For the EAB, the postulated DBA doses and χ/Q values are calculated for a short-term (i.e., 0 to 2 hours). For the LPZ, doses and χ/Q values are calculated for the accident duration (i.e., 0 to 2 hours, 2 to 8 hours, 8 to 24 hours, 1 to 4 days and 4 to 30 days). No credit for building wake was taken for the accident χ/Q values for the EAB and LPZ determined for either the generic U.S. EPR or the site-specific CCNPP Unit 3 χ/Q s. For the generic U.S. EPR and the site-specific CCNPP Unit 3 EAB and LPZ, ground level releases were assumed; therefore, according to Regulatory Guide 1.145, the release point and receptor elevations were assumed to be the same (i.e., no terrain heights were input for the receptors). Since the growing season is taken into account for normal effluent χ/Qs and doses (i.e., not for accident scenarios), annual data were used to generate both sets of accident χ/Q values. Table 7.1-5 contains the 50th percentile **{**CCNPP**}** site-specific and U.S. EPR accident χ/Q values at the EAB and LPZ, and the **{**CCNPP**}** site to U.S. EPR atmospheric dispersion ratios. For DBAs applicable to the U.S. EPR, the time-dependent, postulated doses at the EAB and LPZ for the subject site are provided in Table 7.1-6 to Table 7.1-13. Table 7.1-14 summarizes the **{**CCNPP**}** site-specific TEDE doses and the applicable regulatory TEDE dose acceptance criteria.

As indicated by Table 7.1-5, considering that the χ/Q values for the subject site are bounded by those for the U.S. EPR, **{**CCNPP**}** site-specific TEDE doses are bounded by the U.S. EPR TEDE doses. Referring to Table 7.1-14, **{**CCNPP**}** site-specific accident doses are below regulatory 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 3)

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 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)	Feedwater Pipe Break	Using the analytical approach and acceptance criteria described in Section 15.0.3 of NUREG-0800 and Regulatory Guide 1.183 for a Main Steam Line Break, the radiological consequences of a Feedwater Line Break (FWLB) were evaluated. The limiting FWLB accident was determined to be a double-ended guillotine break of a feedwater line to one of the steam generators inside Containment.
Reactor Coolant Pump Rotor Seizure	Reactor Coolant Pump Locked Rotor Accident	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	Reactor Coolant Pump Shaft Break	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. U.S EPR radiological consequences are the same as those for the locked rotor accident.
Radiological Consequences of Control Rod Drop Accident (BWR)	Not Applicable	The U.S. EPR is a pressurized water reactor.

Table 7.1-1 Design Basis Accidents
(Page 2 of 3)

NUREG-1555 DBA Description	U.S. EPR DBA Description	Remarks
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 was determined to be a double-ended guillotine break in the Fuel Building.
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 STARDOSE 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 STARDOSE 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 Main Steam Isolation Valve Leakage Control System (BWR)	Not Applicable	The U.S. EPR is a pressurized water reactor.

Table 7.1-1 Design Basis Accidents
(Page 3 of 3)

NUREG-1555 DBA Description	U.S. EPR DBA Description	Remarks
Radiological Consequences of Fuel Handling Accidents	Fuel Handling Accident	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.
Not Applicable	Rod Ejection Accident	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[a, b]
(Page 1 of 3)

Radionuclide	Activity µCi/gm (Bq/gm)	Radionuclide	Activity µCi/gm (Bq/gm)
Noble Gases		**Tellurium Group**	
Kr-83m	1.28E-01 (4.74E+03)	Sb-125	1.56E-06 (5.77E-02)
Kr-85m	5.71E-01 (2.11E+04)	Sb-127	6.99E-06 (2.59E-01)
Kr-85	5.31E+00 (1.96E+05)	Sb-129	8.53E-06 (3.16E-01)
Kr-87	3.26E-01 (1.21E+04)	Te-127m	6.19E-04 (2.29E+01)
Kr-88	1.03 E+00 (3.81E+04)	Te-127	3.05E-03 (1.13E+02)
Kr-89	2.42E-02 (8.95E+02)	Te-129m	1.79E-03 (6.62E+01)
Xe-131m	1.08E+00 (4.00E+04)	Te-129	3.00E-03 (1.11E+02)
Xe-133m	1.35E+00 (5.00E+04)	Te-131m	4.36E-03 (1.61E+02)
Xe-133	9.47E+01 (3.50E+06)	Te-131	3.01E-03 (1.11E+02)
Xe-135m	1.95E-01 (7.22E+03)	Te-132	4.70E-02 (1.74E+03)
Xe-135	3.40E+00 (1.26E+05)	Te-134	6.80E-03 2.52E+02)
Xe-137	4.57E-02 (1.69E+03)	**Barium/Strontium Group**	
Xe-138	1.64E-01 (6.07E+03)	Sr-89	6.35E-04 (2.35E+01)
Halogens		Sr-90	4.32E-05 (1.60E+00)
Br-83	3.16E-02 (1.17E+03)	Sr-91	1.02E-03 (3.77E+01)
Br-84	1.67E-02 (6.18E+02)	Sr-92	1.73E-04 (6.40E+00)
Br-85	2.01E-03 (7.44E+01)	Ba-137m	1.50E-01 (5.55E+03)
I-129	4.59E-08 (1.70E-03)	Ba-139	2.30E-02 (8.51E+02)
I-130	4.97E-02 (1.84E+03)	Ba-140	6.74E-04 (2.49E+01)

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

Radionuclide	Activity µCi/gm (Bq/gm)	Radionuclide	Activity µCi/gm (Bq/gm)
I-131	7.43E-01 (2.75E+04)	**Noble Metals**	
I-132	3.71E-01 (1.37E+04)	Mo-99	1.21E-01 (4.48E+03)
I-133	1.25E+00 (4.63E+04)	Tc-99m	5.24E-02 (1.94E+03)
I-134	2.40E-01 (8.88E+03)	Ru-103	1.00E-04 (3.70E+00)
I-135	7.90E-01 (2.92E+04)	Ru-105	1.47E-04 (5.44E+00)
Alkalis		Ru-106	5.83E-05 (2.16E+00)
Rb-86m	5.32E-07 (1.97E-02)	Rh-103m	8.85E-05 (3.27E+00)
Rb-86	3.66E-03 (1.35E+02)	Rh-105	6.62E-05 (2.45E+00)
Rb-88	1.02E+00 (3.77E+04)	Rh-106	5.84E-05 (2.16E+00)
Rb-89	4.72E-02 (1.75E+03)	**Cerium Group**	
Cs-134	4.18E-01 (1.55E+04)	Ce-141	9.12E-05 (3.37E+00)
Cs-136	1.00E-01 (3.70E+03)	Ce-143	7.96E-05 (2.95E+00)
Cs-137	1.60E-01 (5.92E+03)	Ce-144	6.93E-05 (2.56E+00)
Cs-138	2.35E-01 (8.07E+03)	Pu-238	5.97E-07 (2.21E-02)
Cerium Group (cont'd)		**Lanthanides** (cont'd)	
Pu-239	2.51E-08 (9.29E-04)	Cm--242	5.35E-06 (1.98E-01)
Pu-240	5.72E-08 (2.12E-03)	Cm--244	2.83E-06 (1.05E-01)
Pu-241	1.03E-05 (3.81E-01)	**Activation Products**	
Np-239	1.41E-03 (5.22E+01)	Na-24	3.7E-02 (1.37E+03)
Lanthanides		Cr-51	2.0E-03 (7.40E+01)
Y-90	1.03E-05 (3.81E-01)	Mn-54	1.0E-03 (3.70E+01)
Y-91m	5.23E-04 (1.94E+01)	Fe-55	7.6E-04 (2.81E+01)

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

Radionuclide	Activity µCi/gm (Bq/gm)	Radionuclide	Activity µCi/gm (Bq/gm)
Y-91	8.10E-05 (3.00E+00)	Fe-59	1.9E-04 (7.03E+00)
Y-92	1.41E-04 (5.22E+00)	Co-58	2.9E-03 (1.07E+02)
Y-93	6.50E-05 (2.41E+00)	Co-60	3.4E-04 (1.26E+01)
Zr-95	9.31E-05 (3.44E+00)	Zn-65	3.2E-04 (1.18E+01)
Zr-97	7.37E-05 (2.73E+00)	W-187	1.8E-03 (6.66E+01)
Nb-95	9.35E-05 (3.46E+00)	**Tritium**	
Ag-110m	9.87E-07 (3.65E-02)	H-3	1.0E+00 (3.70E+04)
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)		

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[a,b]
(Page 1 of 2)

Radionuclide	Activity µCi/gm (Bq/gm)	Radionuclide	Activity µCi/gm (Bq/gm)
Halogens		**Barium/Strontium Group** (cont'd)	
Br-83	1.61E-03 (5.96E+01)	Sr-90	4.81E-08 (1.78E-03)
Br-84	3.05E-04 (1.13E+01)	Sr-91	9.01E-07 (3.33E-02)
Br-85	3.93E-06 (1.45E-01)	Sr-92	1.00E-07 (3.70E-03)
I-129	4.81E-09 (1.78E-04)	Ba-137m	3.01E-04 (1.11E+01)
I-130	4.33E-03 (1.60E+02)	Ba-139	1.03E-05 (3.81E-01)
I-131	7.67E-02 (2.84E+03)	Ba-140	7.45E-07 (2.76E-02)
I-132	2.27E-02 (8.40E+02)	**Noble Metals**	
I-133	1.17E-01 (4.33E+03)	Mo-99	1.30E-04 (4.81E+00)
I-134	6.68E-03 (2.47E+02)	Tc-99m	7.47E-05 (2.76E+00)
I-135	5.99E-02 (2.22E+03)	Ru-103	1.11E-07 (4.11E-03)
Alkalis		Ru-105	1.09E-07 (4.03E-03)
Rb-86m	3.99E-12 (1.48E-07)	Ru-106	6.49E-08 (2.40E-03)
Rb-86	7.27E-06 (2.69E-01)	Rh-103m	9.97E-08 (3.69E-03)
Rb-88	1.26E-04 (4.66E+00)	Rh-105	7.58E-08 (2.80E-03)
Rb-89	5.02E-06 (1.86E-01)	Rh-106	6.49E-08 (2.40E-03)
Cs-134	8.38E-04 (3.10E+01)	**Cerium Group**	
Cs-136	1.98E-04 (7.33E+00)	Ce-141	1.01E-07 (3.74E-03)
Cs-137	3.21E-04 (1.19E+01)	Ce-143	8.24E-08 (3.05E-03)
Cs-138	5.00E-05 (1.85E+00)	Ce-144	7.72E-08 (2.86E-03)
Tellurium Group		Pu-238	6.65E-10 (2.46E-05)
Sb-125	1.74E-09 (6.44E-05)	Pu-239	2.80E-11 (1.04E-06)
Sb-127	7.60E-09 (2.81E-04)	Pu-240	6.37E-11 (2.36E-06)
Sb-129	6.01E-09 (2.22E-04)	Pu-241	1.15E-08 (4.26E-04)
Te-127m	6.89E-07 (2.55E-02)	Np-239	1.50E-06 (5.55E-02)
Te-127	2.82E-06 (1.04E-01)	**Lanthanides**	
Te-129m	1.99E-06 (7.36E-02)	Y-90	1.29E-08 (4.77E-04)
Te-129	1.94E-06 (7.18E-02)	Y-91m	5.38E-07 (1.99E-02)
Te-131m	4.48E-06 (1.66E-01)	Y-91	9.17E-08 (3.39E-03)
Te-131	1.33E-06 (4.92E-02)	Y-92	1.33E-07 (4.92E-03)
Te-132	5.07E-05 (1.88E+00)	Y-93	5.81E-08 (2.15E-03)
Te-134	1.64E-06 (6.07E+00)	Zr-95	1.04E-07 (3.85E-03)

Table 7.1-3 U.S. EPR Design Basis Secondary Coolant Activity[a, b]
(Page 2 of 2)

Radionuclide	Activity µCi/gm (Bq/gm)	Radionuclide	Activity µCi/gm (Bq/gm)
Barium/Strontium Group		Zr-97	7.15E-08 (2.65E-03)
Sr-89	7.16E-07 (2.64E-02)	Nb-95	1.04E-07 (3.85E-03)
Lanthanides (cont'd)		**Activation Products**	
Ag-110m	1.10E-09 (4.07E-05)	Na-24	3.53E-05 (1.31E+00)
Ag-110	1.47E-11 (5.44E-07)	Cr-51	2.22E-06 (8.21E-02)
La-140	2.28E-07 (8.44E-03)	Mn-54	1.11E-06 (4.11E-02)
La-141	4.06E-08 (1.50E-03)	Fe-55	8.47E-07 (3.13E-02)
La-142	1.51E-08 (5.59E-04)	Fe-59	2.11E-07 (7.81E-03)
Pr-143	1.02E-07 (3.77E-03)	Co-58	3.23E-06 (1.20E-01)
Pr-144	7.72E-08 (2.86E-03)	Co-60	3.79E-07 (1.40E-02)
Nd-147	4.16E-08 (1.54E-03)	Zn-65	3.56E-07 (1.32E-02)
Am-241	1.32E-11 (4.88E-07)	W-187	1.81E-06 (6.70E-02)
Cm-242	5.96E-09 (2.21E-04)	**Tritium**	
Cm-244	3.15E-09 (1.17E-04)	H-3	1.0E-03 (3.70E+01)

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 Bounding Core Inventory[a,b,c]
(Page 1 of 2)

Radionuclide	Inventory Ci (Bq)	Radionuclide	Inventory Ci (Bq)
Noble Gases		**Tellurium Group**	
Kr-83m	1.96E+07 (7.25E+17)	Sb-125	3.83E+06 (1.42E+17)
Kr-85m	4.50E+07 (1.67E+18)	Sb-127	1.80E+07 (6.66E+17)
Kr-85	2.10E+06 (7.77E+16)	Sb-129	4.85E+07 (1.79E+18)
Kr-87	9.02E+07 (3.34E+18)	Te-127m	2.43E+06 (8.99E+16)
Kr-88	1.28E+08 4.74E+18)	Te-127	1.79E+07 (6.62E+17)
Kr-89	1.61E+08 (5.96E+18)	Te-129m	7.08E+06 (2.62E+17)
Xe-131m	1.54E+06 (5.70E+16)	Te-129	4.78E+07 (1.77E+18)
Xe-133m	8.92E+06 (3.30E+17)	Te-131m	2.04E+07 (7.55E+17)
Xe-133	2.89E+08 (1.07E+19)	Te-131	1.24E+08 (4.59E+18)
Xe-135m	5.49E+07 (2.03E+18)	Te-132	1.98E+08 (7.33E+18)
Xe-135	9.26E+07 (3.43E+18)	Te-134	2.50E+08 (9.25E+18)
Xe-137	2.52E+08 (9,32E+18)	**Barium/Strontium Group**	
Xe-138	2.45E+08 (9.07E+18)	Sr-89	1.61E+08 (5.96E+18)
Halogens		Sr-90	1.69E+07 (6.25E+17)
Br-83	1.96E+07 (7.25E+17)	Sr-91	2.07E+08 (7.66E+18)
Br-84	3.62E+07 (1.34E+18)	Sr-92	2.14E+08 (7.92E+18)
Br-85	4.45E+07 (1.65E+18)	Ba-137m	2.34E+07 (8.66E+17)
I-129	8.33E+00 (3.08E+11)	Ba-139	2.62E+08 (9.69E+18)
I-130	1.32E+07 (4.88E+17)	Ba-140	2.52E+08 (9.32E+18)
I-131	1.39E+08 (5.14E+18)	**Noble Metals**	
I-132	2.01E+08 (7.44E+18)	Mo-99	2.59E+08 9.58E+18)
I-133	2.90E+08 (1.07E+19)	Tc-99m	2.27E+08 (8.40E+18)
I-134	3.18E+08 (1.18E+19)	Ru-103	2.42E+08 (8.95E+18)
I-135	2.69E+08 (9.95E+18)	Ru-105	1.96E+08 (7.25E+18)
Alkalis		Ru-106	1.43E+08 (5.29E+18)
Rb-86m	5.53E+04 (2.05E+15)	Rh-103m	2.18E+08 (8.07E+18)
Rb-86	5.80E+05 (2.15E+16)	Rh-105	1.75E+08 (6.48E+18)
Rb-88	1.29E+08 (4.77E+18)	Rh-106	1.58E+08 (5.85E+18)
Rb-89	1.67E+08 (6.18E+18)	**Cerium Group**	
Cs-134	6.48E+07 (2.40E+18)	Ce-141	2.24E+08 (8.29E+18)
Cs-136	1.61E+07 (5.96E+17)	Ce-143	2.28E+08 (8.44E+18)
Cs-137	2.47E+07 (9.14E+17)	Ce-144	1.70E+08 (6.29E+18)

Table 7.1-4 U.S. EPR Bounding Core Inventory[a, b,c]
(Page 2 of 2)

Radionuclide	Inventory Ci (Bq)	Radionuclide	Inventory Ci (Bq)
Cerium Group (cont'd)		**Lanthanides** (cont'd)	
Cs-138	2.69E+08 (9.95E+18)	Pu-238	1.46E+06 (5.40E+16)
Pu-239	6.14E+04 (2.27E+15)	Ag-110m	2.42E+06 (8.95E+16)
Pu-240	1.40E+05 (5.18E+15)	Ag-110	7.15E+07 (2.65E+18)
Pu-241	2.53E+07 (9.36E+17)	La-140	2.54E+08 (9.40E+18)
Np-239	3.82E+09 (1.41E+20)	La-141	2.41E+08 (8.92E+18)
Lanthanides		La-142	2.35E+08 (8.70E+18)
Y-90	1.79E+07 (6.62E+17)	Pr-143	2.26E+08 (8.36E+18)
Y-91m	1.20E+08 (4.44E+18)	Pr-144	1.72E+08 (6.36E+18)
Y-91	1.96E+08 (7.25E+18)	Nd-147	9.44E+07 (3.49E+18)
Y-92	2.14E+08 (7.92E+18)	Am-241	2.88E+04 (1.07E+15)
Y-93	2.34E+08 (8.66E+18)	Cm-242	1.31E+07 (4.85E+17)
Zr-95	2.29E+08 (8.47E+18)	Cm-244	6.94E+06 (2.57E+17)
Zr-97	2.43E+08 (8.99E+18)		
Nb-95	2.29E+08 (8.47E+18)		

Key:
Ci – curies
Bq - Becquerels

Notes:

(a) This table lists the design basis source term inventories for radiological consequences for the U.S. EPR core.

(b) Core inventories are bounding for U-235 fuel enrichment ranging between 2% and 5% and burnups up to 62,000 MWd/MTU.

(c) The design basis power level is 4,612 MWt.

Table 7.1-5 {CCNPP} Site and U.S. EPR Atmospheric Dispersion Factors (Page 1 of 1)

Location	Time Period[c] (hours)	50th Percentile {CCNPP} Site χ/Q[a] (sec/m^3)	U.S. EPR Accident χ/Q[b] (sec/m^3)	χ/Q Ratio CCNPP Site / U.S. EPR
EAB 0.5 mi (0.80 km)	0 to 2	{8.035E-05}	1.00E-03	{8.04E-02}
LPZ {1.5 mi (2.4 km)}	0 to 2	{1.542E-05}	1.75E-04	{8.80E-02}
	2 to 8	{1.183E-05}	1.35E-04	{8.76E-02}
	8 to 24	{9.337E-06}	1.00E-04	{9.34E-02}
	24 to 96	{6.496E-06}	5.40E-05	{1.20E-01}
	96 to 720	{3.858E-06}	2.20E-05	{1.75E-01[c]}

Key:

χ/Q – atmospheric dispersion factor

sec/m^3 – seconds per cubic meter

Notes:

(a) For the 50th percentile χ/Q values, refer to Section 2.7.

(b) The indicated U.S. EPR χ/Q values are those used in the accident radiological evaluations.

(c) Bounding value used for the entire 720 hours (i.e., 30 day) interval for LOCA. At the LPZ, the worst χ/Q ratio at the end of the accident release applies.

Table 7.1-6 Steam System Piping Failure
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[(a)] (rem / Sieverts)	χ/Q Ratio[(b)] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[(c)] (rem / Sieverts)
Pre-accident Iodine Spike				
EAB	0 to 2	2.40E-01 / 2.40E-03	{8.04E-02}	{1.93E-02/1.93E-04}
LPZ	0 to 9	6.00E-02 / 6.00E-04	{9.34E-02}	{5.60E-03/5.60E-05}
Concurrent Iodine Spike				
EAB	0 to 2	2.70E-01 / 2.70E-03	{8.04E-02}	{2.17E-02/2.17E-04}
LPZ	0 to 9	2.00E-01 / 2.00E-03	{9.34E-02}	{1.87E-02/1.87E-04}
3.3% Fuel-Rod Clad Failure				
EAB	0 to 2	5.30E+00 / 5.30E-02	{8.04E-02}	{4.26E-01/4.26E-03}
LPZ	0 to 9	2.60E+00 / 2.60E-02	{9.34E-02}	{2.43E-01/2.43E-03}
0.58% Full-Rod Fuel Melt				
EAB	0 to 2	5.80E+00 / 5.80E-02	{8.04E-02}	{4.66E-01/4.66E-03}
LPZ	0 to 9	2.80E+00 / 2.80E-02	{9.34E-02}	{2.62E-01/2.62E-03}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB were calculated for a 2 hour period. Doses at the LPZ were calculated for the duration of the releases (i.e., 9 hours).

(b) Obtained from Table 7.1-5.

(c) Per Regulatory Guide 1.183 (Table 6) the regulatory TEDE dose acceptance criteria for this accident is 25 rems (0.25 Sieverts) for the pre-accident iodine spike, fuel-rod clad failure and full-rod fuel melt and 2.5 rems (0.025 Sieverts) for the concurrent iodine spike.

Table 7.1-7 Feedwater System Line Break
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[(a)] (rem / Sieverts)	χ/Q Ratio[(b)] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[(c)] (rem / Sieverts)
Coolant Concentrations at TS Limits				
EAB	0 to 2	2.90E-01 / 2.90E-03	{8.04E-02}	{2.33E-02/2.33E-04}
LPZ	0 to 8	5.00E-02 / 5.00E-04	{8.80E-02}	{4.40E-03/4.40E-05}
Pre-accident Iodine Spike				
EAB	0 to 2	4.10E-01/ 4.10E-03	{8.04E-02}	{3.30E-02/3.30E-04}
LPZ	0 to 8	7.00E-02 / 7.00E-04	{8.80E-02}	{6.16E-03/6.16E-05}
Concurrent Iodine Spike				
EAB	0 to 2	5.00E-01 / 5.00E-03	{8.04E-02}	{4.02E-02/4.02E-04}
LPZ	0 to 8	9.00E-02 / 9.00E-04	{8.80E-02}	{7.92E-03/7.92E-05}
4.4% Fuel-Rod Clad Failure				
EAB	0 to 2	1.57E+01 / 1.57E-01	{8.04E-02}	{1.26E+00/1.26E-02}
LPZ	0 to 8	2.90E+00 / 2.90E-02	{8.80E-02}	{2.55E-01/2.55E-03}
0.76% Full-Rod Fuel Melt				
EAB	0 to 2	1.61E+01 / 1.61E-01	{8.04E-02}	{1.29E+00/1.29E-02}
LPZ	0 to 8	3.10E+00 / 3.10E-02	{8.80E-02}	{2.73E-01/2.73E-03}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent
TS – U.S. EPR Standard Technical Specifications

Notes:
(a) Doses for the U.S. EPR at the EAB were calculated for a 2 hour period. At the LPZ, the doses were calculated for an 8 hour release of primary coolant activity.
(b) Obtained from Table 7.1-5.
(c) Per Regulatory Guide 1.183 (Table 6), the regulatory TEDE dose acceptance criteria for this accident is 25 rems (0.25 Sieverts) for the pre-accident iodine spike, fuel-rod clad failure and full-rod fuel melt and 2.5 rems (0.025 Sieverts) for the coolant concentrations at TS limits and concurrent iodine spike.

Table 7.1-8 Reactor Coolant Pump Locked Rotor Accident / Broken Shaft (Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[a] (rem / Sieverts)	χ/Q Ratio[b] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[c,d] (rem / Sieverts)
EAB	0 to 2	2.25E+00 / 2.25E-02	**{**8.04E-02**}**	**{**1.81E-01/1.81E-03**}**
LPZ	0 to 8	8.70E-01/ 8.70E-03	**{**8.80E-02**}**	**{**7.66E-02/7.66E-04**}**

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB were calculated for a 2 hour period starting at t=0 hours (i.e., the assumed time at which releases to the atmosphere commence). At the LPZ, the doses were calculated for 8 hours.

(b) Obtained from Table 7.1-5.

(c) Per Regulatory Guide 1.183 (Table 6), the regulatory TEDE dose acceptance criterion for this accident is 2.5 rem (0.025 Sieverts).

Table 7.1-9 Failure of Small Lines Carrying Primary Coolant Outside Containment (Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[a] (rem / Sieverts)	χ/Q Ratio[b] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[c] (rem / Sieverts)
Nuclear Sampling System Line Break (1/4 inch line)				
EAB	0 to 0.5	1.80E+00 / 1.80E-02	{8.04E-02}	{1.45E-01/1.45E-03}
LPZ	0 to 0.5	3.16E-01 / 3.16E-03	{8.80E-02}	{2.78E-02/2.78E-04}
Chemical and Volume Control System Line Break (6 inch line)				
EAB	0 to 0.5	7.15E-02 / 7.15E-04	{8.04E-02}	{5.75E-03/5.75E-05}
LPZ	0 to 0.5	1.25E-02 / 1.25E-04	{8.80E-02}	{1.10E-03/1.10E-05}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB and LPZ are for the accident duration of 0.5 hour.

(b) Obtained from Table 7.1-5.

(c) Per NUREG-0800 (Section 15.6.2, II) the regulatory TEDE dose acceptance criterion for this accident is 2.5 rem (0.25 Sieverts).

Table 7.1-10 Steam Generator Tube Rupture
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose(a) (rem / Sieverts)	χ/Q Ratio(b) (Site / U.S. EPR)	CCNPP Site TEDE Dose(c) (rem / Sieverts)
Pre-Accident Iodine Spike				
EAB	0 to 2	2.51E+00 / 2.51E-02	{8.04E-02}	{2.02E-01/2.02E-03}
LPZ	0 to 8	6.20E-01 / 6.20E-03	{8.80E-02}	{5.46E-02/5.46E-04}
Concurrent Iodine Spike				
EAB	0 to 2	2.39E+00 /2.39E-02	{8.04E-02}	{1.92E-01/1.92E-03}
LPZ	0 to 8	1.06E+00 / 1.06E-02	{8.80E-02}	{9.33E-02/9.33E-04}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB and LPZ are for the accident duration of 8 hours.

(b) Obtained from Table 7.1-5.

(c) The regulatory TEDE limits are 25 rem (0.25 Sieverts) and 2.5 rem (0.025 Sieverts) for the pre-accident iodine spike and 8 hour concurrent iodine spike, respectively.

Table 7.1-11 Loss of Coolant Accident
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[a] (rem / Sieverts)	χ/Q Ratio[b] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[c] (rem / Sieverts)
EAB	0 to 2	1.37E+01 / 1.37E-01	{8.04E-02}	{1.10E+00/1.10E-02}
LPZ	0 to 720	2.14E+01 / 2.14E-01	{1.75E-01}	{3.75E+00/3.75E-02}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB are for the worst two-hour period (i.e., 1.5 to 3.5 hours) and those at the LPZ are for 30 days (i.e., 720 hours).

(b) Obtained from Table 7.1-5.

(c) The regulatory TEDE limit is 25 rem (0.25 Sieverts).

Table 7.1-12 Fuel Handling Accident
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[a] (rem / Sieverts)	χ/Q Ratio[b] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[c] (rem / Sieverts)
EAB	0 to 2	5.62E+00 / 5.62E-02	{8.04E-02}	{4.52E-01/4.52E-03}
LPZ	0 to 2	1.04E+00 / 1.04E-02	{8.80E-02}	{9.15E-02/9.15E-04}

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB and LPZ are for an accident scenario where the release of all activity from the Reactor Building or the Fuel Building occurs within a 2 hour interval.

(b) Obtained from Table 7.1-5.

(c) The regulatory TEDE limit is 6.3 rem (0.063 Sieverts).

Table 7.1-13 Rod Ejection Accident
(Page 1 of 1)

Location	Time Period (hours)	U.S. EPR TEDE Dose[a] (rem / Sieverts)	χ/Q Ratio[b] (Site / U.S. EPR)	{CCNPP} Site TEDE Dose[c] (rem / Sieverts)
EAB	0 to 2	5.67E+00 / 5.67E-02	**{**8.04E-02**}**	**{**4.56E-01/4.56E-03**}**
LPZ	0 to 8	3.25E+00 / 3.25E-02	**{**8.80E-02**}**	**{**2.86E-01/2.86E-03**}**

Key:
χ/Q – atmospheric dispersion factor
TEDE – Total effective dose equivalent

Notes:

(a) Doses for the U.S. EPR at the EAB are for a two hour period and those at the LPZ are for the accident duration (i.e., 8 hour Reactor Coolant System leakage).

(b) Obtained from Table 7.1-5.

(c) The regulatory TEDE limit is 6.3 rem (0.063 Sieverts).

Table 7.1-14 Summary of DBA {CCNPP} Site-Specific Doses
(Page 1 of 1)

Design Basis Accident	EAB TEDE Dose[a] (rem / Sieverts)	LPZ TEDE Dose[a] (rem / Sieverts)	Regulatory TEDE Dose Acceptance Criteria[b] (rem / Sieverts)
Steam System Piping Failures			
Pre-accident Iodine Spike	{1.93E-02/1.93E-04}	{5.60E-03/5.60E-05}	25 / 0.25
Concurrent Iodine Spike	{2.17E-02/2.17E-04}	{1.87E-02/1.87E-04}	2.5 / 0.025
3.3% Fuel-Rod Clad Failure	{4.26E-01/4.26E-03}	{2.43E-01/2.43E-03}	25 / 0.25
0.58% Full-Rod Fuel Melt	{4.66E-01/4.66E-03}	{2.62E-01/2.62E-03}	25 / 0.25
Feedwater System Line Break			
Coolant Concentrations at TS Limits	{2.33E-02/2.33E-04}	{4.40E-03/4.40E-05}	2.5 / 0.025
Pre-accident Iodine Spike	{3.30E-02/3.30E-04}	{6.16E-03/6.16E-05}	25 / 0.25
Concurrent Iodine Spike	{4.02E-02/4.02E-04}	{7.92E-03/7.92E-05}	2.5 / 0.025
4.4% Fuel-Rod Clad Failure	{1.26E+00/1.26E-02}	{2.55E-01/2.55E-03}	25 / 0.25
0.76% Full-Rod Fuel Melt	{1.29E+00/1.29E-02}	{2.73E-01/2.73E-03}	25 / 0.25
Reactor Coolant Pump Locked Rotor Accident / Broken Shaft	{1.81E-01/1.81E-03	{7.66E-02/7.66E-04}	2.5 / 0.025
Failure of Small Lines Carrying Primary Coolant Outside Containment			
NSS Line Break (1/4 inch line)	{1.45E-01/1.45E-03}	{2.78E-02/2.78E-04}	2.5 / 0.025
CVCS Line Break (6 inch line)	{5.75E-03/5.75E-05}	{1.10E-03/1.10E-05}	2.5 / 0.025
Steam Generator Tube Rupture			
Pre-accident Iodine Spike	{2.02E-01/2.02E-03}	{5.46E-02/5.46E-04}	25 / 0.25
Concurrent Iodine Spike	{1.92E-01/1.92E-03}	{9.33E-02/9.33E-04}	2.5 / 0.025
LOCA	{1.10E+00/1.10E-02}	{3.75E+00/3.75E-02}	25 / 0.25
Fuel Handling Accident	{4.52E-01/4.52E-03}	{9.15E-02/9.15E-04}	6.3 / 0.063
Rod Ejection Accident	{4.56E-01/4.56E-03}	{2.86E-01/2.86E-03}	6.3 / 0.063

Key:
TEDE – Total effective dose equivalent
TBD – To be determined
NSS – Nuclear Sampling System
CVCS – Chemical and Volume Control System

Notes:

(a) For EAB and LPZ TEDE dose, see Tables 7.1-6 through 7.1-13.
(b) For Regulatory TEDE dose acceptance criteria, refer to Note (c) of appropriate table (Table 7.1-6 to Table 7.1-13).

7.2 SEVERE ACCIDENTS

This section evaluates the potential environmental impacts of severe accidents on the {Calvert Cliffs Nuclear Power Plant (CCNPP)} site from the proposed U.S. EPR plant. The environmental impacts from a postulated severe accident have been estimated using {CCNPP} 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 {CCNPP} 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: R 1

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 consequence (RC) categories are assigned to represent all potential severe accident release scenarios. 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 results from the U.S. EPR base case are applicable to {CCNPP Unit 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 {CCNPP} meteorological monitoring data from the years {1995 through 2004} and a site characteristics file, which uses site-specific population data, land usage, watershed index, and regions.

7.2.1.2 Population Data

{Several sources of historical and projected population data were referenced before deciding on appropriate data, including SECPOP1990, SECPOP2000, and 2030 projected data. These data included the 50-mile region surrounding the CCNPP 3 site.

Population growth rate was first determined by comparing SECPOP1990 and SECPOP2000 data, which were each adjusted to include transient population. This resulted in an exponential growth rate of 1.103 per decade. For comparison, population growth rate was also determined by comparing the SECPOP2000 data with the 2030 data. This resulted in an exponential growth rate of 1.146 per decade. The 1.146 per decade rate was chosen as it produces higher populations and more conservative severe accident consequence results. R 1

There are several plausible year 2000 population distributions that could be used in this analysis, including the SECPOP2000 data, and the 2030 data (which is descaled to the year 2000 using the 1.146 per decade exponential growth rate from above). The 2030 descaled data was chosen to represent the 2000 population distribution, because a severe accident was shown to have more severe affects on this population.

In summary, the 2000 population distribution is modeled by taking the 2030 population distribution data, and descaling it by a growth rate of 1.146 per decade. The population growth rate is modeled as 1.146 per decade. Populations at any point in time are then modeled by scaling the assumed 2000 population by the assumed population growth rate.

The consequences of a severe accident at CCNPP Unit 3 were determined using 2050 population. The population for 2050 was chosen because CCNPP Unit 3 has an expected start-up date of 2015 and operating life of 60 years. Recognizing that consequences increase with time (i.e., increasing population), a time-averaged consequence can be estimated by looking at the midpoint of the U.S. EPR operational life, 2045. To be conservative, this was rounded up to 2050. As a sensitivity case, the endpoint of the U.S. EPR operational life, 2075, round up to 2080, is also evaluated. The 2080 population was not checked for realistic results, such as unattainable population densities and is believed to be overly conservative due to the projection of a conservative growth rate over a time period of 80 years.}

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 time window for the analysis is 24 hours following core damage.

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 {CCNPP} 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. R 1

- 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 {6.39 E-03} person-rem per year for the U.S. EPR and for {CCNPP Unit 3}. This value is the sum of all 23 release categories. R 1

Surface water pathways involving swimming, fishing, and boating are not modeled by MACCS2. {Surface water bodies within the 50-mile region of the CCNPP site include the Chesapeake Bay, Patuxent River, Potomac River, 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 {(including the existing CCNPP Units 1 and 2 on Chesapeake Bay)} 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 includes CCNPP Units 1 and 2)}. 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 and for {CCNPP Unit 3}, 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 {unnamed stream identified as Branch 2 and the Chesapeake Bay}.

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, {two of which (Hope Creek and Indian Point) were estuary sites similar to the CCNPP site}. The conclusion for those sites is that the uninterdicted population doses are significantly less than the NUREG-0440 generic site. {The proposed location for CCNPP Unit 3 has the same groundwater characteristics as the location of the existing units and the CDF for the U.S. EPR is lower than that of the existing units. Therefore, the doses from the U.S. EPR and CCNPP Unit 3 groundwater pathway would be smaller than from the existing CCNPP Units 1 and 2.}

7.2.3 CONCLUSIONS

The total calculated dose-risk to the {50-mile, year 2050} estimated population from airborne releases from a U.S. EPR reactor at {CCNPP} is expected to be approximately {0.41} 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 2 Sv (200 rem) and 0.25 Sv (25 rem) are 3.38E-5 and 4.71E-4, 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. R 1

The U.S. EPR dose-risk at the {CCNPP} site is less than the population risk for all current reactors that have undergone license renewal, and less than that for the five reactors analyzed in NUREG-1150 (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 qualitative analysis indicates that risk from the surface water pathway is small. The risks of groundwater contamination from a U.S. EPR accident are several orders of magnitude less than the risk from surface water contamination for currently licensed reactors. The risk of groundwater contamination from an U.S. EPR accident is smaller than the risk from currently licensed reactors. Additionally, interdiction could substantially reduce the groundwater pathway risks.

For comparison, as reported in ER Section 5.4, the total collective dose from normal operations is expected to be {5.7 person-rem per year for CCNPP Unit 3 (based on liquid and gaseous effluent for the projected 50-mile population for year 2080)}. As previously described, dose-risk is dose times frequency. Normal operation has a frequency of one. Therefore, the dose-risk for normal operation is {5.7 person-rem per year}. Comparing this value to the severe accident dose-risk of approximately {0.61} person-rem per year {(2080 conservative estimate)} indicates that the dose risk from severe accidents is less than {11} percent of dose risk from normal operations.

The probability-weighted number of cancer fatalities from a severe accident for the U.S. EPR at {CCNPP} is reported in Table 7.2-3 as {2.5 E-04} per year. The lifetime probability of an individual dying from any cancer is {2.3 E-01} (NCHS, 2007).

7.2.4 REFERENCES

FR, 1985. NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, 50 FR 32138, Nuclear Regulatory Commission, August 8, 1985.

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, Vol. 55, No. 19, dated August 21, 2007, National Center for Health Statistics. Available online at http://www.cdc.gov/nchs/data/nvsr/nvsr55/nvsr55_19.pdf, accessed December 8, 2007. R 1

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

NRC, 1978. Liquid Pathway Generic Study, NUREG 0440, Nuclear Regulatory Commission, February 1978. R 1

NRC, 1990. Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, Nuclear Regulatory Commission, December 1990.

NRC, 1996. Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, Vol. 1, Nuclear Regulatory Commission, May 1996.

NRC, 2006. Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site, NUREG-1811, Nuclear Regulatory Commission, December 2006. R 1

Table 7.2-1 Release Category Descriptions

(Page 1 of 1)

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" 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 through debris quench due to containment rupture, with MCCI, without debris flooding, with containment spray
RC402	Containment failures after breach and up through debris quench due to containment rupture, with MCCI, without debris flooding, without containment spray
RC403	Containment failures after breach and up through debris quench due to containment rupture, without MCCI, with debris flooding, with containment spray
RC404	Containment failures after breach and up through debris quench due to containment rupture, without MCCI, with debris flooding, without containment spray
RC501	Long term containment failure after debris quench due to rupture, with MCCI, without debris flooding, with containment spray
RC502	Long term containment failure after debris quench due to rupture, with MCCI, without debris flooding, without containment spray
RC503	Long term containment failure after debris quench due to rupture, without MCCI, with debris flooding, with containment spray
RC504	Long term containment failure 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

R 1

Table 7.2-2 Source Term Release Fractions

(Page 1 of 1)

Release Category	Xe + Kr + Non-Radioactive Inert Aerosols	CsI + RbI	TeO2	SrO	MoO2	CsOH + RbOH	BaO	La2O3 + Pr2O3 + Nd2O3 + Sm2O3 + Y2O3	CeO2	Sb	Te2	UO2 + NpO0 + PuO2
RC101	1.89E-03	3.05E-05	5.70E-05	8.02E-06	5.82E-05	2.57E-05	2.66E-05	4.14E-07	9.73E-07	6.73E-05	3.48E-08	1.77E-09
RC201	3.22E-01	8.14E-02	2.41E-02	6.89E-05	4.37E-03	8.45E-02	1.04E-03	6.41E-06	1.07E-05	1.18E-02	0.00E+00	0.00E+00
RC202	8.44E-01	3.05E-02	2.93E-02	7.59E-04	8.60E-03	2.22E-02	2.07E-03	1.09E-04	1.72E-04	5.34E-02	4.85E-06	1.42E-07
RC203	8.77E-01	3.56E-02	3.71E-02	2.04E-04	1.37E-02	2.28E-02	4.15E-03	2.98E-05	8.46E-05	8.32E-02	1.82E-06	2.08E-06
RC204	9.19E-01	2.98E-02	1.84E-02	5.49E-04	6.08E-03	1.93E-02	3.32E-03	7.88E-05	1.46E-04	8.14E-02	2.82E-06	6.64E-08
RC205	9.86E-01	4.08E-02	2.92E-02	4.75E-04	7.17E-03	2.56E-02	7.20E-03	7.58E-05	2.16E-04	2.68E-01	1.58E-07	1.20E-05
RC206	1.80E-01	8.85E-03	1.01E-02	2.19E-03	9.05E-03	8.16E-03	5.45E-03	9.71E-05	3.11E-04	1.05E-02	2.40E-06	2.83E-07
RC301	8.44E-01	3.05E-02	2.93E-02	7.59E-04	8.60E-03	2.22E-02	2.07E-03	1.09E-04	1.72E-04	5.34E-02	4.85E-06	1.42E-07
RC302	8.77E-01	3.56E-02	3.71E-02	2.04E-04	1.37E-02	2.28E-02	4.15E-03	2.98E-05	8.46E-05	8.32E-02	1.82E-06	2.08E-06
RC303	9.19E-01	2.98E-02	1.84E-02	5.49E-04	6.08E-03	1.93E-02	3.32E-03	7.88E-05	1.46E-04	8.14E-02	2.82E-06	6.64E-08
RC304	9.86E-01	4.08E-02	2.92E-02	4.75E-04	7.17E-03	2.56E-02	7.20E-03	7.58E-05	2.16E-04	2.68E-01	1.58E-07	1.20E-05
RC401	8.02E-01	6.76E-03	3.21E-03	2.88E-03	2.81E-03	2.81E-03	5.80E-03	1.09E-04	2.64E-04	4.81E-03	9.11E-07	5.95E-06
RC402	9.72E-01	2.81E-02	7.82E-03	4.08E-03	3.95E-03	1.30E-02	8.22E-03	1.57E-04	3.94E-04	1.06E-02	2.78E-05	1.21E-05
RC403	8.02E-01	6.76E-03	3.21E-03	2.88E-03	2.81E-03	2.81E-03	5.80E-03	1.09E-04	2.64E-04	4.81E-03	9.11E-07	5.95E-06
RC404	9.72E-01	2.81E-02	7.82E-03	4.08E-03	3.95E-03	1.30E-02	8.22E-03	1.57E-04	3.94E-04	1.06E-02	2.78E-05	1.21E-05
RC501	9.82E-01	1.93E-03	6.25E-04	1.12E-05	5.81E-05	1.93E-03	6.10E-05	4.92E-07	9.46E-07	3.79E-03	2.37E-05	4.65E-09
RC502	9.82E-01	1.93E-03	6.25E-04	1.12E-05	5.81E-05	1.93E-03	6.10E-05	4.92E-07	9.46E-07	3.79E-03	2.37E-05	4.65E-09
RC503	1.00E+00	8.52E-03	4.27E-02	8.02E-06	5.82E-05	7.92E-04	2.66E-05	4.14E-07	9.73E-07	9.46E-04	1.90E-04	1.78E-09
RC504	1.00E+00	8.52E-03	4.27E-02	8.02E-06	5.82E-05	7.92E-04	2.66E-05	4.14E-07	9.73E-07	9.46E-04	1.90E-04	1.78E-09
RC602	1.09E-13	4.06E-33	6.40E-33	5.13E-43	5.23E-33	4.12E-33	2.77E-33	4.28E-53	1.63E-43	6.63E-33	1.66E-73	4.56E-83
RC701	1.09E-01	8.12E-02	1.28E-01	1.03E-02	1.05E-01	8.25E-02	5.55E-02	8.56E-04	3.25E-03	1.33E-01	3.32E-06	9.13E-07
RC702	8.86E-01	9.30E-01	8.93E-01	1.33E-01	7.83E-01	8.83E-01	4.24E-01	6.90E-03	2.17E-02	9.13E-01	4.75E-05	2.29E-05
RC802	3.22E-01	8.14E-02	2.41E-02	6.89E-05	4.37E-03	8.45E-02	1.04E-03	6.41E-06	1.07E-05	1.18E-02	0.00E+00	0.00E+00

R 1

Table 7.2-3 {U.S. EPR Severe Accidents Analysis Impacts – 50-mile Radius and 2050 Population}

(Page 1 of 1)

{

		Number of Fatalities (per year)		Environmental Risk		
Release Category	**Release Category Frequency (per year)**	**Early Fatalities**	**Late Cancers**	**Population Dose-Risk (person-rem per year)**	**Cost (dollars per year)**	**Land Requiring Decontamination (acres per year)**
RC101	0.00E+00	0.00E+00	8.31E-06	1.60E-02	4.64E+00	2.81E-04
RC201	9.51E-12	3.68E-11	1.09E-06	2.13E-03	4.29E+00	5.41E-05
RC202	1.56E-17	4.04E-15	1.00E-10	1.91E-07	2.28E-04	5.89E-09
RC203	1.61E-16	8.15E-14	2.44E-09	4.29E-06	5.29E-03	1.30E-07
RC204	1.05E-14	2.51E-12	5.54E-08	1.10E-04	1.20E-01	3.52E-06
RC205	3.92E-12	1.39E-10	1.27E-06	2.46E-03	2.68E+00	6.48E-05
RC206	1.32E-08	1.95E-08	3.03E-05	5.18E-02	5.46E+01	1.60E-03
RC301	6.34E-16	1.65E-13	4.08E-09	7.79E-06	9.29E-03	2.40E-07
RC302	2.87E-15	1.46E-12	4.36E-08	7.67E-05	9.45E-02	2.32E-06
RC303	9.96E-13	2.38E-10	5.27E-06	1.05E-02	1.14E+01	3.34E-04
RC304	1.68E-10	5.97E-09	5.46E-05	1.06E-01	1.15E+02	2.78E-03
RC401	0.00E+00	0.00E+00	1.38E-08	2.84E-05	2.13E-02	1.07E-06
RC402	0.00E+00	0.00E+00	4.90E-07	1.03E-03	1.16E+00	3.84E-05
RC403	0.00E+00	0.00E+00	6.82E-07	1.40E-03	1.05E+00	5.29E-05
RC404	0.00E+00	0.00E+00	2.40E-05	5.04E-02	5.70E+01	1.88E-03
RC501	0.00E+00	0.00E+00	1.04E-10	2.32E-07	2.78E-05	1.09E-08
RC502	0.00E+00	0.00E+00	4.37E-08	9.73E-05	1.17E-02	4.55E-06
RC503	0.00E+00	0.00E+00	7.88E-08	1.75E-04	2.31E-02	2.53E-05
RC504	0.00E+00	0.00E+00	2.57E-05	5.70E-02	7.53E+00	8.23E-03
RC602	0.00E+00	0.00E+00	1.42E-07	3.17E-04	3.79E-02	1.48E-05
RC701	0.00E+00	2.23E-11	1.36E-05	2.37E-02	1.90E+01	7.99E-04
RC702	2.74E-08	3.37E-08	7.84E-05	8.48E-02	8.86E+01	1.06E-03
RC802	1.00E-12	6.63E-11	5.04E-07	1.07E-03	1.48E+00	3.67E-05
Total	**4.08E-08**	**5.97E-08**	**2.45E-04**	**4.09E-01**	**3.69E+02**	**1.73E-02**

}

R 1

7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES

The purpose of the severe accident mitigation alternatives (SAMA) 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, 2007) for the U.S. EPR 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 {CCNPP Unit 3}.

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, 2007) for the U.S. EPR.

The comprehensive list of SAMDA candidates was developed for the U.S. EPR 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). R 1

The top 100 U.S. EPR Level 1 PRA cutsets were evaluated to identify plant-specific modifications for inclusion in the comprehensive list of SAMDA candidates. The top 100 cutsets represent approximately 50 percent of the total core damage frequency (CDF) for the U.S. EPR. The percentage of contribution to the total CDF for the cutsets below the top 100 was minimal. Therefore, these cutsets were not likely contributors for identification of cost beneficial enhancements for the U.S. EPR design.

An extensive evaluation of the top 100 cutsets was completed in order to establish that all possible design alternatives for the U.S. EPR were addressed. Through the evaluation, numerous U.S. EPR specific operator actions and hardware-based SAMDA candidates were developed. The U.S. EPR DC ER (AREVA, 2007) provides a detailed list of the SAMDA candidates for the U.S. EPR. The SAMDA candidates identified in the U.S. EPR DC ER are applicable to {CCNPP Unit 3}.

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. 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. Potential enhancements that are not considered applicable to the U.S. EPR 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. R 1

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 {CCNPP Unit 3}.

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 {CCNPP UNIT 3}

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-1084 (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 {CCNPP Unit 3} uses the economic impact and whole body dose for a 2050 population (Table 7.3-1). The best estimate and upper bound severe accident impact cost for {CCNPP Unit 3} is also shown in Table 7.3-1.

The severe accident impact cost calculated in Table 7.3-1 accounts for the risk of internal events, internal flooding, and internal fires. To determine the total cost of severe accident risk the contribution of external events (e.g., seismic risk, as other external event contributors are small) needs to be included. Assuming that fire risk is the dominant contributor to external events risk, the seismic risk contribution was conservatively accounted for by assuming that it is equivalent to the internal fire risk. A scaling factor was calculated by dividing the internal fire CDF (1.76 E-07 per year) by the total CDF (5.30 E-07 per year) resulting in an increase of 33 percent (AREVA, 2007). Increasing the severe accident impact by 33 percent includes the seismic risk and is the maximum benefit for {CCNPP Unit 3}.

The maximum benefit for {CCNPP Unit 3 is $62,038 (best estimate) and $107,253 (upper bound). The best estimate maximum benefit for CCNPP Unit 3 is approximately $11,000} higher than the maximum benefit calculated in the U.S. EPR DC ER. The minimum implementation cost for a SAMDA candidate was determined to be $150,000 (AREVA, 2007), which exceeds the maximum benefit calculated for {CCNPP Unit 3}. Therefore, the U.S. EPR DC ER analysis is applicable to {CCNPP Unit 3}.

7.3.3 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 {CCNPP Unit 3}.

- Twenty-one SAMDA candidates were not applicable to the U.S. EPR design.
- Sixty-seven 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 7.3-2. R 1
- Four SAMDA candidates were combined with another SAMDA because they had the same intent.
- Fifty-one SAMDA candidates were categorized as not required for design certification because they were related to a procedural or surveillance action. Evaluation of any potential administrative SAMDA candidates (i.e., those candidates related to procedures and training) is not appropriate until the plant design is finalized and plant administrative processes, procedures and training program are being developed. However, the plant administrative processes, procedures, and training program will be developed to address appropriate maintenance and use of the U.S. EPR design features which have been credited with the reduction of risk associated with postulated severe accidents. As such, appropriate administrative controls on plant operations will be incorporated into the {CCNPP Unit 3} management systems as part of the initial administrative processes, procedures and training program development process.
- One SAMDA candidate was categorized as very low benefit.
- Twenty-three 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, 2007). 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 {CCNPP Unit 3}. The detailed analysis and conclusions in the U.S. EPR DC ER remain applicable for {CCNPP Unit 3}.

7.3.4 REFERENCES

R 1

AREVA, 2007. "AREVA NP Environmental Report Standard Design Certification," ANP-10290, Revision 0, AREVA NP, November 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-1084, Nuclear Regulatory Commission, January 1997.

Table 7.3-1 Severe Accident Cost Impact
(Page 1 of 1)

{

	Best Estimate (7% Discount Rate)	Upper Bound (3% Discount Rate)
Averted Occupational Exposure	$264	$607
Averted On-site Costs	$29,680	$47,011
Averted Public Exposure	$11,509	$22,757
Averted Off-site Property Damage Costs	$5,192	$10,266
Severe Accident Cost Impact Internal Events, Internal Flooding, Internal Fire	$46,645	$80,641
Maximum Benefit Internal Events, Internal Flooding, Internal Fire, Seismic	**$62,038**	**$107,253**

}

R 1

Table 7.3-2 SAMDA Candidates – Already Implemented

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-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.

R 1

Table 7.3-2 SAMDA Candidates – Already Implemented

(Page 2 of 2)

SAMDA ID	Potential Enhancement
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.
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-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 CO_2 tanks.
OT-01	Install digital large break LOCA protection system.

R 1

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.

R 1

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 Tables 7.4-2 through 7.4-7 and Tables 7.4-9 through 7.4-11.

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 7.4-12. The dose impact from all postulated transportation accident sources is **{**2.0E-4 person-rem/year (2.0E-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 **{**Calvert Cliffs Nuclear Plant (CCNPP)**}** site to the proposed repository is **{**2,680 mi (4,313 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 **{**5.14E-06 person-rem/MTU (5.14E-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 **{**1.9E-04 person-rem/year (1.9E-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 **{**8.2E-06 person-rem/yr (8.2E-08 person-Sv/year).**}** 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 **{**2,733 mi (4,399 km)**}**. The distances through each state are listed in Table 7.4-11. The distances and population densities through the rural, suburban and urban settings are listed in Table 7.4-9 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 2.54E+03 Ci (9.41E+13 Bq), is forecast to be shipped. This is described in Table 7.4-8 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 7.4-10 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 7.4-11 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 **{**8.2E-06 person-rem/yr (8.2E-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 **{**CCNPP 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 **{**2,723 mi (4,381 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 7.4-7 of **{**1.63E-08 fatalities/truck-mi (1.01E-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 7.4-7 of **{**3.68E-07 nonfatal injuries/truck-mi (2.29E-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**}** 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 **{**CCNPP**}** 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 **{**1.78E-01**}** per 100 reactor years.

Based on the above and the accident rates from Table 7.4-6, the non-radiological nonfatal injury rate impact associated with postulated accidents as a result of irradiated fuel shipments is **{**4.08E-01**}** nonfatal injuries per 10 reactor years.

7.4.2.3 Radioactive Waste

The fatal injury rate for accidents associated with radwaste shipments is **{**1.06E-01**}** 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 CCNPP site to the farthest potential disposal repository located in **{**Hanford, WA 2,733 mi (4,399 km)**}** and that the truck conservatively returns to the plant empty (doubling the traveled distance.) The state-specific fatality rates are in Table 7.4-11. 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 7.4-11 is **{**1.29E-08 fatalities/truck-mi (8.00E-09 fatalities/truck-km)**}**. The Radwaste Fatality (SFF) rate was calculated to be **{**1.06E-01 fatal injuries/100 reactor years.**}**

The nonfatal injury rate associated with radwaste shipments is **{**3.06E-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 2,733 mi (4,399 km)**}** and that the truck conservatively returns to plant empty (doubling the traveled distance.) The state-specific fatality rates are in Table 7.4-11. 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 7.4-11 is **{**3.73E-07 injuries/truck-mi (2.32E-07 injuries/truck-km)**}**. The nonfatal Radwaste Injury rate was calculated to be **{**3.06E-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 **{**CCNPP**}** site has been performed in accordance with 10 CFR 51.52(b) (CFR, 2007).

Table 7.4-12 summarizes the radiological impact, and Table 7.4-13 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 7.4-13, 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

AEC, 1972. Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238, Atomic Energy Commission, December 1972.

ANL, 1999. State-Level Accident Rates of Surface Freight Transportation: A Reexamination, ANL/ESD/TM-150, Argonne National Laboratory, 1999.

CFR, 2007. Title 10, Code of Federal Regulations, Part 51, Environmental Protection Regulations for Domestic Licensing and related Regulatory Functions, 2007.

DOE, 2002. Final Environmental Impact Statement for a Geologic Repository for the Disposal of Irradiated Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, DOE/EIS-0250, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, 2002.

DOE, 1981. Radioactive Decay Data Tables, DOE/TIC-11026, U.S. Department of Energy, D. Kocher, 1981

NRC, 1975. Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, Supplement 1, NUREG-75/038, Nuclear Regulatory Commission, April 1975.

NRC, 2000. Re-Examination of Spent Fuel Shipment Risk Estimates, NUREG/CR-6672, Nuclear Regulatory Commission, 2000.

NRC, 2006. Environmental Impact Statement for an Early Site Permit (ESP) at the Exelon ESP Site Final Report, NUREG-1815, Nuclear Regulatory Commission, July, 2006.

ORNL, 2003. Transportation Routing Analysis Geographic Information System (TRAGIS) User's Manual, ORNL/NTRC-006, P. Johnson, and R. Michelhaugh, Oak Ridge National Laboratory, 2003.

SNL, 2006. RADCAT 2.3 User Guide, SAND2006-6315, Sandia National Laboratories, R. Weiner, D. Osborn, G. Mills, D. Hinojosa, T. Heames, and D. Orcutt, 2006.

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 (Page 1 of 1)

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 (Page 1 of 1)

Parameter	CCNPP Model U.S. EPR (English Units)	CCNPP Model U.S. EPR (SI Units)
TRAGIS Input:		
Route Mode	HRCQ	
Route Origin	CCNPP	
Route Destination	Yucca Mt, NV	
RADTRAN Input TRAGIS:		
Total Shipping Distance	2,680 mi	4,312.7 km
Travel Distance – Rural	2,036 mi	3,275.2 km
Travel Distance – Suburban	568 mi	914 km
Travel Distance – Urban	77 mi	123.8 km
Population Density – Rural	30 person/mi^2	11.5 person/km^2
Population Density – Rural	817 person/mi^2	315.5 person/km^2
Population Density – Rural	6,166 person/mi^2	2,381.8 person/km^2
Stop Time, hr/trip	5.0 (1)	
RADTRAN Input from NRC Models		
Vehicle Speed	55 mph	88.49 km/hr
Traffic Count – Rural	329 mph	530 km/hr
Traffic Count – Suburban	472 mph	760 km/hr
Traffic Count – Urban	1,492 mph	2,400 km/hr
Dose Rate at 3.3 ft (1 m) from Vehicle	14 mrem/hr	0.14 mSv/hr
Packaging Length	17 ft	5.2 m
Packaging Diameter	3 ft	1.0 m
Number of Truck Crew	2	
Population Density at Stops (radii: 3.3 to 33 ft (1 to 10 m))	77,666 person/mi^2	30,000 person/km^2
Population Density at Stops (radii: 33 to 2,625 ft (10 to 800 m))	880 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	

Note:

(1) Based on TRAGIS output: 10 stops at 30 minutes each.

Table 7.4-3 Irradiated Fuel Source Term
(Page 1 of 1)

Radionuclide (a)	CCNPP Model U.S. EPR 5 Year Decay Ci/MTU	CCNPP 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 CCNPP Model Accident and Fatality Rates (Page 1 of 1)

State	Accident Rate Accidents / truck-mi (Accidents / truck-km)	Fatality Rate Fatalities / truck-mi (Fatalities / truck-km)	Distance mi (km)	Accident Rate distance weighted fraction accident / truck-mi (accident / truck-km)	Fatality Rate distance weighted fraction fatality / truck-mi (fatality / truck-km)
AZ	2.12E-07 (1.32E-07)	1.51E-08 (9.40E-09)	29.3 (47.1)	2.32E-09 (1.44E-09)	1.66E-10 (1.03E-10)
IL	3.57E-07 (2.22E-07)	1.34E-08 (8.30E-09)	162.2 (261.7)	2.17E-08 (1.35E-08)	8.11E-10 (5.04E-10)
IN	3.62E-07 (2.25E-07)	1.08E-08 (6.70E-09)	151.2 (243.4)	2.04E-08 (1.27E-08)	6.08E-10 (3.78E-10)
IA	1.80E-07 (1.12E-07)	1.51E-08 (9.40E-09)	307.0 (494.1)	2.06E-08 (1.28E-08)	1.74E-09 (1.08E-09)
MD	8.69E-07 (5.40E-07)	1.05E-08 (6.50E-09)	235.4 (378.8)	7.63E-08 (4.74E-08)	9.19E-10 (5.71E-10)
NE	5.13E-07 (3.19E-07)	2.20E-08 (1.37E-08)	456.6 (734.8)	8.75E-08 (5.44E-08)	3.75E-09 (2.33E-09)
NV	3.62E-07 (2.25E-07)	1.06E-08 (6.60E-09)	167.5 (269.5)	2.27E-08 (1.41E-08)	6.63E-10 (4.12E-10)
OH	2.64E-07 (1.64E-07)	6.28E-09 (3.90E-09)	239.9 (386.1)	2.37E-08 (1.47E-08)	5.62E-10 (3.49E-10)
PA	8.27E-07 (5.14E-07)	2.17E-08 (1.35E-08)	107.2 (172.6)	3.32E-08 (2.06E-08)	8.69E-10 (5.40E-10)
UT	4.67E-07 (2.90E-07)	1.92E-08 (1.19E-08)	379.2 (610.3)	6.6E-08 (4.10E-08)	2.70E-09 (1.68E-09)
WV	2.77E-07 (1.72E-07)	2.70E-08 (1.68E-08)	43.3 (69.7)	4.47E-09 (2.78E-09)	4.28E-10 (2.72E-10)
WY	1.08E-06 (6.74E-07)	1.74E-08 (1.08E-08)	400.5 (644.6)	1.63E-07 (1.01E-07)	2.59E-09 (1.61E-09)
Sum:			2680 (4312.7)	5.41E-07 (3.36E-07)	1.58E-08 (9.84E-09)
				Fatalities per Accident (1) :	2.93E-02

Note:

(1) Fatalities per accident = Fatality Rate / Accident Rate.

Table 7.4-5 Irradiated Fuel and Radioactive Waste Models Severity and Release Fractions (Page 1 of 1)

			Release Fractions			
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

Note:

Aerosol and Respirable Fractions set to 1.0.

Table 7.4-6 Irradiated Fuel CCNPP Transportation Injury Rates
(Page 1 of 1)

State	Injury Rate Injury / truck-mi (Injury / truck-km)	Distance mi (km)	Injury Rate Distance Weighted Fraction Injury / truck-mi (Injury / truck-km)
AZ	1.88E-07 (1.17E-07)	29.3 (47.1)	2.06E-09 (1.28E-09)
IL	2.41E-07 (1.50E-07)	162.6 (261.7)	1.46E-08 (9.10E-09)
IN	2.25E-07 (1.40E-07)	151.2 (243.4)	1.27E-08 (7.90E-09)
IA	1.38E-07 (8.60E-08)	307.0 (494.1)	1.59E-08 (9.85E-09)
MD	7.39E-07 (4.59E-07)	235.4 (378.8)	6.49E-08 (4.03E-08)
NE	3.17E-07 (1.97E-07)	456.6 (734.8)	5.41E-08 (3.36E-08)
NV	2.38E-07 (1.48E-07)	167.5 (269.5)	1.49E-08 (9.25E-09)
OH	2.25E-07 (1.40E-07)	239.9 (386.1)	2.01E-08 (1.25E-08)
PA	6.16E-07 (3.83E-07)	107.2 (172.6)	2.46E-08 (1.53E-08)
UT	4.07E-07 (2.53E-07)	379.2 (610.3)	5.76E-08 (3.58E-08)
WV	1.80E-07 (1.12E-07)	43.3 (69.7)	2.91E-09 (1.81E-09)
WY	5.20E-07 (3.23E-07)	400.5 (644.6)	7.77E-08 (4.83E-08)
	Sum:	2680 (4312.7)	3.62E-07 (2.25E-07)

Table 7.4-7 New Fuel CCNPP Transportation Fatality and Injury Rates (Page 1 of 1)

State	Fatality Rate Fatalities / Truck-mi (fatality / truck-km)	Injury Rate Injuries Truck-mi (injury / truck-km)	Distance mi (km)	Fatality Rate Distance Weighted Fraction Fatality / Truck-mi (fatality / truck-km)	Injury Rate Distance Weighted Fraction Injury / Truck-mi (injury / truck-km)
ID	6.12E-09 (3.80E-09)	4.94E-07 (3.07E-07)	275.6 (443.5)	6.20E-10 (3.85E-10)	5.01E-08 (3.11E-08)
IL	1.34E-08 (8.30E-09)	2.41E-07 (1.50E-07)	162.6 (261.7)	7.98E-10 (4.96E-10)	1.44E-08 (8.96E-09)
IN	1.08E-08 (6.70E-09)	2.25E-07 (1.40E-07)	151.2 (243.4)	5.99E-10 (3.72E-10)	1.25E-08 (7.78E-09)
IA	1.51E-08 (9.40E-09)	1.38E-07 (8.60E-08)	305.3 (491.3)	1.69E-09 (1.05E-09)	1.55E-08 (9.64E-09)
MD	1.05E-08 (6.50E-09)	7.39E-07 (4.59E-07)	153.7 (247.4)	5.91E-10 (3.67E-10)	4.17E-08 (2.59E-08)
NE	2.20E-08 (1.37E-08)	3.17E-07 (1.97E-07)	452.7 (728.5)	3.67E-09 (2.28E-09)	5.28E-08 (3.28E-08)
OH	6.28E-09 (3.90E-09)	2.25E-07 (1.40E-07)	239.9 (386.1)	5.54E-10 (3.44E-10)	1.98E-08 (1.23E-08)
OR(1)	3.28E-08 (2.04E-08)	2.19E-07 (1.36E-07)	208.5 (335.5)	2.51E-09 (1.56E-09)	1.67E-08 (1.04E-08)
PA	2.17E-08 (1.35E-08)	6.16E-07 (3.83E-07)	187.3 (301.5)	1.50E-09 (9.29E-10)	4.25E-08 (2.64E-08)
UT	1.92E-08 (1.19E-08)	4.07E-07 (2.53E-07)	149.1 (240)	1.05E-09 (6.52E-10)	2.24E-08 (1.39E-08)
WA	2.90E-09 (1.80E-09)	2.90E-07 (1.80E-07)	35.7 (57.4)	3.80E-11 (2.36E-11)	3.80E-09 (2.36E-09)
WY	1.74E-08 (1.08E-08)	5.20E-07 (3.23E-07)	400.5 (644.6)	2.56E-09 (1.59E-09)	7.64E-08 (4.75E-08)
		Sum:	2723 (4380.9)	1.63E-08 (1.01E-08)	3.68E-07 (2.29E-07)

Note:

Interstate data not provided.

Table 7.4-8 EPR Radwaste Annual Generation
(Page 1 of 1)

Waste Type	Annual Activity [a] Bq	Annual Activity [a] Ci
Evaporator Concentrates	5.55E+12	1.50E+02
Spent Resins (other)	3.96E+13	1.07E+03
Spent Resins (Radwaste Demineralizer System)	5.55E+12	1.50E+02
Wet Waste from Demineralizers	5.55E+12	1.50E+02
Waste Drum for Solids Collection from Centrifuge System of KPF	5.55E+12	1.50E+02
Filters (quantity)	2.54E+13	6.86E+02
Sludge	5.55E+12	1.50E+02
Mixed Waste	1.48E+09	4.00E-02
Non-Compressible Dry Active Waste (DAW)	1.10E+10	2.97E-01
Compressible DAW	2.22E+11	6.01E+00
Combustible DAW	1.18E+12	3.19E+01
Total	9.41E+13	2.54E+03

Note:

(a) Refer to Section 3.5.

Table 7.4-9 RADTRAN/TRAGIS Radwaste Model Input Parameters (Page 1 of 1)

Parameter		CCNPP Model EPR
TRAGIS Input:		
Route Mode		Commercial
Route Origin		CCNPP
Route Destination		Hanford, WA
RADTRAN Input from TRAGIS:		
Total Shipping Distance, mi (km)		2,733 (4,399)
Travel Distance - Rural, mi (km)		2,063 (3,320)
Travel Distance - Suburban, mi (km)		594 (955.5)
Travel Distance - Urban, mi (km)		76.5 (123.2)
Population Density - Rural, person/mi^2 (person/km^2)		30 (11.6)
Population Density - Suburban, person/mi^2 (person/km^2)		835 (322.4)
Population Density - Urban, person/mi^2 (person/km^2)		6,085 (2,349.5)
Stop Time, hr/trip		5.0 [b]
RADTRAN Input from NRC Models [a]		
Vehicle Speed, mph (km/hr)		55 (88.49)
Traffic Count - Rural, vehicles/hr		530
Traffic Count - Suburban, vehicles/hr		760
Traffic Count - Urban, vehicles/hr		2,400
Dose Rate at 3 ft (1 m) from Vehicle, mrem/hr (mSv/hr)		14 (0.14)
Packaging Length, ft (m)		17 (5.2)
Packaging Diameter, ft (m)		3.3 (1.0)
Number of Truck Crew		2
Population Density at Stops (radii: 3.3 to 33 ft (1 to 10 m)), person/mi^2 (person/km^2)		77,700 (30,000)
Population Density at Stops (radii: 33 to 2,655 ft (10 to 800 m)), person/mi^2 (person/km^2)		881 (340)
Shielding Factor at Stops (radii: 3.3 to 33 ft (1 to 10 m))		1
Shielding Factor at Stops (radii: 33 to 2,655 ft (10 to 800 m))		0.2

Notes:

(a) From NUREG-1815 for spent fuel shipments.
(b) Based on TRAGIS output: 11 stops at 30 minutes each.

Table 7.4-10 Radwaste Annual Source Term
(Page 1 of 1)

Radionuclide	RADTRAN Input Annual Activity Bq	Ci
Ce-144	3.89E+10	1.05E+00
Co-58	5.36E+12	1.45E+02
C0-60	9.87E+12	2.67E+02
Cs-134	1.06E+13	2.85E+02
Cs-137	1.94E+13	5.25E+02
Fe-55	1.98E+13	5.36E+02
I-129	3.35E+07	9.06E-04
I-131	3.39E+08	9.16E-03
Mn-54	1.55E+13	4.18E+02
Pu-241	1.26E+10	3.39E-01
Ru-103	8.04E+11	2.17E+01
Ru-106	1.04E+12	2.80E+01
Sb-124	4.22E+08	1.14E-02
Sb-125	1.38E+09	3.74E-02
Sr-89	4.92E+08	1.33E-02
Sr-90	1.24E+11	3.36E+00
Y-90	1.21E+11	3.27E+00
Zn-65	4.06E+12	1.10E+02

Table 7.4-11 Radwaste CCNPP Transportation Accident, Fatality and Injury Rates (Page 1 of 1)

State	Accident Rate Accidents / truck-mi (Accidents / truck-km)	Fatality Rate Fatalities / truck–mi (Fatalities / truck-km)	Injury Rate Injuries / truck–mi (Injuries / truck-km)	Distance mi (km) (a)	Distance Weighted Fraction: Accident Rate	Distance Weighted Fraction: Fatality Rate	Distance Weighted Fraction: Injury Rate
ID	4.75E-07 (2.95E-07)	6.12E-09 (3.80E-09)	4.94E-07 (3.07E-07)	72 (116.6)	7.82E-09	1.01E-10	8.14E-09
IL	3.57E-07 (2.22E-07)	1.34E-08 (8.30E-09)	2.41E-07 (1.50E-07)	118 (190.7)	9.62E-09	3.60E-10	6.50E-09
IN	3.62E-07 (2.25E-07)	1.08E-08 (6.70E-09)	2.25E-07 (1.40E-07)	151 (243.4)	1.24E-08	3.71E-10	7.75E-09
MD	8.69E-07 (5.40E-07)	1.05E-08 (6.50E-09)	7.39E-07 (4.59E-07)	154 (247.4)	3.04E-08	3.66E-10	2.58E-08
MN	2.75E-07 (1.71E-07)	4.82E-09 (3.00E-09)	1.35E-07 (8.40E-08)	275 (442.4)	1.72E-08	3.02E-10	8.45E-09
MT	9.98E-07 (6.20E-07)	2.19E-08 (1.36E-08)	4.12E-07 (2.56E-07)	552 (888.5)	1.25E-07	2.75E-09	5.17E-08
OH	2.64E-07 (1.64E-07)	6.28E-09 (3.90E-09)	2.25E-07 (1.40E-07)	240 (386.1)	1.44E-08	3.42E-10	1.23E-08
PA	8.27E-07 (5.14E-07)	2.17E-08 (1.35E-08)	6.16E-07 (3.83E-07)	187 (301.5)	3.52E-08	9.25E-10	2.62E-08
SD	3.75E-07 (2.33E-07)	9.82E-09 (6.10E-09)	2.77E-07 (1.72E-07)	412 (662.4)	3.51E-08	9.19E-10	2.59E-08
WA	4.26E-07 (2.65E-07)	2.90E-09 (1.80E-09)	2.90E-07 (1.80E-07)	175 (281.5)	1.70E-08	1.15E-10	1.15E-08
WI	7.23E-07 (4.49E-07)	1.46E-08 9.10E-09	5.36E-07 3.33E-07	188 (301.9)	3.08E-08	6.25E-10	2.29E-08
WY	1.08E-07 (6.74E-07)	1.74E-08 (1.08E-08)	5.20E-07 (3.23E-07)	209 (336.7)	5.16E-08	8.27E-10	2.47E-08
Sum:				2,733 mi (4,399 km)	6.23E-07 (3.87E-07)	1.29E-08 (8.00E-09)	3.73E-07 (2.32E-07)
						Per truck-mi (per truck-km)	
Fatalities per Accident (b) :						2.07E-02 Fatalities per accident	

Notes:

(a) From TRAGIS.

(b) Fatalities per accident = Fatality Rate / Accident Rate.

Table 7.4-12 Population Dose from Transportation Accidents (Page 1 of 1)

Environmental Impact	New Fuel	Irradiated Fuel	Radwaste	Total
U.S. EPR Dose Person-rem/ U.S. EPR-reactor-year (person-Sv/ U.S. EPR-reactor-year)	See below	1.9E-04 (1.9E-06)	8.2E-06 (8.2E-08)	2.0E-04 (2.0E-06)
Normalized Dose Person-rem/1000 MWe reactor-year (person-Sv/1000 MWe reactor-year)	See below	1.1E-04 (1.1E-06)	4.6E-06 (4.6E-08)	1.1E-04 (1.1E-06)
The dose from new fuel accidents is assumed to be negligible compared to the doses from Irradiated Fuel and Radioactive Waste as described in Section 7.4.2.				

Table 7.4-13 U.S. EPR Summary of Annual Transportation Accident Non-Radiological Impact (Page 1 of 1)

Environmental Impact	New Fuel	Irradiated Fuel	Radwaste	Total	10 CFR 51.52 Table S-4
Fatal Injury per 100 reactor years	0.066	0.18	1.1E-01	0,36	1.0
Non-Fatal Injury per 10 reactor years	0.15	0.41	3.1E-01	0.87	1.0