Chapter 7 Table of Contents

Chapter 7 List of Tables

List of Tables (Cont.)

List of Tables (Cont.)

Chapter 7 Environmental Impacts of Postulated Accidents Involving Radioactive Materials

The purpose of this section is to assess the environmental impacts of postulated accidents involving radioactive materials. Section 7.1 evaluates DBAs, Section 7.2 considers the impact of severe accidents, Section 7.3 addresses severe accident mitigation alternatives (SAMA), and Section 7.4 pertains to transportation accidents.

7.1 Design Basis Accidents

7.1.1 Selection of Accidents

The radiological consequences of accidents are assessed to demonstrate that new units could be constructed and operated at the ESP site without undue risk to the health and safety of the public. To analyze the suitability of the ESP site, site-specific accident meteorology is used to evaluate the radiological consequences of DBAs associated with selected reactor designs. The DBAs include a spectrum of events, including those of relatively greater probability of occurrence as well as those that are less probable but have greater severity.

The set of accidents selected focuses on four light water reactor (LWR) designs: AP1000, APWR, ABWR, and ESBWR. Two versions of the ABWR (GE and Toshiba) are being considered for the VCS site, but the evaluation of this section is based upon the source term associated with the ABWR certified design. These four designs have been chosen because these are standard designs that have recognized bases for postulated accident analyses. The mPower technology is still in the early stages of design, thus the accidents are not as well defined as those for the other LWRs. The mPower design is standard LWR technology, and given its relatively small thermal output, the accident consequences associated with the mPower reactor are considered to be bounded by those for the other four reactor types. If the mPower (or another reactor technology not previously evaluated) is selected for the ESP site, the COL application would verify that the accident doses are bounded by those provided in the ESP or would provide a complete evaluation of accident radiological consequences compared with regulatory limits.

The following LWR accidents are identified in NUREG-1555 (U.S. NRC Oct 1999), Section 7.1, Appendix A, as those that should be considered for radiological consequences, based on the SRP, NUREG-0800 (U.S. NRC Mar 2007):

- SRP Section 15.1.5A, Radiological Consequences of Main Steam Line Failure Outside Containment of a PWR
- SRP Section 15.2.8, Feedwater System Pipe Break

- SRP Section 15.3.3, Reactor Coolant Pump Rotor Seizure (Locked Rotor Accident)
- SRP Section 15.3.4, Reactor Coolant Pump Shaft Break
- SRP Section 15.4.9, Spectrum of Rod Drop Accidents
- SRP Section 15.6.2, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment
- SRP Section 15.6.3, Radiological Consequences of Steam Generator Tube Failure
- SRP Section 15.6.5, Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary
- SRP Section 15.7.4, Radiological Consequences of Fuel Handling Accidents

RG 1.183 (U.S. NRC Jul 2000) includes the following additional accidents:

- PWR Rod Ejection Accident (corresponding to SRP Section 15.4.8)
- BWR Main Steam Line Break (corresponding to SRP Section 15.6.4)

In addition, a cleanup water line break is evaluated for the ABWR and the ESBWR.

This set of accidents provides a reasonable basis for evaluating the suitability of the ESP site, and the associated radiological consequences from the above DBAs are analyzed as follows.

7.1.2 Evaluation Methodology

Doses for the representative DBAs are evaluated at the EAB and the LPZ. These doses must meet the site acceptance criteria in 10 CFR 52.17 (a)(1). Although the emergency safety features are expected to prevent core damage and mitigate releases of radioactivity, the loss-of-coolant accidents (LOCAs) analyzed presume substantial core melt with the release of significant amounts of fission products. The postulated LOCAs are expected to more closely approach 10 CFR 52.17 limits than the other DBAs of greater probability of occurrence but lesser magnitude of activity releases. The calculated accident doses are compared to the acceptance criteria in RG 1.183 and NUREG-0800 to demonstrate that the consequences of the postulated accidents are acceptable.

The evaluations use short-term accident atmospheric dispersion factors (X/Qs). The X/Qs are calculated using the methodology of RG 1.145 (U.S. NRC Nov 1982) and site-specific meteorological data. As indicated in Subsection 2.7.5, the RG 1.145 methodology is implemented in the NRC-sponsored PAVAN computer program. This program computes X/Qs at the EAB and the LPZ for each combination of wind speed and atmospheric stability for each of the 16 downwind direction sectors.

Releases are assumed to be at ground level, and the shortest distances between the power block area and the offsite locations are selected to conservatively maximize the X/Q values. The following site-specific 50th percentile X/Q values from Subsection 2.7.5 are used in these evaluations, in accordance with NUREG-1555:

- EAB, 0 to 2 hours: 8.85×10^{-5} sec/m^3
- LPZ, 0 to 8 hours: 5.30×10^{-6} sec/m^3
  8 to 24 hours: 3.92×10^{-6} sec/m^3
  24 to 96 hours: 2.05×10^{-6} sec/m^3
  96 to 720 hours: 8.05×10^{-7} sec/m^3

The accident dose calculations are performed using the activity releases for the following time intervals:

- EAB: 2–hour period yielding the maximum dose
- LPZ: 0 to 8 hours, 8 to 24 hours, 24 to 96 hours, and 96 to 720 hours

The accident doses are expressed as total effective dose equivalent (TEDE), consistent with 10 CFR 52.17. The TEDE consists of the sum of the committed effective dose equivalent (CEDE) from inhalation and either the deep dose equivalent (DDE) or the effective dose equivalent (EDE) from external exposure. The CEDE is determined using the dose conversion factors in Federal Guidance Report 11 (U.S. EPA 1988), while the DDE and the EDE are based on dose conversion factors in Federal Guidance Report 12 (U.S. EPA 1993).

7.1.3 Source Terms

Doses are calculated based on the time-dependent activities released to the environment during each DBA. The activities are based on the analyses used to support the reactor standard safety analysis reports. Different reactor technologies use different source terms and approaches in defining the activity releases. The ABWR source terms, methodologies, and assumptions are based on the guidance in NUREG-0800 and Regulatory Guides 1.3 (U.S. NRC Jun 1974) and 1.25 (U.S. NRC Mar 1972). The AP1000, APWR, and ESBWR source terms, methodologies, and assumptions are based on the alternative source term guidance in RG 1.183.

7.1.4 Radiological Consequences

For the accidents identified in Subsection 7.1.1, site-specific doses are calculated by multiplying the design certification doses by the ratio of site X/Qs to design certification X/Qs. Using the EAB and LPZ site-specific X/Qs provided in Subsection 2.7.5, with the design certification X/Qs

(Westinghouse Sep 2008, MHI Aug 2008, GENE Mar 1997, GEH Aug 2009), the ratios presented in Tables 7.1-1 through 7.1-4 are obtained.

Details about the methodology and assumptions pertaining to each of the accidents, such as activity release paths and the credited mitigation features, may be found in the design certification documents for each of the reactor technologies.

As the ABWR design certification document presents whole body and thyroid doses, an equivalent TEDE value is estimated by multiplying the thyroid dose by 0.03 and adding the product to the whole body dose, in accordance with RG 1.183.

A summary of the resulting accident doses is presented in Table 7.1-5. This table also compares the environmental doses to the recommended limits in RG 1.183 and NUREG-0800 and shows that the evaluated dose consequences are within the recommended limits.

The TEDE dose limits in Table 7.1-5 are taken from RG 1.183, Table 6, for all accidents except PWR Reactor Coolant Pump Shaft Break (SRP Section 15.3.4) and Failure of Small Lines Carrying Primary Coolant Outside Containment (SRP Section 15.6.2). For these two accidents, NUREG-0800 indicates that the dose limit is a “small fraction” or 10 percent of the 10 CFR 100 guideline of 25 rem, meaning a limit of 2.5 rem. No guidance is provided in RG 1.183 or NUREG-0800 for Feedwater Line Break and Cleanup Water Line Break; the regulatory limits shown are based on similar accidents.

The doses summarized in Table 7.1-5 are based on the time-dependent doses for each of the accidents, as shown in Tables 7.1-6 through 7.1-70. In addition to doses, the latter tables also show the activities released to the environment. In these tables, the EAB dose for each accident is shown for the 2-hour period yielding the maximum dose. The LPZ dose is shown for the duration of the accident.

7.1.5 References

GEH Aug 2009. Document 26A6642, *ESBWR Design Control Document*, Tier 2, GE-Hitachi Nuclear Energy, Revision 6, August 2009.

GENE Mar 1997. *ABWR Design Control Document*, Tier 2, GE Nuclear Energy, Revision 4, March 1997.

MHI Oct 2009. Document MUAP-DC001, *APWR Design Control Document for the US-APWR,* Tier 2, Mitsubishi Heavy Industries, Ltd., Revision 2, October 2009.

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

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

U.S. NRC Mar 1972. Regulatory Guide 1.25 (Safety Guide 25), *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, NRC, March 1972.

U.S. NRC Jun 1974. Regulatory Guide 1.3, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*, Revision 2, NRC, June 1974.

U.S. NRC Nov 1982. Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Revision 1, NRC, November 1982.

U.S. NRC Oct 1999. NUREG-1555, *Standard Review Plans for Environmental Reviews for Nuclear Power Plants*, Section 7.1, Design Basis Accidents, U.S. NRC, October 1999,

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

U.S. NRC Mar 2007. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, U.S. NRC, March 2007.

Westinghouse Sep 2008. AP1000 Document No. APP-GW-GL-700, *AP1000 Design Control Document*, Tier 2, Westinghouse, Revision 17, September 2008.

Table 7.1-1
Design Certification X/Q Values and Ratios to Site X/Q Value for AP1000

Accident	Location	Release Time (hr)	X/Q (sec/m^3)		X/Q Ratio (Site/DCD)
			DCD	Site	
LOCA	EAB	0–2	5.1×10^{-4}	8.85×10^{-5}	1.74×10^{-1}
	LPZ	0–8	2.2×10^{-4}	5.30×10^{-6}	2.41×10^{-2}
		8–24	1.6×10^{-4}	3.92×10^{-6}	2.45×10^{-2}
		24–96	1.0×10^{-4}	2.05×10^{-6}	2.05×10^{-2}
		96–720	8.0×10^{-5}	8.05×10^{-7}	1.01×10^{-2}
Other	EAB	0–2	1.0×10^{-3}	8.85×10^{-5}	8.85×10^{-2}
	LPZ	0–8	5.0×10^{-4}	5.30×10^{-6}	1.06×10^{-2}
		8–24	3.0×10^{-4}	3.92×10^{-6}	1.31×10^{-2}
		24–96	1.5×10^{-4}	2.05×10^{-6}	1.37×10^{-2}
		96–720	8.0×10^{-5}	8.05×10^{-7}	1.01×10^{-2}

Reference: AP1000 DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-2
Design Certification X/Q Values and Ratios to Site X/Q Values for APWR

Location	Release Time (hr)	X/Q (sec/m^3)		X/Q Ratio (Site/DCD)
		DCD	Site	
EAB	0–2	5.0×10^{-4}	8.85×10^{-5}	1.77×10^{-1}
LPZ	0–8	2.1×10^{-4}	5.30×10^{-6}	2.52×10^{-2}
	8–24	1.3×10^{-4}	3.92×10^{-6}	3.02×10^{-2}
	24–96	6.9×10^{-5}	2.05×10^{-6}	2.97×10^{-2}
	96–720	2.8×10^{-5}	8.05×10^{-7}	2.88×10^{-2}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-3
Design Certification X/Q Values and Ratios to Site X/Q Values for ABWR

Accident	Location	Release Time (hr)	X/Q (sec/m^3)		X/Q Ratio (Site/DCD)
			DCD	Site	
Cleanup Water Line Break	EAB	0–2	2.29×10^{-2}	8.85×10^{-5}	3.86×10^{-3}
	LPZ	0–8	2.29×10^{-2}	5.30×10^{-6}	2.31×10^{-4}
LOCA	EAB	0–2	1.37×10^{-3}	8.85×10^{-5}	6.46×10^{-2}
	LPZ	0–8	1.56×10^{-4}	5.30×10^{-6}	3.40×10^{-2}
		8–24	9.61×10^{-5}	3.92×10^{-6}	4.08×10^{-2}
		24–96	3.36×10^{-5}	2.05×10^{-6}	6.10×10^{-2}
		96–720	7.42×10^{-6}	8.05×10^{-7}	1.08×10^{-1}
Other	EAB	0–2	1.37×10^{-3}	8.85×10^{-5}	6.46×10^{-2}
	LPZ	0–8	1.37×10^{-3}	5.30×10^{-6}	3.87×10^{-3}

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-4
Design Certification X/Q Values and Ratios to Site X/Q Values for ESBWR

Accident	Location	Release Time (hr)	X/Q (sec/m^3) DCD	X/Q (sec/m^3) Site	X/Q Ratio (Site/DCD)
LOCA	EAB	0–2	2.00×10^{-3}	8.85×10^{-5}	4.43×10^{-2}
	LPZ	0–8	1.90×10^{-4}	5.30×10^{-6}	2.79×10^{-2}
		8–24	1.40×10^{-4}	3.92×10^{-6}	2.80×10^{-2}
		24–96	7.50×10^{-5}	2.05×10^{-6}	2.73×10^{-2}
		96–720	3.00×10^{-5}	8.05×10^{-7}	2.68×10^{-2}
Small Break Outside Containment	EAB	0–2	2.00×10^{-3}	8.85×10^{-5}	4.43×10^{-2}
	LPZ	0–8	1.90×10^{-4}	5.30×10^{-6}	2.79×10^{-2}
		8–24	1.40×10^{-4}	3.92×10^{-6}	2.80×10^{-2}
		24–96	7.50×10^{-5}	2.05×10^{-6}	2.73×10^{-2}
		96–720	3.00×10^{-5}	8.05×10^{-7}	2.68×10^{-2}
Other	EAB	0–2	2.00×10^{-3}	8.85×10^{-5}	4.43×10^{-2}
	LPZ	0–8	1.90×10^{-4}	5.30×10^{-6}	2.79×10^{-2}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-5 (Sheet 1 of 2)
Summary of Design Basis Accident Site Doses

SRP Section	Accident		Design	DCD Section	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ	Limit
15.1.5A	Steam System Piping Failure	Pre-Incident Iodine Spike	AP1000	15.1.5	8.9×10^{-2}	8.6×10^{-3}	25
			APWR	15.1.5	3.4×10^{-2}	2.8×10^{-3}	
		Accident-Initiated Iodine Spike	AP1000	15.1.5	9.7×10^{-2}	2.3×10^{-2}	2.5
			APWR	15.1.5	5.7×10^{-2}	7.3×10^{-3}	
15.3.3	Locked Rotor Accident	No Feedwater	AP1000	15.3.3	7.1×10^{-2}	4.1×10^{-3}	2.5
		Feedwater Available	AP1000	15.3.3	5.3×10^{-2}	8.0×10^{-3}	
	Locked Rotor Accident		APWR	15.3.3	8.7×10^{-2}	1.9×10^{-2}	
15.4.8	Rod Ejection Accident		AP1000	15.4.8	3.2×10^{-1}	5.8×10^{-2}	6.3
			APWR	15.4.8	9.0×10^{-1}	1.2×10^{-1}	
15.6.2	Small Break Outside Containment		AP1000	15.6.2	1.9×10^{-1}	1.1×10^{-2}	2.5
			APWR	15.6.2	2.7×10^{-1}	1.5×10^{-2}	
			ABWR	15.6.2	1.5×10^{-2}	9.2×10^{-4}	
	Small Break Outside Ctmt	Pre-Incident Iodine Spike	ESBWR	15.4.8	1.5×10^{-2}	2.8×10^{-3}	25
		Equilibrium Iodine Activity	ESBWR	15.4.8	4.4×10^{-3}	2.8×10^{-3}	2.5
15.6.3	Steam Generator Tube Rupture	Pre-Incident Iodine Spike	AP1000	15.6.3	1.9×10^{-1}	1.3×10^{-2}	25
			APWR	15.6.3	6.4×10^{-1}	3.8×10^{-2}	
		Accident-Initiated Iodine Spike	AP1000	15.6.3	9.7×10^{-2}	8.7×10^{-3}	2.5
			APWR	15.6.3	1.7×10^{-1}	1.1×10^{-2}	
15.6.4	Main Steam Line Break	Pre-Incident Iodine Spike	ABWR	15.6.4	1.8×10^{-1}	1.1×10^{-2}	25
			ESBWR	15.4.5	1.2×10^{-1}	5.6×10^{-3}	
		Equilibrium Iodine Activity	ABWR	15.6.4	9.0×10^{-3}	5.4×10^{-4}	2.5
			ESBWR	15.4.5	8.9×10^{-3}	2.8×10^{-3}	
15.6.5	Loss-of-Coolant Accident		AP1000	15.6.5	4.3	5.5×10^{-1}	25
			APWR	15.6.5	2.3	3.4×10^{-1}	
			ABWR	15.6.5	6.3×10^{-1}	7.9×10^{-1}	
			ESBWR	15.4.4	9.9×10^{-1}	5.7×10^{-1}	
15.7.4	Fuel Handling Accident		AP1000	15.7.4	4.6×10^{-1}	2.7×10^{-2}	6.3
			APWR	15.7.4	5.8×10^{-1}	3.5×10^{-2}	
			ABWR	15.7.4	2.2×10^{-1}	1.3×10^{-2}	
			ESBWR	15.4.1	1.8×10^{-1}	1.1×10^{-2}	

Table 7.1-5 (Sheet 2 of 2)
Summary of Design Basis Accident Site Doses

SRP Section	Accident		Design	DCD Section	Site Dose (rem TEDE) EAB	LPZ	Limit
None	Feedwater Line Break	Pre-Incident Iodine Spike	ESBWR	15.4.7	8.0×10^{-1}	4.7×10^{-2}	25
		Equilibrium Iodine Activity	ESBWR	15.4.7	4.9×10^{-2}	2.8×10^{-3}	2.5
None	Cleanup Water Line Break		ABWR	15.6.6	4.6×10^{-3}	2.7×10^{-4}	25
	Cleanup Water Line Break	Pre-Incident Iodine Spike	ESBWR	15.4.9	3.1×10^{-1}	2.0×10^{-2}	25
		Equilibrium Iodine Activity	ESBWR	15.4.9	1.8×10^{-2}	2.8×10^{-3}	2.5

Note: This table summarizes the doses from the tables that follow. For the ABWR, the whole body and thyroid doses within Tables 7.1-44 to 7.1-53 are converted into TEDE values using weighting factors as indicated in Subsection 7.1.4. For the designs being considered, there are no radiological consequences for the accidents described in SRP Sections 15.2.8, 15.3.4, and 15.4.9.

References: AP1000 DCD Rev. 17 (Westinghouse Sep 2008)
APWR DCD Rev. 2 (MHI Oct 2009)
ABWR DCD Rev. 4 (GENE Mar 1997)
ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-6
Activity Releases for AP1000 Main Steam System Piping Failure with Pre-Incident Iodine Spike

Isotope	Activity Release (Ci)				
	0–2 hr	2–8 hr	8–24 hr	24–72 hr	Total
Kr-85m	6.86×10^{-2}	1.14×10^{-1}	6.80×10^{-2}	6.20×10^{-3}	2.57×10^{-1}
Kr-85	2.82×10^{-1}	8.47×10^{-1}	2.25	6.68	1.01×10^{1}
Kr-87	2.76×10^{-2}	1.34×10^{-2}	5.20×10^{-4}	0.00	4.15×10^{-2}
Kr-88	1.12×10^{-1}	1.37×10^{-1}	4.04×10^{-2}	8.00×10^{-4}	2.90×10^{-1}
Xe-131m	1.28×10^{-1}	3.79×10^{-1}	9.81×10^{-1}	2.70	4.19
Xe-133m	1.59×10^{-1}	4.51×10^{-1}	1.04	2.05	3.70
Xe-133	1.18×10^{1}	3.45×10^{1}	8.65×10^{1}	2.16×10^{2}	3.49×10^{2}
Xe-135m	3.04×10^{-3}	1.30×10^{-5}	0.00	0.00	3.05×10^{-3}
Xe-135	3.10×10^{-1}	6.90×10^{-1}	8.35×10^{-1}	3.39×10^{-1}	2.17
Xe-138	3.99×10^{-3}	1.10×10^{-5}	0.00	0.00	4.00×10^{-3}
I-130	3.59×10^{-1}	1.42×10^{-1}	2.09×10^{-1}	1.33×10^{-1}	8.43×10^{-1}
I-131	2.40×10^{1}	1.21×10^{1}	3.10×10^{1}	8.21×10^{1}	1.49×10^{2}
I-132	3.05×10^{1}	4.14	8.07×10^{-1}	6.00×10^{-3}	3.55×10^{1}
I-133	4.34×10^{1}	1.90×10^{1}	3.53×10^{1}	3.98×10^{1}	1.38×10^{2}
I-134	6.74	1.63×10^{-1}	1.40×10^{-3}	0.00	6.90
I-135	2.60×10^{1}	8.16	7.54	1.71	4.34×10^{1}
Cs-134	1.90×10^{1}	1.95×10^{-1}	5.19×10^{-1}	1.54	2.13×10^{1}
Cs-136	2.82×10^{1}	2.86×10^{-1}	7.42×10^{-1}	2.06	3.13×10^{1}
Cs-137	1.37×10^{1}	1.41×10^{-1}	3.74×10^{-1}	1.11	1.53×10^{1}
Cs-138	1.01×10^{1}	1.02×10^{-3}	0.00	0.00	1.01×10^{1}
Total	$\mathbf{2.15 \times 10^{2}}$	$\mathbf{8.15 \times 10^{1}}$	$\mathbf{1.68 \times 10^{2}}$	$\mathbf{3.56 \times 10^{2}}$	$\mathbf{8.21 \times 10^{2}}$

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-7
Doses for AP1000 Steam System Piping Failure with Pre-Incident Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	1.00	—	8.85×10^{-2}	8.85×10^{-2}	—
0–8	—	5.81×10^{-1}	1.06×10^{-2}	—	6.16×10^{-3}
8–24	—	7.18×10^{-2}	1.31×10^{-2}	—	9.38×10^{-4}
24–96	—	1.08×10^{-1}	1.37×10^{-2}	—	1.48×10^{-3}
Total	1.00	7.61×10^{-1}	—	8.85×10^{-2}	8.57×10^{-3}
Limit	—	—	—	25	25

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-8
Activity Releases for AP1000 Main Steam System Piping Failure with Accident-Initiated Iodine Spike

Isotope	Activity Release (Ci)				
	0–2 hr	**2–8 hr**	**8–24 hr**	**24–72 hr**	**Total**
Kr-85m	6.86×10^{-2}	1.14×10^{-1}	6.80×10^{-2}	6.20×10^{-3}	2.57×10^{-1}
Kr-85	2.82×10^{-1}	8.47×10^{-1}	2.25	6.68	1.01×10^{1}
Kr-87	2.76×10^{-2}	1.34×10^{-2}	5.20×10^{-4}	0.00	4.15×10^{-2}
Kr-88	1.12×10^{-1}	1.37×10^{-1}	4.04×10^{-2}	8.00×10^{-4}	2.90×10^{-1}
Xe-131m	1.28×10^{-1}	3.79×10^{-1}	9.81×10^{-1}	2.70	4.19
Xe-133m	1.59×10^{-1}	4.51×10^{-1}	1.04	2.05	3.70
Xe-133	1.18×10^{1}	3.45×10^{1}	8.65×10^{1}	2.16×10^{2}	3.49×10^{2}
Xe-135m	3.04×10^{-3}	1.30×10^{-5}	0.00	0.00	3.05×10^{-3}
Xe-135	3.10×10^{-1}	6.90×10^{-1}	8.35×10^{-1}	3.39×10^{-1}	2.17
Xe-138	3.99×10^{-3}	1.10×10^{-5}	0.00	0.00	4.00×10^{-3}
I-130	4.15×10^{-1}	9.95×10^{-1}	1.58	1.01	4.00
I-131	2.57×10^{1}	5.73×10^{1}	1.56×10^{2}	4.13×10^{2}	6.52×10^{2}
I-132	4.57×10^{1}	9.74×10^{1}	2.23×10^{1}	2.00×10^{-1}	1.66×10^{2}
I-133	4.85×10^{1}	1.14×10^{2}	2.27×10^{2}	2.55×10^{2}	6.45×10^{2}
I-134	1.33×10^{1}	1.86×10^{1}	2.60×10^{-1}	0.00	3.22×10^{1}
I-135	3.20×10^{1}	7.74×10^{1}	7.83×10^{1}	1.77×10^{1}	2.05×10^{2}
Cs-134	1.90×10^{1}	1.95×10^{-1}	5.19×10^{-1}	1.54	2.13×10^{1}
Cs-136	2.82×10^{1}	2.86×10^{-1}	7.42×10^{-1}	2.06	3.13×10^{1}
Cs-137	1.37×10^{1}	1.41×10^{-1}	3.74×10^{-1}	1.11	1.53×10^{1}
Cs-138	1.01×10^{1}	1.02×10^{-3}	0.00	0.00	1.01×10^{1}
Total	$\mathbf{2.50 \times 10^{2}}$	$\mathbf{4.03 \times 10^{2}}$	$\mathbf{5.79 \times 10^{2}}$	$\mathbf{9.19 \times 10^{2}}$	$\mathbf{2.15 \times 10^{3}}$

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-9
Doses for AP1000 Steam System Piping Failure
with Accident-Initiated Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	1.10	—	8.85×10^{-2}	9.74×10^{-2}	—
0–8	—	1.02	1.06×10^{-2}	—	1.08×10^{-2}
8–24	—	3.77×10^{-1}	1.31×10^{-2}	—	4.93×10^{-3}
24–96	—	5.36×10^{-1}	1.37×10^{-2}	—	7.33×10^{-3}
Total	1.10	1.93	—	9.74×10^{-2}	2.31×10^{-2}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-10
Activity Releases for AP1000 Locked Rotor Accident

	Activity Release (Ci)			
	No Feedwater	With Feedwater		
Isotope	0–2 hr	0–6 hr	6–8 hr	Total
Kr-85m	8.16×10^{1}	2.38×10^{2}	4.12×10^{1}	2.79×10^{2}
Kr-85	7.58	3.03×10^{1}	1.01×10^{1}	4.04×10^{1}
Kr-87	1.20×10^{2}	2.07×10^{2}	5.40	2.12×10^{2}
Kr-88	2.08×10^{2}	5.21×10^{2}	6.04×10^{1}	5.81×10^{2}
Xe-131m	3.77	1.50×10^{1}	4.94	1.99×10^{1}
Xe-133m	2.02×10^{1}	7.85×10^{1}	2.48×10^{1}	1.03×10^{2}
Xe-133	6.66×10^{2}	2.63×10^{3}	8.57×10^{2}	3.49×10^{3}
Xe-135m	3.24×10^{1}	3.30×10^{1}	0.00	3.30×10^{1}
Xe-135	1.59×10^{2}	5.40×10^{2}	1.32×10^{2}	6.72×10^{2}
Xe-138	1.29×10^{2}	1.30×10^{2}	0.00	1.30×10^{2}
I-130	8.45×10^{-1}	8.81×10^{-1}	5.65×10^{-1}	1.45
I-131	3.77×10^{1}	4.60×10^{1}	3.46×10^{1}	8.06×10^{1}
I-132	2.79×10^{1}	1.43×10^{1}	3.95	1.83×10^{1}
I-133	4.86×10^{1}	5.33×10^{1}	3.64×10^{1}	8.97×10^{1}
I-134	2.88×10^{1}	5.53	2.09×10^{-1}	5.74
I-135	4.19×10^{1}	3.74×10^{1}	2.05×10^{1}	5.79×10^{1}
Cs-134	1.29	1.48	1.11	2.59
Cs-136	5.63×10^{-1}	5.17×10^{-1}	3.47×10^{-1}	8.64×10^{-1}
Cs-137	7.74×10^{-1}	8.71×10^{-1}	6.50×10^{-1}	1.52
Cs-138	6.08	2.95	1.13	4.08
Rb-86	1.33×10^{-2}	1.64×10^{-2}	1.27×10^{-2}	2.91×10^{-2}
Total	**1.62×10^{3}**	**4.59×10^{3}**	**1.24×10^{3}**	**5.82×10^{3}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-11
Doses for AP1000 Locked Rotor Accident with No Feedwater

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	8.00×10^{-1}	—	8.85×10^{-2}	7.08×10^{-2}	—
0–8	—	3.89×10^{-1}	1.06×10^{-2}	—	4.12×10^{-3}
Total	8.00×10^{-1}	3.89×10^{-1}	—	7.08×10^{-2}	4.12×10^{-3}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-12
Doses for AP1000 Locked Rotor Accident with Feedwater Available

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
6–8[a]	6.00×10^{-1}	—	8.85×10^{-2}	5.31×10^{-2}	—
0–8	—	7.52×10^{-1}	1.06×10^{-2}	—	7.97×10^{-3}
Total	6.00×10^{-1}	7.52×10^{-1}	—	5.31×10^{-2}	7.97×10^{-3}
Limit	—	—	—	2.5	2.5

(a) Worst case 2-hour period is 6–8 hours.

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-13
Activity Release for AP1000 Rod Ejection Accident

Isotope	Activity Release (Ci)					
	0–2 hr	2–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Kr-85m	1.12×10^{2}	6.48×10^{1}	3.87×10^{1}	1.80	0.00	2.17×10^{2}
Kr-85	5.01	5.60	1.49×10^{1}	3.35×10^{1}	2.88×10^{2}	3.47×10^{2}
Kr-87	1.82×10^{2}	2.60×10^{1}	1.03	0.00	0.00	2.09×10^{2}
Kr-88	2.91×10^{2}	1.18×10^{2}	3.49×10^{1}	3.00×10^{-1}	0.00	4.44×10^{2}
Xe-131m	4.94	5.46	1.42×10^{1}	2.86×10^{1}	1.16×10^{2}	1.69×10^{2}
Xe-133m	2.67×10^{1}	2.81×10^{1}	6.49×10^{1}	8.45×10^{1}	5.31×10^{1}	2.57×10^{2}
Xe-133	8.79×10^{2}	9.59×10^{2}	2.40×10^{3}	4.27×10^{3}	8.44×10^{3}	1.69×10^{4}
Xe-135m	7.34×10^{1}	5.00×10^{-2}	0.00	0.00	0.00	7.35×10^{1}
Xe-135	2.15×10^{2}	1.72×10^{2}	2.09×10^{2}	4.34×10^{1}	2.00×10^{-1}	6.40×10^{2}
Xe-138	2.99×10^{2}	1.40×10^{-1}	0.00	0.00	0.00	2.99×10^{2}
I-130	4.90	7.28	4.32	2.00×10^{-1}	0.00	1.67×10^{1}
I-131	1.36×10^{2}	2.45×10^{2}	2.31×10^{2}	3.10×10^{1}	1.68×10^{1}	6.60×10^{2}
I-132	1.53×10^{2}	9.94×10^{1}	9.80	0.00	0.00	2.62×10^{2}
I-133	2.72×10^{2}	4.40×10^{2}	3.18×10^{2}	2.30×10^{1}	0.00	1.05×10^{3}
I-134	1.66×10^{2}	2.85×10^{1}	1.00×10^{-1}	0.00	0.00	1.95×10^{2}
I-135	2.39×10^{2}	2.97×10^{2}	1.19×10^{2}	2.40	0.00	6.57×10^{2}
Cs-134	3.08×10^{1}	6.22×10^{1}	6.03×10^{1}	7.70	5.20	1.66×10^{2}
Cs-136	8.79	1.75×10^{1}	1.67×10^{1}	2.05	6.50×10^{-1}	4.57×10^{1}
Cs-137	1.79×10^{1}	3.62×10^{1}	3.51×10^{1}	4.52	3.05	9.68×10^{1}
Cs-138	1.09×10^{2}	7.00	0.00	0.00	0.00	1.16×10^{2}
Rb-86	3.62×10^{-1}	7.27×10^{-1}	6.96×10^{-1}	8.40×10^{-2}	3.70×10^{-2}	1.91
Total	**3.23×10^{3}**	**2.62×10^{3}**	**3.57×10^{3}**	**4.53×10^{3}**	**8.92×10^{3}**	**2.29×10^{4}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-14
Doses for AP1000 Rod Ejection Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	3.60	—	8.85×10^{-2}	3.19×10^{-1}	—
0–8	—	4.38	1.06×10^{-2}	—	4.64×10^{-2}
8–24	—	7.85×10^{-1}	1.31×10^{-2}	—	1.03×10^{-2}
24–96	—	6.34×10^{-2}	1.37×10^{-2}	—	8.66×10^{-4}
96–720	—	2.02×10^{-2}	1.01×10^{-2}	—	2.03×10^{-4}
Total	3.60	5.25	—	3.19×10^{-1}	5.78×10^{-2}
Limit	—	—	—	6.3	6.3

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-15
Activity Releases for AP1000 Small Break Outside Containment

Isotope	Activity Release (Ci)
	0–0.5 hr
Kr-85m	1.24×10^{1}
Kr-85	4.40×10^{1}
Kr-87	7.05
Kr-88	2.21×10^{1}
Xe-131m	1.99×10^{1}
Xe-133m	2.50×10^{1}
Xe-133	1.84×10^{3}
Xe-135m	2.59
Xe-135	5.20×10^{1}
Xe-138	3.65
I-130	1.89
I-131	9.26×10^{1}
I-132	3.49×10^{2}
I-133	2.01×10^{2}
I-134	1.58×10^{2}
I-135	1.68×10^{2}
Cs-134	4.16
Cs-136	6.16
Cs-137	3.00
Cs-138	2.21
Rb-86	0.00
Total	**3.01×10^{3}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-16
Doses for AP1000 Small Break Outside Containment

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	2.10	—	8.85×10^{-2}	1.86×10^{-1}	—
0–8	—	1.03	1.06×10^{-2}	—	1.09×10^{-2}
Total	2.10	1.03	—	1.86×10^{-1}	1.09×10^{-2}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-17
Activity Releases for AP1000 Steam Generator Tube Rupture with Pre-Incident Iodine Spike

Isotope	Activity Release (Ci)			
	0–2 hr	2–8 hr	8–14 hr	Total
Kr-85m	5.53×10^{1}	1.93×10^{1}	0.00	7.46×10^{1}
Kr-85	2.20×10^{2}	1.08×10^{2}	2.00×10^{-1}	3.28×10^{2}
Kr-87	2.39×10^{1}	3.61	0.00	2.75×10^{1}
Kr-88	9.22×10^{1}	2.65×10^{1}	0.00	1.19×10^{2}
Xe-131m	9.96×10^{1}	4.88×10^{1}	0.00	1.48×10^{2}
Xe-133m	1.24×10^{2}	5.91×10^{1}	1.00×10^{-1}	1.83×10^{2}
Xe-133	9.19×10^{3}	4.47×10^{3}	1.00×10^{1}	1.37×10^{4}
Xe-135m	3.44	6.00×10^{-3}	0.00	3.45
Xe-135	2.46×10^{2}	1.02×10^{2}	1.00×10^{-1}	3.48×10^{2}
Xe-138	4.56	5.00×10^{-3}	0.00	4.57
I-130	1.79	5.39×10^{-2}	2.67×10^{-1}	2.11
I-131	1.21×10^{2}	5.27	3.05×10^{1}	1.57×10^{2}
I-132	1.42×10^{2}	7.86×10^{-1}	1.91	1.45×10^{2}
I-133	2.16×10^{2}	7.63	4.06×10^{1}	2.64×10^{2}
I-134	2.74×10^{1}	1.06×10^{-2}	2.00×10^{-4}	2.74×10^{1}
I-135	1.27×10^{2}	2.70	1.17×10^{1}	1.41×10^{2}
Cs-134	1.63	6.10×10^{-2}	2.16×10^{-1}	1.91
Cs-136	2.42	8.80×10^{-2}	3.15×10^{-1}	2.82
Cs-137	1.17	4.40×10^{-2}	1.56×10^{-1}	1.37
Cs-138	5.64×10^{-1}	0.00	0.00	5.64×10^{-1}
Total	**1.07×10^{4}**	**4.85×10^{3}**	**9.61×10^{1}**	**1.57×10^{4}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-18
Doses for AP1000 Steam Generator Tube Rupture with Pre-Incident Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	2.20	—	8.85×10^{-2}	1.95×10^{-1}	—
0–8	—	1.16	1.06×10^{-2}	—	1.23×10^{-2}
8–24	—	7.20×10^{-2}	1.31×10^{-2}	—	9.41×10^{-4}
Total	2.20	1.23	—	1.95×10^{-1}	1.32×10^{-2}
Limit	—	—	—	25	25

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-19
Activity Releases for AP1000 Steam Generator Tube Rupture with Accident Initiated Iodine Spike

Isotope	Activity Release (Ci)			
	0–2 hr	2–8 hr	8–14 hr	Total
Kr-85m	5.53×10^{1}	1.93×10^{1}	0.00	7.46×10^{1}
Kr-85	2.20×10^{2}	1.08×10^{2}	2.00×10^{-1}	3.28×10^{2}
Kr-87	2.39×10^{1}	3.61	0.00	2.75×10^{1}
Kr-88	9.22×10^{1}	2.65×10^{1}	0.00	1.19×10^{2}
Xe-131m	9.96×10^{1}	4.88×10^{1}	0.00	1.48×10^{2}
Xe-133m	1.24×10^{2}	5.91×10^{1}	1.00×10^{-1}	1.83×10^{2}
Xe-133	9.19×10^{3}	4.47×10^{3}	1.00×10^{1}	1.37×10^{4}
Xe-135m	3.44	6.00×10^{-3}	0.00	3.45
Xe-135	2.46×10^{2}	1.02×10^{2}	1.00×10^{-1}	3.48×10^{2}
Xe-138	4.56	5.00×10^{-3}	0.00	4.57
I-130	8.87×10^{-1}	1.62×10^{-1}	8.23×10^{-1}	1.87
I-131	4.36×10^{1}	1.14×10^{1}	6.76×10^{1}	1.23×10^{2}
I-132	1.47×10^{2}	4.89	1.29×10^{1}	1.65×10^{2}
I-133	9.33×10^{1}	1.99×10^{1}	1.08×10^{2}	2.21×10^{2}
I-134	5.59×10^{1}	6.06×10^{-2}	6.02×10^{-2}	5.60×10^{1}
I-135	7.61×10^{1}	9.89	4.38×10^{1}	1.30×10^{2}
Cs-134	1.63	6.10×10^{-2}	2.16×10^{-1}	1.91
Cs-136	2.42	8.80×10^{-2}	3.15×10^{-1}	2.82
Cs-137	1.17	4.40×10^{-2}	1.56×10^{-1}	1.37
Cs-138	5.64×10^{-1}	0.00	0.00	5.64×10^{-1}
Total	**1.05×10^{4}**	**4.88×10^{3}**	**2.44×10^{2}**	**1.56×10^{4}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-20
Doses for AP1000 Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	1.10	—	8.85×10^{-2}	9.74×10^{-2}	—
0–8	—	6.10×10^{-1}	1.06×10^{-2}	—	6.47×10^{-3}
8–24	—	1.68×10^{-1}	1.31×10^{-2}	—	2.20×10^{-3}
Total	1.10	7.78×10^{-1}	—	9.74×10^{-2}	8.66×10^{-3}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-21 (Sheet 1 of 2)
Activity Releases for AP1000 Loss-of-Coolant Accident

Isotope	Activity Release (Ci)							
	1.4–3.4 hr	0–2 hr	2–8 hr	8–24 hr	24–72 hr	72–96 hr	96–720 hr	Total
I-130	5.64×10^{1}	3.24×10^{1}	7.85×10^{1}	6.21	5.11×10^{-1}	1.17×10^{-1}	6.00×10^{-3}	1.18×10^{2}
I-131	1.68×10^{3}	9.19×10^{2}	2.57×10^{3}	2.56×10^{2}	1.33×10^{2}	5.84×10^{1}	5.79×10^{2}	4.52×10^{3}
I-132	1.23×10^{3}	8.79×10^{2}	1.26×10^{3}	1.62×10^{1}	6.00×10^{-3}	0	0	2.16×10^{3}
I-133	3.23×10^{3}	1.82×10^{3}	4.72×10^{3}	3.71×10^{2}	7.41×10^{1}	9.90	7.80	7.00×10^{3}
I-134	6.60×10^{2}	7.09×10^{2}	4.29×10^{2}	3.07×10^{-2}	0	0	0	1.14×10^{3}
I-135	2.56×10^{3}	1.54×10^{3}	3.36×10^{3}	1.56×10^{2}	4.79	1.00×10^{-2}	0	5.06×10^{3}
Kr-85m	1.42×10^{3}	6.32×10^{2}	3.14×10^{3}	1.87×10^{3}	8.60×10^{1}	0	0	5.73×10^{3}
Kr-85	8.31×10^{1}	3.22×10^{1}	2.65×10^{2}	7.06×10^{2}	1.06×10^{3}	5.28×10^{2}	1.36×10^{4}	1.62×10^{4}
Kr-87	1.10×10^{3}	6.88×10^{2}	1.26×10^{3}	5.00×10^{1}	0	0	0	2.00×10^{3}
Kr-88	3.11×10^{3}	1.50×10^{3}	5.76×10^{3}	1.70×10^{3}	1.70×10^{1}	0	0	8.98×10^{3}
Xe-131m	8.26×10^{1}	3.21×10^{1}	2.62×10^{2}	6.79×10^{2}	9.42×10^{2}	4.31×10^{2}	5.57×10^{3}	7.92×10^{3}
Xe-133m	4.43×10^{2}	1.74×10^{2}	1.37×10^{3}	3.15×10^{3}	3.14×10^{3}	9.65×10^{2}	2.58×10^{3}	1.14×10^{4}
Xe-133	1.47×10^{4}	5.71×10^{3}	4.62×10^{4}	1.16×10^{5}	1.46×10^{5}	5.97×10^{4}	4.07×10^{5}	7.81×10^{5}
Xe-135m	1.06×10^{1}	3.33×10^{1}	2.62	0	0	0	0	3.59×10^{1}
Xe-135	3.15×10^{3}	1.31×10^{3}	8.33×10^{3}	1.01×10^{4}	2.06×10^{3}	4.00×10^{1}	1.00×10^{1}	2.19×10^{4}
Xe-138	3.11×10^{1}	1.14×10^{2}	6.90	0	0	0	0	1.21×10^{2}
Rb-86	3.04	1.72	4.60	2.80×10^{-1}	1.00×10^{-3}	0	8.00×10^{-3}	6.61
Cs-134	2.58×10^{2}	1.46×10^{2}	3.92×10^{2}	2.40×10^{1}	1.00×10^{-1}	0	1.20	5.63×10^{2}
Cs-136	7.33×10^{1}	4.14×10^{1}	1.11×10^{2}	6.70	0	0	2.00×10^{-1}	1.59×10^{2}
Cs-137	1.51×10^{2}	8.49×10^{1}	2.28×10^{2}	1.41×10^{1}	0	0	7.00×10^{-1}	3.28×10^{2}
Cs-138	1.50×10^{2}	2.60×10^{2}	6.96×10^{1}	0	0	0	0	3.30×10^{2}
Sb-127	2.42×10^{1}	1.14×10^{1}	3.67×10^{1}	2.14	1.00×10^{-2}	0	1.00×10^{-2}	5.03×10^{1}
Sb-129	5.10×10^{1}	2.71×10^{1}	6.23×10^{1}	1.48	0	0	0	9.09×10^{1}
Te-127m	3.15	1.47	4.83	2.95×10^{-1}	2.00×10^{-3}	0	1.30×10^{-2}	6.61
Te-127	2.05×10^{1}	1.02×10^{1}	2.81×10^{1}	1.11	0	0	0	3.94×10^{1}
Te-129m	1.07×10^{1}	5.01	1.64×10^{1}	1.00	1.00×10^{-2}	0	3.00×10^{-2}	2.25×10^{1}
Te-129	1.88×10^{1}	1.39×10^{1}	1.45×10^{1}	3.00×10^{-2}	0	0	0	2.84×10^{1}
Te-131	3.17×10^{1}	1.51×10^{1}	4.69×10^{1}	2.51	0	0	1.00×10^{-2}	6.45×10^{1}
Te-132	3.23×10^{2}	1.52×10^{2}	4.89×10^{2}	2.84×10^{1}	1.00×10^{-1}	0	1.00×10^{-1}	6.70×10^{2}
Sr-89	9.23×10^{1}	4.31×10^{1}	1.45×10^{2}	5.40	1.00×10^{-1}	0	3.00×10^{-1}	1.94×10^{2}
Sr-90	7.95	3.71	1.22×10^{1}	7.50×10^{-1}	0	0	4.00×10^{-2}	1.67×10^{1}
Sr-91	9.68×10^{1}	4.79×10^{1}	1.33×10^{2}	5.30	0	0	0	1.86×10^{2}
Sr-92	6.83×10^{1}	3.91×10^{1}	7.40×10^{1}	1.00	0	0	0	1.14×10^{2}

Table 7.1-21 (Sheet 2 of 2)
Activity Releases for AP1000 Loss-of-Coolant Accident

Isotope	Activity Release (Ci)							
	1.4–3.4 hr	0–2 hr	2–8 hr	8–24 hr	24–72 hr	72–96 hr	96–720 hr	Total
Ba-139	5.44×10^{1}	3.74×10^{1}	4.56×10^{1}	1.50×10^{-1}	0	0	0	8.32×10^{1}
Ba-140	1.63×10^{2}	7.61×10^{1}	2.49×10^{2}	1.51×10^{1}	0	0	4.00×10^{-1}	3.41×10^{2}
Mo-99	2.15×10^{1}	1.01×10^{1}	3.24×10^{1}	1.86	1.00×10^{-2}	0	0	4.44×10^{1}
Tc-99m	1.47×10^{1}	7.54	1.91×10^{1}	5.90×10^{-1}	0	0	0	2.72×10^{1}
Ru-103	1.73×10^{1}	8.08	2.65×10^{1}	1.62	0	1.00×10^{-2}	6.00×10^{-2}	3.63×10^{1}
Ru-105	8.18	4.33	1.00×10^{1}	2.40×10^{-1}	0	0	0	1.46×10^{1}
Ru-106	5.70	2.66	8.75	5.40×10^{-1}	0	0	3.00×10^{-2}	1.20×10^{1}
Rh-105	1.03×10^{1}	4.88	1.53×10^{1}	8.30×10^{-1}	0	0	0	2.10×10^{1}
Ce-141	3.89	1.82	5.96	3.64×10^{-1}	1.00×10^{-3}	1.00×10^{-3}	1.20×10^{-2}	8.16
Ce-143	3.46	1.64	5.14	2.78×10^{-1}	1.00×10^{-3}	0	0	7.06
Ce-144	2.94	1.37	4.51	2.76×10^{-1}	1.00×10^{-3}	1.00×10^{-3}	1.30×10^{-2}	6.17
Pu-238	9.16×10^{-3}	4.28×10^{-3}	1.41×10^{-2}	8.60×10^{-4}	0	0	4.00×10^{-5}	1.93×10^{-2}
Pu-239	8.06×10^{-4}	3.76×10^{-4}	1.24×10^{-3}	7.60×10^{-5}	0	1.00×10^{-6}	3.00×10^{-6}	1.70×10^{-3}
Pu-240	1.18×10^{-3}	5.52×10^{-4}	1.81×10^{-3}	1.11×10^{-4}	1.00×10^{-6}	0	5.00×10^{-6}	2.48×10^{-3}
Pu-241	2.65×10^{-1}	1.24×10^{-1}	4.08×10^{-1}	2.50×10^{-2}	1.00×10^{-4}	0	1.20×10^{-3}	5.58×10^{-1}
Np-239	4.48×10^{1}	2.12×10^{1}	6.75×10^{1}	3.84	1.00×10^{-2}	1.00×10^{-2}	1.00×10^{-2}	9.26×10^{1}
Y-90	8.08×10^{-2}	3.81×10^{-2}	1.22×10^{-1}	7.00×10^{-3}	0	0	0	1.67×10^{-1}
Y-91	1.19	5.54×10^{-1}	1.82	1.11×10^{-1}	1.00×10^{-3}	0	4.00×10^{-3}	2.49
Y-92	7.89×10^{-1}	4.32×10^{-1}	9.19×10^{-1}	1.80×10^{-2}	0	0	0	1.37
Y-93	1.21	6.00×10^{-1}	1.68	6.80×10^{-2}	0	0	0	2.35
Nb-95	1.59	7.46×10^{-1}	2.44	1.49×10^{-1}	1.00×10^{-3}	0	5.00×10^{-3}	3.34
Zr-95	1.59	7.41×10^{-1}	2.43	1.49×10^{-1}	0	0	6.00×10^{-3}	3.33
Zr-97	1.43	6.89×10^{-1}	2.05	9.80×10^{-2}	0	0	0	2.84
La-140	1.67	7.92×10^{-1}	2.50	1.39×10^{-1}	0	0	0	3.43
La-141	1.03	5.54×10^{-1}	1.23	2.70×10^{-2}	0	0	0	1.81
La-142	5.38×10^{-1}	3.57×10^{-1}	4.74×10^{-1}	2.00×10^{-3}	0	0	0	8.33×10^{-1}
Nd-147	6.16×10^{-1}	2.89×10^{-1}	9.42×10^{-1}	5.70×10^{-2}	0	0	1.00×10^{-3}	1.29
Pr-143	1.39	6.50×10^{-1}	2.13	1.28×10^{-1}	1.00×10^{-3}	0	3.00×10^{-3}	2.91
Am-241	1.20×10^{-4}	5.59×10^{-5}	1.84×10^{-4}	1.13×10^{-5}	0	0	6.00×10^{-7}	2.52×10^{-4}
Cm-242	2.82×10^{-2}	1.32×10^{-2}	4.33×10^{-2}	2.65×10^{-3}	1.00×10^{-5}	1.00×10^{-5}	1.20×10^{-4}	5.93×10^{-2}
Cm-244	3.46×10^{-3}	1.62×10^{-3}	5.32×10^{-3}	3.26×10^{-4}	1.00×10^{-6}	0	1.60×10^{-5}	7.28×10^{-3}
Total	$\mathbf{3.53 \times 10^{4}}$	$\mathbf{1.72 \times 10^{4}}$	$\mathbf{8.14 \times 10^{4}}$	$\mathbf{1.35 \times 10^{5}}$	$\mathbf{1.54 \times 10^{5}}$	$\mathbf{6.17 \times 10^{4}}$	$\mathbf{4.29 \times 10^{5}}$	$\mathbf{8.78 \times 10^{5}}$

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-22
Doses for AP1000 Loss-of-Coolant Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
1.4–3.4[a]	2.46×10^{1}	—	1.74×10^{-1}	4.27	—
0–8	—	2.17×10^{1}	2.41×10^{-2}	—	5.23×10^{-1}
8–24	—	7.50×10^{-1}	2.45×10^{-2}	—	1.84×10^{-2}
24–96	—	2.93×10^{-1}	2.05×10^{-2}	—	6.01×10^{-3}
96–720	—	5.49×10^{-1}	1.01×10^{-2}	—	5.52×10^{-3}
Total	2.46×10^{1}	2.33×10^{1}	—	4.27	5.53×10^{-1}
Limit	—	—	—	25	25

(a) Worst case 2-hour period is 1.4–3.4 hours.

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-23
Activity Releases for AP1000 Fuel Handling Accident

Isotope	Activity Release (Ci)
	0–2 hr
Kr-85m	8.40
Kr-85	1.10×10^{3}
Kr-88	3.00×10^{-1}
Xe-131m	5.52×10^{2}
Xe-133m	2.30×10^{3}
Xe-133	8.88×10^{4}
Xe-135m	1.02×10^{2}
Xe-135	5.68×10^{3}
I-130	7.00×10^{-1}
I-131	3.47×10^{2}
I-132	2.44×10^{2}
I-133	1.08×10^{2}
I-135	3.20
Total	**9.92×10^{4}**

Note: Source terms are for the radiation dose values recorded in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-24
Doses for AP1000 Fuel Handling Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	5.20	—	8.85×10^{-2}	4.60×10^{-1}	—
0–8	—	2.59	1.06×10^{-2}	—	2.75×10^{-2}
Total	5.20	2.59	—	4.60×10^{-1}	2.75×10^{-2}
Limit	—	—	—	6.3	6.3

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 17 (Westinghouse Sep 2008)

Table 7.1-25
Activity Releases for APWR Steam System Piping Failure with Pre-Incident Iodine Spike

Isotope	Activity Release (Ci)		
	0–8 hr	8–24 hr	Total
Kr-85	3.21×10^{1}	2.40×10^{1}	5.61×10^{1}
Kr-85m	3.56×10^{-1}	8.77×10^{-2}	4.43×10^{-1}
Kr-87	9.12×10^{-2}	1.13×10^{-3}	9.23×10^{-2}
Kr-88	5.10×10^{-1}	6.46×10^{-2}	5.74×10^{-1}
Xe-133	1.07×10^{2}	7.75×10^{1}	1.85×10^{2}
Xe-135	4.38	3.39	7.78
I-131	1.72×10^{1}	7.25	2.44×10^{1}
I-132	6.18	1.66×10^{-1}	6.35
I-133	2.79×10^{1}	9.03	3.69×10^{1}
I-134	3.49	1.01×10^{-3}	3.49
I-135	1.62×10^{1}	2.73	1.89×10^{1}
Rb-86	8.64×10^{-2}	1.62×10^{-3}	8.80×10^{-2}
Cs-134	8.80	1.68×10^{-1}	8.97
Cs-136	2.32	4.33×10^{-2}	2.37
Cs-137	5.01	9.56×10^{-2}	5.11
Total	2.32×10^{2}	1.25×10^{2}	3.56×10^{2}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-26
Doses for APWR Steam System Piping Failure with Pre-Incident Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	1.9×10^{-1}	—	1.77×10^{-1}	3.4×10^{-2}	—
0–8	—	1.0×10^{-1}	2.52×10^{-2}	—	2.5×10^{-3}
8–24	—	7.6×10^{-3}	3.02×10^{-2}	—	2.3×10^{-4}
Total	1.9×10^{-1}	1.1×10^{-1}	—	3.4×10^{-2}	2.8×10^{-3}
Limit	—	—	—	25	25

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-27
Activity Releases for APWR Steam System Piping Failure with Accident-Initiated Iodine Spike

Isotope	Activity Release (Ci)		
	0–8 hr	8–24 hr	Total
Kr-85	3.21×10^{1}	2.40×10^{1}	5.61×10^{1}
Kr-85m	3.56×10^{-1}	8.77×10^{-2}	4.43×10^{-1}
Kr-87	9.12×10^{-2}	1.13×10^{-3}	9.23×10^{-2}
Kr-88	5.10×10^{-1}	6.46×10^{-2}	5.74×10^{-1}
Xe-133	1.08×10^{2}	8.03×10^{1}	1.88×10^{2}
Xe-135	7.61	1.33×10^{1}	2.09×10^{1}
I-131	5.05×10^{1}	6.50×10^{1}	1.16×10^{2}
I-132	9.89	1.49	1.14×10^{1}
I-133	7.65×10^{1}	8.09×10^{1}	1.57×10^{2}
I-134	3.77	9.11×10^{-3}	3.78
I-135	3.77×10^{1}	2.45×10^{1}	6.21×10^{1}
Rb-86	8.64×10^{-2}	1.62×10^{-3}	8.80×10^{-2}
Cs-134	8.80	1.68×10^{-1}	8.97
Cs-136	2.32	4.33×10^{-2}	2.37
Cs-137	5.01	9.56×10^{-2}	5.11
Total	3.43×10^{2}	2.90×10^{2}	6.33×10^{2}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-28
Doses for APWR Steam System Piping Failure with Accident-Initiated Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	3.2×10^{-1}	—	1.77×10^{-1}	5.7×10^{-2}	—
0–8	—	2.1×10^{-1}	2.52×10^{-2}	—	5.3×10^{-3}
8–24	—	6.5×10^{-2}	3.02×10^{-2}	—	2.0×10^{-3}
Total	3.2×10^{-1}	2.8×10^{-1}	—	5.7×10^{-2}	7.3×10^{-3}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-29
Activity Releases for APWR Locked Rotor Accident

Isotope	Activity Release (Ci)		
	0–8 hr	8–24 hr	Total
Kr-85	1.12×10^{2}	8.40×10^{1}	1.96×10^{2}
Kr-85m	6.40×10^{2}	1.58×10^{2}	7.98×10^{2}
Kr-87	5.02×10^{2}	6.21	5.08×10^{2}
Kr-88	1.37×10^{3}	1.74×10^{2}	1.55×10^{3}
Xe-133	6.87×10^{3}	4.96×10^{3}	1.18×10^{4}
Xe-135	1.61×10^{3}	7.67×10^{2}	2.37×10^{3}
I-131	8.81×10^{1}	2.32×10^{2}	3.20×10^{2}
I-132	1.94×10^{1}	8.35	2.77×10^{1}
I-133	9.85×10^{1}	2.17×10^{2}	3.15×10^{2}
I-134	6.46	1.10×10^{-1}	6.57
I-135	6.38×10^{1}	9.16×10^{1}	1.55×10^{2}
Rb-86	3.23×10^{-2}	8.66×10^{-2}	1.19×10^{-1}
Cs-134	3.24	8.78	1.20×10^{1}
Cs-136	8.72×10^{-1}	2.33	3.21
Cs-137	1.84	5.00	6.84
Total	1.14×10^{4}	6.71×10^{3}	1.81×10^{4}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-30
Doses for APWR Locked Rotor Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
10–12[(a)]	4.9×10^{-1}	—	1.77×10^{-1}	8.7×10^{-2}	—
0–8	—	4.4×10^{-1}	2.52×10^{-2}	—	1.1×10^{-2}
8–24	—	2.5×10^{-1}	3.02×10^{-2}	—	7.5×10^{-3}
Total	4.9×10^{-1}	6.9×10^{-1}	—	8.7×10^{-2}	1.9×10^{-2}
Limit	—	—	—	2.5	2.5

(a) Worst case 2-hour period is 10–12 hours.

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-31
Activity Releases for APWR Rod Ejection Accident

Isotope	Activity Release (Ci)				
	0–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Kr-85	2.63×10^{2}	2.50×10^{2}	1.90×10^{2}	1.63×10^{3}	2.33×10^{3}
Kr-85m	3.59×10^{3}	9.58×10^{2}	9.86	0	4.56×10^{3}
Kr-87	2.81×10^{3}	3.50×10^{1}	0	0	2.85×10^{3}
Kr-88	7.70×10^{3}	1.02×10^{3}	2.05	0	8.72×10^{3}
Xe-133	3.81×10^{4}	3.46×10^{4}	2.11×10^{4}	4.22×10^{4}	1.36×10^{5}
Xe-135	9.31×10^{3}	5.32×10^{3}	5.40×10^{2}	2.81	1.52×10^{4}
I-131	5.82×10^{2}	7.17×10^{2}	2.58×10^{2}	7.79×10^{2}	2.34×10^{3}
I-132	4.62×10^{2}	3.93×10^{1}	1.40×10^{-2}	0	5.01×10^{2}
I-133	1.12×10^{3}	1.06×10^{3}	1.13×10^{2}	1.13×10^{1}	2.30×10^{3}
I-134	4.95×10^{2}	5.15×10^{-1}	0	0	4.95×10^{2}
I-135	8.75×10^{2}	4.39×10^{2}	6.60	4.00×10^{-3}	1.32×10^{3}
Rb-86	4.16×10^{-1}	9.65×10^{-2}	0	0	5.13×10^{-1}
Cs-134	4.15×10^{1}	9.79	1.01×10^{-3}	0	5.13×10^{1}
Cs-136	1.13×10^{1}	2.60	1.00×10^{-6}	0	1.39×10^{1}
Cs-137	2.36×10^{1}	5.57	0	0	2.92×10^{1}
Total	6.53×10^{4}	4.45×10^{4}	2.22×10^{4}	4.46×10^{4}	1.77×10^{5}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-32
Doses for APWR Rod Ejection Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	5.1	—	1.77×10^{-1}	9.0×10^{-1}	—
0–8	—	3.2	2.52×10^{-2}	—	8.1×10^{-2}
8–24	—	8.8×10^{-1}	3.02×10^{-2}	—	2.7×10^{-2}
24–96	—	1.6×10^{-1}	2.97×10^{-2}	—	4.8×10^{-3}
96–720	—	1.7×10^{-1}	2.88×10^{-2}	—	4.9×10^{-3}
Total	5.1	4.4	—	9.0×10^{-1}	1.2×10^{-1}
Limit	—	—	—	6.3	6.3

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-33
Activity Releases for APWR Small Break Outside Containment

Isotope	Activity Release (Ci), 0-8 hr
Kr-85	6.84×10^{2}
Kr-85m	1.25×10^{1}
Kr-87	7.05
Kr-88	2.26×10^{1}
Xe-133	2.32×10^{3}
Xe-135	7.70×10^{1}
I-131	1.72×10^{2}
I-132	7.98×10^{1}
I-133	2.93×10^{2}
I-134	4.33×10^{1}
I-135	1.85×10^{2}
Total	3.90×10^{3}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-34
Doses for APWR Small Break Outside Containment

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	1.5	—	1.77×10^{-1}	2.7×10^{-1}	—
0–8	—	6.0×10^{-1}	2.52×10^{-2}	—	1.5×10^{-2}
Total	1.5	6.0×10^{-1}	—	2.7×10^{-1}	1.5×10^{-2}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-35
Activity Releases for APWR Steam Generator Tube Rupture with Pre-Incident Iodine Spike

Isotope	Activity Release (Ci)				
	0–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Kr-85	3.43×10^{3}	4.64×10^{1}	2.06×10^{2}	1.59×10^{3}	5.27×10^{3}
Kr-85m	6.17×10^{1}	9.70×10^{-2}	8.00×10^{-3}	0	6.18×10^{1}
Kr-87	3.40×10^{1}	0	0	0	3.40×10^{1}
Kr-88	1.11×10^{2}	6.00×10^{-2}	1.00×10^{-2}	0	1.11×10^{2}
Xe-133	1.16×10^{4}	1.44×10^{2}	5.06×10^{2}	9.44×10^{2}	1.32×10^{4}
Xe-135	3.75×10^{2}	2.18	6.70×10^{-1}	0	3.78×10^{2}
I-131	4.18×10^{2}	1.81	0	0	4.20×10^{2}
I-132	2.09×10^{2}	3.92×10^{-2}	0	0	2.09×10^{2}
I-133	7.16×10^{2}	2.24	0	0	7.18×10^{2}
I-134	1.28×10^{2}	6.00×10^{-5}	0	0	1.28×10^{2}
I-135	4.61×10^{2}	6.70×10^{-1}	0	0	4.62×10^{2}
Rb-86	4.54×10^{-3}	5.44×10^{-4}	0	0	5.09×10^{-3}
Cs-134	4.63×10^{-1}	5.63×10^{-2}	0	0	5.19×10^{-1}
Cs-136	1.22×10^{-1}	1.45×10^{-2}	0	0	1.37×10^{-1}
Cs-137	2.64×10^{-1}	3.21×10^{-2}	0	0	2.96×10^{-1}
Total	1.76×10^{4}	1.98×10^{2}	7.12×10^{2}	2.53×10^{3}	2.10×10^{4}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-36
Doses for APWR Steam Generator Tube Rupture with Pre-Incident Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	3.6	—	1.77×10^{-1}	6.4×10^{-1}	—
0–8	—	1.5	2.52×10^{-2}	—	3.8×10^{-2}
8–24	—	2.0×10^{-3}	3.02×10^{-2}	—	6.0×10^{-5}
24–96	—	2.1×10^{-4}	2.97×10^{-2}	—	6.2×10^{-6}
96–720	—	1.7×10^{-4}	2.88×10^{-2}	—	4.9×10^{-6}
Total	3.6	1.5	—	6.4×10^{-1}	3.8×10^{-2}
Limit	—	—	—	25	25

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-37
Activity Releases for APWR Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

Isotope	Activity Release (Ci)				
	0–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Kr-85	3.43×10^{3}	4.64×10^{1}	2.06×10^{2}	1.59×10^{3}	5.27×10^{3}
Kr-85m	6.17×10^{1}	9.70×10^{-2}	8.00×10^{-3}	0	6.18×10^{1}
Kr-87	3.40×10^{1}	0	0	0	3.40×10^{1}
Kr-88	1.11×10^{2}	6.00×10^{-2}	1.00×10^{-2}	0	1.11×10^{2}
Xe-133	1.16×10^{4}	1.45×10^{2}	5.06×10^{2}	9.44×10^{2}	1.32×10^{4}
Xe-135	3.70×10^{2}	3.82	6.70×10^{-1}	0	3.74×10^{2}
I-131	1.10×10^{2}	1.03×10^{1}	0	0	1.20×10^{2}
I-132	5.24×10^{1}	2.12×10^{-1}	0	0	5.26×10^{1}
I-133	1.87×10^{2}	1.27×10^{1}	0	0	2.00×10^{2}
I-134	3.05×10^{1}	1.06×10^{-3}	0	0	3.05×10^{1}
I-135	1.19×10^{2}	3.74	0	0	1.23×10^{2}
Rb-86	4.54×10^{-3}	5.44×10^{-4}	0	0	5.09×10^{-3}
Cs-134	4.63×10^{-1}	5.63×10^{-2}	0	0	5.19×10^{-1}
Cs-136	1.22×10^{-1}	1.45×10^{-2}	0	0	1.37×10^{-1}
Cs-137	2.64×10^{-1}	3.21×10^{-2}	0	0	2.96×10^{-1}
Total	1.61×10^{4}	2.22×10^{2}	7.12×10^{2}	2.53×10^{3}	1.96×10^{4}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-38
Doses for APWR Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	9.6×10^{-1}	—	1.77×10^{-1}	1.7×10^{-1}	—
0–8	—	4.1×10^{-1}	2.52×10^{-2}	—	1.0×10^{-2}
8–24	—	1.0×10^{-2}	3.02×10^{-2}	—	3.0×10^{-4}
24–96	—	2.1×10^{-4}	2.97×10^{-2}	—	6.2×10^{-6}
96–720	—	1.7×10^{-4}	2.88×10^{-2}	—	4.9×10^{-6}
Total	9.6×10^{-1}	4.2×10^{-1}	—	1.7×10^{-1}	1.1×10^{-2}
Limit	—	—	—	2.5	2.5

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-39 (Sheet 1 of 2)
Activity Releases for APWR Loss-of-Coolant Accident

Isotope	Activity Release (Ci)				
	0–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Kr-85	7.75×10^{2}	1.74×10^{3}	3.92×10^{3}	3.35×10^{4}	3.99×10^{4}
Kr-85m	9.16×10^{3}	4.37×10^{3}	1.99×10^{2}	0	1.37×10^{4}
Kr-87	3.54×10^{3}	7.83×10^{1}	0	0	3.62×10^{3}
Kr-88	1.68×10^{4}	3.68×10^{3}	3.70×10^{1}	0	2.05×10^{4}
Xe-133	1.26×10^{5}	2.76×10^{5}	4.93×10^{5}	9.77×10^{5}	1.87×10^{6}
Xe-135	3.79×10^{4}	4.05×10^{4}	9.60×10^{3}	4.41×10^{1}	8.80×10^{4}
I-131	1.42×10^{3}	5.61×10^{2}	1.85×10^{3}	5.60×10^{3}	9.43×10^{3}
I-132	1.50×10^{3}	1.01×10^{2}	2.22×10^{2}	2.48×10^{2}	2.07×10^{3}
I-133	2.67×10^{3}	7.37×10^{2}	8.09×10^{2}	8.07×10^{1}	4.30×10^{3}
I-134	4.22×10^{2}	1.84×10^{-1}	0	0	4.22×10^{2}
I-135	1.95×10^{3}	2.44×10^{2}	4.67×10^{1}	1.20×10^{-1}	2.24×10^{3}
Rb-86	1.44	1.60×10^{-2}	0	0	1.45
Cs-134	1.44×10^{2}	1.62	0	0	1.46×10^{2}
Cs-136	3.90×10^{1}	4.31×10^{-1}	0	0	3.94×10^{1}
Cs-137	8.19×10^{1}	9.21×10^{-1}	1.00×10^{-3}	0	8.28×10^{1}
Sb-127	1.04×10^{1}	1.26×10^{-1}	1.00×10^{-5}	0	1.05×10^{1}
Sb-129	1.99×10^{1}	6.87×10^{-2}	0	0	2.00×10^{1}
Te-127	1.04×10^{1}	1.30×10^{-1}	0	0	1.05×10^{1}
Te-127m	1.39	1.80×10^{-2}	0	0	1.40
Te-129	2.30×10^{1}	1.12×10^{-1}	0	0	2.31×10^{1}
Te-129m	4.75	6.13×10^{-2}	0	0	4.81
Te-131m	1.36×10^{1}	1.44×10^{-1}	0	0	1.37×10^{1}
Te-132	1.41×10^{2}	1.71	1.00×10^{-4}	0	1.43×10^{2}
Sr-89	4.74×10^{1}	6.12×10^{-1}	0	0	4.80×10^{1}
Sr-90	3.93	5.10×10^{-2}	0	0	3.98
Sr-91	5.01×10^{1}	3.54×10^{-1}	1.00×10^{-3}	0	5.05×10^{1}
Sr-92	3.11×10^{1}	4.95×10^{-2}	0	0	3.11×10^{1}
Ba-139	1.96×10^{1}	5.04×10^{-3}	0	0	1.96×10^{1}
Ba-140	7.49×10^{1}	9.53×10^{-1}	0	0	7.59×10^{1}
Co-58	3.36×10^{-3}	4.50×10^{-8}	0	0	3.36×10^{-3}
Co-60	1.59×10^{-2}	2.00×10^{-4}	1.01×10^{-6}	0	1.61×10^{-2}
Mo-99	9.57	1.11×10^{-1}	1.00×10^{-4}	0	9.68
Tc-99m	8.50	1.04×10^{-1}	1.00×10^{-4}	0	8.60

Table 7.1-39 (Sheet 2 of 2)
Activity Releases for APWR Loss-of-Coolant Accident

Isotope	Activity Release (Ci)				
	0–8 hr	8–24 hr	24–96 hr	96–720 hr	Total
Ru-103	7.62	9.83×10^{-2}	1.01×10^{-4}	0	7.72
Ru-105	3.14	1.12×10^{-2}	0	0	3.15
Ru-106	2.67	3.46×10^{-2}	0	0	2.70
Rh-105	4.61	5.41×10^{-2}	0	0	4.67
Y-90	7.44×10^{-2}	5.12×10^{-3}	6.06×10^{-6}	0	7.96×10^{-2}
Y-91	6.00×10^{-1}	8.54×10^{-3}	0	0	6.09×10^{-1}
Y-92	4.13	1.04×10^{-1}	0	0	4.24
Y-93	5.90×10^{-1}	4.32×10^{-3}	0	0	5.94×10^{-1}
Zr-95	7.55×10^{-1}	9.76×10^{-3}	0	0	7.65×10^{-1}
Zr-97	6.65×10^{-1}	6.12×10^{-3}	0	0	6.71×10^{-1}
Nb-95	7.60×10^{-1}	9.85×10^{-3}	1.01×10^{-5}	0	7.69×10^{-1}
La-140	1.76	1.43×10^{-1}	2.02×10^{-4}	0	1.90
La-141	4.25×10^{-1}	1.29×10^{-3}	0	0	4.27×10^{-1}
La-142	2.01×10^{-1}	7.07×10^{-5}	0	0	2.01×10^{-1}
Pr-143	6.74×10^{-1}	8.91×10^{-3}	1.00×10^{-5}	0	6.83×10^{-1}
Nd-147	2.80×10^{-1}	3.55×10^{-3}	0	0	2.83×10^{-1}
Am-241	7.51×10^{-5}	9.77×10^{-7}	0	0	7.60×10^{-5}
Cm-242	1.86×10^{-2}	2.41×10^{-4}	0	0	1.88×10^{-2}
Cm-244	2.26×10^{-3}	2.93×10^{-5}	0	0	2.29×10^{-3}
Ce-141	1.78	2.29×10^{-2}	0	0	1.80
Ce-143	1.63	1.78×10^{-2}	0	0	1.65
Ce-144	1.35	1.75×10^{-2}	0	0	1.36
Np-239	1.85×10^{1}	2.16×10^{-1}	1.00×10^{-5}	0	1.87×10^{1}
Pu-238	5.30×10^{-3}	6.88×10^{-5}	0	0	5.37×10^{-3}
Pu-239	4.00×10^{-4}	5.19×10^{-6}	0	0	4.05×10^{-4}
Pu-240	6.28×10^{-4}	8.14×10^{-6}	1.01×10^{-8}	0	6.36×10^{-4}
Pu-241	1.39×10^{-1}	1.81×10^{-3}	0	0	1.41×10^{-1}
Total	2.03×10^{5}	3.28×10^{5}	5.09×10^{5}	1.02×10^{6}	2.06×10^{6}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-40
Doses for APWR Loss-of-Coolant Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0.5–2.5[a]	1.3×10^{1}	—	1.77×10^{-1}	2.3	—
0–8	—	9.0	2.52×10^{-2}	—	2.3×10^{-1}
8–24	—	1.2	3.02×10^{-2}	—	3.6×10^{-2}
24–96	—	1.3	2.97×10^{-2}	—	3.9×10^{-2}
96–720	—	1.4	2.88×10^{-2}	—	4.0×10^{-2}
Total	1.3×10^{1}	1.3×10^{1}	—	2.3	3.4×10^{-1}
Limit	—	—	—	25	25

(a) Worst case 2-hour period is 0.5–2.5 hours

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-41
Activity Releases for APWR Fuel Handling Accident

Isotope	Activity Release (Ci), 0–8 hr
Kr-85	1.20×10^{3}
Kr-85m	3.90×10^{2}
Kr-87	5.98×10^{-2}
Kr-88	1.25×10^{2}
Xe-133	9.90×10^{4}
Xe-135	2.21×10^{4}
I-131	3.67×10^{2}
I-132	2.75×10^{2}
I-133	2.31×10^{2}
I-134	2.71×10^{-6}
I-135	3.80×10^{1}
Total	1.24×10^{5}

Reference: APWR DCD Rev. 2 (MHI Oct 2009)

Table 7.1-42
Doses for APWR Fuel Handling Accident

Time (hr)	DCD Dose (rem TEDE)		X/Q Ratio	Site Dose (rem TEDE)	
	EAB	LPZ	Site to DCD	EAB	LPZ
0–2	3.3	—	1.77×10^{-1}	5.8×10^{-1}	—
0–8	—	1.4	2.52×10^{-2}	—	3.5×10^{-2}
Total	3.3	1.4	—	5.8×10^{-1}	3.5×10^{-2}
Limit	—	—	—	6.3	6.3

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 2 (MHI Oct 2009)

Table 7.1-43
Activity Releases for ABWR Main Steam Line Break

Isotope	Activity Release (MBq)	
	Equilibrium Activity	Pre-Incident Spike
I-131	7.29×10^{4}	1.46×10^{6}
I-132	7.10×10^{5}	1.42×10^{7}
I-133	5.00×10^{5}	9.99×10^{6}
I-134	1.40×10^{6}	2.79×10^{7}
I-135	7.29×10^{5}	1.46×10^{7}
Kr-83m	4.07×10^{2}	2.44×10^{3}
Kr-85m	7.18×10^{2}	4.29×10^{3}
Kr-85	2.26	1.36×10^{1}
Kr-87	2.44×10^{3}	1.47×10^{4}
Kr-88	2.46×10^{3}	1.48×10^{4}
Kr-89	9.88×10^{3}	5.92×10^{4}
Kr-90	2.55×10^{3}	1.55×10^{4}
Xe-131m	1.76	1.06×10^{1}
Xe-133m	3.39×10^{1}	2.04×10^{2}
Xe-133	9.47×10^{2}	5.70×10^{3}
Xe-135m	2.89×10^{3}	1.74×10^{4}
Xe-135	2.70×10^{3}	1.62×10^{4}
Xe-137	1.23×10^{4}	7.40×10^{4}
Xe-138	9.44×10^{3}	5.66×10^{4}
Xe-139	4.33×10^{3}	2.59×10^{4}

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-44
Doses for ABWR Main Steam Line Break with Pre-Incident Iodine Spike

Location	Time (hr)	DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
		W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	1.3×10^{-2}	5.1×10^{-1}	6.46×10^{-2}	8.4×10^{-2}	3.3
LPZ	0–8	—	—	3.87×10^{-3}	5.0×10^{-3}	2.0×10^{-1}
Limit	—	—	—	—	25	300

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-45
Doses for ABWR Main Steam Line Break with Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
		W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	6.2×10^{-4}	2.6×10^{-2}	6.46×10^{-2}	4.0×10^{-3}	1.7×10^{-1}
LPZ	0–8	—	—	3.87×10^{-3}	2.4×10^{-4}	1.0×10^{-2}
Limit	—	—	—	—	2.5	30

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-46
Activity Releases for ABWR Small Break Outside Containment

Isotope	Activity Release (MBq)					
	1 min	10 min	1 hr	2 hr	4 hr	8 hr
I-131	6.36×10^{-1}	5.77×10^{1}	2.77×10^{4}	6.81×10^{4}	1.27×10^{5}	1.41×10^{5}
I-132	6.18	5.51×10^{2}	2.52×10^{5}	5.96×10^{5}	1.09×10^{6}	1.19×10^{6}
I-133	4.37	3.96×10^{2}	1.87×10^{5}	4.59×10^{5}	8.51×10^{5}	9.44×10^{5}
I-134	1.21×10^{1}	1.06×10^{3}	4.44×10^{5}	9.92×10^{5}	1.76×10^{6}	1.90×10^{6}
I-135	6.36	5.74×10^{2}	2.71×10^{5}	6.59×10^{5}	1.21×10^{6}	1.34×10^{6}

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-47
Doses for ABWR Small Break Outside Containment

Location	Time (hr)	DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
		W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	9.4×10^{-4}	4.8×10^{-2}	6.46×10^{-2}	6.1×10^{-3}	3.1×10^{-1}
LPZ	0–8	—	—	3.87×10^{-3}	3.6×10^{-4}	1.9×10^{-2}
Limit	—	—	—	—	2.5	30

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-48 (Sheet 1 of 2)
Activity Releases for ABWR Loss-of-Coolant Accident

Isotope	Activity Release from Reactor Building (MBq)									
	1 min	10 min	1 hr	2 hr	4 hr	8 hr	12 hr	24 hr	96 hr	720 hr
I-131	2.9×10^4	2.6×10^6	9.6×10^6	9.6×10^6	1.0×10^7	1.3×10^7	1.7×10^7	3.6×10^7	1.9×10^8	6.7×10^8
I-132	4.1×10^4	3.7×10^6	1.3×10^7	1.3×10^7	1.4×10^7	1.4×10^7	1.5×10^7	1.5×10^7	1.5×10^7	1.5×10^7
I-133	5.9×10^4	5.6×10^6	2.0×10^7	2.0×10^7	2.1×10^7	2.6×10^7	3.3×10^7	5.6×10^7	1.2×10^8	1.3×10^8
I-134	6.7×10^4	5.6×10^6	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7
I-135	5.6×10^4	5.2×10^6	1.9×10^7	1.9×10^7	2.0×10^7	2.3×10^7	2.6×10^7	3.1×10^7	3.5×10^7	3.5×10^7
Kr-83m	2.7×10^4	2.3×10^6	9.3×10^6	1.2×10^7	1.9×10^7	2.8×10^7	3.2×10^7	3.3×10^7	3.3×10^7	3.3×10^7
Kr-85	2.6×10^3	2.3×10^5	1.0×10^6	1.5×10^6	3.6×10^6	1.2×10^7	2.4×10^7	8.1×10^7	6.7×10^8	5.6×10^9
Kr-85m	5.6×10^4	5.2×10^6	2.1×10^7	3.1×10^7	5.9×10^7	1.3×10^8	1.9×10^8	2.7×10^8	2.9×10^8	2.9×10^8
Kr-87	1.1×10^5	9.3×10^6	3.6×10^7	4.4×10^7	6.3×10^7	7.8×10^7	8.1×10^7	8.1×10^7	8.1×10^7	8.1×10^7
Kr-88	1.6×10^5	1.4×10^7	5.6×10^7	7.8×10^7	1.4×10^8	2.5×10^8	3.1×10^8	3.6×10^8	3.7×10^8	3.7×10^8
Kr-89	1.7×10^5	4.8×10^6	6.7×10^6	6.7×10^6	6.7×10^6	6.7×10^6	6.7×10^6	6.7×10^6	6.7×10^6	6.7×10^6
Xe-131m	1.3×10^3	1.2×10^5	5.2×10^5	7.8×10^5	1.9×10^6	5.9×10^6	1.3×10^7	4.1×10^7	3.0×10^8	1.4×10^9
Xe-133	4.8×10^5	4.1×10^7	1.8×10^8	2.8×10^8	6.7×10^8	2.1×10^9	4.4×10^9	1.4×10^{10}	8.9×10^{10}	2.5×10^{11}
Xe-133m	2.0×10^4	1.8×10^6	7.4×10^6	1.1×10^7	2.7×10^7	8.5×10^7	1.7×10^8	5.2×10^8	2.6×10^9	4.1×10^9
Xe-135	5.9×10^4	5.6×10^6	2.3×10^7	3.4×10^7	7.4×10^7	1.9×10^8	3.3×10^8	6.7×10^8	1.0×10^9	1.0×10^9
Xe-135m	8.5×10^4	5.9×10^6	1.7×10^7	1.8×10^7	1.8×10^7	1.8×10^7	1.8×10^7	1.8×10^7	1.8×10^7	1.8×10^7
Xe-137	3.7×10^5	1.3×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7	1.9×10^7
Xe-138	3.7×10^5	2.6×10^7	7.4×10^7	7.4×10^7	7.4×10^7	7.4×10^7	7.4×10^7	7.4×10^7	7.4×10^7	7.4×10^7

Table 7.1-48 (Sheet 2 of 2)
Activity Releases for ABWR Loss-of-Coolant Accident

Isotope	Activity Release from Condenser (MBq)									
	1 min	**10 min**	**1 hr**	**2 hr**	**4 hr**	**8 hr**	**12 hr**	**24 hr**	**96 hr**	**720 hr**
I-131	0	0	7.0×10^{2}	1.2×10^{4}	1.2×10^{5}	8.5×10^{5}	2.4×10^{6}	1.2×10^{7}	1.8×10^{8}	2.0×10^{9}
I-132	0	0	8.1×10^{2}	1.1×10^{4}	7.0×10^{4}	2.4×10^{5}	3.5×10^{5}	4.4×10^{5}	4.4×10^{5}	4.4×10^{5}
I-133	0	0	1.5×10^{3}	2.4×10^{4}	2.3×10^{5}	1.5×10^{6}	3.7×10^{6}	1.5 x 107	7.4×10^{7}	8.9×10^{7}
I-134	0	0	8.5×10^{2}	8.5×10^{3}	3.0×10^{4}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
I-135	0	0	1.3×10^{3}	2.1×10^{4}	1.7×10^{5}	9.3×10^{5}	2.0×10^{6}	5.2×10^{6}	7.4×10^{6}	7.4×10^{6}
Kr-83m	0	0	5.9×10^{3}	7.8×10^{4}	4.4×10^{5}	1.3×10^{6}	1.7×10^{6}	1.9×10^{6}	1.9×10^{6}	1.9×10^{6}
Kr-85	0	0	7.4×10^{2}	1.3×10^{4}	1.3×10^{5}	9.3×10^{5}	2.6×10^{6}	1.3×10^{7}	2.3×10^{8}	5.9×10^{9}
Kr-85m	0	0	1.5×10^{4}	2.3×10^{5}	1.8×10^{6}	8.5×10^{6}	1.6×10^{7}	3.0×10^{7}	3.6×10^{7}	3.6×10^{7}
Kr-87	0	0	2.0×10^{4}	2.4×10^{5}	1.1×10^{6}	2.4×10^{6}	2.7×10^{6}	2.8×10^{6}	2.8×10^{6}	2.8×10^{6}
Kr-88	0	0	3.7×10^{4}	5.6×10^{5}	3.7×10^{6}	1.4×10^{7}	2.3×10^{7}	3.1×10^{7}	3.2×10^{7}	3.2×10^{7}
Kr-89	0	0	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1
Xe-131m	0	0	4.1×10^{2}	6.7×10^{3}	6.7×10^{4}	4.8×10^{5}	1.3×10^{6}	6.7×10^{6}	1.0×10^{8}	1.3×10^{9}
Xe-133	0	0	1.4×10^{5}	2.4×10^{6}	2.3×10^{7}	1.6×10^{8}	4.4×10^{8}	2.2×10^{9}	3.0×10^{10}	1.8×10^{11}
Xe-133m	0	0	5.6×10^{3}	1.0×10^{5}	9.3×10^{5}	6.7×10^{6}	1.8×10^{7}	8.1×10^{7}	8.1×10^{8}	2.0×10^{9}
Xe-135	0	0	1.7×10^{4}	2.7×10^{5}	2.4×10^{6}	1.4×10^{7}	3.3×10^{7}	9.6×10^{7}	2.0×10^{8}	2.0 x 108
Xe-135m	0	0	2.9×10^{3}	1.0×10^{4}	1.3×10^{4}	1.3×10^{4}	1.3×10^{4}	1.3×10^{4}	1.3×10^{4}	1.3×10^{4}
Xe-137	0	0	3.4×10^{1}	3.5×10^{1}	3.5×10^{1}	3.5×10^{1}	3.5×10^{1}	3.5×10^{1}	3.5×10^{1}	3.5×10^{1}
Xe-138	0	0	1.0×10^{4}	3.2×10^{4}	3.7×10^{4}	3.7×10^{4}	3.7×10^{4}	3.7×10^{4}	3.7×10^{4}	3.7×10^{4}

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-49
Doses for ABWR Loss-of-Coolant Accident

Location	Time (hr)	DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
		W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	4.1×10^{-2}	1.9	6.46×10^{-2}	2.6×10^{-1}	1.2×10^{1}
	0–8	1.0×10^{-2}	3.1×10^{-1}	3.40×10^{-2}	3.4×10^{-2}	1.1
	8–24	8.0×10^{-3}	2.0×10^{-1}	4.08×10^{-2}	3.3×10^{-2}	8.2×10^{-1}
	24–96	1.1×10^{-2}	7.9×10^{-1}	6.10×10^{-2}	6.7×10^{-2}	4.8
	96–720	9.0×10^{-3}	1.1	1.08×10^{-1}	9.8×10^{-2}	1.2×10^{1}
LPZ	Total	3.8×10^{-2}	2.4	—	2.3×10^{-1}	1.9×10^{1}
Limit	—	—	—	—	25	300

Reference: ABWR DCD Rev. 4 (GENE Mar 1997) |

Table 7.1-50
Activity Releases for ABWR Cleanup Water Line Break

Isotope	Activity Release (MBq)
I-131	8.1×10^4
I-132	1.9×10^5
I-133	2.3×10^5
I-134	3.2×10^5
I-135	2.5×10^5

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-51
Doses for ABWR Cleanup Water Line Break

		DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
Location	Time (hr)	W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	2.8×10^{-3}	3.0×10^{-1}	3.86×10^{-3}	1.1×10^{-3}	1.2×10^{-1}
LPZ	0–8	—	—	2.31×10^{-4}	6.5×10^{-5}	6.9×10^{-3}
Limit	—	—	—	—	25	300

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-52
Activity Releases for ABWR Fuel Handling Accident

Isotope	Activity Release (MBq)			
	1 min	10 min	1 hr	2 hr
I-131	2.85×10^{5}	2.56×10^{6}	4.55×10^{6}	4.55×10^{6}
I-132	3.67×10^{5}	3.22×10^{6}	5.62×10^{6}	5.62×10^{6}
I-133	2.95×10^{5}	2.64×10^{6}	4.70×10^{6}	4.70×10^{6}
I-134	1.60×10^{-2}	1.36×10^{-1}	2.28×10^{-1}	2.28×10^{-1}
I-135	4.85×10^{4}	4.29×10^{5}	7.62×10^{5}	7.62×10^{5}
TOTAL	9.96×10^{5}	8.85×10^{6}	1.56×10^{7}	1.56×10^{7}
Kr-83m	1.52×10^{4}	1.32×10^{5}	2.33×10^{5}	2.38×10^{5}
Kr-85m	1.94×10^{5}	1.72×10^{6}	3.08×10^{6}	3.16×10^{6}
Kr-85	1.05×10^{6}	9.47×10^{6}	1.72×10^{7}	1.77×10^{7}
Kr-87	3.00×10^{1}	2.59×10^{2}	4.51×10^{2}	4.55×10^{2}
Kr-88	5.62×10^{4}	4.92×10^{5}	8.81×10^{5}	8.99×10^{5}
Kr-89	6.55×10^{-7}	2.77×10^{-6}	3.01×10^{-6}	3.01×10^{-6}
Xe-131m	1.84×10^{5}	1.65×10^{6}	3.00×10^{6}	3.09×10^{6}
Xe-133m	2.44×10^{6}	2.18×10^{7}	3.96×10^{7}	4.07×10^{7}
Xe-133	6.22×10^{7}	5.59×10^{8}	1.01×10^{9}	1.04×10^{9}
Xe-135m	7.25×10^{5}	5.44×10^{6}	8.18×10^{6}	8.18×10^{6}
Xe-135	1.42×10^{7}	1.27×10^{8}	2.29×10^{8}	2.36×10^{8}
Xe-137	1.45×10^{-6}	6.77×10^{-6}	7.66×10^{-6}	7.66×10^{-6}
Xe-138	1.46×10^{-6}	1.07×10^{-5}	1.59×10^{-5}	1.59×10^{-5}

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-53
Doses for ABWR Fuel Handling Accident

Location	Time (hr)	DCD Dose (Sv)		X/Q Ratio	Site Dose (rem)	
		W. Body	Thyroid	Site to DCD	W. Body	Thyroid
EAB	0–2	1.2×10^{-2}	7.5×10^{-1}	6.46×10^{-2}	7.8×10^{-2}	4.8
LPZ	0–8	—	—	3.87×10^{-3}	4.6×10^{-3}	2.9×10^{-1}
Limit	—	—	—	—	6	75

Reference: ABWR DCD Rev. 4 (GENE Mar 1997)

Table 7.1-54
Activity Releases for ESBWR Main Steam Line Break

Isotope	Activity Release (Ci)	
	Equilibrium Iodine	Pre-Incident Iodine
Co-58	8.95×10^{-3}	8.95×10^{-3}
Co-60	1.79×10^{-2}	1.79×10^{-2}
Kr-85	9.48×10^{-4}	9.48×10^{-4}
Kr-85m	2.42×10^{-1}	2.42×10^{-1}
Kr-87	7.84×10^{-1}	7.84×10^{-1}
Kr-88	7.84×10^{-1}	7.84×10^{-1}
Sr-89	4.11×10^{-2}	4.11×10^{-2}
Sr-90	2.85×10^{-3}	2.85×10^{-3}
Sr-91	1.59	1.59
Sr-92	3.60	3.60
Y-90	2.85×10^{-3}	2.85×10^{-3}
Y-91	1.68×10^{-2}	1.68×10^{-2}
Y-92	2.18	2.18
Y-93	1.59	1.59
Zr-95	3.27×10^{-3}	3.27×10^{-3}
Nb-95	3.27×10^{-3}	3.27×10^{-3}
Mo-99	8.13×10^{-1}	8.13×10^{-1}
Tc-99m	8.13×10^{-1}	8.13×10^{-1}
Ru-103	8.21×10^{-3}	8.21×10^{-3}
Ru-106	1.26×10^{-3}	1.26×10^{-3}
Te-129m	1.68×10^{-2}	1.68×10^{-2}
Te-131m	4.02×10^{-2}	4.02×10^{-2}
Te-132	4.11×10^{-3}	4.11×10^{-3}
I-131	1.55	3.10×10^{1}
I-132	1.08×10^{1}	2.15×10^{2}
I-133	1.01×10^{1}	2.02×10^{2}
I-134	1.68×10^{1}	3.36×10^{2}
I-135	1.35×10^{1}	2.69×10^{2}
Xe-133	3.29×10^{-1}	3.29×10^{-1}
Xe-135	9.10×10^{-1}	9.10×10^{-1}
Cs-134	1.09×10^{-2}	1.09×10^{-2}
Cs-136	7.37×10^{-3}	7.37×10^{-3}
Cs-137	2.93×10^{-2}	2.93×10^{-2}
Ba-140	1.68×10^{-1}	1.68×10^{-1}
La-140	1.68×10^{-1}	1.68×10^{-1}
Ce-141	1.26×10^{-2}	1.26×10^{-2}
Ce-144	1.26×10^{-3}	1.26×10^{-3}
Np-239	3.27	3.27

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-55
Doses for ESBWR Main Steam Line Break with Pre-Incident Iodine Spike

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	2.6	4.43×10^{-2}	1.2×10^{-1}
LPZ	0–8	2.0×10^{-1}	2.79×10^{-2}	5.6×10^{-3}
Limit	—	—	—	25

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-56
Doses for ESBWR Main Steam Line Break with Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	2.0×10^{-1}	4.43×10^{-2}	8.9×10^{-3}
LPZ	0–8	1.0×10^{-1}	2.79×10^{-2}	2.8×10^{-3}
Limit	—	—	—	2.5

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-57
Activity Releases for ESBWR Feedwater Line Break

Isotope	Activity Release (Ci)	
	Equilibrium Iodine	Pre-Incident Spike
I-131	1.08×10^{1}	2.16×10^{2}
I-132	7.50×10^{1}	1.50×10^{3}
I-133	7.03×10^{1}	1.41×10^{3}
I-134	1.17×10^{2}	2.34×10^{3}
I-135	9.37×10^{1}	1.87×10^{3}
Cs-134	7.75×10^{-2}	7.75×10^{-2}
Cs-136	5.25×10^{-2}	5.25×10^{-2}
Cs-137	2.09×10^{-1}	2.09×10^{-1}
Co-58	6.37×10^{-2}	6.37×10^{-2}
Co-60	1.27×10^{-1}	1.27×10^{-1}
Sr-89	2.92×10^{-1}	2.92×10^{-1}
Sr-90	2.03×10^{-2}	2.03×10^{-2}
Y-90	2.03×10^{-2}	2.03×10^{-2}
Sr-91	1.13×10^{1}	1.13×10^{1}
Sr-92	2.56×10^{1}	2.56×10^{1}
Y-91	1.19×10^{-1}	1.19×10^{-1}
Y-92	1.55×10^{1}	1.55×10^{1}
Y-93	1.13×10^{1}	1.13×10^{1}
Zr-95	2.33×10^{-2}	2.33×10^{-2}
Nb-95	2.33×10^{-2}	2.33×10^{-2}
Mo-99	5.78	5.78
Tc-99m	5.78	5.78
Ru-103	5.84×10^{-2}	5.84×10^{-2}
Ru-106	8.94×10^{-3}	8.94×10^{-3}
Te-129m	1.19×10^{-1}	1.19×10^{-1}
Te-131m	2.86×10^{-1}	2.86×10^{-1}
Te-132	2.92×10^{-2}	2.92×10^{-2}
Ba-140	1.19	1.19
La-140	1.19	1.19
Ce141	8.94×10^{-2}	8.94×10^{-2}
Ce-144	8.94×10^{-3}	8.94×10^{-3}
Np-239	2.33×10^{1}	2.33×10^{1}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-58
Doses for ESBWR Feedwater Line Break with Pre-Incident Iodine Spike

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	1.8×10^{1}	4.43×10^{-2}	8.0×10^{-1}
LPZ	0–8	1.7	2.79×10^{-2}	4.7×10^{-2}
Limit	—	—	—	25

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-59
Doses for ESBWR Feedwater Line Break with Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	1.1	4.43×10^{-2}	4.9×10^{-2}
LPZ	0–8	1.0×10^{-1}	2.79×10^{-2}	2.8×10^{-3}
Limit	—	—	—	2.5

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-60
Activity Releases for ESBWR Small Break Outside Containment with Pre-Incident Iodine Spike

Isotope	Activity Release (Ci)				
	0.5 hr	1 hr	2 hr	4 hr	6 hr
Co-58	4.32×10^{-4}	6.91×10^{-4}	1.21×10^{-3}	1.90×10^{-3}	1.96×10^{-3}
Co-60	8.64×10^{-4}	1.38×10^{-3}	2.42×10^{-3}	3.81×10^{-3}	3.93×10^{-3}
Sr-89	1.98×10^{-3}	3.17×10^{-3}	5.55×10^{-3}	8.74×10^{-3}	9.02×10^{-3}
Sr-90	1.38×10^{-4}	2.20×10^{-4}	3.85×10^{-4}	6.06×10^{-4}	6.26×10^{-4}
Sr-91	7.69×10^{-2}	1.23×10^{-1}	2.15×10^{-1}	3.39×10^{-1}	3.50×10^{-1}
Sr-92	1.74×10^{-1}	2.78×10^{-1}	4.87×10^{-1}	7.67×10^{-1}	7.91×10^{-1}
Y-90	1.38×10^{-4}	2.20×10^{-4}	3.85×10^{-4}	6.06×10^{-4}	6.26×10^{-4}
Y-91	8.09×10^{-4}	1.29×10^{-3}	2.26×10^{-3}	3.57×10^{-3}	3.68×10^{-3}
Y-92	1.05×10^{-1}	1.68×10^{-1}	2.94×10^{-1}	4.64×10^{-1}	4.78×10^{-1}
Y-93	7.69×10^{-2}	1.23×10^{-1}	2.15×10^{-1}	3.39×10^{-1}	3.50×10^{-1}
Zr-95	1.58×10^{-4}	2.52×10^{-4}	4.42×10^{-4}	6.95×10^{-4}	7.18×10^{-4}
Nb-95	1.58×10^{-4}	2.52×10^{-4}	4.42×10^{-4}	6.95×10^{-4}	7.18×10^{-4}
Mo-99	3.93×10^{-2}	6.28×10^{-2}	1.10×10^{-1}	1.73×10^{-1}	1.78×10^{-1}
Tc-99m	3.93×10^{-2}	6.28×10^{-2}	1.10×10^{-1}	1.73×10^{-1}	1.78×10^{-1}
Ru-103	3.97×10^{-4}	6.34×10^{-4}	1.11×10^{-3}	1.75×10^{-3}	1.80×10^{-3}
Ru-106	6.07×10^{-5}	9.71×10^{-5}	1.70×10^{-4}	2.67×10^{-4}	2.76×10^{-4}
Te-129m	8.09×10^{-4}	1.29×10^{-3}	2.26×10^{-3}	3.57×10^{-3}	3.68×10^{-3}
Te-131m	1.94×10^{-3}	3.11×10^{-3}	5.44×10^{-3}	8.56×10^{-3}	8.83×10^{-3}
Te-132	1.98×10^{-4}	3.17×10^{-4}	5.55×10^{-4}	8.74×10^{-4}	9.02×10^{-4}
I-131	1.46	2.34	4.09	6.44	6.65
I-132	1.02×10^{1}	1.63×10^{1}	2.85×10^{1}	4.48×10^{1}	4.63×10^{1}
I-133	9.54	1.53×10^{1}	2.67×10^{1}	4.20×10^{1}	4.34×10^{1}
I-134	1.59×10^{1}	2.54×10^{1}	4.45×10^{1}	7.01×10^{1}	7.23×10^{1}
I-135	1.27×10^{1}	2.03×10^{1}	3.56×10^{1}	5.60×10^{1}	5.78×10^{1}
Cs-134	5.26×10^{-4}	8.41×10^{-4}	1.47×10^{-3}	2.32×10^{-3}	2.39×10^{-3}
Cs-136	3.56×10^{-4}	5.70×10^{-4}	9.96×10^{-4}	1.57×10^{-3}	1.62×10^{-3}
Cs-137	1.42×10^{-3}	2.27×10^{-3}	3.96×10^{-3}	6.24×10^{-3}	6.44×10^{-3}
Ba-140	8.09×10^{-3}	1.29×10^{-2}	2.26×10^{-2}	3.57×10^{-2}	3.68×10^{-2}
La-140	8.09×10^{-3}	1.29×10^{-2}	2.26×10^{-2}	3.57×10^{-2}	3.68×10^{-2}
Ce-141	6.07×10^{-4}	9.71×10^{-4}	1.70×10^{-3}	2.67×10^{-3}	2.76×10^{-3}
Ce-144	6.07×10^{-5}	9.71×10^{-5}	1.70×10^{-4}	2.67×10^{-4}	2.76×10^{-4}
Np-239	1.58×10^{-1}	2.52×10^{-1}	4.42×10^{-1}	6.95×10^{-1}	7.18×10^{-1}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-61
Doses for ESBWR Small Break Outside Containment with Pre-Incident Iodine Spike

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	3.4×10^{-1}	4.43×10^{-2}	1.5×10^{-2}
LPZ	0–720	1.0×10^{-1}	2.80×10^{-2}	2.8×10^{-3}
Limit	—	—	—	25

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Note: Although the DCD indicates that the LPZ dose extends to 720 hr, it does not provide the dose as a function of time. The site LPZ dose is estimated by multiplying the total DCD dose by the maximum X/Q ratio from Table 7.1-4.

Table 7.1-62
Activity Releases for ESBWR Small Break Outside Containment with Equilibrium Iodine Activity

Isotope	Activity Release (Ci)				
	0.5 hr	1 hr	2 hr	4 hr	6 hr
Co-58	4.32×10^{-4}	6.91×10^{-4}	1.21×10^{-3}	1.90×10^{-3}	1.96×10^{-3}
Co-60	8.64×10^{-4}	1.38×10^{-3}	2.42×10^{-3}	3.81×10^{-3}	3.93×10^{-3}
Sr-89	1.98×10^{-3}	3.17×10^{-3}	5.55×10^{-3}	8.74×10^{-3}	9.02×10^{-3}
Sr-90	1.38×10^{-4}	2.20×10^{-4}	3.85×10^{-4}	6.06×10^{-4}	6.26×10^{-4}
Sr-91	7.69×10^{-2}	1.23×10^{-1}	2.15×10^{-1}	3.39×10^{-1}	3.50×10^{-1}
Sr-92	1.74×10^{-1}	2.78×10^{-1}	4.87×10^{-1}	7.67×10^{-1}	7.91×10^{-1}
Y-90	1.38×10^{-4}	2.20×10^{-4}	3.85×10^{-4}	6.06×10^{-4}	6.26×10^{-4}
Y-91	8.09×10^{-4}	1.29×10^{-3}	2.26×10^{-3}	3.57×10^{-3}	3.68×10^{-3}
Y-92	1.05×10^{-1}	1.68×10^{-1}	2.94×10^{-1}	4.64×10^{-1}	4.78×10^{-1}
Y-93	7.69×10^{-2}	1.23×10^{-1}	2.15×10^{-1}	3.39×10^{-1}	3.50×10^{-1}
Zr-95	1.58×10^{-4}	2.52×10^{-4}	4.42×10^{-4}	6.95×10^{-4}	7.18×10^{-4}
Nb-95	1.58×10^{-4}	2.52×10^{-4}	4.42×10^{-4}	6.95×10^{-4}	7.18×10^{-4}
Mo-99	3.93×10^{-2}	6.28×10^{-2}	1.10×10^{-1}	1.73×10^{-1}	1.78×10^{-1}
Tc-99m	3.93×10^{-2}	6.28×10^{-2}	1.10×10^{-1}	1.73×10^{-1}	1.78×10^{-1}
Ru-103	3.97×10^{-4}	6.34×10^{-4}	1.11×10^{-3}	1.75×10^{-3}	1.80×10^{-3}
Ru-106	6.07×10^{-5}	9.71×10^{-5}	1.70×10^{-4}	2.67×10^{-4}	2.76×10^{-4}
Te-129m	8.09×10^{-4}	1.29×10^{-3}	2.26×10^{-3}	3.57×10^{-3}	3.68×10^{-3}
Te-131m	1.94×10^{-3}	3.11×10^{-3}	5.44×10^{-3}	8.56×10^{-3}	8.83×10^{-3}
Te-132	1.98×10^{-4}	3.17×10^{-4}	5.55×10^{-4}	8.74×10^{-4}	9.02×10^{-4}
I-131	7.31×10^{-2}	1.17×10^{-1}	2.05×10^{-1}	3.22×10^{-1}	3.33×10^{-1}
I-132	5.09×10^{-1}	8.14×10^{-1}	1.42	2.24	2.31
I-133	4.77×10^{-1}	7.63×10^{-1}	1.33	2.10	2.17
I-134	7.95×10^{-1}	1.27	2.22	3.50	3.61
I-135	6.36×10^{-1}	1.02	1.78	2.80	2.89
Cs-134	5.26×10^{-4}	8.41×10^{-4}	1.47×10^{-3}	2.32×10^{-3}	2.39×10^{-3}
Cs-136	3.56×10^{-4}	5.70×10^{-4}	9.96×10^{-4}	1.57×10^{-3}	1.62×10^{-3}
Cs-137	1.42×10^{-3}	2.27×10^{-3}	3.96×10^{-3}	6.24×10^{-3}	6.44×10^{-3}
Ba-140	8.09×10^{-3}	1.29×10^{-2}	2.26×10^{-2}	3.57×10^{-2}	3.68×10^{-2}
La-140	8.09×10^{-3}	1.29×10^{-2}	2.26×10^{-2}	3.57×10^{-2}	3.68×10^{-2}
Ce-141	6.07×10^{-4}	9.71×10^{-4}	1.70×10^{-3}	2.67×10^{-3}	2.76×10^{-3}
Ce-144	6.07×10^{-5}	9.71×10^{-5}	1.70×10^{-4}	2.67×10^{-4}	2.76×10^{-4}
Np-239	1.58×10^{-1}	2.52×10^{-1}	4.42×10^{-1}	6.95×10^{-1}	7.18×10^{-1}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-63
Doses for ESBWR Small Break Outside Containment with Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	1.0×10^{-1}	4.43×10^{-2}	4.4×10^{-3}
LPZ	0–720	1.0×10^{-1}	2.80×10^{-2}	2.8×10^{-3}
Limit	—	—	—	2.5

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Note: Although the DCD indicates that the LPZ dose extends to 720 hr, it does not provide the dose as a function of time. The site LPZ dose is estimated by multiplying the total DCD dose by the maximum X/Q ratio from Table 7.1-4.

Table 7.1-64
Activity Releases for ESBWR Cleanup Water Line Break

Isotope	Activity Release (Ci)	
	Equilibrium Iodine	Pre-Incident Spike
I-131	4.10	8.21×10^{1}
I-132	2.85×10^{1}	5.71×10^{2}
I-133	2.68×10^{1}	5.35×10^{2}
I-134	4.46×10^{1}	8.92×10^{2}
I-135	3.57×10^{1}	7.14×10^{2}
Cs-134	2.95×10^{-2}	2.95×10^{-2}
Cs-136	2.00×10^{-2}	2.00×10^{-2}
Cs-137	7.95×10^{-2}	7.95×10^{-2}
Co-58	2.42×10^{-2}	2.42×10^{-2}
Co-60	4.85×10^{-2}	4.85×10^{-2}
Sr-89	1.11×10^{-1}	1.11×10^{-1}
Sr-90	7.72×10^{-3}	7.72×10^{-3}
Y-90	7.72×10^{-3}	7.72×10^{-3}
Sr-91	4.31	4.31
Sr-92	9.76	9.76
Y-91	4.54×10^{-2}	4.54×10^{-2}
Y-92	5.90	5.90
Y-93	4.31	4.31
Zr-95	8.86×10^{-3}	8.86×10^{-3}
Nb-95	8.86×10^{-3}	8.86×10^{-3}
Mo-99	2.20	2.20
Tc-99m	2.20	2.20
Ru-103	2.23×10^{-2}	2.23×10^{-2}
Ru-106	3.41×10^{-3}	3.41×10^{-3}
Te-129m	4.54×10^{-2}	4.54×10^{-2}
Te-131m	1.09×10^{-1}	1.09×10^{-1}
Te-132	1.11×10^{-2}	1.11×10^{-2}
Ba-140	4.54×10^{-1}	4.54×10^{-1}
La-140	4.54×10^{-1}	4.54×10^{-1}
Ce141	3.41×10^{-2}	3.41×10^{-2}
Ce-144	3.41×10^{-3}	3.41×10^{-3}
Np-239	8.86	8.86

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-65
Doses for ESBWR Cleanup Water Line Break with Pre-Incident Iodine Spike

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	6.9	4.43×10^{-2}	3.1×10^{-1}
LPZ	0–8	7.0×10^{-1}	2.79×10^{-2}	2.0×10^{-2}
Limit	—	—	—	25

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-66
Doses for ESBWR Cleanup Water Line Break with Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	4.0×10^{-1}	4.43×10^{-2}	1.8×10^{-2}
LPZ	0–8	1.0×10^{-1}	2.79×10^{-2}	2.8×10^{-3}
Limit	—	—	—	2.5

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-67 (Sheet 1 of 2)
Activity Releases for ESBWR Loss-of-Coolant Accident

Isotope	Activity Release (Ci)								
	0.5 hr	2 hr	8 hr	12 hr	24 hr	72 hr	96 hr	168 hr	720 hr
Co-58	0	8.0×10^{-3}	6.7×10^{-2}	1.0×10^{-1}	1.8×10^{-1}	3.0×10^{-1}	3.3×10^{-1}	4.0×10^{-1}	7.6×10^{-1}
Co-60	0	1.9×10^{-2}	1.6×10^{-1}	2.4×10^{-1}	4.2×10^{-1}	7.2×10^{-1}	7.9×10^{-1}	9.5×10^{-1}	2.0
Kr-85	1.7×10^{-1}	1.5×10^{1}	3.6×10^{2}	8.0×10^{2}	2.9×10^{3}	1.8×10^{4}	2.7×10^{4}	5.9×10^{4}	3.5×10^{5}
Kr-85m	3.2	2.3×10^{2}	3.1×10^{3}	5.0×10^{3}	7.5×10^{3}	8.2×10^{3}	8.2×10^{3}	8.2×10^{3}	8.2×10^{3}
Kr-87	5.2	2.4×10^{2}	1.1×10^{3}	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}
Kr-88	8.3	5.4×10^{2}	5.5×10^{3}	7.5×10^{3}	9.1×10^{3}	9.2×10^{3}	9.2×10^{3}	9.2×10^{3}	9.2×10^{3}
Rb-86	1.3×10^{-2}	2.8×10^{-1}	2.1	3.1	5.3	8.9	9.7	1.1×10^{1}	1.8×10^{1}
Sr-89	0	8.9	7.4×10^{1}	1.1×10^{2}	2.0×10^{2}	3.4×10^{2}	3.7×10^{2}	4.4×10^{2}	8.2×10^{2}
Sr-90	0	1.0	8.4	1.3×10^{1}	2.2×10^{1}	3.8×10^{1}	4.2×10^{1}	5.1×10^{1}	1.1×10^{2}
Sr-91	0	1.0×10^{1}	6.8×10^{1}	9.1×10^{1}	1.2×10^{2}	1.3×10^{2}	1.3×10^{2}	1.3×10^{2}	1.3×10^{2}
Sr-92	0	8.3	3.6×10^{1}	4.0×10^{1}	4.1×10^{1}	4.2×10^{1}	4.2×10^{1}	4.2×10^{1}	4.2×10^{1}
Y-90	0	1.6×10^{-2}	3.8×10^{-1}	8.0×10^{-1}	2.5	8.4	1.1×10^{1}	1.7×10^{1}	7.0×10^{1}
Y-91	0	1.2×10^{-1}	1.0	1.5	2.8	5.1	5.6	6.8	1.3×10^{1}
Y-92	0	9.7×10^{-1}	1.9×10^{1}	2.8×10^{1}	3.6×10^{1}	3.7×10^{1}	3.7×10^{1}	3.7×10^{1}	3.7×10^{1}
Y-93	0	1.3×10^{-1}	8.7×10^{-1}	1.2	1.6	1.7	1.7	1.7	1.7
Zr-95	0	1.7×10^{-1}	1.4	2.1	3.7	6.4	7.0	8.4	1.6×10^{1}
Zr-97	0	1.6×10^{-1}	1.2	1.7	2.5	3.1	3.1	3.1	3.1
Nb-95	0	1.7×10^{-1}	1.4	2.1	3.8	6.5	7.1	8.6	1.7×10^{1}
Mo-99	0	2.2	1.8×10^{1}	2.6×10^{1}	4.4×10^{1}	6.7×10^{1}	7.1×10^{1}	7.6×10^{1}	8.0×10^{1}
Tc-99m	0	2.0	1.7×10^{1}	2.5×10^{1}	4.2×10^{1}	6.6×10^{1}	7.0×10^{1}	7.5×10^{1}	7.9×10^{1}
Ru-103	0	1.8	1.5×10^{1}	2.3×10^{1}	4.0×10^{1}	6.9×10^{1}	7.5×10^{1}	8.9×10^{1}	1.6×10^{2}
Ru-105	0	1.0	5.4	6.6	7.4	7.5	7.5	7.5	7.5
Ru-106	0	6.9×10^{-1}	5.8	8.7	1.5×10^{1}	2.6×10^{1}	2.9×10^{1}	3.5×10^{1}	7.1×10^{1}
Rh-105	0	1.1	9.4	1.4×10^{1}	2.3×10^{1}	3.2×10^{1}	3.3×10^{1}	3.4×10^{1}	3.4×10^{1}
Sb-127	0	2.5	2.0×10^{1}	3.0×10^{1}	5.2×10^{1}	8.2×10^{1}	8.7×10^{1}	9.5×10^{1}	1.0×10^{2}
Sb-129	0	6.0	3.2×10^{1}	3.9×10^{1}	4.3×10^{1}	4.4×10^{1}	4.4×10^{1}	4.4×10^{1}	4.4×10^{1}
Te-127	0	2.5	2.1×10^{1}	3.1×10^{1}	5.4×10^{1}	8.7×10^{1}	9.3×10^{1}	1.0×10^{2}	1.3×10^{2}
Te-127m	0	3.4×10^{-1}	2.8	4.3	7.6	1.3×10^{1}	1.4×10^{1}	1.7×10^{1}	3.5×10^{1}
Te-129	0	6.6	4.0×10^{1}	5.0×10^{1}	6.5×10^{1}	8.1×10^{1}	8.4×10^{1}	9.1×10^{1}	1.3×10^{2}
Te-129m	0	1.1	9.3	1.4×10^{1}	2.5×10^{1}	4.2×10^{1}	4.6×10^{1}	5.5×10^{1}	9.7×10^{1}
Te-131m	0	3.3	2.6×10^{1}	3.7×10^{1}	5.9×10^{1}	8.0×10^{1}	8.2×10^{1}	8.4×10^{1}	8.4×10^{1}
Te-132	0	3.3×10^{1}	2.7×10^{2}	4.0×10^{2}	6.7×10^{2}	1.0×10^{3}	1.1×10^{3}	1.2×10^{3}	1.3×10^{3}
I-131	5.8	1.5×10^{2}	1.1×10^{3}	1.6×10^{3}	2.8×10^{3}	5.1×10^{3}	5.7×10^{3}	7.1×10^{3}	1.1×10^{4}

Table 7.1-67 (Sheet 2 of 2)
Activity Releases for ESBWR Loss-of-Coolant Accident

Isotope	Activity Release (Ci)								
	0.5 hr	2 hr	8 hr	12 hr	24 hr	72 hr	96 hr	168 hr	720 hr
I-132	8.0	1.9×10^{2}	8.5×10^{2}	1.0×10^{3}	1.4×10^{3}	1.8×10^{3}	1.9×10^{3}	2.1×10^{3}	2.2×10^{3}
I-133	1.2×10^{1}	2.8×10^{2}	1.9×10^{3}	2.7×10^{3}	4.2×10^{3}	5.5×10^{3}	5.6×10^{3}	5.7×10^{3}	5.7×10^{3}
I-134	9.9	1.1×10^{2}	2.0×10^{2}	2.1×10^{2}	2.1×10^{2}	2.1×10^{2}	2.1×10^{2}	2.1×10^{2}	2.1×10^{2}
I-135	1.1×10^{1}	2.4×10^{2}	1.3×10^{3}	1.7×10^{3}	2.1×10^{3}	2.2×10^{3}	2.2×10^{3}	2.2×10^{3}	2.2×10^{3}
Xe-133	2.6×10^{1}	2.2×10^{3}	5.2×10^{4}	1.2×10^{5}	4.0×10^{5}	2.1×10^{6}	2.9×10^{6}	5.2×10^{6}	1.1×10^{7}
Xe-135	9.4	8.2×10^{2}	1.6×10^{4}	3.0×10^{4}	6.5×10^{4}	1.0×10^{5}	1.0×10^{5}	1.0×10^{5}	1.0×10^{5}
Cs-134	1.2	2.7×10^{1}	2.0×10^{2}	2.9×10^{2}	5.1×10^{2}	8.7×10^{2}	9.6×10^{2}	1.1×10^{3}	2.3×10^{3}
Cs-136	4.1×10^{-1}	8.7	6.3×10^{1}	9.4×10^{1}	1.6×10^{2}	2.7×10^{2}	2.9×10^{2}	3.4×10^{2}	5.0×10^{2}
Cs-137	7.9×10^{-1}	1.7×10^{1}	1.2×10^{2}	1.9×10^{2}	3.3×10^{2}	5.6×10^{2}	6.1×10^{2}	7.3×10^{2}	1.5×10^{3}
Ba-139	0	8.2	2.3×10^{1}	2.3×10^{1}	2.3×10^{1}	2.3×10^{1}	2.3×10^{1}	2.3×10^{1}	2.3×10^{1}
Ba-140	0	1.6×10^{1}	1.4×10^{2}	2.1×10^{2}	3.6×10^{2}	6.0×10^{2}	6.5×10^{2}	7.6×10^{2}	1.1×10^{3}
La-140	0	3.2×10^{-1}	8.8	1.9×10^{1}	5.9×10^{1}	1.9×10^{2}	2.3×10^{2}	3.3×10^{2}	7.4×10^{2}
La-141	0	1.2×10^{-1}	6.2×10^{-1}	7.4×10^{-1}	8.1×10^{-1}	8.2×10^{-1}	8.2×10^{-1}	8.2×10^{-1}	8.2×10^{-1}
La-142	0	7.8×10^{-2}	2.3×10^{-1}	2.4×10^{-1}	2.4×10^{-1}	2.4×10^{-1}	2.4×10^{-1}	2.4×10^{-1}	2.4×10^{-1}
Ce-141	0	3.9×10^{-1}	3.3	4.9	8.6	1.5×10^{1}	1.6×10^{1}	1.9×10^{1}	3.4×10^{1}
Ce-143	0	3.5×10^{-1}	2.8	4.0	6.4	8.9	9.1	9.3	9.4
Ce-144	0	3.2×10^{-1}	2.7	4.0	7.2	1.2×10^{1}	1.4×10^{1}	1.6×10^{1}	3.3×10^{1}
Pr-143	0	1.4×10^{-1}	1.2	1.8	3.2	5.6	6.1	7.3	1.1×10^{1}
Nd-147	0	6.3×10^{-2}	5.2×10^{-1}	7.8×10^{-1}	1.4	2.3	2.4	2.8	4.0
Np-239	0	4.6	3.7×10^{1}	5.4×10^{1}	9.1×10^{1}	1.4×10^{2}	1.4×10^{2}	1.5×10^{2}	1.6×10^{2}
Pu-238	0	9.6×10^{-4}	8.0×10^{-3}	1.2×10^{-2}	2.1×10^{-2}	3.7×10^{-2}	4.0×10^{-2}	4.9×10^{-2}	1.0×10^{-1}
Pu-239	0	1.1×10^{-4}	8.9×10^{-4}	1.3×10^{-3}	2.4×10^{-3}	4.1×10^{-3}	4.5×10^{-3}	5.4×10^{-3}	1.1×10^{-2}
Pu-240	0	1.4×10^{-4}	1.2×10^{-3}	1.7×10^{-3}	3.1×10^{-3}	5.3×10^{-3}	5.8×10^{-3}	7.0×10^{-3}	1.5×10^{-2}
Pu-241	0	4.4×10^{-2}	3.7×10^{-1}	5.6×10^{-1}	9.8×10^{-1}	1.7	1.9	2.2	4.6
Am-241	0	2.1×10^{-5}	1.8×10^{-4}	2.7×10^{-4}	4.7×10^{-4}	8.2×10^{-4}	9.1×10^{-4}	1.1×10^{-3}	2.4×10^{-3}
Cm-242	0	5.0×10^{-3}	4.2×10^{-2}	6.3×10^{-2}	1.1×10^{-1}	1.9×10^{-1}	2.1×10^{-1}	2.5×10^{-1}	5.1×10^{-1}
Cm-244	0	2.6×10^{-4}	2.2×10^{-3}	3.3×10^{-3}	5.8×10^{-3}	1.0×10^{-2}	1.1×10^{-2}	1.3×10^{-2}	2.8×10^{-2}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-68
Doses for ESBWR Loss-of-Coolant Accident

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	2.3–4.3[(a)]	2.24×10^{1}	4.43×10^{-2}	9.91×10^{-1}
LPZ	0–8	6.93	2.79×10^{-2}	1.93×10^{-1}
	8–24	4.54	2.80×10^{-2}	1.27×10^{-1}
	24–96	4.72	2.73×10^{-2}	1.29×10^{-1}
	96–720	4.56	2.68×10^{-2}	1.22×10^{-1}
	Total	2.08×10^{1}	—	5.72×10^{-1}
Limit	—	—	—	25

(a) Worst case 2-hour period is 2.3–4.3 hours.

Note: Time-dependent LPZ doses and total LPZ and EAB doses correspond to the radiation doses in DCD Rev. 6 (GEH Aug 2009)

Table 7.1-69
Activity Releases for ESBWR Fuel Handling Accident

Isotope	Activity Release (Ci), 0–2 hr
I-131	1.37×10^{2}
I-132	7.01×10^{-2}
I-133	8.21×10^{1}
I-134	5.24×10^{-7}
I-135	1.28×10^{1}
Kr-85m	9.96×10^{1}
Kr-85	4.98×10^{2}
Kr-87	1.07×10^{-2}
Kr-88	2.85×10^{1}
Xe-133	3.23×10^{4}
Xe-135	2.01×10^{3}

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

Table 7.1-70
Doses for ESBWR Fuel Handling Accident

Location	Time (hr)	DCD Dose (rem TEDE)	X/Q Ratio (Site to DCD)	Site Dose (rem TEDE)
EAB	0–2	4.1	4.43×10^{-2}	1.8×10^{-1}
LPZ	0–8	4.0×10^{-1}	2.79×10^{-2}	1.1×10^{-2}
Limit	—	—	—	6.3

Reference: ESBWR DCD Rev. 6 (GEH Aug 2009)

7.2 Severe Accidents

Severe accidents are those involving multiple failures of equipment to function. The likelihood of occurrence is lower for severe accidents than for design basis accidents, but the consequences of such accidents may be higher. Although severe accidents are not part of the design basis for the plant, NRC, in its policy statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50 FR 32138), requires the completion of a probabilistic risk assessment (PRA) for severe accidents for new reactor designs. This requirement is codified under 10 CFR 52.47.

As described in Chapter 3, Exelon's ESP analyses are based on the following reactor types: ESBWR, ABWR, AP1000, APWR, and mPower. For the severe accident analysis, Exelon selected the ESBWR and ABWR to represent the entire suite of advanced light water reactor technologies. Exelon believes this representative approach is appropriate because:

- Unlike for safety analyses, a representative analysis is acceptable under the National Environmental Policy Act.
- In its 1985 policy statement on severe accidents (50 FR 32138), NRC stated its expectation that new reactors would achieve a higher standard of severe accident safety performance than the earlier designs. This has proven to be true as severe accident risks from new reactor design certification applications and ESP/COL applications are compared to license renewal applications. The greatest risk associated with a new generation reactor design (for which data is available) is well below that of the already low risk associated with the existing fleet undergoing license renewal.

General Electric-Hitachi (GEH) completed a PRA for the ESBWR design (GEH Jun 2009) as part of the application for design certification. GE prepared a PRA as part of the Standard Safety Analysis Report (SSAR) Amendment 35 for the ABWR design. NRC reviewed the ABWR SSAR, and the review was documented in NUREG-1503 (U.S. NRC Jul 1994). NRC has certified the ABWR design, concluding that the ABWR is of a robust design, and that the design meets NRC's safety goals.

In this section, Exelon presents an update of the two generic PRA analyses described above. Exelon's analysis includes VCS site-specific characteristics. The analysis evaluates the impacts of a severe accident at VCS to demonstrate that the impacts are bounded by the generic certification analyses.

7.2.1 ESBWR and ABWR Reactor Vendor Methodology

ESBWR

The GEH PRA for the ESBWR established a containment event tree that defined the possible end states of the containment following a severe accident. Using EPRI's Modular Accident Analysis Program code, GEH determined that 10 release categories would represent the entire suite of potential severe accidents. Five of the release categories were represented by dual source term categories. An accident frequency was assigned to each of the 10 source term categories (Table 7.2-2). For the dual source term release categories, GEH assigned the entire frequency of the release category to each of the source terms. (The site specific analysis described further on in this section made the conservative assumption that the entire frequency of each of the dual source term release categories was attributed to the source term resulting in the greater population dose.)

The 10 release categories and associated source term categories are as follows:

Break Outside of Containment (BOC) — Radioactivity is released through an unisolated break outside the containment in the shutdown cooling piping allowing direct communication between the reactor pressure vessel and the environment outside the containment. This is followed by no injection of cooling water into the reactor pressure vessel. Two separate locations of a break in the piping were selected for determining source term categories in this release category, one mid-level in the reactor pressure vessel (BOC mid) and the other at the lower level (BOC low).

Containment Bypass (BYP) — Radioactivity is released directly to the atmosphere from the containment due to a failure of the containment isolation system to function. Sequences in which the reactor pressure vessel is depressurized generally result in the core being uncovered earlier than those with a failure to depressurize. Both a low pressure sequence (BYP low) and a high pressure sequence (BYP high) were selected for determining the source term categories for this release category.

Core-Concrete Interaction Dry (CCID) — This release category applies to sequences in which the containment fails due to interaction between the core and the containment concrete. The deluge function is assumed to fail, and the lower drywell debris bed is uncovered. Sequences in which the containment vessel is not depressurized may result in earlier containment vessel failure. A low pressure sequence (CCID low) and a high pressure sequence (CCID high) were selected for determining the source term categories in this release category.

Core-Concrete Interaction Wet (CCIW) — This release category applies to sequences in which the containment fails due to interaction between the core and containment concrete. The deluge function works; however, the basemat internal melt arrest and coolability device is not effective in providing debris bed cooling. Unlike the CCID category, cooling water is present and provides the potential for scrubbing of the radionuclides that evolve from the debris bed, thus reducing the magnitude of the source term. Sequences in which the reactor vessel is not depressurized may result in earlier reactor vessel failure. A low pressure sequence (CCIW low) and a high pressure sequence (CCIW high) were selected for determining the source term categories associated with this release category.

Ex-Vessel Steam Explosion (EVE) — This release category applies to sequences in which the reactor vessel fails at low pressure and a significant steam explosion occurs. Containment depressurization is assumed to occur when the vessel fails, at which time there is direct communication with the environment. Due to the uncertainties associated with equipment damage and water availability, no credit is taken for lower drywell water to reduce the source term.

Filtered Release (FR) — Radioactivity is released by manually venting the containment from the suppression chamber air space. This action may be implemented to limit the containment pressure increase if containment heat removal fails or the containment is over pressurized. Venting the suppression chamber forces the radionuclides through the suppression pool, which reduces the magnitude of the source term.

Overpressure-Vacuum Breaker (OPVB) — This release category applies to sequences in which the vacuum breaker failure has occurred (either by failing to close or by remaining open in a preexisting condition), resulting in failure of the containment pressure function, which in turn causes failure in containment heat removal. Two sequences are associated with this release category; both high (OPVB high) and low pressure (OPVB low) sequences were selected for source term categories.

Overpressure-Early Containment Heat Removal Loss (OPW1) — This release category applies to sequences in which containment heat removal fails within 24 hours after event initiation. A sequence with the reactor pressure vessel failure at high pressure was selected because it has an earlier failure and higher probability of the loss of containment heat removal. Containment heat removal is assumed to be unavailable for the duration of the sequence.

Overpressure-Late Containment Heat Removal Loss (OPW2) — This release category applies to sequences in which containment heat removal fails in the period after that addressed by OPW1, above, until 72 hours after onset of core damage. The passive containment cooling system is assumed to be unavailable 24 hours after event initiation and the availability of the fuel and auxiliary pool cooling system is determined. A sequence with the reactor pressure vessel failure at high pressure was selected because it has an earlier failure and higher probability of the loss of containment heat removal. Containment heat removal is terminated 24 hours after the event initiation.

Technical Specification Leakage (TSL) — This category applies to sequences in which the containment is intact and the only release is due to the maximum leak rate allowed by technical specifications. For additional conservatism, the area of containment leakage corresponding to the maximum allowable technical specification leak rate was doubled to produce the representative source term used for this release category.

In addition, a direct containment heating (DCH) category was evaluated. The DCH category applies to sequences in which the reactor fails at high pressure and a significant DCH event occurs. GEH subsequently determined that catastrophic containment failure due to DCH is physically unreasonable and studied local damage to the liner in the lower drywell as a sensitivity case. Thus, no DCH sequence was evaluated for the baseline case.

GEH then used the MACCS2 (MELCOR Accident Consequence Code System) to model the environmental consequences of severe accidents using generic, but conservative, meteorological and population parameters to represent a generic ESBWR site. The analysis focused on the 24-hour period following core damage as a measure of the consequences from a large release and, therefore, did not address the chronic pathways such as ingestion, inhalation of re-suspended material, or groundshine subsequent to plume passage. GEH also considered the releases for the first 72 hours after core damage. Additional details of analysis are found in the ESBWR PRA (GEH Jun 2009) and are reported in the ESBWR Design Control Document (GEH Aug 2009).

ABWR

The GE PRA for the ABWR established a containment event tree that defined the possible end states of the containment following a severe accident. These end states can logically be grouped to produce 10 source term categories that represent the entire suite of potential severe accidents. An accident frequency was assigned to each of the 10 source term categories (Table 7.2-2).

The 10 source term categories can be characterized as follows (GE Jan 1995):

NCL — Normal Containment Leakage to Reactor Building

Case 1 — Core melt arrested in vessel or in containment with actuation of containment rupture disk.

Case 2 — Low pressure core melt with suppression pool bypass and actuation of containment rupture disk.

Case 3 — High pressure core melt with drywell head failure and fire water spray initiation.

Case 4 — Suppression pool decontamination reduction.

Case 5 — Large break LOCA (loss of cooling accident) without recovery and actuation of containment rupture disk.

Case 6 — High pressure core melt with drywell head failure and no fire water spray initiation.

Case 7 — Low pressure core melt with drywell head failure and no mitigation.

Case 8 — High pressure core melt with early containment failure.

Case 9 — ATWS (anticipated transient without scram) event with drywell head failure.

GE then used the CRAC-2 code to model the environmental consequences of the severe accidents using the generic meteorology, population, and evacuation characteristics as described in Section 19E.3 of the Design Control Document (DCD). CRAC-2 is a revision of the CRAC program developed in support of NRC's Reactor Safety Study (often referred to as WASH-1400) to assess the risk from potential accidents at nuclear power plants.

GE only included severe accidents that were initiated from internal events. External events, including tornado strikes and earthquakes, were evaluated and GE concluded that the accident frequency for these external initiated events was much less than the accident frequency for the internally initiated events.

Details of the analysis are reported in Chapter 19 of the ABWR DCD (GE Mar 1997).

7.2.2 Exelon Methodology

Exelon used the MACCS2 computer code (Version 1.13.1), which was developed for the NRC as a successor to CRAC-2, to evaluate offsite risks and consequences of severe accidents, using VCS site-specific information. MACCS2 simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS2 include atmospheric transport, mitigation actions based on dose projection, dose accumulation by a number of pathways including food and water ingestion, early and latent human health effects, and economic costs. The specific 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 re-suspended 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 to determine a drinking water risk from airborne releases. The code also evaluates the extent of contamination. Exelon used site-specific meteorology and population data to model both the early exposure pathways, such as inhalation of the passing plume and direct exposure to the passing plume, and the long-term exposure pathways, such as ingestion, over the life cycle of the accident. Ingestion exposure was determined using the COMIDA2 food model option of MACCS2.

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

Five files provide input to a MACCS2 analysis: EARLY, ATMOS, CHRONC, MET, and SITE.

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 inputs in this file include the core inventory, release fractions, and geometry of the reactor and associated buildings. This input data is taken from the generic PRAs.

A second file, EARLY, provides inputs to calculations regarding exposure in the time period immediately following the release, including parameters describing breathing rates and sheltering. Important site-specific information includes emergency response information such as evacuation time; in this case, the conservative assumption of no evacuation was made.

The third input file, 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 that uses actual VCS site meteorological monitoring data and a site characteristics file which is built using SECPOP2000 (U.S. NRC Aug 2003).

MACCS2 requires a calendar year of meteorological data for the MET file. One year of VCS site meteorological data (July 2008 to June 2009) was used to create the meteorological data file. Sensitivity studies considered VCS meteorological data from July 2007 through June 2008, along with five years of National Weather Service Victoria Regional Airport data (years 2003–2007 and July 2007 through June 2008). The sensitivity studies indicate that the site data is representative of recent conditions near the site.

The SITE file requires the 50-mile population distribution as well as agricultural-economic data. SECPOP2000 incorporates 2000 census data for the 50-mile region around the VCS site. For this analysis, the census data were modified to include transient populations and projected to the year 2060 (included in the population projections in Subsection 2.5.1), using county-specific growth rates. MACCS2 also requires the spatial distribution of certain agriculture and economic data (fraction of land devoted to farming, annual farm sales, fraction of farm sales resulting from dairy production, and property value of farm and nonfarmland) in the same manner as the population. This data was prepared using the 2007 National Census of Agriculture (USDA Sep 2009).

Exelon used the resulting MACCS2 calculations and accident frequency information to determine risk. The sum of the accident frequencies is known as the core damage frequency and includes only internally initiated events during reactor operation. Risk is the product of frequency of an accident times the consequences of the accident. The consequence can be any measure of release impacts such as radiation dose and economic cost. Dose-risk is the product of the collective dose times the accident frequency. Because the severe accident analysis addressed a suite of accidents, the individual risks were summed to provide a total risk. The same process was applied to estimating cost-risk. Risk from these consequences can be reported as person-rem per reactor year or dollars per reactor year.

Exelon used the source term parameters (e.g., core inventory, release height at top of containment, release heat, nuclide release fractions and durations) applied in the generic PRAs. Similar to the vendors' analyses, only internal events were analyzed. Exelon assumed perpetual rainfall in the last spatial segment (40-50 miles) of the model domain so that a conservatively large quantity of the nuclides released in each accident scenario were deposited (via wet deposition) within the model domain.

7.2.3 Consequences to Population Groups

The pathway consequences to population groups including air pathways, surface water, and groundwater pathways are described in the following sections. The presence of threatened and endangered species and federally designated critical habitat are described in Subsections 2.4.1 and 2.4.2. The impacts on biota due to the previously calculated radiation exposure levels are described in Subsection 5.4.4. Risk values in the text of this section below are for the reactor technology resulting in the highest value.

7.2.3.1 Air Pathways

Each of the accident categories was analyzed with MACCS2 to estimate population dose, number of early and latent fatalities, cost, and farmland requiring decontamination. The analysis conservatively assumed that none of the 10-mile emergency planning zone population was evacuated following declaration of a general emergency. For each accident category, the risk for each analytical endpoint was calculated by multiplying the analytical endpoint by the accident category frequency and adding across all accident categories. The results are provided in Tables 7.2-1 and 7.2-2.

7.2.3.2 Surface Water Pathways

People can be exposed to radiation when airborne radioactivity is deposited onto the ground and runs off into surface water or is deposited directly onto surface water. The exposure pathway can be from drinking the water, submersion in the water (swimming), undertaking activities near the shoreline (fishing and boating), or ingestion of fish or shellfish. For the surface water pathway, MACCS2 only calculates the dose from drinking the water. It is conservatively assumed that all water within 50 miles of the site is drinkable (even though most of it is saltwater). The maximum MACCS2 code severe accident dose-risk to the 50-mile population from drinking the water is 1.8×10^{-4} person-rem per year of ESBWR operation. As shown in Table 7.2-1, this value is the sum of all accident category risks.

Surface water bodies within the 50-mile region of the VCS site that are accessible to the public include the Guadalupe River, San Antonio River, Matagorda Bay, Hynes Bay, Guadalupe Bay, San Antonio Bay, Lake Texana, the Gulf of Mexico, and other smaller water bodies. In NUREG-1437, the NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet of reactors (U.S. NRC May 1996). For sites discharging to small rivers, the NRC evaluation estimated the un-interdicted aquatic food pathway dose risk as 0.4 person-rem per reactor year. For sites near large water bodies, values ranged from 270 person-rem per reactor year (Hope Creek on Delaware Bay) to 5500 person-rem per reactor year (Calvert Cliffs on Chesapeake Bay). The VCS site would more likely fall between the small river analysis and the least impacting large water body analysis (Hope Creek), given the VCS site's distance from nearby major water bodies (approximately 20 miles to the Gulf of Mexico). Actual dose-risk values would be expected to be much less (by a factor of 2 to 10) due to interdiction of contaminated foods (U.S. NRC May 1996). Both the ESBWR and ABWR atmospheric pathway doses are significantly lower than those of the current nuclear fleet (Subsection 7.2.5). Given the dependency of surface water doses on airborne releases, it is reasonable to conclude that the doses from surface water sources would be consistently lower than those reported above for the surface water pathway.

Doses associated with submersion in the water and undertaking activities near the shoreline are not modeled by MACCS2, and NUREG-1437 does not provide specific data on submersion and shoreline activities. However, it does indicate that these contributors to dose are much less than for drinking water and consuming aquatic foods, especially at estuary sites.

7.2.3.3 Groundwater Pathways

People can also receive dose from groundwater pathways. Radioactivity released during a severe accident can enter groundwater and may move through an aquifer and eventually be discharged to surface water. The consequences of a radioactive spill are evaluated in Subsection 2.4.13 of the Site Safety Evaluation Report, and the results show that if radioactive liquids were released directly to groundwater, the isotopic concentrations would be below 10 CFR 20 effluent limits before they reached a drinking water receptor.

NUREG-1437 evaluated the groundwater pathway dose, based on the analysis in NUREG-0440, the Liquid Pathway Generic Study (U.S. NRC Feb 1978). NUREG-0440 analyzed a core meltdown that contaminated groundwater, which subsequently contaminated surface water; NUREG-0440 did not analyze direct consumption of groundwater because it assumed a limited number of potable groundwater wells and limited accessibility.

The Liquid Pathway Generic Study results provide conservative, un-interdicted 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. Therefore, although VCS was not specifically included in the Liquid Pathway Generic Study, the doses from the VCS site groundwater pathway would be expected to be much less than the doses from the atmospheric pathway, given that all categories of plant locations showed the same trend.

7.2.4 Comparison to NRC Safety Goals

Exelon evaluated performance of the ESBWR and the ABWR relative to two safety goals: (1) individual risk goal, and (2) societal risk goal. These goals are defined in the following subsections. Table 7.2-3 provides the quantitative evaluation of these safety goals and the VCS site-specific calculation of these risk values.

7.2.4.1 Individual Risk Goal

The risk to an average individual in the vicinity of a nuclear power plant of experiencing a prompt fatality resulting from a severe reactor accident should not exceed one-tenth of one percent (0.1 percent) of the sum of "prompt fatality risks" resulting from other accidents to which members of the United States population are generally exposed. As defined in the Safety Goals Policy statement (51 FR 30028), "vicinity" is the area within one mile of the plant site boundary. The population within one mile of the proposed VCS reactors is zero. Exelon conservatively assumed a uniformly distributed synthetic population surrounding the site within 1 mile of the reactor. "Prompt Fatality Risks" are defined as the sum of risks which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities (driving, household chores, occupational activities, etc). For this evaluation, the sum of prompt fatality risks was taken as the United States accidental death risk value of 39.8 deaths per 100,000 people per year (CDC Apr 2009).

7.2.4.2 Societal Risk Goal

The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from its operation should not exceed one-tenth of one percent (0.1 percent) of the sum of the cancer fatality risks resulting from all other causes. As defined in the Safety Goal Policy Statement (51 FR 30028), “near” is within 10 miles of the plant. The cancer fatality risk was taken as 180.7 deaths per 100,000 people per year based upon National Center for Health Statistics data (CDC Apr 2009).

7.2.5 Conclusions

The total calculated dose-risk to the 50-mile population from airborne releases from an ABWR reactor at the VCS site would be 0.0020 person-rem per reactor year (Table 7.2-2). This value 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 (U.S. NRC Jun 1989).

Comparisons with the existing nuclear reactor fleet (Subsection 7.2.3.2) indicate that risk from the surface water pathway is small. Under the severe accident scenarios, surface water is primarily contaminated by atmospheric deposition. The atmospheric pathway doses are significantly lower than those of the current nuclear fleet. Therefore, it is reasonable to conclude that the doses from the surface water pathway at the VCS site would be consistently lower than those reported in Subsection 7.2.3.2 for the current fleet.

The risks of groundwater contamination from a severe accident (see Subsection 7.2.3.3) would be much less than the risk from currently licensed reactors. Additionally, interdiction could substantially reduce the groundwater pathway risks.

For comparison, as reported in Section 5.4, the total collective whole body dose from the VCS site normal airborne releases is expected to be 1.1 person-rem annually. As previously described, dose-risk is dose times frequency. Normal operations have a frequency of one. Therefore, the dose-risk for normal operations is 1.1 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.0020 person-rem per reactor year indicates that the dose risk from severe accidents is approximately 0.2 percent of the dose risk from normal operations.

The probability-weighted risk of fatalities (early and latent cancer) from a severe accident for the VCS site is reported in Table 7.2-2 as 1.2×10^{-6} fatalities per reactor year. The probability of an individual dying from any cancer from any cause is approximately 0.24 over a lifetime (ACS Mar 2008). Comparing this value to the 1.2×10^{-6} fatalities per reactor year indicates that individual risk is 0.0005 percent of the background risk. As reported in Table 7.2-3, the individual and societal risks for a severe accident from both the ESBWR and the ABWR reactors at the VCS site would be less than the NRC risk goals described in Subsection 7.2.4.

7.2.6 References

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

Accident Category	Accident Frequency	Population Dose-Risk (person-rem/ reactor year)	Cost in Dollars (per reactor year)	Number of Early Fatalities (per reactor year)	Number of Latent Fatalities (per reactor year)	Water Ingestion Dose (person-rem per reactor year)	Land Requiring Decontamination (acres per reactor year)
BOC	7.9×10^{-11}	1.5×10^{-4}	2.9×10^{-1}	1.6×10^{-10}	1.2×10^{-7}	2.0×10^{-5}	4.0×10^{-5}
BYP	5.7×10^{-11}	1.2×10^{-4}	1.5×10^{-1}	2.9×10^{-11}	1.2×10^{-7}	3.8×10^{-6}	2.7×10^{-5}
CCID	1.5×10^{-12}	1.5×10^{-6}	4.5×10^{-3}	8.1×10^{-14}	9.5×10^{-10}	2.1×10^{-7}	7.1×10^{-7}
CCIW	2.9×10^{-12}	1.3×10^{-6}	3.5×10^{-3}	0.0	7.7×10^{-10}	5.1×10^{-8}	7.9×10^{-7}
EVE	1.1×10^{-9}	1.2×10^{-3}	3.5	9.3×10^{-11}	7.6×10^{-7}	1.5×10^{-4}	5.5×10^{-4}
FR	9.2×10^{-11}	1.9×10^{-5}	4.0×10^{-2}	0.0	1.1×10^{-8}	2.7×10^{-7}	8.3×10^{-6}
OPVB	2.1×10^{-12}	6.4×10^{-7}	1.6×10^{-3}	0.0	3.8×10^{-10}	1.5×10^{-8}	3.8×10^{-7}
OPW1	2.0×10^{-12}	6.0×10^{-7}	1.6×10^{-3}	0.0	3.6×10^{-10}	1.6×10^{-8}	3.9×10^{-7}
OPW2	8.5×10^{-12}	8.0×10^{-7}	7.8×10^{-4}	0.0	4.8×10^{-10}	4.7×10^{-9}	6.0×10^{-8}
TSL	1.5×10^{-8}	3.0×10^{-4}	2.3×10^{-1}	0.0	1.8×10^{-7}	6.5×10^{-7}	1.1×10^{-5}
Total	1.7×10^{-8}	1.8×10^{-3}	4.2	2.8×10^{-10}	1.2×10^{-6}	1.8×10^{-4}	6.4×10^{-4}

Table 7.2-2
Impacts to the Population and Land From Severe ABWR Accident Analysis

Accident Category	Accident Frequency	Population Dose-Risk (person-rem/ reactor year)	Cost in Dollars (per reactor year)	Number of Early Fatalities (per reactor year)	Number of Latent Fatalities (per reactor year)	Water Ingestion Dose (person-rem/ reactor year)	Land Requiring Decontamination (acres per reactor year)
NCL	1.3×10^{-07}	1.0×10^{-3}	2.8×10^{-1}	0.0	6.1×10^{-7}	1.7×10^{-6}	5.4×10^{-07}
Case 1	2.1×10^{-08}	9.7×10^{-5}	2.3×10^{-2}	0.0	5.8×10^{-8}	1.5×10^{-7}	5.5×10^{-09}
Case 2	1.0×10^{-10}	2.7×10^{-7}	4.1×10^{-5}	0.0	1.6×10^{-10}	2.8×10^{-10}	$0.0 \times 10^{+00}$
Case 3	1.0×10^{-10}	1.4×10^{-5}	1.9×10^{-2}	0.0	8.6×10^{-9}	1.2×10^{-7}	2.1×10^{-06}
Case 4	1.0×10^{-10}	1.0×10^{-5}	1.3×10^{-2}	0.0	6.1×10^{-9}	8.8×10^{-8}	1.5×10^{-06}
Case 5	1.0×10^{-10}	5.1×10^{-6}	8.8×10^{-3}	0.0	3.1×10^{-9}	2.9×10^{-8}	5.1×10^{-07}
Case 6	1.0×10^{-10}	5.8×10^{-5}	1.9×10^{-1}	0.0	3.5×10^{-8}	4.2×10^{-6}	3.8×10^{-05}
Case 7	3.9×10^{-10}	2.6×10^{-4}	8.3×10^{-1}	0.0	1.6×10^{-7}	2.1×10^{-5}	1.6×10^{-04}
Case 8	4.1×10^{-10}	3.9×10^{-4}	1.3	5.1×10^{-14}	2.5×10^{-7}	5.7×10^{-5}	2.0×10^{-04}
Case 9	1.7×10^{-10}	1.8×10^{-4}	5.9×10^{-1}	1.2×10^{-13}	1.2×10^{-7}	3.3×10^{-5}	8.3×10^{-05}
Total	1.6×10^{-7}	2.0×10^{-3}	3.2	1.7×10^{-13}	1.2×10^{-6}	1.2×10^{-4}	4.8×10^{-04}

Note: $<1.0 \times 10^{-10}$ frequencies taken as 1.0×10^{-10} for analysis.

Table 7.2-3
Comparison to NRC Safety Goals

Goal	Risk Goal	Exelon ESBWR per unit	Exelon ABWR per unit
Individual Risk (0-1 mile)	4.0×10^{-7}	5.3×10^{-11}	1.4×10^{-12}
Societal Risk (0-10 miles)	1.8×10^{-6}	1.4×10^{-13}	1.5×10^{-13}

7.3 Severe Accident Mitigation Alternatives

The purpose of severe accident mitigation alternatives (SAMAs) is to review and evaluate plant design alternatives that could significantly reduce the radiological risk from a severe accident by preventing substantial core damage or by limiting releases from containment in the event that substantial core damage does occur.

SAMAs depend on design issues evaluated during the development and review of standard design certifications and COL applications. The design of the reactor and analyses of projected severe accidents are major contributing factors in the determination of SAMAs. To determine whether mitigation alternatives are cost beneficial, severe accident analyses must be included in these evaluations. SAMA would be evaluated for the new units in the COL application.

7.4 Transportation Accidents

Subsection 5.7.2.2 describes the methodology used to analyze the impacts of transportation of radioactive materials. Subsection 7.4.1 describes the radiological impacts of transportation accidents. The nonradiological impacts of transportation accidents are addressed in Subsection 7.4.2. The data currently available for the mPower reactor will not support an evaluation of radioactive materials transportation. Should Exelon select the mPower technology for VCS, an evaluation of radioactive materials transportation will be provided as part of the COL application.

7.4.1 Radiological Impacts of Transportation Accidents

7.4.1.1 Transportation of Unirradiated Fuel

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 CFR 51.52. Unirradiated fuel would be transported to the site via truck. Accident risks are calculated as frequency multiplied by consequence. Accident frequencies for transportation of fuel to future reactors are expected to be lower than those used in the analysis in WASH-1238 (AEC Dec 1972), which forms the basis for Table S-4 of 10 CFR 51.52, because of improvements in highway safety and security. Traffic accident, injury, and fatality rates have decreased over the past 30 years. Because fuel forms, cladding, and packages for the advanced light water reactor (LWR) technologies are similar to those of current generation LWRs, the consequences of accidents that are severe enough to result in a release of radioactivity to the environment are also similar. Accordingly, the risks of accidents during transportation of unirradiated fuel to the VCS site would be expected to be smaller than the reference LWR consequences listed in Table S-4.

7.4.1.2 Transportation of Spent Fuel

The RADTRAN 5 computer code was used to estimate impacts of transportation accidents involving spent fuel shipments. RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences (i.e., “fender benders”) to those with low frequencies and high consequences (i.e., accidents in which the shipping container is exposed to severe mechanical and thermal conditions).

The radionuclide inventory of the advanced LWR spent fuel after 5 years of decay was estimated using the ORIGEN code. A screening analysis was performed to select the dominant contributors to accident risks and to simplify the RADTRAN 5 calculations. This screening identified the radionuclides that would collectively contribute more than 99.999 percent of the dose from inhalation of radionuclides released following a transportation accident (U.S. NRC Dec 2006). The spent fuel inventory used in this analysis is presented in Table 7.4-1. The spent fuel transportation accident risks for each advanced LWR were calculated assuming the entire Co-60 inventory (Table 7.4-1) is in the form of crud.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance features required by 10 CFR 71, “Packaging and Transportation of Radioactive Material.” Spent fuel shipping casks must be certified Type B packaging systems, meaning they must withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability[1]. As stated in NUREG/CR-6672 (Sprung et al. Mar 2000), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). This analysis assumed that shipping casks for the advanced LWR spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo, in accordance with the requirements of 10 CFR 71.

The RADTRAN 5 accident risk calculations were performed using an assumption of 0.5 metric tons of uranium (MTU) per shipment for radionuclide inventories. The resulting risk estimates were multiplied by the expected annual spent fuel shipment amounts (in MTU per year) to derive estimates of the annual accident risks associated with the advanced LWR spent fuel shipments. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 30 MTU per year for the ABWR, 38.5 MTU per year for the ESBWR, 23 MTU per year for the AP1000, and 34.8 MTU per year for the APWR. (This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2.)

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

Using RADTRAN 5, the population dose from the released radioactive material was calculated for five possible exposure pathways:

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

1. Requirements for Type B packaging are set forth in 49 CFR § 173.413 and 10 CFR §§ 71.41 through 47 and § 71.51.

- Internal dose from resuspension of radioactive materials that were deposited on the ground (the radiation exposures from this pathway were included even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures).
- Internal dose from ingestion of contaminated food. No internal dose due to ingestion of contaminated foods was calculated because the analysis assumed interdiction of foodstuffs and evacuation of people after an accident would prevent such ingestion.

A sixth pathway, external doses from increased radiation fields surrounding a shipping cask with damaged shielding, was considered. It is possible that shielding materials incorporated into the cask structures could become damaged because of an accident; however, the loss of shielding events was not included in the analysis because their contribution to spent fuel transportation risk is much smaller than the dispersal accident risks from the pathways listed above.

Calculations were performed to assess the environmental consequences of transportation accidents when shipping spent fuel from the VCS to a spent fuel repository assumed to be at Yucca Mountain, Nevada. The shipping distances and population distribution information for the route were the same as those used for the “incident-free” transportation impacts analysis described in Subsection 5.7.2.2.

Table 7.4-2 presents accident risks associated with transportation of spent fuel from VCS to the proposed Yucca Mountain repository. The accident risks are provided in the form of a collective population dose (i.e., person-rem per year over the shipping campaign). The table also presents estimates of accident risk per reactor year normalized to the reference reactor analyzed in WASH-1238. The transportation accident impacts were also calculated for the alternative sites (Matagorda County, Buckeye, Alpha, and Bravo) in the region of interest.

The risk to the public from radiation exposure was estimated using the nominal probability coefficient for total detrimental health effects (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 1×10^6 person-rem) per reference reactor year from the International Commission on Radiological Protection Publication 60 (ICRP 1991). These values are presented in Table 7.4-2. These estimated risks are small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation. Therefore, negligible increases in environmental risk effects are expected from accidents that may result during shipping spent fuel from the site to a spent fuel disposal repository. The risks of accidents during transportation of spent fuel from the VCS site or an alternate site would be consistent with the environmental impacts presented in Table S-4.

7.4.2 Nonradiological Impacts of Transportation Accidents

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

7.4.2.1 Transportation of Unirradiated Fuel

The nonradiological accident impacts that could result from shipments of unirradiated fuel to the VCS site and return of empty containers from the site are presented in Table 7.4-3. The nonradiological impacts for the reference LWR analyzed in WASH-1238 are also shown for comparison. Nationwide median rates for interstate highway transportation from Saricks and Tompkins (Apr 1999) were used to estimate the annual impacts. Consistent with the incident-free transportation analysis described in Subsection 5.7.2, an average one-way shipping distance of 2000 miles was used to evaluate the unirradiated fuel shipments. The differences between the reference LWR and the advanced LWR results are due to the lower number of shipments per year (when normalized for electrical output) projected for the units at VCS. The values presented in Table 7.4-3 would be doubled for a two-unit plant.

7.4.2.2 Transportation of Spent Fuel

The general approach to calculating the nonradiological impacts for spent fuel shipments is similar to that for other radioactive materials shipments. The main difference is that the spent fuel shipping route characteristics are better defined, allowing the state-specific accident statistics in Saricks and Tompkins (Apr 1999) to be used in the analysis. State-by-state shipping distances and road types were obtained from the TRAGIS output file (see Subsection 5.7.2.2.2 for a discussion of the TRAGIS routing model). The shipping distances were doubled to allow for return shipments of empty containers to VCS. This information, the annual number of shipments, and state-specific accident statistics were used to estimate the nonradiological impacts presented in Table 7.4-4.

7.4.2.3 Transportation of Radioactive Waste

Nonradiological impacts of radioactive waste shipments were calculated using the same general approach as the unirradiated fuel shipments. A shipping distance of 500 miles was assumed consistent with the analysis in WASH-1238. Because the destination of the waste shipments is not known, the national median accident, injury, and fatality rates from Saricks and Tompkins (Apr 1999) were used to calculate the values presented in Table 7.4-5. The nonradiological impacts for the reference LWR analyzed in WASH-1238 are also shown for comparison. The differences between the reference LWR and the advanced LWR results are due to the higher number of radioactive waste shipments projected for the reactors except for the AP1000. The AP1000 design control document provides estimated shipped waste volumes assuming onsite processing to reduce the waste volume by a factor of three. Waste estimates for the other advanced LWR technologies do not reflect onsite processing. The values presented in Table 7.4-5 would be doubled for a two-unit plant.

7.4.3 Conclusion

Based on this analysis, the overall transportation accident risks associated with spent fuel shipments from the advanced LWR technologies being considered to demonstrate VCS site suitability are consistent with the risks associated with transportation of spent fuel from current generation reactors presented in WASH-1238 and Table S-4 of 10 CFR 51.52 (reproduced in Table 5.7-3) and thus will be SMALL.

7.4.4 References

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

ICRP 1991. International Commission on Radiological Protection, *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, 1991, Pergamon Press.

Saricks and Tompkins Apr 1999. Saricks, C. L. and M. M. Tompkins, *State-Level Accident Rates for Surface Freight Transportation: A Reexamination,* ANL/ESD/TM-150, April 1999, Argonne National Laboratory.

Sprung et al. Mar 2000. Sprung, J. L., D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, *Reexamination of Spent Fuel Shipment Risk Estimates*, NUREG/CR-6672, Volume 1, March 2000, U.S. Nuclear Regulatory Commission.

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

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

Radionuclide	ABWR Inventory (curies per MTU)	ESBWR Inventory (curies per MTU)	AP1000 Inventory (curies per MTU)	APWR Inventory (curies per MTU)
Am-241	1.44×10^3	1.30×10^3	7.27×10^2	1.81×10^3
Am-242m	3.32×10^1	2.79×10^1	1.31×10^1	2.04×10^1
Am-243	5.95×10^1	3.26×10^1	3.34×10^1	7.45×10^1
Ce-144	1.32×10^4	1.35×10^4	8.87×10^3	1.39×10^4
Cm-242	6.22×10^1	4.86×10^1	2.83×10^1	6.08×10^1
Cm-243	6.17×10^1	3.47×10^1	3.07×10^1	5.76×10^1
Cm-244	1.35×10^4	4.96×10^3	7.75×10^3	1.25×10^4
Cm-245	2.25	6.75×10^{-1}	1.21	–
Co-60	3.63×10^3	2.86×10^3	4.09	8.58×10^1
Cs-134	7.76×10^4	5.19×10^4	4.80×10^4	6.41×10^4
Cs-137	1.58×10^5	1.27×10^5	9.31×10^4	1.76×10^5
Eu-154	1.56×10^4	1.05×10^4	9.13×10^3	1.03×10^4
Eu-155	8.27×10^3	5.47×10^3	4.62×10^3	2.74×10^3
Pm-147	3.13×10^4	3.53×10^4	1.76×10^4	5.17×10^4
Pu-238	1.09×10^4	6.15×10^3	6.07×10^3	9.50×10^3
Np-239	–	–	–	7.45×10^1
Pu-239	4.27×10^2	3.86×10^2	2.55×10^2	4.08×10^2
Pu-240	8.52×10^2	6.22×10^2	5.43×10^2	6.97×10^2
Pu-241	1.35×10^5	1.22×10^5	6.96×10^4	1.68×10^5
Pu-242	3.19	2.24	1.82	–
Ru-106	2.29×10^4	1.86×10^4	1.55×10^4	2.46×10^4
Sb-125	7.17×10^3	5.80×10^3	3.83×10^3	3.39×10^3
Sr-90	1.06×10^5	9.08×10^4	6.19×10^4	1.20×10^5
Y-90	1.06×10^5	9.09×10^4	6.19×10^4	1.20×10^5

Table 7.4-2
Spent Fuel Transportation Accident Risks

Site	Victoria County	Matagorda County	Buckeye	Alpha	Bravo
ABWR					
Unit Population Dose (person-rem per MTU)[a]	2.78×10^{-7}	2.60×10^{-7}	2.60×10^{-7}	2.58×10^{-7}	2.92×10^{-7}
MTU per Reference Reactor Year	21.5	21.5	21.5	21.5	21.5
Population Dose (person-rem per reference reactor year)[a]	5.97×10^{-6}	5.59×10^{-6}	5.59×10^{-6}	5.54×10^{-6}	6.27×10^{-6}
Total Detrimental Health Effects per Reference Reactor Year	4.36×10^{-9}	4.08×10^{-9}	4.08×10^{-9}	4.05×10^{-9}	4.58×10^{-9}
ESBWR					
Unit Population Dose (person-rem per MTU)[a]	1.03×10^{-7}	9.70×10^{-8}	9.70×10^{-8}	9.62×10^{-8}	1.09×10^{-7}
MTU per Reference Reactor Year	23.0	23.0	23.0	23.0	23.0
Population Dose (person-rem per reference reactor year)[a]	2.38×10^{-6}	2.23×10^{-6}	2.23×10^{-6}	2.21×10^{-6}	2.50×10^{-6}
Total Detrimental Health Effects per Reference Reactor Year	1.73×10^{-9}	1.63×10^{-9}	1.63×10^{-9}	1.62×10^{-9}	1.82×10^{-9}
AP1000					
Unit Population Dose (person-rem per MTU)[a]	3.62×10^{-8}	3.40×10^{-8}	3.40×10^{-8}	3.38×10^{-8}	3.82×10^{-8}
MTU per Reference Reactor Year	19.5	19.5	19.5	19.5	19.5
Population Dose (person-rem per reference reactor year)[a]	7.04×10^{-7}	6.62×10^{-7}	6.62×10^{-7}	6.58×10^{-7}	7.43×10^{-7}
Total Detrimental Health Effects per Reference Reactor Year	5.14×10^{-10}	4.83×10^{-10}	4.83×10^{-10}	4.80×10^{-10}	5.43×10^{-10}
APWR					
Unit Population Dose (person-rem per MTU)[a]	6.34×10^{-8}	5.96×10^{-8}	5.96×10^{-8}	5.92×10^{-8}	6.68×10^{-8}
MTU per Reference Reactor Year	19.9	19.9	19.9	19.9	19.9
Population Dose (person-rem per reference reactor year)[a]	1.26×10^{-6}	1.19×10^{-6}	1.19×10^{-6}	1.18×10^{-6}	1.33×10^{-6}
Total Detrimental Health Effects per Reference Reactor Year	9.20×10^{-10}	8.65×10^{-10}	8.65×10^{-10}	8.59×10^{-10}	9.70×10^{-10}

(a) Value presented is the product of probability multiplied by collective dose.

Table 7.4-3
Nonradiological Impacts of Transporting Unirradiated Fuel to the Victoria County Station

Reactor	Total Shipments Normalized to Reference LWR	One-Way Shipping Distance (miles)	Total Round-trip Shipping Distance (miles)	Annual Impacts		
				Fatalities per Year	Injuries per Year	Accidents per Year
Reference LWR	252	2000	1.01×10^{6}	3.7×10^{-4}	0.0078	0.011
ABWR	195	2000	7.80×10^{5}	2.9×10^{-4}	0.0060	0.0088
ESBWR	221	2000	8.84×10^{5}	3.3×10^{-4}	0.0069	0.010
AP1000	172	2000	6.88×10^{5}	2.5×10^{-4}	0.0053	0.0078
APWR	132	2000	5.28×10^{5}	2.0×10^{-4}	0.0041	0.0060

Table 7.4-4 (Sheet 1 of 2)
Nonradiological Impacts of Transporting Spent Fuel from the Victoria County Station

State	Highway Type	One-Way Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
ABWR					
Arizona	Interstate	391	5.1×10^{-4}	0.0063	0.0071
California	Interstate	367	3.6×10^{-4}	0.0063	0.0081
Nevada	Interstate	66	6.0×10^{-5}	0.0013	0.0020
	Primary	79	1.8×10^{-4}	0.0028	0.0042
New Mexico	Interstate	164	2.7×10^{-4}	0.0026	0.0026
Texas	Interstate	670	0.0012	0.051	0.056
	Primary	71	2.8×10^{-4}	0.0051	0.0068
Totals	–	1,807	0.0029	0.075	0.086
ESBWR					
Arizona	Interstate	391	5.6×10^{-4}	0.0069	0.0078
California	Interstate	367	3.9×10^{-4}	0.0069	0.0089
Nevada	Interstate	66	6.5×10^{-5}	0.0015	0.0022
	Primary	79	2.0×10^{-4}	0.0030	0.0046
New Mexico	Interstate	164	2.9×10^{-4}	0.0029	0.0028
Texas	Interstate	670	0.0013	0.055	0.061
	Primary	71	3.1×10^{-4}	0.0056	0.0074
Totals	–	1,807	0.0031	0.082	0.094
AP1000					
Arizona	Interstate	391	4.6×10^{-4}	0.0057	0.0065
California	Interstate	367	3.2×10^{-4}	0.0057	0.0074
Nevada	Interstate	66	5.4×10^{-5}	0.0012	0.0019
	Primary	79	1.7×10^{-4}	0.0025	0.0038
New Mexico	Interstate	164	2.4×10^{-4}	0.0024	0.0023
Texas	Interstate	670	0.0011	0.046	0.050
	Primary	71	2.5×10^{-4}	0.0047	0.0062
Totals	–	1,807	0.0026	0.068	0.078

Table 7.4-4 (Sheet 2 of 2)
Nonradiological Impacts of Transporting Spent Fuel from the Victoria County Station

State	Highway Type	One-Way Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
APWR					
Arizona	Interstate	391	4.7×10^{-4}	0.0059	0.0066
California	Interstate	367	3.3×10^{-4}	0.0059	0.0076
Nevada	Interstate	66	5.6×10^{-5}	0.0013	0.0019
	Primary	79	1.7×10^{-4}	0.0026	0.0039
New Mexico	Interstate	164	2.5×10^{-4}	0.0024	0.0024
Texas	Interstate	670	0.0011	0.047	0.052
	Primary	71	2.6×10^{-4}	0.0048	0.0063
Totals	–	1,807	0.0027	0.070	0.080

Table 7.4-5
Nonradiological Impacts of Transporting Radioactive Waste from the Victoria County Station

Reactor	Shipments per Year Normalized to Reference LWR	One-Way Shipping Distance (miles)	Annual Impacts		
			Fatalities per Year	Injuries per Year	Accidents per Year
Reference LWR	46	500	6.8×10^{-4}	0.014	0.021
ABWR	144	500	2.1×10^{-3}	0.045	0.066
ESBWR	115	500	1.7×10^{-3}	0.036	0.052
AP1000	20	500	3.0×10^{-4}	0.0062	0.0091
APWR	106	500	1.6×10^{-3}	0.033	0.048