# CHAPTER 7
# ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS
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# CHAPTER 7 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter assesses the environmental impacts of postulated accidents involving radioactive materials. Section 7.1 evaluates design basis accidents. Section 7.2 considers the impact of severe accidents, Section 7.3 addresses severe accident mitigation alternatives, and Section 7.4 addresses transportation accidents.

## 7.1 DESIGN BASIS ACCIDENTS

This section evaluates the radiological consequences of design basis accidents. Subsection 7.1.1 lists the accidents considered, Subsection 7.1.2 outlines the evaluation methodology, Subsection 7.1.3 describes the source terms, and Subsection 7.1.4 presents the resulting consequences.

### 7.1.1 SELECTION OF ACCIDENTS

The design basis accidents considered in this section are from the DCD (WEC 2011). Table 7.1-1 lists the design basis accidents having the potential for releases to the environment, and shows the NUREG-0800 Standard Review Plan (SRP) section numbers and accident descriptions as well as the corresponding accidents as defined in the DCD. The radiological consequences of the accidents listed in Table 7.1-1 are assessed to demonstrate that new units can be sited at Turkey Point without undue risk to the health and safety of the public.

### 7.1.2 EVALUATION METHODOLOGY

The DCD presents the radiological consequences of the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with site data to demonstrate that the DCD analyses are bounding for the Turkey Point site. The basic scenario for each accident is that some quantity of activity is released at the accident location inside a building and this activity is eventually released to the environment. The transport of activity within the plant is independent of the site and specific to the AP1000 design. Details about the methodologies and assumptions pertaining to each of the accidents, such as activity release pathways and credited mitigation features, are provided in the DCD.

The dose to an individual located at the exclusion area boundary (EAB) or the low population zone (LPZ) is calculated based on the amount of activity released to the environment, the atmospheric dispersion of the activity during the transport from the release point to the offsite location, the breathing rate of the individual at the offsite location, and activity-to-dose conversion factors. The only variable parameter is atmospheric dispersion. Site-specific doses were obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (X/Q) values. Since the site-specific X/Q values are bounded by the DCD X/Q values, this approach demonstrates that the site-specific doses are within those calculated in the DCD.

The DCD uses conservative assumptions to perform bounding safety analyses that substantially overstate the environmental impact of the identified accidents. Among the conservative assumptions in the DCD is the use of time-dependent X/Q values corresponding to the top 5th percentile meteorology during the 2-hour accident period that yields the maximum dose, meaning that conditions would be more favorable for dispersion 95 percent of the time. In this environmental report, the maximum 2-hour dose is calculated based on the 50th percentile site-specific X/Q values, reflecting more realistic meteorological conditions.

The X/Q values were calculated using the methodology of RG 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,* (Rev. 1, Nov. 1982) with site-specific meteorological data. As described in Subsection 2.7.5, the methodology of RG 1.145 is implemented in the NRC-sponsored PAVAN computer program. This program computes X/Q values at the EAB and the LPZ for each combination of wind speed and atmospheric stability for each of the 16 downwind direction sectors and then calculates overall (nondirection-specific) X/Q values. For a given location, either the EAB or the LPZ, the initial maximum X/Q value is the 50th percentile overall value calculated by PAVAN. For the LPZ, the X/Q values for all subsequent times were calculated by logarithmic interpolation between the 50th percentile X/Q value and the annual average X/Q value. Releases were assumed to be at ground level, and the shortest distances between the power block and the offsite locations were selected to conservatively maximize the X/Q values.

The accident doses are expressed as total effective dose equivalent (TEDE), consistent with 10 CFR 50.34. The TEDE consists of the sum of the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent from external exposure. The CEDE is determined using the dose conversion factors in Federal Guidance Report 11 (U.S. EPA 1988), while the effective dose equivalent is based on the dose conversion factors in Federal Guidance Report 12 (U.S. EPA 1993). Appendix 15A of the DCD provides information on the methodologies used to calculate CEDE and effective dose equivalent values. As described in RG 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors* (Rev. 0, Jul 2000) the dose conversion factors in Federal Guidance Reports 11 and 12 are acceptable to the NRC Staff.

### 7.1.3 SOURCE TERMS

The design basis accident source terms in the DCD were calculated in accordance with RG 1.183, based on 102 percent of the rated core thermal power of 3400 MW (WEC 2011). The time-dependent isotopic activities released to the environment from each of the evaluated accidents are presented in Tables 7.1-2 to 7.1-10.

### 7.1.4 RADIOLOGICAL CONSEQUENCES

For each of the accidents identified in Table 7.1-1, the site-specific dose for a given time interval was calculated by multiplying the DCD dose by the ratio of the site X/Q value from Subsection 2.7.5.2 to the DCD X/Q value. The time-dependent DCD X/Q values and the time-dependent site X/Q values and their ratios are shown in Table 7.1-11. As all site X/Q values are bounded by DCD X/Q values, site-specific doses for all accidents are also bounded by DCD doses. The total doses are summarized in Table 7.1-12, based on individual accident doses presented in Tables 7.1-13 to 7.1-22. For each accident, the EAB dose shown is for the 2-hour period that yields the maximum dose, in accordance with RG 1.183.

The results of the site analysis contained in the referenced tables demonstrate that all accident doses meet the site acceptance criteria of 10 CFR 50.34. The acceptance criteria in 10 CFR 50.34 apply to accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. For events with a higher probability of occurrence, more restrictive dose limits are specified in RG 1.183. Where applied, the more restrictive dose limit is either 10 percent or 25 percent of the 10 CFR 50.34 limit of 25 rem TEDE. Although conformance to these more restrictive dose limits is not required for an environmental report, they are included in the tables for comparison purposes, and shown to result in doses that meet the more restrictive limits.

The TEDE dose limits shown in Tables 7.1-12 to 7.1-22 are from RG 1.183, Table 6, for all accidents except reactor coolant pump shaft break (NUREG-0800 SRP Section 15.3.4, Rev. 3, Mar 2007) and failure of small lines carrying primary coolant outside containment (NUREG-0800 SRP Section 15.6.2, Rev. 2, Jul 1981). Although RG 1.183 does not address these two accidents, NUREG-0800 identified a dose limit of 2.5 rem for these accidents. All doses are within the acceptance criteria. Because the dose criteria of 10 CFR 50.34 are intended to provide assurance of low risk to the public under postulated accidents, any health effects resulting from the design basis accidents are negligible.

**Section 7.1 References**

U.S. EPA 1988. U.S. Environmental Protection Agency, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11, EPA-520/1-88-020, 1988.

U.S. EPA 1993. *External Exposure to Radionuclides in Air, Water, and Soil*, Federal Guidance Report No. 12, EPA-402-R-93-081, 1993.

WEC 2011. Westinghouse Electric Company, LLC, *AP1000 Design Control Document*, Document No. APP-GW-GL-700, Tier 2 Material, Rev. 19, June 2011.

**Table 7.1-1**
**Selection of Accidents**

| SRP/DCD Section | SRP Description | DCD Description | Identified in NUREG-1555[(a)] Section 7.1 Appendix A | Comment |
|---|---|---|---|---|
| 15.1.5A | Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR | Steam System Piping Failure | Yes | Addressed in DCD Section 15.1.5 |
| 15.2.8 | Feedwater System Pipe Breaks Inside and Outside Containment (PWR) | Feedwater System Pipe Break | Yes | In the DCD, this is bounded by Section 15.1.5 accident |
| 15.3.3 | Reactor Coolant Pump Rotor Seizure | Reactor Coolant Pump Shaft Seizure (Locked Rotor) | Yes | |
| 15.3.4 | Reactor Coolant Pump Shaft Break | Reactor Coolant Pump Shaft Break | Yes | In the DCD, this is bounded by Section 15.3.3 accident |
| 15.4.8 | Spectrum of Rod Ejection Accidents (PWR) | Spectrum of Rod Cluster Control Assembly Ejection Accidents | No | Evaluated for completeness |
| 15.6.2 | Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment | Failure of Small Lines Carrying Primary Coolant Outside Containment | Yes | |
| 15.6.3 | Radiological Consequences of Steam Generator Tube Failure (PWR) | Steam Generator Tube Rupture | Yes | |
| 15.6.5A | Radiological Consequences of a Design Basis Loss of Coolant Accident Including Containment Leakage Contribution | Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary | Yes | Addressed in DCD Section 15.6.5 |
| 15.6.5B | Radiological Consequences of a Design Basis Loss of Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment | Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary | Yes | Addressed in DCD Section 15.6.5 |
| 15.7.4 | Radiological Consequences of Fuel Handling Accidents | Fuel Handling Accident | Yes | |

(a) Oct 1999

**Table 7.1-2**
**Activity Releases for Steam System Piping Failure with Preexisting Iodine Spike**

| Isotope | Activity Release (Ci) | | | | |
|---|---|---|---|---|---|
| | 0-2 hr | 2-8 hr | 8-24 hr | 24-72 hr | Total |
| Kr-85m | 6.86E-02 | 1.14E-01 | 6.80E-02 | 6.20E-03 | 2.57E-01 |
| Kr-85 | 2.82E-01 | 8.47E-01 | 2.25E+00 | 6.68E+00 | 1.01E+01 |
| Kr-87 | 2.76E-02 | 1.34E-02 | 5.20E-04 | 0.00E+00 | 4.15E-02 |
| Kr-88 | 1.12E-01 | 1.37E-01 | 4.04E-02 | 8.00E-04 | 2.90E-01 |
| Xe-131m | 1.28E-01 | 3.79E-01 | 9.81E-01 | 2.70E+00 | 4.19E+00 |
| Xe-133m | 1.59E-01 | 4.51E-01 | 1.04E+00 | 2.05E+00 | 3.70E+00 |
| Xe-133 | 1.18E+01 | 3.45E+01 | 8.65E+01 | 2.16E+02 | 3.49E+02 |
| Xe-135m | 3.04E-03 | 1.30E-05 | 0.00E+00 | 0.00E+00 | 3.05E-03 |
| Xe-135 | 3.10E-01 | 6.90E-01 | 8.35E-01 | 3.39E-01 | 2.17E+00 |
| Xe-138 | 3.99E-03 | 1.10E-05 | 0.00E+00 | 0.00E+00 | 4.00E-03 |
| I-130 | 3.59E-01 | 1.42E-01 | 2.09E-01 | 1.33E-01 | 8.43E-01 |
| I-131 | 2.40E+01 | 1.21E+01 | 3.10E+01 | 8.21E+01 | 1.49E+02 |
| I-132 | 3.05E+01 | 4.14E+00 | 8.07E-01 | 6.00E-03 | 3.55E+01 |
| I-133 | 4.34E+01 | 1.90E+01 | 3.53E+01 | 3.98E+01 | 1.38E+02 |
| I-134 | 6.74E+00 | 1.63E-01 | 1.40E-03 | 0.00E+00 | 6.90E+00 |
| I-135 | 2.60E+01 | 8.16E+00 | 7.54E+00 | 1.71E+00 | 4.34E+01 |
| Cs-134 | 1.90E+01 | 1.95E-01 | 5.19E-01 | 1.54E+00 | 2.13E+01 |
| Cs-136 | 2.82E+01 | 2.86E-01 | 7.42E-01 | 2.06E+00 | 3.13E+01 |
| Cs-137 | 1.37E+01 | 1.41E-01 | 3.74E-01 | 1.11E+00 | 1.53E+01 |
| Cs-138 | 1.01E+01 | 1.02E-03 | 0.00E+00 | 0.00E+00 | 1.01E+01 |
| Total | 2.15E+02 | 8.15E+01 | 1.68E+02 | 3.56E+02 | 8.21E+02 |

**Table 7.1-3**
**Activity Releases for Steam System Piping Failure with Accident-Initiated Iodine Spike**

| Isotope | Activity Release (Ci) | | | | |
|---|---|---|---|---|---|
| | 0-2 hr | 2-8 hr | 8-24 hr | 24-72 hr | Total |
| Kr-85m | 6.86E-02 | 1.14E-01 | 6.80E-02 | 6.20E-03 | 2.57E-01 |
| Kr-85 | 2.82E-01 | 8.47E-01 | 2.25E+00 | 6.68E+00 | 1.01E+01 |
| Kr-87 | 2.76E-02 | 1.34E-02 | 5.20E-04 | 0.00E+00 | 4.15E-02 |
| Kr-88 | 1.12E-01 | 1.37E-01 | 4.04E-02 | 8.00E-04 | 2.90E-01 |
| Xe-131m | 1.28E-01 | 3.79E-01 | 9.81E-01 | 2.70E+00 | 4.19E+00 |
| Xe-133m | 1.59E-01 | 4.51E-01 | 1.04E+00 | 2.05E+00 | 3.70E+00 |
| Xe-133 | 1.18E+01 | 3.45E+01 | 8.65E+01 | 2.16E+02 | 3.49E+02 |
| Xe-135m | 3.04E-03 | 1.30E-05 | 0.00E+00 | 0.00E+00 | 3.05E-03 |
| Xe-135 | 3.10E-01 | 6.90E-01 | 8.35E-01 | 3.39E-01 | 2.17E+00 |
| Xe-138 | 3.99E-03 | 1.10E-05 | 0.00E+00 | 0.00E+00 | 4.00E-03 |
| I-130 | 4.15E-01 | 9.95E-01 | 1.58E+00 | 1.01E+00 | 4.00E+00 |
| I-131 | 2.57E+01 | 5.73E+01 | 1.56E+02 | 4.13E+02 | 6.52E+02 |
| I-132 | 4.57E+01 | 9.74E+01 | 2.23E+01 | 2.00E-01 | 1.66E+02 |
| I-133 | 4.85E+01 | 1.14E+02 | 2.27E+02 | 2.55E+02 | 6.45E+02 |
| I-134 | 1.33E+01 | 1.86E+01 | 2.60E-01 | 0.00E+00 | 3.22E+01 |
| I-135 | 3.20E+01 | 7.74E+01 | 7.83E+01 | 1.77E+01 | 2.05E+02 |
| Cs-134 | 1.90E+01 | 1.95E-01 | 5.19E-01 | 1.54E+00 | 2.13E+01 |
| Cs-136 | 2.82E+01 | 2.86E-01 | 7.42E-01 | 2.06E+00 | 3.13E+01 |
| Cs-137 | 1.37E+01 | 1.41E-01 | 3.74E-01 | 1.11E+00 | 1.53E+01 |
| Cs-138 | 1.01E+01 | 1.02E-03 | 0.00E+00 | 0.00E+00 | 1.01E+01 |
| Total | 2.50E+02 | 4.03E+02 | 5.79E+02 | 9.19E+02 | 2.15E+03 |

**Table 7.1-4**
**Activity Releases for Reactor Coolant Pump Shaft Seizure**

| Isotope | Activity Release (Ci) | | |
|---|---|---|---|
| | No Feedwater | With Feedwater | |
| | 0-1.5 hr | 0-6 hr | 6-8 hr |
| Kr-85m | 8.15E+01 | 2.37E+02 | 4.10E+01 |
| Kr-85 | 7.58E+00 | 3.03E+01 | 1.01E+01 |
| Kr-87 | 1.20E+02 | 2.05E+02 | 5.28E+00 |
| Kr-88 | 2.07E+02 | 5.16E+02 | 5.94E+01 |
| Xe-131m | 3.77E+00 | 1.50E+01 | 4.94E+00 |
| Xe-133m | 2.02E+01 | 7.85E+01 | 2.48E+01 |
| Xe-133 | 6.67E+02 | 2.63E+03 | 8.57E+02 |
| Xe-135m | 3.19E+01 | 3.25E+01 | 0.00E+00 |
| Xe-135 | 1.59E+02 | 5.39E+02 | 1.31E+02 |
| Xe-138 | 1.27E+02 | 1.28E+02 | 0.00E+00 |
| I-130 | 8.44E-01 | 8.79E-01 | 5.64E-01 |
| I-131 | 3.78E+01 | 4.60E+01 | 3.46E+01 |
| I-132 | 2.80E+01 | 1.42E+01 | 3.90E+00 |
| I-133 | 4.87E+01 | 5.34E+01 | 3.65E+01 |
| I-134 | 2.87E+01 | 5.43E+00 | 2.03E-01 |
| I-135 | 4.18E+01 | 3.72E+01 | 2.03E+01 |
| Cs-134 | 2.99E+00 | 4.42E+00 | 3.32E+00 |
| Cs-136 | 1.43E+00 | 1.55E+00 | 1.03E+00 |
| Cs-137 | 1.81E+00 | 2.61E+00 | 1.95E+00 |
| Cs-138 | 8.30E+00 | 1.29E+00 | 4.11E-03 |
| Rb-86 | 2.95E-02 | 4.89E-02 | 3.78E-02 |
| Total | 1.63E+03 | 4.58E+03 | 1.24E+03 |

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

| Isotope | Activity Release (Ci) | | | | | |
|---|---|---|---|---|---|---|
| | 0-2 hr | 2-8 hr | 8-24 hr | 24-72 hr | 96-720 hr | Total |
| Kr-85m | 1.12E+02 | 6.48E+01 | 3.87E+01 | 1.77E+00 | 2.51E-05 | 2.17E+02 |
| Kr-85 | 5.01E+00 | 5.60E+00 | 1.49E+01 | 3.35E+01 | 2.88E+02 | 3.47E+02 |
| Kr-87 | 1.82E+02 | 2.60E+01 | 1.03E+00 | 8.37E-05 | 0.00E+00 | 2.09E+02 |
| Kr-88 | 2.91E+02 | 1.18E+02 | 3.49E+01 | 3.59E-01 | 8.41E-09 | 4.44E+02 |
| Xe-131m | 4.94E+00 | 5.46E+00 | 1.42E+01 | 2.86E+01 | 1.16E+02 | 1.69E+02 |
| Xe-133m | 2.67E+01 | 2.81E+01 | 6.49E+01 | 8.45E+01 | 5.31E+01 | 2.57E+02 |
| Xe-133 | 8.79E+02 | 9.58E+02 | 2.40E+03 | 4.27E+03 | 8.45E+03 | 1.70E+04 |
| Xe-135m | 7.34E+01 | 5.30E-02 | 4.33E-09 | 0.00E+00 | 0.00E+00 | 7.35E+01 |
| Xe-135 | 2.15E+02 | 1.72E+02 | 2.09E+02 | 4.35E+01 | 1.79E-01 | 6.40E+02 |
| Xe-138 | 2.99E+02 | 1.38E-01 | 3.19E-09 | 0.00E+00 | 0.00E+00 | 2.99E+02 |
| I-130 | 4.90E+00 | 7.28E+00 | 4.32E+00 | 2.03E-01 | 2.95E-04 | 1.67E+01 |
| I-131 | 1.36E+02 | 2.45E+02 | 2.31E+02 | 3.10E+01 | 1.68E+01 | 6.60E+02 |
| I-132 | 1.53E+02 | 9.94E+01 | 9.85E+00 | 8.24E-03 | 0.00E+00 | 2.62E+02 |
| I-133 | 2.72E+02 | 4.40E+02 | 3.18E+02 | 2.28E+01 | 2.41E-01 | 1.05E+03 |
| I-134 | 1.66E+02 | 2.85E+01 | 1.37E-01 | 4.48E-08 | 0.00E+00 | 1.95E+02 |
| I-135 | 2.39E+02 | 2.97E+02 | 1.19E+02 | 2.39E+00 | 7.32E-05 | 6.57E+02 |
| Cs-134 | 3.10E+01 | 6.22E+01 | 6.03E+01 | 7.76E+00 | 5.16E+00 | 1.66E+02 |
| Cs-136 | 8.89E+00 | 1.75E+01 | 1.67E+01 | 2.05E+00 | 6.58E-01 | 4.58E+01 |
| Cs-137 | 1.80E+01 | 3.62E+01 | 3.51E+01 | 4.52E+00 | 3.05E+00 | 9.69E+01 |
| Cs-138 | 1.09E+02 | 7.05E+00 | 1.68E-03 | 0.00E+00 | 0.00E+00 | 1.16E+02 |
| Rb-86 | 3.63E-01 | 7.27E-01 | 6.96E-01 | 8.67E-02 | 3.42E-02 | 1.91E+00 |
| Total | 3.23E+03 | 2.62E+03 | 3.57E+03 | 4.53E+03 | 8.93E+03 | 2.29E+04 |

**Table 7.1-6**
**Activity Releases for Failure of Small Lines Carrying Primary Coolant Outside Containment**

| Isotope | Activity Release (Ci) 0–2 hr |
|---|---|
| Kr-85m | 1.24E+01 |
| Kr-85 | 4.40E+01 |
| Kr-87 | 7.05E+00 |
| Kr-88 | 2.21E+01 |
| Xe-131m | 1.99E+01 |
| Xe-133m | 2.50E+01 |
| Xe-133 | 1.84E+03 |
| Xe-135m | 2.59E+00 |
| Xe-135 | 5.20E+01 |
| Xe-138 | 3.65E+00 |
| I-130 | 1.89E+00 |
| I-131 | 9.26E+01 |
| I-132 | 3.49E+02 |
| I-133 | 2.01E+02 |
| I-134 | 1.58E+02 |
| I-135 | 1.68E+02 |
| Cs-134 | 4.16E+00 |
| Cs-136 | 6.16E+00 |
| Cs-137 | 3.00E+00 |
| Cs-138 | 2.21E+00 |
| Total | 3.02E+03 |

**Table 7.1-7**
**Activity Releases for Steam Generator Tube Rupture with Preexisting Iodine Spike**

| Isotope | Activity Release (Ci) | | | |
|---|---|---|---|---|
| | 0-2 hr | 2-8 hr | 8-24 hr | Total |
| Kr-85m | 5.50E+01 | 2.14E+01 | 7.00E-03 | 7.64E+01 |
| Kr-85 | 2.19E+02 | 1.24E+02 | 1.30E-01 | 3.43E+02 |
| Kr-87 | 2.40E+01 | 3.76E+00 | 0.00E+00 | 2.78E+01 |
| Kr-88 | 9.20E+01 | 2.90E+01 | 0.00E+00 | 1.21E+02 |
| Xe-131m | 9.90E+01 | 5.56E+01 | 6.00E-02 | 1.55E+02 |
| Xe-133m | 1.23E+02 | 6.75E+01 | 6.00E-02 | 1.91E+02 |
| Xe-133 | 9.13E+03 | 5.09E+03 | 5.00E+00 | 1.42E+04 |
| Xe-135m | 3.51E+00 | 5.00E-03 | 0.00E+00 | 3.52E+00 |
| Xe-135 | 2.44E+02 | 1.15E+02 | 7.00E-02 | 3.59E+02 |
| Xe-138 | 4.66E+00 | 4.20E-03 | 0.00E+00 | 4.66E+00 |
| I-130 | 2.19E+00 | 7.48E-02 | 2.79E-01 | 2.54E+00 |
| I-131 | 1.47E+02 | 7.02E+00 | 3.21E+01 | 1.86E+02 |
| I-132 | 1.75E+02 | 1.42E+00 | 1.96E+00 | 1.78E+02 |
| I-133 | 2.64E+02 | 1.04E+01 | 4.24E+01 | 3.17E+02 |
| I-134 | 3.41E+01 | 3.19E-02 | 4.38E-03 | 3.41E+01 |
| I-135 | 1.56E+02 | 3.94E+00 | 1.22E+01 | 1.72E+02 |
| Cs-134 | 2.10E+00 | 2.52E-01 | 6.32E-01 | 2.98E+00 |
| Cs-136 | 3.14E+00 | 3.70E-01 | 9.20E-01 | 4.43E+00 |
| Cs-137 | 1.52E+00 | 1.82E-01 | 4.56E-01 | 2.16E+00 |
| Cs-138 | 7.33E-01 | 4.80E-04 | 1.00E-06 | 7.33E-01 |
| Total | 1.08E+04 | 5.53E+03 | 9.63E+01 | 1.64E+04 |

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

| Isotope | Activity Release (Ci) | | | |
|---|---|---|---|---|
| | 0-2 hr | 2-8 hr | 8-24 hr | Total |
| Kr-85m | 5.50E+01 | 2.14E+01 | 7.00E-03 | 7.64E+01 |
| Kr-85 | 2.19E+02 | 1.24E+02 | 1.30E-01 | 3.43E+02 |
| Kr-87 | 2.40E+01 | 3.76E+00 | 0.00E+00 | 2.78E+01 |
| Kr-88 | 9.20E+01 | 2.90E+01 | 0.00E+00 | 1.21E+02 |
| Xe-131m | 9.90E+01 | 5.56E+01 | 6.00E-02 | 1.55E+02 |
| Xe-133m | 1.23E+02 | 6.75E+01 | 6.00E-02 | 1.91E+02 |
| Xe-133 | 9.13E+03 | 5.09E+03 | 5.00E+00 | 1.42E+04 |
| Xe-135m | 3.51E+00 | 5.00E-03 | 0.00E+00 | 3.52E+00 |
| Xe-135 | 2.44E+02 | 1.15E+02 | 7.00E-02 | 3.59E+02 |
| Xe-138 | 4.66E+00 | 4.20E-03 | 0.00E+00 | 4.66E+00 |
| I-130 | 9.80E-01 | 2.19E-01 | 8.95E-01 | 2.09E+00 |
| I-131 | 4.92E+01 | 1.54E+01 | 7.57E+01 | 1.40E+02 |
| I-132 | 1.66E+02 | 8.36E+00 | 1.40E+01 | 1.88E+02 |
| I-133 | 1.05E+02 | 2.71E+01 | 1.20E+02 | 2.52E+02 |
| I-134 | 6.32E+01 | 3.02E-01 | 6.33E-02 | 6.36E+01 |
| I-135 | 8.58E+01 | 1.41E+01 | 4.84E+01 | 1.48E+02 |
| Cs-134 | 2.10E+00 | 2.52E-01 | 6.32E-01 | 2.98E+00 |
| Cs-136 | 3.14E+00 | 3.70E-01 | 9.20E-01 | 4.43E+00 |
| Cs-137 | 1.52E+00 | 1.82E-01 | 4.56E-01 | 2.16E+00 |
| Cs-138 | 7.33E-01 | 4.80E-04 | 1.00E-06 | 7.33E-01 |
| Total | 1.05E+04 | 5.57E+03 | 2.66E+02 | 1.63E+04 |

## Table 7.1-9 (Sheet 1 of 2)
## Activity Releases for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

| Isotope | Activity Release (Ci) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 1.4-3.4 hr | 0-2 hr | 2-8 hr | 8-24 hr | 24-96 hr | 96-720 hr | Total |
| I-130 | 5.64E+01 | 3.24E+01 | 7.95E+01 | 5.24E+00 | 6.28E-01 | 6.00E-03 | 1.18E+02 |
| I-131 | 1.68E+03 | 9.19E+02 | 2.57E+03 | 2.56E+02 | 1.92E+02 | 5.79E+02 | 4.52E+03 |
| I-132 | 1.23E+03 | 8.79E+02 | 1.26E+03 | 1.62E+01 | 6.00E-03 | 0.00E+00 | 2.16E+03 |
| I-133 | 3.23E+03 | 1.82E+03 | 4.72E+03 | 3.71E+02 | 8.40E+01 | 7.80E+00 | 7.00E+03 |
| I-134 | 6.60E+02 | 7.09E+02 | 4.29E+02 | 3.07E-02 | 0.00E+00 | 0.00E+00 | 1.14E+03 |
| I-135 | 2.56E+03 | 1.54E+03 | 3.36E+03 | 1.56E+02 | 4.80E+00 | 0.00E+00 | 5.06E+03 |
| Kr-85m | 1.42E+03 | 6.32E+02 | 3.14E+03 | 1.87E+03 | 8.60E+01 | 0.00E+00 | 5.73E+03 |
| Kr-85 | 8.31E+01 | 3.22E+01 | 2.65E+02 | 7.06E+02 | 1.59E+03 | 1.36E+04 | 1.62E+04 |
| Kr-87 | 1.10E+03 | 6.88E+02 | 1.26E+03 | 5.00E+01 | 0.00E+00 | 0.00E+00 | 2.00E+03 |
| Kr-88 | 3.11E+03 | 1.50E+03 | 5.76E+03 | 1.70E+03 | 1.70E+01 | 0.00E+00 | 8.98E+03 |
| Xe-131m | 8.26E+01 | 3.21E+01 | 2.62E+02 | 6.79E+02 | 1.37E+03 | 5.57E+03 | 7.91E+03 |
| Xe-133m | 4.43E+02 | 1.74E+02 | 1.37E+03 | 3.15E+03 | 4.11E+03 | 2.58E+03 | 1.14E+04 |
| Xe-133 | 1.47E+04 | 5.71E+03 | 4.62E+04 | 1.16E+05 | 2.06E+05 | 4.07E+05 | 7.81E+05 |
| Xe-135m | 1.06E+01 | 3.33E+01 | 2.62E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 3.59E+01 |
| Xe-135 | 3.15E+03 | 1.31E+03 | 8.33E+03 | 1.01E+04 | 2.10E+03 | 1.00E+01 | 2.19E+04 |
| Xe-138 | 3.11E+01 | 1.14E+02 | 6.90E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 1.21E+02 |
| Rb-86 | 3.04E+00 | 1.72E+00 | 4.60E+00 | 2.80E-01 | 1.00E-03 | 8.00E-03 | 6.61E+00 |
| Cs-134 | 2.58E+02 | 1.46E+02 | 3.92E+02 | 2.40E+01 | 1.00E-01 | 1.20E+00 | 5.63E+02 |
| Cs-136 | 7.33E+01 | 4.14E+01 | 1.11E+02 | 6.70E+00 | 0.00E+00 | 2.00E-01 | 1.59E+02 |
| Cs-137 | 1.51E+02 | 8.49E+01 | 2.28E+02 | 1.41E+01 | 0.00E+00 | 7.00E-01 | 3.28E+02 |
| Cs-138 | 1.50E+02 | 2.60E+02 | 6.96E+01 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 3.30E+02 |
| Sb-127 | 2.42E+01 | 1.14E+01 | 3.67E+01 | 2.14E+00 | 1.00E-02 | 1.00E-02 | 5.03E+01 |
| Sb-129 | 5.10E+01 | 2.71E+01 | 6.23E+01 | 1.48E+00 | 0.00E+00 | 0.00E+00 | 9.09E+01 |
| Te-127m | 3.15E+00 | 1.47E+00 | 4.83E+00 | 2.95E-01 | 2.00E-03 | 1.30E-02 | 6.61E+00 |
| Te-127 | 2.05E+01 | 1.02E+01 | 2.81E+01 | 1.11E+00 | 0.00E+00 | 0.00E+00 | 3.94E+01 |
| Te-129m | 1.07E+01 | 5.01E+00 | 1.64E+01 | 1.00E+00 | 1.00E-02 | 3.00E-02 | 2.25E+01 |
| Te-129 | 1.88E+01 | 1.39E+01 | 1.45E+01 | 3.00E-02 | 0.00E+00 | 0.00E+00 | 2.84E+01 |
| Te-131m | 3.17E+01 | 1.51E+01 | 4.69E+01 | 2.51E+00 | 0.00E+00 | 1.00E-02 | 6.45E+01 |
| Te-132 | 3.23E+02 | 1.52E+02 | 4.89E+02 | 2.84E+01 | 1.00E-01 | 1.00E-01 | 6.70E+02 |
| Sr-89 | 9.23E+01 | 4.31E+01 | 1.42E+02 | 8.60E+00 | 1.00E-01 | 3.00E-01 | 1.94E+02 |
| Sr-90 | 7.95E+00 | 3.71E+00 | 1.22E+01 | 7.50E-01 | 0.00E+00 | 4.00E-02 | 1.67E+01 |
| Sr-91 | 9.68E+01 | 4.79E+01 | 1.33E+02 | 5.30E+00 | 0.00E+00 | 0.00E+00 | 1.86E+02 |
| Sr-92 | 6.83E+01 | 3.91E+01 | 7.40E+01 | 1.00E+00 | 0.00E+00 | 0.00E+00 | 1.14E+02 |
| Ba-139 | 5.44E+01 | 3.74E+01 | 4.56E+01 | 1.50E-01 | 0.00E+00 | 0.00E+00 | 8.32E+01 |
| Ba-140 | 1.63E+02 | 7.61E+01 | 2.49E+02 | 1.51E+01 | 0.00E+00 | 4.00E-01 | 3.41E+02 |
| Mo-99 | 2.15E+01 | 1.01E+01 | 3.24E+01 | 1.86E+00 | 1.00E-02 | 0.00E+00 | 4.44E+01 |
| Tc-99m | 1.47E+01 | 7.54E+00 | 1.91E+01 | 5.90E-01 | 0.00E+00 | 0.00E+00 | 2.72E+01 |
| Ru-103 | 1.73E+01 | 8.08E+00 | 2.65E+01 | 1.62E+00 | 1.00E-02 | 6.00E-02 | 3.63E+01 |
| Ru-105 | 8.18E+00 | 4.33E+00 | 1.00E+01 | 2.40E-01 | 0.00E+00 | 0.00E+00 | 1.46E+01 |
| Ru-106 | 5.70E+00 | 2.66E+00 | 8.75E+00 | 5.40E-01 | 0.00E+00 | 3.00E-02 | 1.20E+01 |
| Rh-105 | 1.03E+01 | 4.88E+00 | 1.53E+01 | 8.30E-01 | 0.00E+00 | 0.00E+00 | 2.10E+01 |
| Ce-141 | 3.89E+00 | 1.82E+00 | 5.96E+00 | 3.64E-01 | 2.00E-03 | 1.20E-02 | 8.16E+00 |
| Ce-143 | 3.46E+00 | 1.64E+00 | 5.14E+00 | 2.78E-01 | 1.00E-03 | 0.00E+00 | 7.06E+00 |

**Table 7.1-9 (Sheet 2 of 2)**
**Activity Releases for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary**

| Isotope | Activity Release (Ci) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 1.4-3.4 hr | 0-2 hr | 2-8 hr | 8-24 hr | 24-96 hr | 96-720 hr | Total |
| Ce-144 | 2.94E+00 | 1.37E+00 | 4.51E+00 | 2.76E-01 | 2.00E-03 | 1.30E-02 | 6.17E+00 |
| Pu-238 | 9.16E-03 | 4.28E-03 | 1.41E-02 | 8.60E-04 | 0.00E+00 | 4.00E-05 | 1.93E-02 |
| Pu-239 | 8.06E-04 | 3.76E-04 | 1.24E-03 | 7.60E-05 | 1.00E-06 | 3.00E-06 | 1.70E-03 |
| Pu-240 | 1.18E-03 | 5.52E-04 | 1.81E-03 | 1.11E-04 | 1.00E-06 | 5.00E-06 | 2.48E-03 |
| Pu-241 | 2.65E-01 | 1.24E-01 | 4.08E-01 | 2.50E-02 | 1.00E-04 | 1.20E-03 | 5.58E-01 |
| Np-239 | 4.48E+01 | 2.12E+01 | 6.75E+01 | 3.84E+00 | 2.00E-02 | 1.00E-02 | 9.26E+01 |
| Y-90 | 8.08E-02 | 3.81E-02 | 1.22E-01 | 7.00E-03 | 0.00E+00 | 0.00E+00 | 1.67E-01 |
| Y-91 | 1.19E+00 | 5.54E-01 | 1.82E+00 | 1.11E-01 | 1.00E-03 | 4.00E-03 | 2.49E+00 |
| Y-92 | 7.89E-01 | 4.32E-01 | 9.19E-01 | 1.80E-02 | 0.00E+00 | 0.00E+00 | 1.37E+00 |
| Y-93 | 1.21E+00 | 6.00E-01 | 1.68E+00 | 6.80E-02 | 0.00E+00 | 0.00E+00 | 2.35E+00 |
| Nb-95 | 1.59E+00 | 7.46E-01 | 2.44E+00 | 1.49E-01 | 1.00E-03 | 5.00E-03 | 3.34E+00 |
| Zr-95 | 1.59E+00 | 7.41E-01 | 2.43E+00 | 1.49E-01 | 0.00E+00 | 6.00E-03 | 3.33E+00 |
| Zr-97 | 1.43E+00 | 6.89E-01 | 2.05E+00 | 9.80E-02 | 0.00E+00 | 0.00E+00 | 2.84E+00 |
| La-140 | 1.67E+00 | 7.92E-01 | 2.50E+00 | 1.39E-01 | 0.00E+00 | 0.00E+00 | 3.43E+00 |
| La-141 | 1.03E+00 | 5.54E-01 | 1.23E+00 | 2.70E-02 | 0.00E+00 | 0.00E+00 | 1.81E+00 |
| La-142 | 5.38E-01 | 3.57E-01 | 4.74E-01 | 2.00E-03 | 0.00E+00 | 0.00E+00 | 8.33E-01 |
| Nd-147 | 6.16E-01 | 2.89E-01 | 9.42E-01 | 5.70E-02 | 0.00E+00 | 1.00E-03 | 1.29E+00 |
| Pr-143 | 1.39E+00 | 6.50E-01 | 2.13E+00 | 1.28E-01 | 1.00E-03 | 3.00E-03 | 2.91E+00 |
| Am-241 | 1.20E-04 | 5.59E-05 | 1.84E-04 | 1.13E-05 | 0.00E+00 | 6.00E-07 | 2.52E-04 |
| Cm-242 | 2.82E-02 | 1.32E-02 | 4.33E-02 | 2.65E-03 | 2.00E-05 | 1.20E-04 | 5.93E-02 |
| Cm-244 | 3.46E-03 | 1.62E-03 | 5.32E-03 | 3.26E-04 | 1.00E-06 | 1.60E-05 | 7.28E-03 |
| Total | 3.53E+04 | 1.72E+04 | 8.14E+04 | 1.35E+05 | 2.16E+05 | 4.29E+05 | 8.79E+05 |

**Table 7.1-10**
**Activity Releases for Fuel Handling Accident**

| Isotope | Activity Release (Ci) 0–2 hr |
|---|---|
| Kr-85m | 8.40E+00 |
| Kr-85 | 1.10E+03 |
| Kr-88 | 3.00E-01 |
| Xe-131m | 5.52E+02 |
| Xe-133m | 2.30E+03 |
| Xe-133 | 8.88E+04 |
| Xe-135m | 1.02E+02 |
| Xe-135 | 5.68E+03 |
| I-130 | 7.00E-01 |
| I-131 | 3.47E+02 |
| I-132 | 2.44E+02 |
| I-133 | 1.08E+02 |
| I-135 | 3.20E+00 |
| Total | 9.92E+04 |

**Table 7.1-11**
**Atmospheric Dispersion Factors**

| Location | Time (hr) | χ/Q (sec/m$^3$) DCD | χ/Q (sec/m$^3$) Site | Ratio (Site/DCD) |
|---|---|---|---|---|
| EAB | 0–2 | 5.1E-04 | 1.89E-04 | 3.71E-01 |
| LPZ | 0–8 | 2.2E-04 | 5.29E-06 | 2.40E-02 |
| | 8–24 | 1.6E-04 | 4.02E-06 | 2.51E-02 |
| | 24–96 | 1.0E-04 | 2.21E-06 | 2.21E-02 |
| | 96–720 | 8.0E-05 | 9.39E-07 | 1.17E-02 |

**Table 7.1-12**
**Summary of Design Basis Accident Doses**

| DCD/SRP Section | Accident | Site Dose (rem TEDE) EAB | Site Dose (rem TEDE) LPZ | Limit[a] (rem TEDE) | Dose Table |
|---|---|---|---|---|---|
| 15.1.5A | Steam System Piping Failure | | | | |
| | Preexisting Iodine Spike | 1.9E-01 | 8.8E-03 | 25 | 7.1-13 |
| | Accident-Initiated Iodine Spike | 2.2E-01 | 2.4E-02 | 2.5 | 7.1-14 |
| 15.2.8 | Feedwater System Pipe Break[b] | | | | |
| 15.3.3 | Reactor Coolant Pump Shaft Seizure | | | | |
| | No Feedwater | 1.9E-01 | 4.3E-03 | 2.5 | 7.1-15 |
| | Feedwater Available | 1.5E-01 | 9.1E-03 | 2.5 | 7.1-16 |
| 15.3.4 | Reactor Coolant Pump Shaft Break[c] | | | | |
| 15.4.8 | Spectrum of Rod Cluster Control Assembly Ejection Accidents | 6.7E-01 | 6.0E-02 | 6.3 | 7.1-17 |
| 15.6.2 | Failure of Small Lines Carrying Primary Coolant Outside Containment | 4.1E-01 | 1.1E-02 | 2.5 | 7.1-18 |
| 15.6.3 | Steam Generator Tube Rupture | | | | |
| | Preexisting Iodine Spike | 5.2E-01 | 1.6E-02 | 25 | 7.1-19 |
| | Accident-Initiated Iodine Spike | 2.2E-01 | 1.0E-02 | 2.5 | 7.1-20 |
| 15.6.5A,B | Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary | 9.1E+00 | 5.6E-01 | 25 | 7.1-21 |
| 15.7.4 | Fuel Handling Accident | 1.0E+00 | 2.6E-02 | 6.3 | 7.1-22 |

(a) NUREG-1555 specifies a dose limit of 25 rem TEDE for all design basis accidents. The more restrictive limits shown in the table apply to safety analysis report doses, but are shown here to demonstrate that even these more restrictive limits are met.
(b) Feedwater System Pipe Break is bounded by Steam System Piping Failure, as indicated in the DCD.
(c) Reactor Coolant Pump Shaft Break is bounded by Reactor Coolant Pump Shaft Seizure, as indicated in the DCD.

## Table 7.1-13
## Doses for Steam System Piping Failure with Preexisting Iodine Spike

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 5.0E-01 | — | 3.71E-01 | 1.9E-01 | — |
| 0–8 hr | — | 2.6E-01 | 2.40E-02 | — | 6.3E-03 |
| 8–24 hr | — | 3.8E-02 | 2.51E-02 | — | 1.0E-03 |
| 24–96 hr | — | 7.2E-02 | 2.21E-02 | — | 1.6E-03 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 5.0E-01 | 3.7E-01 | — | 1.9E-01 | 8.8E-03 |
| Limit | — | — | — | 25 | 25 |

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

| Time | DCD Dose (rem TEDE) | | χ/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 6.0E-01 | — | 3.71E-01 | 2.2E-01 | — |
| 0–8 hr | — | 4.5E-01 | 2.40E-02 | — | 1.1E-02 |
| 8–24 hr | — | 2.0E-01 | 2.51E-02 | — | 5.0E-03 |
| 24–96 hr | — | 3.6E-01 | 2.21E-02 | — | 8.0E-03 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 6.0E-01 | 1.0E+00 | — | 2.2E-01 | 2.4E-02 |
| Limit | — | — | — | 2.5 | 2.5 |

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

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 5.0E-01 | — | 3.71E-01 | 1.9E-01 | — |
| 0–8 hr | — | 1.8E-01 | 2.40E-02 | — | 4.3E-03 |
| 8–24 hr | — | 0 | 2.51E-02 | — | 0 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 5.0E-01 | 1.8E-01 | — | 1.9E-01 | 4.3E-03 |
| Limit | — | — | — | 2.5 | 2.5 |

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

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 6–8 hr | 4.0E-01 | — | 3.71E-01 | 1.5E-01 | — |
| 0–8 hr | — | 3.8E-01 | 2.40E-02 | — | 9.1E-03 |
| 8–24 hr | — | 0 | 2.51E-02 | — | 0 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 4.0E-01 | 3.8E-01 | — | 1.5E-01 | 9.1E-03 |
| Limit | — | — | — | 2.5 | 2.5 |

Note: Maximum 2-hour EAB dose occurs between 6 and 8 hours.

## Table 7.1-17
## Doses for Spectrum of Rod Cluster Control Assembly Ejection Accidents

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 1.8E+00 | — | 3.71E-01 | 6.7E-01 | — |
| 0–8 hr | — | 2.0E+00 | 2.40E-02 | — | 4.8E-02 |
| 8–24 hr | — | 4.2E-01 | 2.51E-02 | — | 1.1E-02 |
| 24–96 hr | — | 4.2E-02 | 2.21E-02 | — | 9.3E-04 |
| 96–720 hr | — | 2.1E-02 | 1.17E-02 | — | 2.5E-04 |
| Total | 1.8E+00 | 2.5E+00 | — | 6.7E-01 | 6.0E-02 |
| Limit | — | — | — | 6.3 | 6.3 |

## Table 7.1-18
## Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 1.1E+00 | — | 3.71E-01 | 4.1E-01 | — |
| 0–8 hr | — | 4.5E-01 | 2.40E-02 | — | 1.1E-02 |
| 8–24 hr | — | 0 | 2.51E-02 | — | 0 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 1.1E+00 | 4.5E-01 | — | 4.1E-01 | 1.1E-02 |
| Limit | — | — | — | 2.5 | 2.5 |

**Table 7.1-19**
**Doses for Steam Generator Tube Rupture with Preexisting Iodine Spike**

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 1.4E+00 | — | 3.71E-01 | 5.2E-01 | — |
| 0–8 hr | — | 6.2E+01 | 2.40E-02 | — | 1.5E-02 |
| 8–24 hr | — | 4.1E-02 | 2.51E-02 | — | 1.0E-03 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 1.4E+00 | 6.6E+01 | — | 5.2E-01 | 1.6E-02 |
| Limit | — | — | — | 25 | 25 |

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

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 6.0E-01 | — | 3.71E-01 | 2.2E-01 | — |
| 0–8 hr | — | 3.2E-01 | 2.40E-02 | — | 7.7E-03 |
| 8–24 hr | — | 1.0E-01 | 2.51E-02 | — | 2.5E-03 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 6.0E-01 | 4.2E-01 | — | 2.2E-01 | 1.0E-02 |
| Limit | — | — | — | 2.5 | 2.5 |

## Table 7.1-21
## Doses for Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 1.4–3.4 hr | 2.46E+01 | — | 3.71E-01 | 9.1E+00 | — |
| 0–8 hr | — | 2.2E+01 | 2.40E-02 | — | 5.3E-01 |
| 8–24 hr | — | 7.5E-01 | 2.51E-02 | — | 1.9E-02 |
| 24–96 hr | — | 2.9E-01 | 2.21E-02 | — | 6.4E-03 |
| 96–720 hr | — | 5.5E-01 | 1.17E-02 | — | 6.5E-03 |
| Total | 2.46E+01 | 2.4E+01 | — | 9.1E+00 | 5.6E-01 |
| Limit | — | — | — | 25 | 25 |

Note: Maximum 2-hour EAB dose occurs between 1.4 and 3.4 hours.

## Table 7.1-22
## Doses for Fuel Handling Accident

| Time | DCD Dose (rem TEDE) | | X/Q Ratio (Site/DCD) | Site Dose (rem TEDE) | |
|---|---|---|---|---|---|
| | EAB | LPZ | | EAB | LPZ |
| 0–2 hr | 2.7E+00 | — | 3.71E-01 | 1.0E+00 | — |
| 0–8 hr | — | 1.1E+00 | 2.40E-02 | — | 2.6E-02 |
| 8–24 hr | — | 0 | 2.51E-02 | — | 0 |
| 24–96 hr | — | 0 | 2.21E-02 | — | 0 |
| 96–720 hr | — | 0 | 1.17E-02 | — | 0 |
| Total | 2.7E+00 | 1.1E+00 | — | 1.0E+00 | 2.6E-02 |
| Limit | — | — | — | 6.3 | 6.3 |

## 7.2 SEVERE ACCIDENTS

Severe accidents are defined as accidents with substantial damage to the reactor core and degradation of containment systems. Because the probability of a severe accident is very low for the AP1000, such accidents are not part of the design basis for the plant. However, the NRC requires, in its *Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants* (50 FR 32138), the completion of a probabilistic risk assessment for severe accidents for new reactor designs. This requirement is codified in 10 CFR 52.47, *Contents of Applications*.

Westinghouse completed a probabilistic risk assessment for the AP1000 design (WEC 2004) as part of their application for design certification. The AP1000 design was reviewed by the NRC, and the review was documented in NUREG-1793, *Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design*. Subsequently, the NRC certified the design, concluding that this advanced design meets the NRC's safety goals and represents an improvement in safety over currently operating reactors in the United States.

The Westinghouse analysis used generic, but conservative, meteorology and regional characteristics. FPL presents in this section an update of the generic probabilistic risk assessment analysis of severe accidents to include Turkey Point site-specific characteristics and impacts over the entire life cycle of a severe accident. The purpose of this section is to show the complete impacts of a severe accident, demonstrate that the impacts are less than NRC safety goals, and support the severe accident mitigation alternatives analyses in Section 7.3.

### 7.2.1 WESTINGHOUSE METHODOLOGY

The Westinghouse probabilistic risk assessment for the AP1000 established an event tree that defined the possible functional end states of the containment following a severe accident initiated by internal events. These end states are grouped into three categories: (1) an intact containment with normal leakage or a larger leak with a containment isolation failure, (2) a containment breach, possibly a result of high containment pressure or a hydrogen detonation, and (3) containment bypass such as a steam generator tube rupture. Using the EPRI code Modular Accident Analysis Program, Westinghouse determined that six source term categories would represent the entire suite of potential severe accidents from these three end state categories. An accident frequency was assigned to each of the six categories (Table 7.2-1).

The six source term categories or accident categories are:

1. Intact Containment — Containment integrity is maintained throughout the accident. The release of radioactivity to the environment is a result of nominal design leakage.

2. Containment Bypass — Radioactivity is released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs before the onset of core damage. This accident category contributes to the large, early release frequency.

3. Containment Isolation Failure — Radioactivity is released through a failure of the valves that close the penetrations between containment and the environment. Containment failure occurs before the onset of core damage. This accident category contributes to the large, early release frequency.

4. Early Containment Failure — Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after the onset of core damage but before core relocation. Such phenomena could include hydrogen detonation, hydrogen diffusion flame, steam explosions, or vessel failures. This accident category contributes to the large, early release frequency.

5. Intermediate Containment Failure — Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after core relocation but before 24 hours have passed since initiation of the accident. Such phenomena could include hydrogen detonation and hydrogen deflagration. This accident category contributes to large releases but does not occur early in the accident life cycle.

6. Late Containment Failure — Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon more than 24 hours after initiation of the accident. Such phenomena could include the failure of containment heat removal. This accident category contributes to large releases but does not occur early in the accident life cycle.

Westinghouse then used the NRC code MACCS2 (Chanin and Young May 1997) to model the environmental consequences of the severe accidents described above. The MELCOR Accident Consequence Code System (MACCS) and its successor MACCS2 were developed specifically for the NRC to evaluate severe accidents at nuclear power plants. The meteorology Westinghouse used to represent a generic AP1000 site is specified in EPRI's Utility Requirements Document (EPRI Mar 1999). The meteorology is from a database selected because it is expected to result in calculated impacts greater than those that would be expected at 80 to 90 percent of U.S. operating plants. The population considered also was selected to provide impacts greater than those that would be expected at 80 to 90 percent of the plants. The Westinghouse analysis focused on 24 hours following core damage as a measure of the consequences from a large release and, therefore, did not address the chronic exposure pathways such as ingestion, inhalation of resuspended material, or groundshine subsequent to plume passage.

Additional details on the Westinghouse analysis are found in (WEC 2004) and reported in the DCD (WEC 2011).

### 7.2.2 FPL METHODOLOGY

FPL also used the MACCS2 computer code to evaluate consequences of severe accidents. The exposure pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water. The MACCS2 code primarily addresses dose from the air exposure pathway, but also calculates dose from surface runoff and deposits on surface water. The code also evaluates the extent of contamination. A difference between the Westinghouse generic analysis and the Turkey Point site-specific analysis is that FPL used site-specific meteorology and population data and extended the analysis to include 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, FPL determined the collective dose to the 50-mile population, number of latent cancer fatalities, and number of early fatalities associated with each severe accident category. 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. One file 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. A second file provides inputs to calculations regarding exposure in the time period immediately following the release. Important site-specific information includes emergency response information such as evacuation time. A third input file provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors. These three files access both a meteorological file, which uses actual Turkey Point meteorological monitoring data and a site characteristics file which is built using SECPOP2000 (NUREG/CR-6525) as a template.

Three years of meteorological data (2002, 2005, and 2006) from the existing Units 3 and 4 60-meter meteorological tower were analyzed. MACCS2 requires an entire calendar year of meteorological data. The year 2002 meteorology data was selected for subsequent analyses because it resulted in the largest consequences of the years analyzed, and, therefore, is the most conservative meteorological dataset of the 3 years.

For this analysis, the census data were modified to include transient populations and projected to the year 2080, as described in Subsection 2.5.1. 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 nonfarm land) in the same manner as the population. Agricultural production and economic parameters were taken from the 2007 National Census of Agriculture. Nonfarm land property values were taken from 2010 Florida property tax records for the portion of the counties within 50 miles of Turkey Point.

The resultant MACCS2 calculations and accident frequency information was used to determine risk. The consequence risk is the product of frequency of an accident times the consequences of the accident. The consequence can be either radiation dose or economic cost. Dose-risk is the product of the collective dose times the accident frequency. Because the AP1000's severe accident analysis addressed a suite of accidents, the individual risks were summed to provide a total risk. Similarly, cost-risk is the product of economic cost times the accident frequency, and the individual risks were summed to provide a total cost-risk. Therefore, risk can be reported as person-rem per reactor year or dollars per reactor year.

A ground-level release height and no release heat for each accident release hypothesized was assumed. A sensitivity analysis was performed on each of those assumptions; release heights of middle and top of containment and release heat of 1 and 10 megawatt per release segment were considered. The dose-risk varied by less than 3.3 percent for each of the sensitivity calculations.

An evacuation time estimate for the population surrounding the Turkey Point site which assumed evacuation to a 10-mile radius was also performed. The evacuation time estimate was used in the MACCS2 analysis to estimate the evacuation of transient and resident populations within the 10-mile radius.

As described above, the resulting MACCS2 calculations include only internally initiated events, consistent with the Westinghouse analysis. The external event core damage frequencies are slightly greater than the internal event core damage frequencies. An approach to qualitatively estimate the total event core damage frequency (internal and external events) could be to double the internal event core damage frequency, which would double the resulting dose-risk or cost-risk.

### 7.2.3 CONSEQUENCES TO POPULATION GROUPS

The exposure pathway consequences to population groups including air exposure pathways, surface water exposure pathways, and groundwater exposure pathways are addressed in the following sections.

#### 7.2.3.1 Air Exposure Pathways

Each of the six accident categories was analyzed with MACCS2 to estimate population dose, number of early and latent cancer fatalities, cost, and farmland requiring decontamination. The analysis assumed that 95 percent of the population was evacuated following declaration of a general emergency. For each accident category, FPL calculated the risk for each analytical endpoint (population dose, fatalities, cost, and contaminated land) by multiplying it by the accident category frequency. The results are provided in Table 7.2-1.

#### 7.2.3.2 Surface Water Exposure Pathways

People can be exposed to radiation when deposited airborne radioactivity runs off into or is deposited onto surface water. The exposure pathway can be from drinking the water, external radiation from submersion in the water, external radiation from human activities near the shoreline, or ingestion of fish or shellfish. MACCS2 only calculates the dose from drinking the water. The MACCS2 severe accident dose-risk to the 50-mile population from drinking water is 0.0079 person-rem per year of AP1000 operation. This value is included with the air exposure pathways dose and is the sum of all six accident category risks.

Surface water exposure pathways involving swimming, fishing, boating, and performing activities near the shoreline are not modeled by MACCS2. Surface water bodies within the 50-mile region of Turkey Point include the Biscayne Bay, Atlantic Ocean, Card Sound, the Everglades, canals, ponds, and other smaller water bodies. 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. NUREG-1437 evaluated doses from the aquatic food exposure pathway (fishing) for the existing licensed power reactors. For sites near large water bodies, the NRC evaluation estimated the uninterdicted aquatic food exposure pathway dose risk which ranged from 270 person-rem per reactor year (Hope Creek on the Delaware Bay) to 5500 person-rem per reactor year (Calvert Cliffs on the Chesapeake Bay). The Units 6 & 7 site would more likely be similar to Calvert Cliffs on the Chesapeake Bay. Actual dose-risk values would be expected to be much less (by a factor of 2 to 10) due to interdiction of contaminated foods (NUREG-1437). Furthermore, because the AP1000 atmospheric exposure pathway doses are lower than those of the existing licensed power reactors, it is reasonable to conclude that the doses from surface water sources would be considerably lower than those reported above for the surface water exposure pathway.

#### 7.2.3.3 Groundwater Exposure Pathways

Radioactivity released during an accident can directly and indirectly enter groundwater that serves as a source of drinking water or irrigation, or can move through an aquifer that eventually discharges to surface water. NUREG-1437 evaluated the groundwater exposure pathway dose, based on the analysis in NUREG-0440, *Liquid Pathway Generic Study*. NUREG-0440 analyzed a core meltdown that contaminated groundwater which subsequently contaminated surface water. However, NUREG-0440 did not analyze direct drinking of groundwater because of the limited number of potable groundwater wells and limited accessibility.

The *Liquid Pathway Generic Study* results provide conservative, uninterdicted population dose estimates for six generic categories of plants. These dose estimates were one or more orders-of-magnitude less than those attributed to the atmospheric exposure pathway. The Units 6 & 7 site is represented by one of these categories and would be bounded by this analysis. Therefore, the doses from the Units 6 & 7 site groundwater exposure pathway would be much less than the doses from the atmospheric exposure pathway.

## 7.2.4 COMPARISON TO NRC SAFETY GOALS

FPL compared the severe accident risks from Units 6 & 7 against two risk goals identified by the NRC (51 FR 30028) as described below. The results are presented in Table 7.2-2.

### 7.2.4.1 Individual Risk Goal

The risk of prompt fatalities that might result from reactor accidents to an average individual in the vicinity of a nuclear power plant should not exceed 0.1 percent of the sum of “prompt fatality risks” resulting from other accidents to which members of the U.S. population are generally exposed. As noted in the Safety Goals Policy statement (51 FR 30028), “vicinity” is defined as the area within 1 mile of the plant site boundary. “Prompt Fatality Risks” are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. Such risks are the sum of risks that result in fatalities from such activities as driving, household chores, occupational activities, etc. For this evaluation, the sum of prompt fatality risks was taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year for 2005 (CDC Apr 2008).

### 7.2.4.2 Societal Risk Goal

The risk of cancer fatalities that might result from nuclear power plant operations to the population in the area near a nuclear power plant should not exceed 0.1 percent of the sum of the cancer fatality risks resulting from all other causes. As noted in the Safety Goal Policy Statement (51 FR 30028), “near” is defined as within 10 miles of the plant. The cancer fatality risk from all other sources was taken as 186.6 deaths per 100,000 people per year for 2003 to 2005 (CDC Apr 2008).

## 7.2.5 CONCLUSIONS

The total calculated dose-risk to the 50-mile population from airborne releases from an AP1000 reactor at Turkey Point would be 0.27 person-rem per reactor year (Table 7.2-1). This value is greater than the 0.043 person-rem per reactor year reported by Westinghouse in the DCD (WEC 2011). The FPL analysis included long-term (chronic) exposure pathways in the dose-risk. The equivalent short-term exposure pathway dose from a single AP1000 reactor at Turkey Point would be 0.083 person-rem per reactor year. This value is also greater than the dose-risk reported in the DCD. This is a result of the large population within 50 miles surrounding Units 6 & 7.

The AP1000 dose-risk at the Units 6 & 7 site is less than the population risk for all current reactors that have performed severe accident mitigation alternatives (SAMA) analysis through 2008 as part of license renewal, and less than that for the five reactors analyzed in NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*.

Comparisons with the existing licensed power reactors indicate that risk from the surface water exposure pathway is small. Under the severe accident scenarios, surface water is primarily contaminated by atmospheric deposition. The AP1000 atmospheric exposure pathway doses are significantly lower than those of the existing licensed power reactors. Therefore, it is reasonable to conclude that the doses from the surface water exposure pathway at the Units 6 & 7 site would be consistently lower than those for the currently licensed power reactors.

The risks of groundwater contamination from a severe AP1000 accident (see Subsection 7.2.3.3) would be much less than the risk from currently licensed power reactors. Additionally, interdiction could substantially reduce the groundwater exposure pathway risks.

For comparison, as reported in Section 5.4, the total collective dose from Units 6 & 7 normal operations is expected to be 4.0 person-rem per year. As previously described, dose-risk is dose times frequency. Normal operations have a frequency no greater than one. Therefore, the dose-risk for normal operations is 4.0 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.27 person-rem per reactor year indicates that the dose-risk from severe accidents is approximately 7 percent of the dose-risk from normal operations.

The risk of cancer fatalities from a severe accident for the Units 6 & 7 site is reported in Table 7.2-2 as 2.1E-10 for early fatality risk per reactor year and 2.6E-12 late (cancer) fatalities per year per reactor year. Comparing these values to the NRC safety goals indicates that the risk is less than 0.1 percent of the NRC safety goals.

The impacts from an AP1000 reactor at the Units 6 & 7 site would be SMALL because the probability-weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts from severe accidents are small and because the early and late fatality risks meet the NRC safety goals.

**Section 7.2 References**

CDC Apr 2008. Centers for Disease Control, *Deaths: Final Data for 2005*, National Vital Statistics Reports, Volume 56, Number 10, April 24, 2008.

Chanin and Young May 1997. *Code Manual for MACCS2: User's Guide*, NUREG/CR-6613, Volume 1, SAND97-0594, May, 1997.

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

WEC 2004. *AP1000 Probabilistic Risk Assessment*, Revision 8. Pittsburgh, Pennsylvania, 2004.

WEC 2011. *Design Control Document,* Revision 19, Appendix 1B, "Severe Accident Mitigation Design Alternatives." June 2011.

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

| | | | Environmental Risk | | | |
|---|---|---|---|---|---|---|
| Accident Category | Accident Frequency (per reactor year)[(a)] | Population Dose-Risk (person-rem/ reactor year) | Number of Fatalities (per reactor year) | | Cost-Risk in Dollars[(b)] (per reactor year) | Land Requiring Decontamination (acres/reactor year) |
| | | | Early | Late | | |
| Intact containment | 2.2E-07 | 4.0E-03 | 0.0E+0 | 2.4E-06 | 0.78 | 1.6E-07 |
| Containment bypass | 1.1E-08 | 2.0E-01 | 3.0E-07 | 1.4E-04 | 497 | 2.8E-04 |
| Containment isolation failure | 1.3E-09 | 8.3E-03 | 1.3E-09 | 5.4E-06 | 18 | 1.3E-05 |
| Early containment failure | 7.5E-09 | 5.0E-02 | 2.5E-08 | 3.4E-05 | 116 | 7.9E-05 |
| Intermediate containment failure | 1.9E-09 | 1.5E-03 | 5.0E-11 | 9.9E-07 | 4.2 | 3.5E-06 |
| Late containment failure | 3.5E-13 | 4.3E-06 | 0.0E+0 | 2.7E-09 | 0.014 | 9.0E-09 |
| **Total** | **2.4E-07** | **2.7E-01** | **3.2E-07** | **1.8E-04** | **636** | **3.8E-04** |

(a) (WEC 2004).
(b) Presented in 2012 dollars.

## Table 7.2-2
## Comparison to NRC Safety Goals

| | Safety Risk | |
|---|---|---|
| | Early Fatality Risk (individual 0-1 mile) (deaths per reactor year) | Late Fatalities (0-10 mile cancers) (deaths per year per reactor year) |
| Safety Goal[(a)] | 3.9E-07 | 1.9E-06 |
| Unit 6 or 7 | 2.0E-10 | 2.6E-12 |

(a) (CDC Apr 2008)

## 7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES

As described in Section 7.2, Westinghouse performed a generic severe accident analysis for the AP1000 as part of the design certification process (WEC 2011). The Westinghouse analysis determined that severe accident impacts are small and that no potential mitigating design alternatives are cost-effective, that is, appropriate mitigating measures are already incorporated into the plant design. Section 7.2 extends the Westinghouse generic severe accident analysis to examine the proposed new nuclear units at Turkey Point and determined that the generic conclusions remain valid for the Units 6 & 7 site. The analysis in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented at the Units 6 & 7 site to mitigate these small impacts.

### 7.3.1 THE SEVERE ACCIDENT MITIGATION ALTERNATIVE ANALYSIS PROCESS

Design or procedural modifications that could mitigate the consequences of a severe accident are known as severe accident mitigation alternatives (SAMAs). In the past, SAMAs were known as severe accident mitigation design alternatives (SAMDAs) that primarily focused on design changes and did not consider procedural modification SAMAs. The Westinghouse DCD analysis is an SAMDA analysis. For an existing plant with a well-defined design and established procedural controls, the normal evaluation process for identifying potential SAMAs includes four steps:

1. Define the base case — The base case is the dose-risk and cost-risk of a severe accident before implementation of any SAMAs. A plant's probabilistic risk assessment is a primary source of data in calculating the base case. The base case risks are converted to a monetary value to use for screening SAMAs. Section 7.2 presents the base case for a single AP1000 unit at the Units 6 & 7 site, without the monetization step.

2. Identify and screen potential SAMAs — Potential SAMAs can be identified from the plant's individual plant examination, the plant's probabilistic risk assessment, and the results of other plants' SAMA analyses. This list of potential SAMAs is assigned a conservatively low implementation cost based on historical costs, similar design changes, and/or engineering judgment, then compared to the base case screening value. SAMAs with higher implementation cost than the base case are not evaluated further.

3. Determine the cost and net value of each SAMA — Each SAMA remaining after Step 2 has a detailed engineering cost evaluation developed using current plant engineering processes. If the SAMA continues to pass the screening value, Step 4 is performed.

4. Determine the benefit associated with each screened SAMA — Each SAMA that passes the screening in Step 3 is evaluated using the probabilistic risk assessment model to

determine the reduction in risk associated with implementation of the proposed SAMA. The reduction in risk benefit is then monetized and compared to the detailed cost estimate. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

The base case benefit value is calculated by assuming the current dose-risk of the unit could be reduced to zero and assigning a defined dollar value for this change in risk. Any design or procedural change cost that exceeded the benefit value would not be considered cost-effective. The dose-risk and cost-risk results (Section 7.2 analyses) are monetized in accordance with methods established in NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*. NUREG/BR-0184 presents methods for determining the value of decreases in risk using four types of attributes: public health, occupational health, offsite property, and onsite property. Any SAMAs in which the conservatively low implementation cost exceeds the base case monetization would not be expected to pass the screening in Step 2. If the FPL baseline analysis produces a value that is below that expected for implementing any reasonable SAMA, no matter how inexpensive, the remaining steps of the SAMA analysis are not necessary.

### 7.3.2 THE AP1000 SAMDA ANALYSIS

The Westinghouse SAMDA analysis is presented in Appendix 1B of the DCD. Westinghouse compiled a list of potential SAMDAs based on the AP600 analysis and other plant designs and suggestions from the AP600/AP1000 design staff. Some SAMDAs were then screened out based on their inapplicability to the AP1000 or the fact that they were already included in the AP1000 design. Rough implementation costs that far exceeded any reasonable benefit were also excluded. The 13 SAMDAs that passed the screening process are as follows and are described more fully in the DCD.

- Chemical volume and control system upgrade to mitigate small loss-of-coolant accidents
- Filtered containment vent
- Self-actuating containment isolation valves
- Passive containment spray
- Steam generator shell-side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank
- Increased steam generator secondary side pressure capacity
- Secondary containment filtered ventilation

- Diverse in-containment refueling water storage tank injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High-pressure containment design
- Improved reliability of diverse actuation system

These remaining SAMDAs were quantified by the probabilistic risk assessment model to determine the reduction in risk for implementing the SAMDA. Each SAMDA was assumed to reduce the risk of the accident sequences that they address to zero, a conservative assumption. Using the cost-benefit methodology of NUREG/BR-0184, the maximum averted cost risk was calculated for each SAMDA. The maximum averted cost risk calculation used the dose-risks and cost-risks calculated for the severe accidents described in Subsection 7.2.1. Westinghouse calculated the base case maximum averted cost risk to be $21,000 (2007 dollars) using a 7 percent discount rate.

Westinghouse next compared the implementation costs for each SAMDA to the $21,000 value and found that none of the SAMDAs would be cost-effective. The least costly SAMDA, self-actuating containment isolation valves, had an implementation cost of approximately $30,000, with the others having costs at least an order of magnitude greater. The one potential SAMDA was further evaluated but not found to be cost-effective.

In its *Finding of No Significant Impact* relating to the certification of the AP1000 design, the NRC (U.S. NRC Jan 2005) concluded, "none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that any other design changes would be justified in the future on the basis of person-rem exposure because the estimated core damage frequencies are very low on an absolute scale."

Pursuant to 10 CFR 51.55(b), it was confirmed that the design changes that are incorporated into the referenced DCD, as defined in Section 1.1, did not change the SAMDA screening or evaluation results or conclusions. Specifically, the SAMDAs assessed as being rejected for the certified AP1000 design, as documented in DCD Revision 19, Appendix 1B, have not become cost-beneficial for Units 6 & 7, nor have any new SAMDAs been identified for Units 6 & 7.

### 7.3.3 MONETIZATION OF THE UNITS 6 & 7 BASE CASE

The principal inputs to the calculations are the core damage frequency (reported in Section 7.2), dose-risk and cost-risk (reported in Table 7.2-1), dollars per person-rem ($2000 as provided by NRC in NUREG/BR-0184), plant operating life (60 years), and economic discount rate (7 percent and 3 percent are NRC precedents). Both the Westinghouse and FPL severe accident analyses described in Section 7.2 calculate risks from internal events. For this SAMDA analysis, the base-case core damage frequency, dose-risk, and cost-risk for internal events were escalated to account for external events, both at power and at shutdown. As explained in the DCD, dose-risk and cost-risk were scaled up by the ratio of the total (internal and external events) frequency divided by the internal events frequency (5.0E-07/2.4E-07 per reactor year). With these inputs, the monetized value of reducing the base case core damage frequency to zero is presented in Table 7.3-1. The monetized value, known as the maximum averted cost-risk, is conservative because no SAMA can reduce the core damage frequency to zero.

The maximum averted cost risk of $55,513 for a single proposed AP1000 at Turkey Point is so low that FPL does not believe there are any design changes, over those already incorporated into the advanced reactor design, that could be determined to be cost-effective. With a 3 percent discount rate, the valuation of the averted risk is $123,602. The least costly SAMDA, the self-actuating containment isolation valves, had an implementation cost of approximately $30,000. The maximum averted cost risk of $55,513 is the total cost risk benefit from the implementation of every SAMDA, and the benefit from implementation of the least costly SAMDA is only a portion of the total (maximum) cost risk benefit. The cost risk benefit from the implementation of the least costly SAMDA is only $994. Each of the remaining SAMDA implementation costs are much greater than the maximum averted cost risk of $55,513.

As demonstrated in WEC 2011, and confirmed for Turkey Point, the benefit of any SAMDA is much less than its implementation cost. The Turkey Point analysis resulted in slightly higher values than the Westinghouse generic analysis results of $21,000 for the 7 percent discount rate and $43,000 for the 3 percent discount rate. This is a result of the larger population and higher property values surrounding the Units 6 & 7 site.

Accordingly, further evaluation of design-related SAMAs is not warranted. FPL does not believe that administrative SAMAs, such as those relating to procedures or training, are appropriate for evaluation at this time because the procedures and training have not been developed. The purpose of this analysis is to demonstrate that the maximum averted cost risk for an AP1000 at the Units 6 & 7 site are not cost-beneficial. Evaluation of administrative SAMAs would not be appropriate until a plant design is finalized and plant administrative processes and procedures are being developed. At that time, appropriate administrative controls on plant operations would be incorporated into the plants' management systems as part of the baseline.

### Section 7.3 References

U.S. NRC Jan 2005. *Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the AP1000 Standard Plant Design*. Docket No. 52-006, SECY 05-0227 (accession number ML053630176). Washington D.C., January 24, 2005.

WEC (Westinghouse Electric Corporation) 2011. *Design Control Document*, Revision 19, Appendix 1B, "Severe Accident Mitigation Design Alternatives," June 2011.

**Table 7.3-1**
**Monetization of the Turkey Point AP1000 Base Case (2012 Dollars)**

| | 7% Discount Rate | 3% Discount Rate |
|---|---|---|
| Offsite exposure cost | 15,821 | 31,283 |
| Offsite economic cost | 18,859 | 37,289 |
| Onsite exposure cost | 253 | 582 |
| Onsite cleanup cost | 7,711 | 18,317 |
| Replacement power cost | 12,869 | 36,131 |
| **Total** | **55,513** | **123,602** |

## 7.4 TRANSPORTATION ACCIDENTS

Subsection 5.7.2.1 addresses the conditions in subparagraphs 10 CFR 51.52(a)(1) through (5) regarding use of Table S-4 to characterize the impacts of radioactive materials transportation in this environmental report. Because the AP1000 does not meet all of the conditions set forth in 10 CFR 51.52(a), a further analysis of the transportation effects was required. Subsection 5.7.2.2 describes the methodology used to analyze the impacts of transporting radioactive materials and addresses the incident-free transport of radioactive materials to and from Units 6 & 7. Subsection 7.4.1 describes the radiological impacts of transportation accidents. The nonradiological impacts of transportation accidents are addressed in Subsection 7.4.2.

### 7.4.1 RADIOLOGICAL IMPACTS OF TRANSPORTATION ACCIDENTS

#### 7.4.1.1 Transporting 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 transporting 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 form, cladding, and packaging for the AP1000 are similar to those of current generation light water reactors (LWRs), the consequences of accidents that are severe enough to result in a release of radioactivity to the environment would also be similar. Accordingly, the risks of accidents during transporting unirradiated fuel to Units 6 & 7 would be expected to be smaller than the reference LWR consequences listed in Table S-4.

#### 7.4.1.2 Transporting 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 AP1000 spent fuel after 5 years of decay was estimated using the ORIGEN code (Version 2.1). 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 (NUREG-1811). The spent fuel inventory used in this analysis for the AP1000 is presented in Table 7.4-1. The specific quantities and characteristics of the crud deposited on AP1000 spent fuel from corrosion products generated elsewhere in the reactor coolant system are unknown at this time because of insufficient operating experience. The spent fuel transportation accident risks were calculated assuming the entire Co-60 inventory (Table 7.4-1) is in the form of crud. Assuming a minimum decay period of 5 years, the expected Co-60 activity is approximately 4.09 Ci/metric tons uranium (MTU). Sb-125 was also included in the crud analysis. However, the total activity of Sb-125 reported as crud was less than 0.003 percent of the total Sb-125 inventory in the fuel. These crud values were included as a separate group in the RADTRAN 5 calculations. The total activity of the crud components is roughly five orders of magnitude lower than the fission and activation products of the fuel. Therefore, from a radiological dose standpoint, the crud contribution is negligible.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance features required by 10 CFR Part 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). The analysis presented in this ER assumed that shipping casks for AP1000 spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo, in accordance with the requirements of 10 CFR Part 71.

For the spent fuel from the AP1000, the RADTRAN 5 accident risk calculations were performed using an assumption of 0.5 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 spent fuel shipments from the AP1000. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 23 MTU per year for the AP1000. (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 four possible exposure pathways:

- External dose from exposure to the passing cloud of radioactive material.

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1. Requirements for Type B packaging are set forth in 49 CFR 173.413 and 10 CFR 71.41 through 51.

- 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.

- 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).

External doses from increased radiation fields surrounding a shipping cask with damaged shielding were also 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 Units 6 & 7 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 transporting spent fuel from Units 6 & 7 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 (St. Lucie, Glades, Martin, and Okeechobee 2) 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 1E+06 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 quite 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 transporting spent fuel from Units 6 & 7 or an alternate site would be consistent with the environmental impacts presented in Table S-4.

#### 7.4.1.3 Transporting Radioactive Waste

As shown in Table 5.7-4, transporting radioactive waste meets the applicable conditions in 10 CFR 51.52(a) and no further analysis is required.

### 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 Units 6 & 7 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 Transporting Unirradiated Fuel

The nonradiological accident impacts that could result from shipments of unirradiated fuel to Units 6 & 7 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 (1999) were used to estimate the annual impacts. Consistent with the incident-free transportation analysis described in Subsection 5.7.2, an average round-trip shipping distance of 4000 miles was used to evaluate the unirradiated fuel shipments. The differences between the reference LWR and AP1000 results are because of the lower number of shipments per year (when normalized for electrical output) projected for the AP1000 units at Units 6 & 7. The values presented in Table 7.4-3 would be doubled for a two-unit plant.

#### 7.4.2.2 Transporting Spent Fuel

The general approach to calculating the nonradiological impacts for spent fuel shipments is similar to that for other radioactive materials shipments. The primary difference is the spent fuel shipping route characteristics and are better defined allowing the state-specific accident statistics in Saricks and Tompkins (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 description of the TRAGIS routing model). The shipping distances were doubled to allow for return shipments of empty containers to Units 6 & 7. 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 Transporting 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 (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 AP1000 are because of the lower number of radioactive waste shipments projected for the AP1000. The values presented in Table 7.4-5 would be doubled for a two-unit plant.

## 7.4.3 CONCLUSION

The transportation accident risk results for the AP1000 for unirradiated and spent fuel and radioactive waste are less than the nonradiological effects of accidents in transportation (one fatal injury in 100 reactor years and one nonfatal injury per ten reactor years) indicated in Table S-4. Based on this analysis, the overall transportation accident risks associated with unirradiated fuel, spent fuel, and radioactive waste shipments from the proposed AP1000 units at Units 6 & 7 are consistent with the risks associated with transporting the radioactive materials from current generation reactors presented in Table S-4 of 10 CFR 51.52 (reproduced in Table 5.7-2) and thus would be SMALL.

**Section 7.4 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, J.L., Ammerman, Breivik, N.L., Dukart, R.J., Kanipe, F.L., Koski, J.A., Mills, G.S., Neuhauser, K.S., Radloff, H.D., Weiner, R.F., and Yoshimura, H.R. *Reexamination of Spent Fuel*

*Shipment Risk Estimates*, NUREG/CR-6672, Volume 1, Office of Nuclear Material Safety and Safeguards, U.S. NRC, Washington, D.C., March 2000.

**Table 7.4-1**
**Radionuclide Inventory Used in Transportation Accident Risk Calculations for One AP1000**

| Radionuclide | AP1000 Inventory (curies per MTU) |
|---|---|
| Am-241 | 7.27E+02 |
| Am-242m | 1.31E+01 |
| Am-243 | 3.34E+01 |
| Ce-144 | 8.87E+03 |
| Cm-242 | 2.83E+01 |
| Cm-243 | 3.07E+01 |
| Cm-244 | 7.75E+03 |
| Cm-245 | 1.21E+00 |
| Co-60 | 4.09E+00 (all as crud) |
| Cs-134 | 4.80E+04 |
| Cs-137 | 9.31E+04 |
| Eu-154 | 9.13E+03 |
| Eu-155 | 4.62E+03 |
| Pm-147 | 1.76E+04 |
| Pu-238 | 6.07E+03 |
| Pu-239 | 2.55E+02 |
| Pu-240 | 5.43E+02 |
| Pu-241 | 6.96E+04 |
| Pu-242 | 1.82E+00 |
| Ru-106 | 1.55E+04 |
| Sb-125 | 1.12E-01 (as crud) |
| Sr-90 | 6.19E+04 |
| Y-90 | 6.19E+04 |

## Table 7.4-2
## Spent Fuel Transportation Accident Risks for One AP1000

| Site | Unit Population Dose (person-rem per MTU)[a] | MTU per Reference Reactor Year | Population Dose (person-rem per reference reactor year)[a] | Total Detrimental Health Effects per Reference Reactor Year |
|---|---|---|---|---|
| Turkey Point | 1.72E-06 | 22 | 3.75E-05 | 2.74E-08 |
| St. Lucie | 1.48E-06 | 22 | 3.22E-05 | 2.35E-08 |
| Glades | 1.46E-06 | 22 | 3.17E-05 | 2.31E-08 |
| Martin | 1.47E-06 | 22 | 3.20E-05 | 2.34E-08 |
| Okeechobee 2 | 1.47E-06 | 22 | 3.20E-05 | 2.34E-08 |

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

## Table 7.4-3
## Nonradiological Impacts of Transporting Unirradiated Fuel for One AP1000

| 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 | 1.01E+06 | 3.7E-04 | 7.8E-03 | 1.1E-02 |
| AP1000 | 176 | 2000 | 7.88E+05 | 2.9E-04 | 6.1E-03 | 9.0E-03 |

**Table 7.4-4**
**Nonradiological Impacts of Transporting Spent Fuel for One AP1000 from Turkey Point to Yucca Mountain**

| State | Highway Type | One-Way Shipping Distance (miles) | Fatalities per Year | Injuries per Year | Accidents per Year |
|---|---|---|---|---|---|
| Alabama | Primary | 7 | 4.0E-05 | 3.0E-04 | 5.0E-04 |
| | Interstate | 73 | 8.9E-05 | 1.5E-03 | 2.9E-03 |
| Arizona | Interstate | 357 | 4.8E-04 | 5.9E-03 | 6.7E-03 |
| California | Interstate | 265 | 2.6E-04 | 4.7E-03 | 6.0E-03 |
| Florida | Primary | 37 | 5.6E-05 | 3.0E-04 | 4.0E-04 |
| | Interstate | 714 | 7.8E-04 | 5.6E-03 | 7.0E-03 |
| Louisiana | Interstate | 372 | 4.9E-04 | 9.7E-03 | 1.16E-02 |
| Mississippi | Interstate | 77 | 2.7E-05 | 4.0E-04 | 5.0E-04 |
| Nevada | Primary | 79 | 1.9E-04 | 2.8E-03 | 4.3E-03 |
| | Interstate | 61 | 5.7E-05 | 1.3E-03 | 1.9E-03 |
| New Mexico | Interstate | 371 | 6.2E-04 | 6.0E-03 | 5.9E-03 |
| Oklahoma | Interstate | 278 | 5.2E-04 | 1.14E-02 | 1.06E-02 |
| Texas | Interstate | 423 | 7.8E-04 | 3.28E-02 | 3.59E-02 |
| Totals | | 3115 | 4.4E-03 | 8.27E-02 | 9.43E-02 |

**Table 7.4-5**
**Nonradiological Impacts of Transporting Radioactive Waste for One AP1000**

| 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 | 6.8E-04 | 1.4E-02 | 2.1E-02 |
| AP1000 | 24 | 500 | 3.3E-04 | 7.0E-03 | 1.0E-02 |