Chapter 7 Environmental Impacts of Postulated Accidents Involving Radioactive Materials

This chapter assesses the environmental impacts of postulated accidents involving radioactive materials at Fermi 3. The chapter is divided into the following four sections that address design basis accidents, severe accidents, severe accident mitigation design alternatives, and transportation accidents:

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

7.1 Design Basis Accidents

The purpose of this section is to assess the environmental risks of accidents involving radioactive material. The scope of this section is limited to a comparison of the offsite dose consequences and resulting health effects for design basis accidents (DBAs) as calculated by Detroit Edison and those contained in DCD Chapter 15 (Reference 7.1-1).

7.1.1 Selection of Accidents

The radiological consequences of accidents are assessed to demonstrate that a new unit could be constructed and operated at the Fermi site without undue risk to the health and safety of the public. The assessment uses site-specific accident meteorology with radiological analyses in DCD Chapter 15 (Reference 7.1-1). 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 the ESBWR design. From Reference 7.1-1, the following DBAs are evaluated for the ESBWR:

- Feedwater Line Break Accident
- Failure of Small Line Carrying Primary Coolant Outside Containment
- Main Steamline Break Accident (MSLBA)
- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident
- RWCU/SDC Line Break Accident
- Control Rod Drop Accident
- Spent Fuel Cask Drop Accident

As discussed in DCD Sections 15.4.6 and 15.4.10, radiological consequence analyses are not required for the control rod drop accident and the spent fuel cask drop accident.

7.1.2 **Evaluation Methodology**

Doses for the representative DBAs are evaluated at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). These doses must meet the site acceptance criteria in 10 CFR 50.34 and 10 CFR 100. Although the analysis of engineered safety features demonstrate that these systems prevent core damage and mitigate releases of radioactivity, the LOCA dose analysis presumes substantial core melt with the release of significant amounts of fission products. The postulated DBA LOCA is expected to more closely approach 10 CFR 50.34 limits than the other DBAs of greater probability of occurrence but lesser magnitude of activity releases. For the accidents evaluated herein, the calculated doses are compared to the acceptance criteria in Regulatory Guide 1.183 and NUREG-0800, to demonstrate that the consequences of the postulated accidents are acceptable.

The evaluations discussed herein use short-term accident atmospheric dispersion factors (X/Q). The X/Qs are calculated using the computer code PAVAN, Version 2.0, following the methodology in Regulatory Guide 1.145 and using site-specific meteorological data. Consistent with NUREG-1555, Section 7.1.III.(2), X/Qs used for this assessment should either be the "50th percentile X/Q value that was based on onsite meteorological data, or 10 percent of the levels given in Regulatory Guide 1.3 or Regulatory Guide 1.4, to represent more realistic dispersion conditions than assumed in the safety evaluation." Determination of the 50th percentile X/Q values is discusssed in Subsection 2.7.6.1. For the Fermi site, the 50th percentile X/Qs are provided in Table 7.1-1.

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 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 (Reference 7.1-2), while the DDE and the EDE are based on dose conversion factors in Federal Guidance Report 12 (Reference 7.1-3).

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 DCD safety analyses reports. The DCD source term, methodologies, and assumptions are based on the alternative source term methods outlined in Regulatory Guide 1.183. The activity releases and doses are based on a power level of 4590 MWt, which represents a core thermal power of 4500 MWt multiplied by an uncertainty factor of 1.02. DBA source terms have been updated and are presented as isotopic activity releases to the environment in the unit of megabecquerel (MBq) in DCD Section 15.4, DCD Tables 15.4-3a, 15.4-7, 15.4-12, 15.4-15, 15.4-18a, 15.4-18b, and 15.4-22.

7.1.4 **Radiological Consequences**

The Fermi 3 specific doses are calculated based on the doses in Reference 7.1-1. For each DBA, the Fermi 3 specific dose is calculated by multiplying the DCD dose (provided in DCD Section 15.4) by the ratio of the Fermi 3 site-specific X/Q value to the associated DCD X/Q value from DCD

Section 15.4. The Fermi 3 site-specific X/Q values are the time-dependent X/Q values in Table 7.1-1. Reference 7.1-1 does not provide time-dependent LPZ doses; thus, the Fermi 3 LPZ dose is determined by multiplying the total dose from Reference 7.1-1 by the maximum X/Q ratio. The resulting X/Q ratios are shown in Table 7.1-2.

Because the Fermi 3 site-specific X/Q values are bounded by the DCD X/Q values, the Fermi 3 site-specific doses are within those calculated in DCD Section 15.4, and, in turn, within regulatory limits. The DBA doses summarized in Table 7.1-3 are based on individual accident doses presented in Table 7.1-4 through Table 7.1-13. For each DBA, the EAB dose shown is for the two-hour period that yields the maximum dose, in accordance with Regulatory Guide 1.183.

The Fermi 3 specific doses summarized in Table 7.1-3 are within the acceptance criteria of Regulatory Guide 1.183 and NUREG-0800. Thus, the potential environmental impacts of DBAs are SMALL. Refer to Section 5.4 for the impacts to the public from anticipated releases during normal operation.

7.1.5 **References**

7.1-1 GE-Hitachi Nuclear Energy, “ESBWR Design Control Document – Tier 2,” Revision 6, August 2009.

7.1-2 U.S. Environmental Protection Agency, Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion,” EPA-520/1-88-020, 1988.

7.1-3 U.S. Environmental Protection Agency, Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water and Soil,” EPA-402-R-93-081, 1993.

Table 7.1-1 Maximum 50th percentile χ/Q Values

Location		χ/Q (sec/m^3)
EAB		5.779E-05
LPZ	0-8 hr	3.046E-06
LPZ	8-24 hr	2.654E-06
LPZ	24-96 hr	1.969E-06
LPZ	96-720 hr	1.282E-06

Table 7.1-2 Determination of Χ/Q Ratios

Accident	Location		ESBWR DCD Χ/Q[1]	Fermi 3 50th % Χ/Q	Ratio (Fermi 3/DCD)
Feedwater Line Break (Pre-Incident Iodine Spike & Equilibrium Iodine Spike)	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ		1.90E-04	3.046E-06	1.60E-02
Failure of Small Line Carrying Primary Coolant Outside Containment (Pre-Incident Iodine Spike & Equilibrium Iodine Spike)	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ	0-8 hr	1.90E-04	3.046E-06	1.60E-02
	LPZ	8-24 hr	1.40E-04	2.654E-06	1.90E-02
	LPZ	24-96 hr	7.50E-05	1.969E-06	2.63E-02
	LPZ	96-720 hr	3.00E-05	1.282E-06	4.27E-02
MSLB (Pre-Incident Iodine Spike & Equilibrium Iodine Spike)	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ		1.90E-04	3.046E-06	1.60E-02
LOCA	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ	0-8 hr	1.90E-04	3.046E-06	1.60E-02
	LPZ	8-24 hr	1.40E-04	2.654E-06	1.90E-02
	LPZ	24-96 hr	7.50E-05	1.969E-06	2.63E-02
	LPZ	96-720 hr	3.00E-05	1.282E-06	4.27E-02
Fuel Handling	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ		1.90E-04	3.046E-06	1.60E-02
RWCU/SDC (Pre-Incident Iodine Spike & Equilibrium Iodine Spike)	EAB		2.00E-03	5.779E-05	2.89E-02
	LPZ		1.90E-04	3.046E-06	1.60E-02

1. DCD Χ/Q values are taken from Reference 7.1-1, Section 15.4.

Table 7.1-3 Summary of Design Bases Accident Doses

Accident	Location	TEDE (rem)	Limit (rem)[1]
FWLB - Pre-Incident Iodine Spike	EAB	5.23E-01	25
	LPZ	2.73E-02	25
FWLB - Equilibrium Iodine Spike	EAB	3.18E-02	2.5
	LPZ	1.60E-03	2.5
SBOC - Pre-Incident Iodine Spike	EAB	9.82E-03	25
	LPZ	4.27E-03	25
SBOC - Equilibrium Iodine Spike	EAB	2.89E-03	2.5
	LPZ	4.27E-03	2.5
MSLB - Pre-Incident Iodine Spike	EAB	7.51E-02	25
	LPZ	3.21E-03	25
MSLB - Equilibrium Iodine Spike	EAB	5.78E-03	2.5
	LPZ	1.60E-03	2.5
LOCA	EAB	6.47E-01	25
	LPZ	8.85E-01	25
FHA	EAB	1.18E-01	6.3
	LPZ	6.41E-03	6.3
RWCU/SDC Line Break - Pre-Incident Iodine Spike	EAB	1.99E-01	25
	LPZ	1.12E-02	25
RWCU/SDC Line Break - Equilibrium Iodine Spike	EAB	1.16E-02	2.5
	LPZ	1.60E-03	2.5
Control Rod Drop	Evaluation of radiological consequences not required		
Spent Fuel Cask Drop	Evaluation of radiological consequences not required		

1. Radiological limits are taken from Regulatory Guide 1.183 and NUREG-0800.

Table 7.1-4 Feedwater Line Break Pre-Incident Iodine Spike (DCD Doses are from DCD Table 15.4-16)

	DCD TEDE (rem)	χ/Q Ratio	Fermi 3 TEDE (rem)
EAB	18.1	2.89E-02	5.23E-01
LPZ	1.7	1.60E-02	2.73E-02
Limit			25

Table 7.1-5 Feedwater Line Break Equilibrium Iodine Spike (DCD Doses are from DCD Table 15.4-16)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	1.10	2.89E-02	3.18E-02
LPZ	0.10	1.60E-02	1.60E-03
Limit			2.5

Table 7.1-6 Small Line Carrying Primary Coolant Outside Containment Pre-Incident Iodine Spike (DCD Doses are from DCD Table 15.4-19)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	0.34	2.89E-02	9.82E-03
LPZ	0.10	4.27E-02	4.27E-03
Limit			25

Reference 7.1-1 does not provide time-dependent LPZ doses for this incident; thus, the site LPZ dose is determined by multiplying the total DCD dose by the maximum χ/Q Ratio for the LPZ.

Table 7.1-7 Small Line Carrying Primary Coolant Outside Containment Equilibrium Iodine Spike (DCD Doses are from DCD Table 15.4-19)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	0.10	2.89E-02	2.89E-03
LPZ	0.10	4.27E-02	4.27E-03
Limit			2.5

Reference 7.1-1 does not provide time-dependent LPZ doses for this incident; thus, the site LPZ dose is determined by multiplying the total DCD dose by the maximum χ/Q Ratio for the LPZ.

Table 7.1-8 Main Steam Line Break Pre-Incident Iodine Spike (DCD Doses are from DCD Table 15.4-13)

	DCD TEDE (rem)	X/Q Ratio	Unit 3 TEDE (rem)
EAB	2.6	2.89E-02	7.51E-02
LPZ	0.2	1.60E-02	3.21E-03
Limit			25

Table 7.1-9 Main Steam Line Break Equilibrium Iodine Spike (DCD Doses are from DCD Table 15.4-13)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	0.2	2.89E-02	5.78E-03
LPZ	0.1	1.60E-02	1.60E-03
Limit			2.5

Table 7.1-10 Loss of Coolant Accident (DCD Doses are from DCD Table 15.4-9)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	22.4	2.89E-02	6.47E-01
LPZ	20.7	4.27E-02	8.85E-01
Limit			25

DCD does not provide time-dependent LPZ doses for this incident; thus, the site LPZ dose is determined by multiplying the total DCD dose by the maximum χ/Q Ratio for the LPZ.

Table 7.1-11 Fuel Handling Accident (Reactor Building or Fuel Building) (DCD Doses are from DCD Table 15.4-4)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	4.10	2.89E-02	1.18E-01
LPZ	0.40	1.60E-02	6.41E-03
Limit			6.3

Table 7.1-12 RWCU/SDC Line Break Pre-Incident Iodine Spike (DCD Doses are from DCD Table 15.4-23)

	DCD TEDE (rem)	X/Q Ratio	Fermi 3 TEDE (rem)
EAB	6.9	2.89E-02	1.99E-01
LPZ	0.7	1.60E-02	1.12E-02
Limit			25

Table 7.1-13 RWCU/SDC Line Break Equilibrium Iodine Spike (DCD Doses are from DCD Table 15.4-23)

	DCD TEDE (rem)	χ/Q Ratio	Unit 3 TEDE (rem)
EAB	0.40	2.89E-02	1.16E-02
LPZ	0.10	1.60E-02	1.60E-03
Limit			2.5

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, the Nuclear Regulatory Commission (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.

General Electric (GE) completed a PRA for the ESBWR design (Reference 7.2-3) as part of the application for design certification. The GE analysis used generic, but conservative, meteorology and regional characteristics and determined that severe accident impacts are within the safety goals established by the NRC.

In this section, Detroit Edison presents an update of the generic PRA analysis, which includes Fermi site-specific characteristics. The analysis evaluates the impacts of a severe accident at Fermi 3 to demonstrate that the impacts are bounded in the generic analysis performed for the ESBWR certification and to support performing the severe accident mitigation alternatives analyses in Section 7.3.

7.2.1 GE Methodology

The GE PRA for the ESBWR established a containment event tree which defined the possible end states of the containment following a severe accident. Using EPRI's Modular Accident Analysis Program (MAAP) code, GE 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. A release frequency was assigned to each of the 10 release categories (Table 7.2-1). For the dual source term release categories, GE assigned the entire frequency of the release category to each of the source terms.

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

1. Break Outside of Containment (BOC) – Radioactivity is released through an unisolated break outside of containment in the shutdown cooling piping allowing direct communication between the reactor pressure vessel and the environment outside of 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 (BOCa) and the other at the lower-level (BOCb).
2. Containment Bypass (BYP) – Radioactivity is released directly to the atmosphere from 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 (BYPa) and a high pressure sequence (BYPb) were selected for determining the source term categories for this release category.

3. 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 (CCIDa) and a high pressure sequence (CCIDb) were selected for determining the source term in this release category.
4. 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 of scrubbing for 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 (CCIWa) and a high pressure sequence (CCIWb) were selected for determining the source term categories associated with this release category.
5. 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.
6. 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.
7. 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 pre-existing condition), resulting in failure of the containment pressure function, which in turn causes failure in containment heat removal. Both high (OPVBa) and low pressure sequences (OPVBb) were selected for source term categories.
8. 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.
9. 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.

10. 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. GE 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.

GE then used the MACCS2 code (MELCOR Accident Consequence Code System) (Reference 7.2-9) 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 long-term exposure pathways such as ingestion, inhalation of re-suspended material, or groundshine subsequent to plume passage. GE also considered the releases for the first 72 hours after core damage. Additional details of analysis are found in the ESBWR PRA (Reference 7.2-3) and are reported in the ESBWR Design Control Document (Reference 7.2-4).

7.2.2 Site Specific Methodology

For Fermi 3, the MACCS2 computer code was used to evaluate offsite risks and consequences of severe accidents, using Fermi 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 water ingestion dose from surface runoff and deposition on surface-water. The MACCS2 code also evaluates the extent of land contamination. For Fermi 3, the analysis used site-specific meteorology and population data (Subsection 2.5.1) 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, the analysis determined the collective dose to the 50-mi region 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: ATMOS, EARLY, 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 GE's generic PRA.

The 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. The exposure model assumed that 95% of the 0-10 mile residents would evacuate. The evacuation time for 2060 was estimated in two parts.

- An 80-minute delay was assumed before evacuation begins after declaration of a General Emergency.
- The evacuation rate was estimated by escalating the 2008 population and reducing the evacuation speed proportionally. This is conservative because it assumes that the existing evacuation routes are saturated in the 2008 evacuation time estimate, and any increases in population growth will result in reduced evacuation speeds. This resulted in an estimated outward speed of 1.12 meters per second.

Exposures to the plume were assumed to terminate when the population were 10 miles from the release point.

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 Fermi meteorological monitoring data and a site characteristics file which is built using SECPOP2000 (Reference 7.2-5).

Seven years of site specific meteorological data (2001 through 2007) were evaluated. MACCS2 requires a calendar year of meteorological data for the MET file. The year 2002 meteorological data was selected for subsequent analysis because it resulted in the greatest population dose and cost risk, and also was the most complete yearly data set. In addition, sensitivities were performed for the other six years of meteorological data.

The SITE file requires the 50-mi population distribution as well as agricultural-economic data. SECPOP2000 (Reference 7.2-5) incorporates 2000 population census data for the 50-mi region around the Fermi site. For this analysis, the 50-mile population projected to the year 2060, as described in Subsection 2.5.1, was used. 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 non-farm land) in the same manner as the population. This was done by applying the SECPOP2000 program, changing the regional economic data format to comply with MACCS2 input requirements. In this case, SECPOP2000 was used to access data from the 1997 National Census of Agriculture. The version 3.12.01 data file accessed by SECPOP2000, County97.dat, was modified to correct two errors (generally known as the missing notes parameter error and the missing county numbers error) in the issued version. The program's specification of crop production parameters for the 50-mi region (e.g., fraction of farmland devoted to grains, vegetables, etc.) was also applied.

The analysis used the resulting MACCS2 calculations and release 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 release frequency. Because the ESBWR's 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.

The base case analysis assumed a ground level release height and no release heat for each hypothesized accident release. A sensitivity analysis was performed for these and other modeling assumptions. A middle of containment and a top of containment release was compared to the ground level release and the dose-risk increased by 0.7 percent and 2.6 percent respectively. The cost-risk for the middle of containment and top of containment release had increases of 1.6 percent and 5.0 percent respectively. A release heat of 1 MW and 10 MW was compared to the base case of no release heat and the dose-risk increased by 0.5 percent and 2.3 percent respectively, while the cost-risk increased by 0.9 percent and 4.9 percent respectively. A sensitivity analysis was performed on the precipitation input where the site specific precipitation rate was doubled and halved. The doubled precipitation resulted in a decrease in both the doserisk and cost-risk of 0.9 percent and 0.7 percent respectively. The halved precipitation resulted in increases in the dose-risk of 0.1 percent and 0.3 percent in the cost-risk. In addition, a sensitivity analysis was performed on the conservative assumption that the final 40- to 50-mi ring has constant meteorology, including constant precipitation. The base case hourly precipitation in the 40- to 50-mi ring was set equal to the 12- hour average site-specific precipitation, forcing additional deposition of the remaining airborne radioactivity which reaches this ring. Allowing the meteorology, including the precipitation, in this ring to follow the hourly site measured meteorology, resulted in a reduction in the dose-risk of 24 percent and a reduction in the cost-risk of 35 percent when compared to the base case.

The site specific analysis assumed a ground level release height and no release heat for each accident release hypothesized. The GE analysis used this same assumption to report impacts during the first 24-hours after the onset of core damage, but reported impacts during the first 72-hours after the onset of core damage from an elevated release with one megawatt of release heat. The latter combination of release height and heat would increase the 50-mi population and cost-risks by 3.5 percent and 6.2 percent, respectively. However, near site risks decrease. An example of the latter would be the 66 percent decrease in early fatality risk from the base case; note, however, the very small base case early fatality risk as given in Table 7.2-1.

For each release category with a dual source term, the site specific analysis conservatively attributes the entire release category frequency to the source term which results in the greater 50-mi population dose. The sensitivity of this assumption to the calculated risks was investigated by considering a case with the entire release category frequency attributed to each of the dual source terms, and the greater calculated risk impact between the two source terms reported for each risk metric. All base case risks were within 0.5 percent of the sensitivity case risk except water ingestion, which was within 1.5 percent.

7.2.3 Consequences to Population Groups

The pathway consequences to population groups including air pathways, surface-water, and groundwater pathways are discussed in the following sections. The presence of threatened and endangered species and federally designated critical habitat are discussed in Subsection 2.4.1 and Subsection 2.4.2. As necessary, the impacts on threatened and endangered species due to the previously calculated radiation exposure levels are discussed in Subsection 5.4.4.

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 farm land requiring decontamination. The analysis conservatively assumed that evacuation occurs during adverse weather conditions following declaration of a General Emergency. It was also conservatively assumed that the evacuation routes were already at full capacity at the time of the Reference 7.2-10 evacuation study. Therefore, the increased population expected in the year 2060 would take longer to evacuate. 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 Table 7.2-1.

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 mi of the site is drinkable. The MACCS2 code severe accident dose-risk to the 50-mi population from drinking the water is 1.3×10^{-3} 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-mi region of the Fermi site that are accessible to the public include Lake Erie, River Raisin, Huron River, Maumee River, Lake St. Clair, Detroit River, 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, including Fermi 2 (Reference 7.2-8). The aquatic food pathway dose for Fermi 2 was 1400 person-rem. Actual dose-risk values would be expected to be much less (by a factor of 2 to 10) due to interdiction of contaminated foods (Reference 7.2-8). Examination of the atmospheric dose-risk from severe accidents to the population within 50 mi of operating nuclear plants resulted in dose-risks ranging from 0.55 to 68 person-rem per reactor year for nuclear plants undergoing license renewal. The Fermi 3 atmospheric pathway dose of 0.032 person-rem per reactor year is significantly lower. 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 that reported above for the Fermi 2 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.

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.

NUREG-1437 evaluated the groundwater pathway dose, based on the analysis in NUREG-0440, the Liquid Pathway Generic Study (LPGS) (Reference 7.2-6). 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 LPGS results provide conservative, uninterdicted population dose estimates for six generic categories of plants. These dose estimates were one or more orders of magnitude less than those attributed to the atmospheric pathway. Therefore, although the Fermi site was not one of the reactors analyzed, the doses from the Fermi 3 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. It is noted that, as discussed in Subsection 2.3.1, the Fermi site is not over or near a sole source aquifer.

7.2.4 **Comparison to U.S. NRC Safety Goals**

The ESBWR PRA evaluates performance of the ESBWR under generic conditions to three safety goals: (1) individual risk goal, (2) societal risk goal, and (3) radiation risk goal (Reference 7.2-3). These goals are defined in the following subsections. Table 7.2-2 provides the quantitative evaluation of these three safety goals and the Fermi 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 U.S. 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. "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 U.S. accidental death risk value of 37.7 deaths per 100,000 people per year (Reference 7.2-2).

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 191.4 deaths per 100,000 people per year based upon National Center for Health Statistics data for 2001–2004 (Reference 7.2-2).

7.2.4.3 Radiation Dose Goal

The probability of an individual exceeding a whole body dose of 25 rem at a distance of 0.5 mile from the reactor shall be less than one in a million per reactor year.

7.2.5 Conclusions

The total calculated dose-risk to the 50-mi population from airborne releases from an ESBWR reactor at the Fermi site would be 0.032 person-rem per reactor year (Table 7.2-1). 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 (Reference 7.2-7).

Seventy-five percent of the Fermi 3 dose-risk is from late phase pathway exposures, especially groundshine and ingestion. The Fermi 3 early phase dose-risk, 0.0071 person-rem per reactor year, can be compared with the GEH generic calculation of 24-hour dose-risk (which does not include late phase exposure) of 0.017 person-rem per reactor year (Reference 7.2-3); GEH did not calcualte late phase consequences.

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 ESBWR 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 Fermi 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 ESBWR 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 Subsection 5.4.3, the whole body dose from the Fermi site normal airborne releases is predicted to be 22.2 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 22.5 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.032 person-rem per reactor year indicates that the dose risk from severe accidents is approximately 0.1 percent of the dose-risk from normal operations.

The probability-weighted risk of early and late cancer fatalities from a severe accident at the Fermi site in the surrounding 50-mile population projected for 2060 of 7.7 million is reported as 1.8 x 10^{-5} fatalities per reactor year in Table 7.2-1. For a 60-year reactor operating life, this population cancer fatality risk becomes 1.1 x 10^{-3}.

The probability of an individual dying from any cancer from any cause is approximately 0.23 for men and 0.20 for women over a lifetime (Reference 7.2-1). This implies that more than 1.5 x 10^{6} members of the 50-mile population will die of cancer.

The cancer fatality risk from a severe accident at Fermi 3 to the 50-mile population is then less than 10^{-7} percent of the background risk, which is much less than the societal risk goal of 0.1 percent of the background risk.

The results from the analysis discussed in this section are used in Section 7.3 to determine if there are any cost-beneficial design alternatives that should be considered to mitigate the impacts described herein.

7.2.6 References

7.2-1 American Cancer Society, “Lifetime Probability of Developing or Dying from Cancer,” http://www.cancer.org/docroot/CRI/content/CRI_2_6x_Lifetime_Probability_of_Developing_or_Dying_From_Cancer.asp, accessed 1 May 2008.

7.2-2 Centers for Disease Control, “Deaths: Final Data for 2004,” National Vital Statistics Reports, Volume 55 Number 19, August 21, 2007.

7.2-3 GE Energy, “ESBWR Probabilistic Risk Assessment,” NEDO-33201, Revision 4, June 2009.

7.2-4 GE-Hitachi Nuclear Energy, “ESBWR Design Control Document - Tier 2,” Revision 6, August 2009.

7.2-5 U.S. Nuclear Regulatory Commission, “SECPOP 2000: Sector Population Land Fraction, and Economic Estimation Program,” NUREG/CR-6525, August 2003.

7.2-6 U.S. Nuclear Regulatory Commission, “Liquid Pathway Generic Study: Impacts of Accidental Radioactive Releases to the Hydrosphere from Floating and Land-Based Nuclear Power Plants,” NUREG-0440, February 1978.

7.2-7 U.S. Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, June 1989.

7.2-8 U.S. Nuclear Regulatory Commission, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants,” NUREG-1437, Volumes 1 and 2, May 1996.

7.2-9 Chanin, D.I., and M.L. Young, “Code Manual for MACCS 2: User’s Guide,” NUREG/CR-6613, SAND97-0594, Volume 1, Sandia National Laboratories, Albuquerque, New Mexico, May 1998.

7.2-10 KLD Associates, Inc., “Fermi Nuclear Power Plant Development of Evacuation Time Estimates,” Revision 0, May 2008.

Table 7.2-1 Impacts to the Population and Land from Fermi 3 Severe Accidents Analysis

Accident Category[1]	Release Frequency (per reactor year)[2]	Population Dose-Risk (person-rem per reactor year)	Number of Fatalities (per reactor year)		Cost-Risk (dollars per reactor year)	Water Ingestion Dose (person-rem per reactor year)	Land Requiring Decontamination (acres per reactor year)
			Early	Late			
BOC	7.9E-11	2.6E-03	2.3E-09	1.8E-06	8.7E+00	1.5E-04	1.6E-05
BYP	5.7E-11	1.7E-03	5.4E-10	1.4E-06	3.5E+00	1.9E-05	1.0E-05
CCID	1.5E-12	3.2E-05	3.6E-12	2.0E-08	1.2E-01	1.6E-06	2.9E-07
CCIW	2.9E-12	2.6E-05	1.3E-13	1.5E-08	7.0E-02	3.9E-07	3.2E-07
EVE	1.1E-09	2.5E-02	3.4E-09	1.6E-05	9.2E+01	1.2E-03	2.2E-04
FR	9.2E-11	4.2E-04	1.5E-14	2.5E-07	4.6E-01	2.1E-06	3.2E-06
OPVB	2.1E-12	1.3E-05	2.6E-14	7.7E-09	2.9E-02	1.2E-07	1.5E-07
OPW1	2.0E-12	1.2E-05	7.6E-17	7.4E-09	3.0E-02	1.2E-07	1.5E-07
OPW2	8.5E-12	1.2E-05	0.0E+00	7.0E-09	2.1E-03	3.6E-08	1.7E-08
TSL	1.5E-08	2.2-03	0.0E+00	1.3E-06	4.9E-01	4.8E-06	4.1E-06
Total	1.7E-08	3.2E-02	6.2E-09	2.0E-05	1.1E+02	1.3E-03	2.6E-04

Notes:

1. Reference 7.2-3, Table 9-1
2. Reference 7.2-3, Table 10.3-3a

Table 7.2-2 Comparison of Fermi 3 Results to U.S. NRC Safety Goals

Year of Fermi Site Meteorological Data	Safety Risk: Prompt Fatality Risk (Individual 0-1 mi) (deaths per reactor year)	Safety Risk: Cancer Fatality Risk (0-10 mi cancers) (deaths per year per reactor year)	Probability of Exceeding 0.25 Sv (25 rem) at 0.5 mi (per reactor-year)
2001	4.46E-12	8.40E-14	1.20E-09
2002	4.07E-12	8.21E-14	1.14E-09
2003	4.82E-12	9.38E-14	1.23E-09
2004	4.05E-12	8.34E-14	1.11E-09
2005	4.30E-12	8.60E-14	1.09E-09
2006	4.31E-12	8.49E-14	1.11E-09
2007	4.47E-12	8.78E-14	1.14E-09
Safety Goal	3.77E-07[(2)]	1.90E-06[(2)]	<1.00E-06[(1)]
Generic ESBWR Analysis [1]	1.61E-10	2.56E-11	2.04E-09

Notes:
1. Reference 7.2-3, Table 10.4-2. Maximum At Power Internal case
2. Reference 7.2-2

7.3 Severe Accident Mitigation Alternatives

U.S. Environmental Protection Agency regulations require that a discussion on environmental consequences include mitigation measures (40 CFR 1502.16(h)). Mitigation measures should be considered even for impacts that would not be significant by themselves, if the overall proposed action could have significant impacts.

As described in Section 7.2, General Electric (GE) performed a probabilistic risk analysis (PRA) for the ESBWR as part of the design certification process (Reference 7.3-1). This analysis determined that severe accident impacts are within the safety goals established by the NRC. Detroit Edison extended the GE generic PRA to examine Detroit Edison's proposed ESBWR unit at the Fermi site and concluded that the generic analysis remains valid for the site. The analysis discussed in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented at Fermi 3 to mitigate the small impacts described in Section 7.2.

7.3.1 The SAMA Analysis Process

Design or procedural modifications that could mitigate the consequences of a severe accident are known as severe accident mitigation alternatives (SAMAs). In the past, SAMAs were known as SAMDAs, severe accident mitigation design alternatives, which primarily focused on design changes and did not consider procedural modifications. 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 defined by the dose-risk and cost-risk of a severe accident before implementation of any SAMAs. A plant's PRA is the primary source of data in calculating the base case. The base case risks are converted to a monetary value for subsequent use in screening SAMAs. Section 7.2 presents the base case dose- and cost-risk for a single ESBWR at the Fermi site.
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. Each potential SAMA in the list is assigned a conservatively low implementation cost based on historical costs for similar design changes and/or engineering judgment, and is then compared to the base case value from Step 1, above. SAMAs with higher implementation cost than the base case value are not evaluated further. SAMAs with a lower implementation cost than the base case screening value go to Step 3.
3. Determine the cost and net value of each SAMA – Each SAMA remaining after Step 2 receives 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 PRA model to determine the reduction in risk associated with implementation of the proposed SAMA. The reduction-in-risk

benefit is converted to a monetary value and is then compared to the detailed cost estimate. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

In the absence of a completed plant with established procedural controls, the analysis process is limited to demonstrating that the severe accident analysis using Fermi-specific parameters is bounded by the GE severe accident analysis and to determining what magnitude of plant-specific design or procedural modification would be cost-effective. Determining the magnitude of cost-effective design or procedural modifications is the same as defining the base case (Step 1) for existing nuclear units. 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 converted to a monetary value in accordance with methods established in NUREG/BR-0184 (Reference 7.3-3). 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 valuation would not be expected to pass the screening in Step 2. If the baseline analysis produces a value that is below that expected for implementation of any reasonable SAMA, no matter how inexpensive, then the remaining two steps of the SAMA process are not necessary.

7.3.2 The GE-Hitachi ESBWR SAMDA Analysis

The GE-Hitachi (GEH) SAMDA analysis was provided to the NRC in Reference 7.3-2. GEH compiled a list of potential SAMDAs based on the Advanced Boiling Water Reactor SAMA study and license renewal environmental reports. Some SAMDAs were then screened out based on their inapplicability to the ESBWR design or because they were already included in the ESBWR design. SAMDAs with implementation costs that far exceeded any reasonable benefit or had very low benefits were also excluded. None of the SAMDAs passed the screening process.

GEH compared the implementation costs for each SAMDA to the maximum severe accident risk reduction value possible and found that none of the SAMDAs would be cost-effective.

7.3.3 Monetary Valuation of the Fermi 3 Base Case

The principal inputs to the calculations are: core damage frequency, dose-risk and cost-risk, dollars per person-rem, licensing period, and economic discount rate.

- The core damage frequency, including both internal and external events, is 1.16×10^{-7} per year (Reference 7.3-1).
- The dose-risk and cost-risk are reported in Table 7.2-1.
- The calculations use $2000 per person-rem, provided in NUREG/BR–0184.
- The licensing period is assumed to be 60 years for the calculations, rather than the 40-year period in the Combined License (COL) application, to be consistent with the GEH analysis.

- The economic discount rate is assumed to be 7 percent, consistent with the GEH analysis. In addition, a sensitivity analysis is included using 3 percent. The NRC recommends using a 7 percent discount rate and performing a sensitivity analysis using 3 percent (Reference 7.3-3).

Using these inputs, the maximum monetary value associated with complete risk reduction is presented in Table 7.3-1. The monetary value (the maximum averted cost-risk) is conservative because no SAMA can reduce the core damage frequency to zero.

The maximum averted cost-risk of $22,804 for a single ESBWR at the Fermi site is sufficiently small that no design changes would be cost-effective to implement. This is consistent with the GEH analysis that demonstrates that cost-effective designs to mitigate severe accidents have already been incorporated into the design submitted for certification. Even with a conservative 3 percent discount rate, the valuation of the averted risk is only $49,557. GEH concluded (Reference 7.3-2) that, even for their upper bound estimate, none of the SAMDA candidates were cost beneficial.

A review was performed of the compilation of SAMAs in NEDO-33206 to identify procedural and administrative measures that were not considered design alternatives (Reference 7.3-2). Most of these items related to PWRs and have no relevance to the ESBWR. **[START: COM ER-7.3-002]** A SAMA analysis to comply with 40 CFR 1502.16(h) shall be conducted of the administrative and procedural measures applicable to Fermi 3 and considered for implementation prior to fuel load if the associated cost does not exceed the maximum value associated with averting all risk of severe accidents. **[END: COM ER-7.3-002]**

Accordingly, no cost-beneficial SAMDAs have been identified. Further, pursuant to 10 CFR 51.30(d), the NRC will, as part of its design certification rulemaking, prepare an environmental assessment evaluating the costs and benefits of SAMDAs for the ESBWR. Pursuant to 10 CFR 51.50(c)(2) and 51.75(c)(2), this environmental assessment may be incorporated by reference into the ER upon completion.

7.3.4 **References**

7.3-1 GE Energy, “ESBWR Probabilistic Risk Assessment,” NEDO-33201, Revision 4, June 2009.

7.3-2 GE-Hitachi Nuclear Energy Americas LLC, “ESBWR Severe Accident Mitigation Design Alternatives,” NEDO-33306, Revision 1, August 2007.

7.3-3 U.S. Nuclear Regulatory Commission, “Regulatory Analysis Technical Evaluation Handbook,” NUREG/BR-0184, January 1997.

Table 7.3-1 Valuation of the Detroit Edison ESBWR Base Case

	7% Discount Rate	3% Discount Rate
Offsite exposure cost	$6,289	$12,435
Offsite economic cost	$10,184	$20,137
Onsite exposure cost	$58	$133
Onsite cleanup cost	$1,761	$4,184
Replacement power cost	$4,512	$12,668
Total	$22,804	$49,557

7.4 Transportation Accidents

This section addresses the environmental impact of transportation accidents involving radioactive materials. The means of transportation of radioactive materials is discussed in Section 3.8.

7.4.1 Transportation of Unirradiated Fuel

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 CFR 51.52, which summarizes the environmental impacts of transportation of fuel and radioactive wastes to and from a reference reactor. Transportation of unirradiated fuel is addressed in Subsection 3.8.1, which demonstrates that the calculated dose for shipping unirradiated fuel to Fermi 3 for an ESBWR reactor is within the conditions shown in Table S-4 of 10 CFR 51.52.

7.4.2 Transportation of Spent Fuel

Subsection 3.8.2 evaluates the number and characteristics of shipments of irradiated fuel from Fermi 3 as compared to the conditions described in 10 CFR 51.52. Any conditions where Fermi 3 is not bounded by the values in 10 CFR 51.52 are identified and evaluated in Section 3.8. In addressing transportation of irradiated fuel, Subsection 3.8.2 concludes that the analyses and results in NUREG-1817 (Reference 7.4-1) bound the Fermi site.

In the analysis documented in Reference 7.4-1, 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 analysis in Reference 7.4-1 obtained the radionuclide inventories of LWR spent fuel after five years decay from Idaho National Engineering and Environmental Laboratory (INEEL) (Reference 7.4-2) and performed a screening analysis to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations. This screening identified the radionuclides that would contribute more than 99.999 percent of the dose from inhalation of radionuclides released following a transportation accident. The NRC found that the dominant radionuclides are similar regardless of the fuel type. The spent fuel radionuclide inventory used in the NRC analysis for the ESBWR is presented in Table 7.4-1.

Robust shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71. Spent fuel shipping casks must be certified as Type B packaging systems, meaning they must withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability. According to NUREG/CR-6672 (Reference 7.4-3), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). The NRC analysis assumed that shipping casks for advanced light water reactor (LWR) spent fuels would provide equivalent mechanical and thermal protection of the spent fuel cargo.

The RADTRAN 5 accident risk calculations documented in Reference 7.4-1 used unit radionuclide inventories (curies/metric ton of uranium [Ci/MTU]) for the spent fuel shipments for the advanced LWRs. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (MTU/yr) to derive estimates of the annual risks associated with spent fuel shipments from each potential advanced LWR. The amounts of spent fuel shipped per year were assumed to be equivalent to the annual discharge quantities: 32.76 MTU/yr for the ESBWR from Reference 7.4-2. The value normalized to the Reference LWR net electrical generation is 20.3 MTU/reference reactor year (Reference 7.4-1).

The analysis in Reference 7.4-1 used the release fractions for current generation LWR fuels to approximate the impacts from the advanced LWR spent fuel shipments. This assumes that the fuel materials and containment systems (i.e., cladding, fuel coatings) behave similarly to current LWR fuel under applied mechanical and thermal conditions.

As discussed in Reference 7.4-3, a bounding value for crud surface activity for boiling water reactor (BWR) fuel rods is 595 x 10^{-6} Ci/cm^2 (2.20 x 10^7 Bq/cm^2). This value is based on measurements taken from operating BWRs. Because ESBWR operational parameters are similar to operating BWRs, this bounding value is appropriate for the ESBWR. Furthermore, based on previous BWR operational experience, the ESBWR design incorporates provisions to minimize crud buildup, described in ESBWR DCD Section 5.2.3.2.2 for “Radiation Buildup”, which further justifies use of this bounding value. The crud surface activity used for the analysis in Reference 7.4-1 was 1.01 x 10^{14} Bq/MTU. Using ESBWR bounding fuel rod dimensions, uranium loading, and the 595 x 10^{-6} Ci/cm^2 (2.20 x 10^7 Bq/cm^2) bounding crud surface activity from NUREG/CR-6672, the ESBWR crud surface activity is calculated to be 1.48 x 10^{13} Bq/MTU, more than a factor of six less than that used in Reference 7.4-1. Therefore, the impacts of crud and activation products on spent fuel transportation accidents are enveloped by the analysis in Reference 7.4-1 and can be considered as SMALL.

Route-specific accident rates (accidents per km) were derived for the RADTRAN 5 accident risk analysis presented in Reference 7.4-1. In Reference 7.4-1, the approach used to develop accident rates for spent fuel shipments is as follows. The TRAGIS data (used in Reference 7.4-1) provide estimates of the distance traveled in each state along a route and the type of highway (interstate, state highway, or other). Reference 7.4-4 provided accident rates for each state that are a function of highway type. The approach taken to estimate route-specific accident rates was to multiply the state-level accident or fatality rates by the distances traveled in each state on the corresponding highway type and then sum over all the states on each route. For example, for interstate highways, the interstate distances and interstate accident rates were used. For non-interstate highway travel, either the “Primary” or “Other” accident rates given in Reference 7.4-4 were used. This approach allowed computation of route-specific accident rates.

The estimated distances used in the RADTRAN analysis in Reference 7.4-1 are bounding for the Fermi site as shown in Section 3.8. Transportation accident risk analysis in RADTRAN 5 is performed using an accident severity and package release model. The user can define up to 30 severity categories, with each category increasing in magnitude. Severity categories are related to fire, puncture, crush, and immersion environments created in vehicular accidents. For this analysis

(Reference 7.4-1), the 19 severity categories defined by Sprung, et. al. were adopted. For accidents that result in a release of radioactive material, RADTRAN 5 assumes the material is dispersed into the environment according to standard Gaussian diffusion models. The code allows the user to choose two different methods for modeling the atmospheric transport of radionuclides after a potential accident. The user can input either Pasquill atmospheric-stability category data or averaged time-integrated concentrations. In the Reference 7.4-1 analysis, the default standard cloud option (using time-integrated concentrations) was used.

Using RADTRAN 5, the analysis in Reference 7.4-1 calculated the population dose from the released radioactive material for five possible exposure pathways:

1. External dose from exposure to the passing cloud of radioactive material.
2. External dose from the radionuclides deposited on the ground by the passing plume. The analysis conservatively included the radiation exposure from this pathway even though the area surrounding a potential accidental release would be evacuated and decontaminated, thus preventing long-term exposures from this pathway.
3. Internal dose from inhalation of airborne radioactive contaminants.
4. Internal dose from resuspension of radioactive materials that were deposited on the ground. The analysis conservatively included the radiation exposures from this pathway even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures.
5. Internal dose from ingestion of contaminated food (the NRC analysis assumed interdiction of foodstuffs and evacuation after an accident so no internal dose due to ingestion of contaminated foods was calculated).

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

The analysis in Reference 7.4-1 calculated the environmental consequences of transportation accidents when shipping spent fuel from other potential new reactor sites to a spent fuel repository assumed to be at Yucca Mountain, Nevada. As discussed in Section 3.8, the consequences for transportation accidents were determined to be bounding of transportation from the Fermi site.

7.4.3 **Transportation of Radioactive Waste**

The regulations in 10 CFR 51.52(a)(4) require that, with the exception of spent fuel, radioactive waste shipped from the reactor is to be packaged and in a solid form. Additionally, existing NRC (10 CFR 71) and Department of Transportation (49 CFR 173, 178) packaging and transportation regulations specify requirements for the shipment of radioactive material. Fermi 3 is also subject to these regulations.

7.4.4 **Conclusions**

Reference 7.4-1 concludes that the overall transportation accident risks associated with advanced LWR spent fuel shipments are SMALL and are consistent with the risks associated with transportation of spent fuel from current generation reactors presented in Table S-4 of 10 CFR 51.52. As discussed above, the analyses in Reference 7.4-1 bound the Fermi site. Therefore, the overall transportation accident risks for the Fermi site are considered to be SMALL, and no mitigation measures are needed.

7.4.5 **References**

7.4-1 U.S. Nuclear Regulatory Commission, “Environmental Impact Statement for an Early Site Permit (ESP) at the Grand Gulf Site Final Report,” NUREG-1817, April 2006.

7.4-2 Idaho National Engineering and Environmental Laboratory, “Early Site Permit Environmental Report Sections and Supporting Documentation,” Engineering Design File Number 3747, Idaho Falls, ID.

7.4-3 U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, “Reexamination of Spent Fuel Shipment Risk Estimates,” NUREG/CR-6672, Volume 1, March 2000.

7.4-4 Saricks, C.L., and M.M. Tompkins, “State-Level Accident Rates of Surface Freight Transportation: A Reexamination,” Argonne National Laboratory, ANL/ESD/TM-150, April 1999.

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

Radionuclide	ESBWR Inventory Ci/MTU
Am-241	1340
Am-242m	33.5
Am-243	32.4
Ce-144	1.14E+04
Cm-242	55.1
Cm-243	37.0
Cm-244	4860
Cm-245	0.66
Co-60	2730
Cs-134	4.81E+04
Cs-137	1.24E+05
Eu-154	1.03E+04
Eu-155	5220
I-129	0.04
Kr-85	8890
Pm-147	3.38E+04
Pu-238	6135
Pu-239	386
Pu-240	616
Pu-241	1.22E+05
Pu-242	2.2
Ru-106	1.64E+04
Sb-125	5380
Sr-90	8.84E+04
Y-90	8.84E+04

Ci/MTU = curies per metric ton of uranium

Source: Reference 7.4-1, Table H-11