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7.1 Design Basis Accidents

This section evaluates the radiological consequences of design basis accidents at the VCS site.

7.1.1 Selection of Accidents

Consistent with the ESBWR DCD (GEH Sep 2007), the following DBAs are evaluated as those having potential for radioactivity releases to the environment:

- Fuel-handling accident
- LOCA
- Main steam line break outside containment (MSLB)
- Feedwater line break outside containment
- Failure of small line carrying primary coolant outside containment
- Reactor water cleanup/shutdown cooling (RWCU/SDC) system line failure outside containment

The radiological consequences of these accidents are assessed to demonstrate that the ESBWR can be sited at the VCS site without undue risk to the health and safety of the public.

The following BWR accidents are not evaluated in the DCD because they are either not applicable to the ESBWR design or have no radiological consequences:

- Reactor Coolant Pump Rotor Seizure
- Reactor Coolant Pump Shaft Break
- Control Rod Drop Accident

7.1.2 Evaluation Methodology

The DCD presents the radiological consequences for the accidents identified in Subsection 7.1.1. 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 is specific to the ESBWR 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 DCD Section 15.4.

The dose to an individual located at the exclusion area boundary (EAB) or in 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 dose conversion factors. The only site-specific parameter is atmospheric dispersion. Site-specific doses are obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (X/Q values).

ER DBA doses are evaluated using 50th percentile X/Q values, which represent more realistic meteorological conditions than those provided in FSAR Chapter 15. Short-term accident X/Q values are calculated using the methodology of RG 1.145 (U.S. NRC Nov 1982) with site-specific meteorological data. As indicated in ER Subsection 2.7.5, the RG 1.145 methodology 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. Releases are assumed to be at ground level, and the shortest distances between the power block and the offsite locations are selected to conservatively maximize the X/Q values.

Consistent with the DCD, the accident doses are expressed as total effective dose equivalent (TEDE) to demonstrate compliance with 10 CFR 50.34. The TEDE consists of the sum of the committed effective dose equivalent from inhalation and either the deep dose equivalent or the effective dose equivalent from external exposure. The committed effective dose equivalent is determined using the dose conversion factors in Federal Guidance Report 11 (U.S. EPA 1988), while the deep dose equivalent and the effective dose equivalent are based on dose conversion factors in Federal Guidance Report 12 (U.S. EPA 1993).

7.1.3 **Source Terms**

The ESBWR DBA source terms are based on a design power of 4590 MWt, which represents 102% of the rated power of 4500 MWt. The source terms are presented as time-dependent isotopic activity releases to the environment in ESBWR DCD Section 15.4, Tables 15.4-3a, 15.4-7, 15.4-12, 15.4-15, 15.4-18, and 15.4-22.

7.1.4 **Radiological Consequences**

For each of the accidents identified in Subsection 7.1.1, the site-specific dose for a given time interval is calculated by multiplying the ESBWR DCD dose by the ratio of the site X/Q value to the DCD X/Q value. The time-dependent DCD and site X/Q values and their ratios are shown in Table 7.1-1. (Because the site X/Q values shown in Table 7.1-1 are bounded by DCD X/Q values, site-specific doses for all accidents would also be bounded by DCD doses.) However, site-specific doses are presented for completeness. Accident doses are presented in Tables 7.1-2 to 7.1-9 and are summarized in Table 7.1-10.

The 10 CFR 50.34 dose limit at the EAB and the LPZ is 25 rem TEDE, as specified in 10 CFR 50.34(a)(1)(ii). As indicated in 10 CFR 50.34(a)(1)(ii), this limit applies to extremely low probability accidents that could result in the release of significant quantities of radioactive fission products. For accidents with higher probabilities, more restrictive dose limits are specified in NUREG-0800, Section 15.0.3 (U.S. NRC 2007). Where applied, these dose limits are either 10% or 25% of the 10 CFR 50.34 limit of 25 rem TEDE. The dose limits are shown in Tables 7.1-2 to 7.1-10 for comparison purposes.

The tables demonstrate that, in addition to meeting the 25 rem TEDE acceptance criterion of 10 CFR 50.34(a)(i)(ii), the accidents also meet the more restrictive limits of NUREG-0800 where applicable. Because all doses are within the acceptance criteria of NUREG-0800, the potential environmental impacts of DBAs will be SMALL.

7.1.5 **References**

GEH 2007. GE-Hitachi Nuclear, *ESBWR Design Control Document*, Tier 2, Document 26A6642, Revision 4, September 2007.

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

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

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

U.S. NRC 2007. U.S. NRC, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, Section 15.0.3, Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors, March 2007.

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

Accident	Location	Release Time[a]	DCD X/Q[b] (sec/m^3)	Site X/Q (sec/m^3)	X/Q Ratio (Site/DCD)
Fuel Handling Accident,	EAB	0–2 hr	2.00×10^{-3}	4.57×10^{-5}	0.023
RWCU/SDC Line Failure	LPZ	Instantaneous	1.90×10^{-4}	1.84×10^{-6}	0.010
LOCA, Failure of Small Line	EAB	2-hr period	2.00×10^{-3}	4.57×10^{-5}	0.023
Outside Containment	LPZ	0–8 hr	1.90×10^{-4}	1.84×10^{-6}	0.010
		8–24 hr	1.40×10^{-4}	1.52×10^{-6}	0.011
		24–96 hr	7.50×10^{-5}	1.01×10^{-6}	0.013
		96–720 hr	3.00×10^{-5}	5.61×10^{-7}	0.019
MSLB	EAB	Instantaneous	2.00×10^{-3}	4.57×10^{-5}	0.023
	LPZ	Instantaneous	2.00×10^{-3}	1.84×10^{-6}	0.00092
Feedwater Line Break	EAB	Instantaneous	1.00×10^{-3}	4.57×10^{-5}	0.046
	LPZ	Instantaneous	1.00×10^{-3}	1.84×10^{-6}	0.0018

a. EAB dose is for the period yielding the maximum 2-hr dose
b. (GEH Sep 2007) Tables 15.4-2, 15.4-9, 15.4-11, 15.4-14, 15.4-17, 15.4-21

Table 7.1-2
Doses for Fuel Handling Accident

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	4.13	0.023	0.094
LPZ	0.39	0.010	0.0038
Limit[b]	6.3	—	6.3

a. (GEH Sep 2007) Table 15.4-4
b. NUREG-0800, Section15.0.3, Table 1

Table 7.1-3
Doses for LOCA

		DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB[b]	2–hr period	13.0	0.024	0.30
LPZ	0–8 hr	3.2	0.010	0.031
	8–24 hr	2.7	0.011	0.029
	24–96 hr	5.2	0.013	0.070
	96–720 hr	6.6	0.019	0.12
	Total	17.7	—	0.25
Limit[c]	—	25	—	25

a. (GEH Sep 2007) Table 15.4-9
b. EAB dose is for the period yielding the maximum 2-hr dose
c. NUREG-0800, Section15.0.3, Table 1

Table 7.1-4
Doses for MSLB, Pre-Incident Iodine Spike

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	12.6	0.023	0.29
LPZ	12.6	0.00092	0.012
Limit[b]	25	—	25

a. Reference 7.1-1, Table 15.4-13
b. NUREG-0800, Section15.0.3, Table 1

Table 7.1-5
Doses for MSLB, Equilibrium Iodine Activity

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	0.7	0.023	0.016
LPZ	0.7	0.00092	0.00064
Limit[b]	2.5	—	2.5

a. (GEH Sep 2007) Table 15.4-13
b. NUREG-0800, Section 15.0.3, Table 1

Table 7.1-6
Doses for Feedwater Line Break Outside Containment

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	1.7×10^{-4}	0.046	7.8×10^{-6}
LPZ	1.7×10^{-4}	0.0018	3.1×10^{-7}
Limit[b]	2.5	—	2.5

a. (GEH Sep 2007) Table 15.4-16
b. No limit specified in NUREG-0800; the most restrictive limit for any accident in NUREG-0800 is assumed

Table 7.1-7
Doses for Failure of Small Line Carrying Primary Coolant Outside Containment

		DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	0–2 hr	0.15	0.023	0.0034
LPZ	0–8 hr	0.04	0.010	0.00039
	8–24 hr	0.01	0.011	0.00011
	24–96 hr	0	0.013	0
	96–720 hr	0	0.019	0
	Total	0.05	—	0.00050
Limit[b]	—	2.5	—	2.5

a. (GEH Sep 2007) Table 15.4-19
b. NUREG-0800, Section15.0.3, Table 1

Table 7.1-8
Doses for RWCU/SDC System Line Failure, Pre-Incident Iodine Spike

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	9.8	0.023	0.22
LPZ	0.93	0.010	0.0090
Limit[b]	25	—	25

a. (GEH Sep 2007) Table 15.4-23
b. No limit specified in NUREG-0800; assumed to be same as MSLB pre-incident spike

Table 7.1-9
Doses for RWCU/SDC System Line Failure, Coincident Iodine Spike

	DCD Dose[a] (rem TEDE)	X/Q Ratio (Site/DCD)	Site Dose (rem TEDE)
EAB	0.49	0.023	0.011
LPZ	0.047	0.010	0.00046
Limit[b]	2.5	—	2.5

a. Reference 7.1-1, Table 15.4-23
b. No limit specified in NUREG-0800; assumed to be same as MSLB equilibrium iodine activity

Table 7.1-10
Summary of Design Basis Accident Doses

Accident	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)	Limit (rem TEDE)
Fuel-Handling Accident	0.094	0.0038	6.3
LOCA	0.30	0.25	25
Main Steam Line Break			
Pre-Incident Iodine Spike	0.29	0.012	25
Equilibrium Iodine Activity	0.016	0.00064	2.5
Feedwater Line Break	$7.8x10^{-6}$	$3.1x10^{-7}$	2.5
Failure of Small Line Carrying Primary Coolant Outside Containment	0.0034	0.00050	2.5
RWCU/SDC System Line Failure			
Pre-Incident Iodine Spike	0.22	0.0090	25
Coincident Iodine Spike	0.011	0.00046	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, 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 (GE Aug 2007) 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 NRC.

In this section, Exelon presents an update of the generic PRA analysis, which includes VCS site-specific characteristics. The analysis evaluates the impacts of a severe accident at VCS to demonstrate that the impacts are bounded by the generic 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 that defined the possible end states of the containment following a severe accident. Using EPRI's Modular Accident Analysis Program code, GE determined that 10 release categories with 15 source term categories would represent the entire suite of potential severe accidents. Some release categories have more than one source term category, depending on the specific release characteristics. An accident frequency was assigned to each of the 15 source term categories (Table 7.2-1).

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

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

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

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

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

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

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

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

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

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

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

In addition, a direct containment heating (DCH) category was evaluated. The DCH category applies to sequences in which the reactor fails at high pressure and a significant DCH event occurs. 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 (MELCOR Accident Consequence Code System) to model the environmental consequences of severe accidents using generic, but conservative, meteorological and population parameters to represent a generic ESBWR site. The analysis focused on the 24-hour period following core damage as a measure of the consequences from a large release and, therefore, did not address the chronic pathways such as ingestion, inhalation of re-suspended material, or groundshine subsequent to plume passage. GE also considered the releases for the first 72 hours after core damage. Additional details of analysis are found in the ESBWR PRA (GE Aug 2007) and are reported in the ESBWR Design Control Document (GEH Sep 2007).

7.2.2 **Exelon Methodology**

Exelon also used the MACCS2 computer code to evaluate offsite risks and consequences of severe accidents, using VCS site-specific information. MACCS2 simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS2 include atmospheric transport, mitigation actions based on dose projection, dose accumulation by a number of pathways including food and water ingestion, early and latent human health effects, and economic costs. The specific pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground, inhalation of material in the passing plume or re-suspended from the ground, and ingestion of contaminated food and surface water. The MACCS2 code primarily addresses dose from the air pathway, but also calculates dose from surface runoff and deposition on surface water to determine a drinking water risk from airborne releases. The code also evaluates the extent of contamination. Exelon used site-specific meteorology and population data 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, Exelon determined the collective dose to the 50-mile population, number of latent cancer fatalities, and number of early fatalities associated with a severe accident. Economic costs were also determined, including the costs associated with short-term relocation of people, decontamination of property and equipment, interdiction of food supplies, and indirect costs resulting from loss of use of the property and incomes derived as a result of the accident.

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

ATMOS provides data to calculate the amount of material released to the atmosphere that is dispersed and deposited. The calculation uses a Gaussian plume model. Important inputs in this file include the core inventory, release fractions, and geometry of the reactor and associated buildings. This input data is taken from GE's generic PRA.

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

The third input file, CHRONC, provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors. These files access a meteorological file that uses actual VCS site meteorological monitoring data and a site characteristics file which is built using SECPOP2000 (U.S. NRC Aug 2003).

MACCS2 requires a calendar year of meteorological data for the MET file. One year of VCS site meteorological data (July 2007 to June 2008) was used to create the meteorological data file. Sensitivity studies using five years of National Weather Service Victoria Regional Airport data indicate that the site data is representative of recent conditions near the site.

The SITE file requires the 50-mile population distribution as well as agricultural-economic data. SECPOP2000 incorporates 2000 census data for the 50-mile region around the VCS site. For this analysis, the census data were modified to include transient populations and projected to the year 2060 (included in the population projections in Subsection 2.5.1), using county-specific growth rates. MACCS2 also requires the spatial distribution of certain agriculture and economic data (fraction of land devoted to farming, annual farm sales, fraction of farm sales resulting from dairy production, and property value of farm and nonfarmland) in the same manner as the population. This 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 program's specification of crop production parameters for the 50-mile region (e.g., fraction of farmland devoted to grains, vegetables, etc.) was also applied.

Exelon used the resulting MACCS2 calculations and accident frequency information to determine risk. The sum of the accident frequencies is known as the core damage frequency and includes only internally initiated events during reactor operation. Risk is the product of frequency of an accident times the consequences of the accident. The consequence can be any measure of release impacts such as radiation dose and economic cost. Dose-risk is the product of the collective dose times the accident frequency. Because the 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.

Exelon assumed a ground level release height and no release heat for each accident release hypothesized, consistent with the GE analysis. Each of those assumptions was investigated using a sensitivity calculation; release heights of middle and top of the containment and release heat of 1 and 10 megawatts per release segment were considered. The dose-risk varied by less than 1.4% for each of those.

7.2.3 **Consequences to Population Groups**

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

7.2.3.1 **Air Pathways**

Each of the accident categories was analyzed with MACCS2 to estimate population dose, number of early and latent fatalities, cost, and farmland requiring decontamination. The analysis conservatively assumed that none of the 10-mile emergency planning zone population was evacuated following declaration of a general emergency. For each accident category, the risk for each analytical endpoint was calculated by multiplying the analytical endpoint by the accident category frequency and adding across all accident categories. The results are provided in 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 miles of the site is drinkable (even though most of it is saltwater). The maximum MACCS2 code severe accident dose-risk to the 50-mile population from drinking the water is 5.0×10^{-4} person-rem per year of ESBWR operation. As shown in Table 7.2-1, this value is the sum of all accident category risks.

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

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

7.2.3.3 **Groundwater Pathways**

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

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

The Liquid Pathway Generic Study results provide conservative, un-interdicted population dose estimates for six generic categories of plants. These dose estimates were one or more orders of magnitude less than those attributed to the atmospheric pathway. Therefore, although VCS was not one of the reactors analyzed, the doses from the VCS site groundwater pathway would be expected to be much less than the doses from the atmospheric pathway, given that all categories of plant locations showed the same trend.

7.2.4 **Comparison to NRC Safety Goals**

The ESBWR PRA evaluates performance of the ESBWR under generic conditions to two safety goals: (1) individual risk goal and (2) societal risk goal (GE Aug 2007). These goals are defined in the following subsections. Table 7.2-2 provides the quantitative evaluation of these safety goals and the VCS site-specific calculation of these risk values.

7.2.4.1 **Individual Risk Goal**

The risk to an average individual in the vicinity of a nuclear power plant of experiencing a prompt fatality resulting from a severe reactor accident should not exceed one-tenth of one percent (0.1%) of the sum of “prompt fatality risks” resulting from other accidents to which members of the United States population are generally exposed. As defined in the Safety Goals Policy statement (51 FR 30028), “vicinity” is the area within one mile of the plant site boundary. The population within one mile of the proposed VCS reactors is zero. Exelon conservatively assumed a uniformly distributed synthetic population surrounding the site within 1 mile of the reactor. “Prompt Fatality Risks” are defined as the sum of risks which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities (driving, household, chores, occupational activities, etc). For this evaluation, the sum of prompt fatality risks was taken as the United States accidental death risk value of 37.7 deaths per 100,000 people per year (CDC Aug 2007).

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%) 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 189.8 deaths per 100,000 people per year based upon National Center for Health Statistics data for 2002–2004 (CDC Aug 2007).

7.2.5 **Conclusions**

The total calculated dose-risk to the 50-mile population from airborne releases from an ESBWR reactor at the VCS site would be 0.0019 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 (U.S. NRC Jun 1989).

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

The risks of groundwater contamination from a severe 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 Section 5.4, the total collective whole body dose from the VCS Unit 1 or 2 normal airborne releases is expected to be 0.63 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 0.63 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.0019 person-rem per reactor year indicates that the dose risk from severe accidents is approximately 0.3% of the dose risk from normal operations.

The probability-weighted risk of fatalities (early and latent cancer) from a severe accident for the VCS site is reported in Table 7.2-1 as 1.2×10^{-6} fatalities per reactor year. The probability of an individual dying from any cancer from any cause is approximately 0.24 over a lifetime (ACS Mar 2008). Comparing this value to the 1.2×10^{-6} fatalities per reactor year indicates that individual risk is 0.0005% of the background risk. As reported in Table 7.2-1, the individual and societal risks for a severe accident from an ESBWR reactor at the VCS site would be less than the NRC risk goals described in Subsection 7.2.4.

7.2.6 **References**

ACS Mar 2008. American Cancer Society, *Lifetime Probability of Developing or Dying From Cancer,* available at http://www.cancer.org/ docroot/CRI/content/ CRI_2_6x_ Lifetime_Probability_ of_Developing _or_Dying_From _Cancer.asp?sitearea=, accessed March 15, 2008.

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

Table 7.2-1
Impacts to the Population and Land from Severe Accident Airborne Releases

Accident Category[a]	Accident Frequency (per reactor year)[b]	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 mid	7.4×10^{-11}	2.1×10^{-4}	4.4×10^{-10}	1.6×10^{-7}	2.9×10^{-1}	8.2×10^{-5}	1.2×10^{-5}
BOC low	7.4×10^{-11}	6.9×10^{-5}	7.9×10^{-12}	4.4×10^{-8}	1.4×10^{-1}	8.7×10^{-6}	1.7×10^{-5}
BYP low	1.5×10^{-12}	2.4×10^{-6}	8.6×10^{-13}	1.6×10^{-9}	4.9×10^{-3}	5.6×10^{-7}	3.7×10^{-7}
BYP high	5.5×10^{-11}	1.3×10^{-4}	5.8×10^{-11}	1.2×10^{-7}	1.4×10^{-1}	1.2×10^{-5}	1.1×10^{-5}
CCID low	1.0×10^{-12}	1.4×10^{-6}	0.0	8.2×10^{-10}	3.4×10^{-3}	3.7×10^{-7}	2.5×10^{-7}
CCID high	1.0×10^{-12}	1.2×10^{-6}	8.6×10^{-16}	7.3×10^{-10}	2.7×10^{-3}	2.3×10^{-7}	2.6×10^{-7}
CCIW low	5.3×10^{-11}	7.2×10^{-7}	0.0	4.3×10^{-10}	7.8×10^{-5}	4.0×10^{-9}	4.3×10^{-9}
CCIW high	4.6×10^{-11}	2.3×10^{-5}	0.0	1.4×10^{-8}	5.1×10^{-2}	3.4×10^{-6}	6.3×10^{-6}
EVE	6.1×10^{-10}	1.2×10^{-3}	5.9×10^{-14}	7.4×10^{-7}	3.0	3.9×10^{-4}	1.9×10^{-4}
FR	1.0×10^{-12}	2.7×10^{-7}	0.0	1.6×10^{-10}	3.6×10^{-4}	1.1×10^{-8}	7.4×10^{-8}
OPVB low	3.0×10^{-13}	1.4×10^{-7}	0.0	8.4×10^{-11}	2.4×10^{-4}	8.6×10^{-9}	4.8×10^{-8}
OPVB high	5.7×10^{-12}	2.2×10^{-6}	0.0	1.3×10^{-9}	3.5×10^{-3}	1.2×10^{-7}	7.3×10^{-7}
OPW1	1.0×10^{-12}	4.2×10^{-7}	0.0	2.5×10^{-10}	6.7×10^{-4}	2.3×10^{-8}	1.5×10^{-7}
OPW2	1.0×10^{-12}	9.1×10^{-8}	0.0	5.4×10^{-11}	5.5×10^{-5}	2.0×10^{-9}	4.3×10^{-9}
TSL	1.1×10^{-8}	2.4×10^{-4}	0.0	1.4×10^{-7}	1.4×10^{-1}	2.4×10^{-6}	4.5×10^{-6}
Total	1.2×10^{-8} [c]	1.9×10^{-3}	5.1×10^{-10}	1.2×10^{-6}	3.8	5.0×10^{-4}	2.4×10^{-4}

[a] GE Aug 2007 Table 9-1
[b] GE Aug 2007 Table 10.3-3a
[c] GE Aug 2007 Table 17.1-1

Table 7.2-2
Comparison to NRC Safety Goals

Goal	Risk Goal	Exelon ESBWR per unit
Individual Risk (0-1 mile)	3.77×10^{-7}	1.9×10^{-11}
Societal Risk (0-10 miles)	1.90×10^{-6}	3.8×10^{-13}

7.3 Severe Accident Mitigation Alternatives

Regulations of the Council on Environmental Quality regarding the National Environmental Policy Act require that a discussion on environmental consequences include mitigation measures (40 CFR 1502.16(h)). The Council on Environmental Quality has stated that 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 (GE Aug 2007). This analysis determined that severe accident impacts are within the safety goals established by the NRC. Exelon extended the GE generic PRA to examine Exelon's proposed ESBWR units at VCS and concluded that the generic analysis remains valid for the site. The analysis discussed in this section demonstrates that there are no cost-beneficial design alternatives that would need to be implemented at VCS to mitigate the small impacts described in Section 7.2.

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 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 VCS 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 screening value. SAMAs with higher implementation cost than the base case screening 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.

The process used in this evaluation is (1) to demonstrate that the severe accident analysis using VCS-specific parameters is bounded by the GE severe accident analysis and (2) to determine 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. Steps 2 through 4 cannot be completed until there is a completed plant with procedural controls.

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, *Regulatory Analysis Technical Evaluation Handbook* (U.S. NRC Jan 1997). 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 Exelon baseline analysis produces a value that is below that expected for implementation of any reasonable SAMA, no matter how inexpensive, 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 in a response to an NRC request for additional information (GEH Aug 2007) during the review of the ESBWR design for certification. 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. Of the original 177 SAMDAs, none survived the Step 2 screening process.

Nevertheless, using the cost-benefit methodology of NUREG/BR-0184, GEH calculated the maximum averted cost-risk for ESBWR severe accidents using conservative population estimates and meteorology. GEH calculated the base case maximum averted cost risk to be $4628, with an upper bound value of $41,383 using more conservative assumptions. GEH concluded, "It is unlikely that any future design changes would be justifiable on the basis of person-rem exposure because the estimated CDF (core damage frequency) changes remain low on an absolute scale."

7.3.3 **Monetary Valuation of the VCS 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 6.61×10^{-8} per year (GE Aug 2007). The CDF, dose-risk, and cost-risk from internal events are reported in Table 7.2-1. The dose-risk and cost-risk associated with external events CDF are assumed to be equal to the dose-risk and cost-risk associated with internal events CDF. 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 COL application, to be consistent with the GEH analysis. Finally, the economic discount rate is assumed to be 7%, consistent with the GEH analysis. The NRC recommends using a 7% discount rate and performing a sensitivity analysis using 3% (U.S. NRC Jan 1997).

Using these inputs, the maximum monetary value (in 2007 dollars) 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 $4,190 for a single ESBWR at the VCS 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% discount rate, the valuation of the averted risk is only $10,830. These values compare well to the GEH generic analysis result of $4,628 for the best estimate and $41,383 for the upper bound estimate.

Accordingly, further evaluation of design-related SAMAs is not warranted. Exelon does not believe that administrative SAMAs, such as those relating to procedures or training, are appropriate for evaluation at this time. The design analysis confirmed that the maximum averted cost risk for an ESBWR at VCS is comparable to or less than that generically calculated by GEH for the reactor certification application. Additionally, emergency procedure guidelines/severe accident guidelines are being developed as part of the ESBWR human factors engineering process. This process requires input from both design basis accident analysis and severe accidents identified in the PRA. Finally, the low value of the maximum averted cost risk precludes any administrative changes being cost-effective.

7.3.4 **References**

GE Aug 2007. GE Energy, *ESBWR Probabilistic Risk Assessment*, NEDO-33201, U.S. NRC Accession Number ML072410543, Revision 2, August 17, 2007, Sections 8, 9, 10, 16 and Marked Up Table 7.2-5.

GEH Aug 2007. GE-Hitachi Nuclear Energy Americas LLC, *ESBWR Severe Accident Mitigation Design Alternatives*, NEDO-33306, U.S. NRC Accession Number ML072390051, Revision 1, August 14, 2007.

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Table 7.3-1
Valuation of the Exelon ESBWR Base Case

	7% Discount Rate	3% Discount Rate
Offsite exposure cost	$293	$580
Offsite economic cost	$289	$572
Onsite exposure cost	$33	$76
Onsite cleanup cost	$1,004	$2,384
Replacement power cost	$2,571	$7,219
Total	$4,190	$10,830

Note: Values are in 2007 dollars.

7.4 Transportation Accidents

Subsection 5.7.2.2 describes the methodology used to analyze the impacts of transportation of radioactive materials. Subsection 7.4.1 describes the radiological impacts of transportation accidents. The nonradiological impacts of transportation accidents are addressed in Subsection 7.4.2.

7.4.1 Radiological Impacts of Transportation Accidents

7.4.1.1 Transportation of Unirradiated Fuel

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

7.4.1.2 Transportation of Spent Fuel

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

The radionuclide inventory of the ESBWR spent fuel after 5 years of decay was estimated using the ORIGEN code. A screening analysis was performed to select the dominant contributors to accident risks and to simplify the RADTRAN 5 calculations. This screening identified the radionuclides that would collectively contribute more than 99.999% of the dose from inhalation of radionuclides released following a transportation accident (USNRC Dec 2006). The spent fuel inventory used in this analysis for the ESBWR is presented in Table 7.4-1.

The specific quantities and characteristics of the crud deposited on ESBWR spent fuel from corrosion products generated elsewhere in the reactor coolant system are unknown at this time due to 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. The highest surface radioactivity of Co-60 in spent fuel crud available for spallation during transportation accidents for the ESBWR is expected to be 579 μCi per cm^2. NUREG/CR-6672 (Sprung et al. Mar 2000) indicates that the total surface area for a BWR fuel rod is approximately 1600 cm^2. The number of fuel rods for an ESBWR assembly is expected to be about 100. As a result, the total surface area of an ESBWR spent fuel assembly would be 160,000 cm^2. The weight of UO_2 for each ESBWR assembly is estimated to be 0.163 metric tons uranium (MTU). Thus, the inventory of Co-60 in ESBWR spent fuel crud available for spallation during transportation accidents is estimated to be 568 Ci per MTU.The inventory of Co-60 in spent fuel crud used for the RADTRAN analysis was 2860 Ci per MTU. As such, the available inventory of Co-60 in the ESBWR spent fuel crud is about a factor of five lower than that used in the RADTRAN 5 analysis.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance features required by 10 CFR 71, "Packaging and Transportation of Radioactive Material." Spent fuel shipping casks must be certified Type B packaging systems, meaning they must withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability[a]. 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% (i.e., more than 99.99% of all accidents would result in no release of radioactive material from the shipping cask). This analysis assumed that shipping casks for ESBWR spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo, in accordance with the requirements of 10 CFR 71.

For the spent fuel from the ESBWR, 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 ESBWR. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 38.5 MTU per year for the ESBWR. (This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2.)

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

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

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

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

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

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

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

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

The risk to the public from radiation exposure was estimated using the nominal probability coefficient for total detrimental health effects (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 1×10^6 person-rem) per reference reactor year from the International Commission on Radiological Protection Publication 60 (ICRP 1991). These values are presented in Table 7.4-2. These estimated risks are 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 transportation of spent fuel from the VCS site or an alternate site would be consistent with the environmental impacts presented in Table S-4.

7.4.2 **Nonradiological Impacts of Transportation Accidents**

Nonradiological impacts would include the projected number of accidents, injuries, and fatalities that could result from shipments of radioactive materials to or from the VCS site and return of empty containers. Nonradiological impacts were estimated using accident, injury, and fatality rates from Table 4 of *State-Level Accident Rates for Surface Freight Transportation: A Reexamination* (Saricks and

Tompkins Apr 1999). This data is representative of the traffic accident, injury, and fatality rates for heavy truck shipments similar to those that would be used to transport radioactive materials to and from the site. These rates (measured in impacts per vehicle-mile traveled) are multiplied by the annual numbers of shipments and estimated travel distances for the shipments to estimate annual impacts. These estimates include the human health impacts projected to result from traffic accidents involving shipments of radioactive materials; they do not consider the radiological or hazardous characteristics of the cargo.

7.4.2.1 **Transportation of Unirradiated Fuel**

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

7.4.2.2 **Transportation of Spent Fuel**

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

7.4.2.3 **Transportation of Radioactive Waste**

Nonradiological impacts of radioactive waste shipments were calculated using the same general approach as the unirradiated fuel shipments. A shipping distance of 500 miles was assumed consistent with the analysis in WASH-1238. Because the destination of the waste shipments is not known, the national median accident, injury, and fatality rates from Saricks and Tompkins (Apr 1999) were used to calculate the values presented in Table 7.4-5. The nonradiological impacts for the reference LWR analyzed in WASH-1238 are also shown for comparison. The differences between the reference LWR and ESBWR are due to the higher number of radioactive waste shipments projected for the ESBWR. The values presented in Table 7.4-5 would be doubled for a two-unit plant.

7.4.3 Conclusion

Based on this analysis, the overall transportation accident risks associated with spent fuel shipments from the proposed ESBWR units at VCS are consistent with the risks associated with transportation of spent fuel from current generation reactors presented in WASH-1238 and Table S-4 of 10 CFR 51.52 (reproduced in Table 5.7-2) and thus will be SMALL.

7.4.4 References

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

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

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

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

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

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

Radionuclide	ESBWR Inventory (curies per MTU)
Am-241	1.30×10^3
Am-242m	2.79×10^1
Am-243	3.26×10^1
Ce-144	1.35×10^4
Cm-242	4.86×10^1
Cm-243	3.47×10^1
Cm-244	4.96×10^3
Cm-245	6.75×10^{-1}
Co-60	2.86×10^3
Cs-134	5.19×10^4
Cs-137	1.27×10^5
Eu-154	1.05×10^4
Eu-155	5.47×10^3
Pm-147	3.53×10^4
Pu-238	6.15×10^3
Pu-239	3.86×10^2
Pu-240	6.22×10^2
Pu-241	1.22×10^5
Pu-242	2.24
Ru-106	1.86×10^4
Sb-125	5.80×10^3
Sr-90	9.08×10^4
Y-90	9.09×10^4

Table 7.4-2
Spent Fuel Transportation Accident Risks for the ESBWR

Site	Unit Population Dose (person-rem per MTU)[1]	MTU per Reference Reactor Year	Population Dose (person-rem per reference reactor year)[1]	Total Detrimental Health Effects per Reference Reactor Year
VCS	1.03×10^{-7}	23	2.38×10^{-6}	1.73×10^{-9}
Matagorda County	9.70×10^{-8}	23	2.23×10^{-6}	1.63×10^{-9}
Buckeye	9.70×10^{-8}	23	2.23×10^{-6}	1.63×10^{-9}
Allens Creek	9.62×10^{-8}	23	2.21×10^{-6}	1.62×10^{-9}
Malakoff	1.09×10^{-7}	23	2.50×10^{-6}	1.82×10^{-9}

[1]Value presented is the product of probability multiplied by collective dose.

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

				Annual Impacts		
Reactor	Total Shipments Normalized to Reference LWR	One-way Shipping Distance (miles)	Total Round-trip Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
Reference LWR	252	2000	1.01×10^{6}	3.7×10^{-4}	0.0078	0.011
ESBWR	221	2000	8.84×10^{5}	3.3×10^{-4}	0.0069	0.010

Table 7.4-4
Nonradiological Impacts of Transporting Spent Fuel from the Victoria County Station

State	Highway Type	One-way Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
Arizona	Interstate	391	5.6×10^{-4}	0.0069	0.0078
California	Interstate	367	3.9×10^{-4}	0.0069	0.0089
Nevada	Interstate	66	6.5×10^{-5}	0.0015	0.0022
	Primary	79	2.0×10^{-4}	0.0030	0.0046
New Mexico	Interstate	164	2.9×10^{-4}	0.0029	0.0028
Texas	Interstate	670	$1.3 \text{ x } 10^{-3}$	0.0550	0.0610
	Primary	71	3.1×10^{-4}	0.0056	0.0074
Totals		1807	$3.1 \text{ x } 10^{-3}$	0.0820	0.0950

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

			Annual Impacts		
Reactor	Shipments per Year Normalized to Reference LWR	One-way Shipping Distance (miles)	Fatalities per Year	Injuries per Year	Accidents per Year
Reference LWR	46	500	6.8×10^{-4}	0.014	0.021
ESBWR	115	500	$1.7 \text{ x } 10^{-3}$	0.036	0.052