CHAPTER 7 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter assesses the environmental impacts of postulated accidents involving radioactive materials at the Clinch River Nuclear (CRN) Site. The chapter is divided into 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 Design Alternatives (Section 7.3)
- Transportation Accidents (Section 7.4)

7.1 DESIGN BASIS ACCIDENTS

7.1.1 Accident Selection

The evaluation of nuclear power plant safety includes analysis of the facility's response to a spectrum of postulated disturbances in process variables and postulated equipment failures.

However, it is neither practical nor necessary to analyze all historically-postulated design basis accidents (DBAs) associated with small modular reactor (SMR) types under consideration for the Clinch River Nuclear (CRN) Site in the early site permit application (ESPA), as discussed below.

As noted in NEI 10-01, *Industry Guideline for Developing a Plant Parameter Envelope (PPE) in Support of an Early Site Permit*, accident analyses model the time-dependent transport of radionuclides out of the reactor core through several pathways, each with different time-dependent removal mechanisms for radionuclides. Different reactor designs have different release pathways, and each pathway has different release rates and different radionuclide removal mechanisms. Given these differences, it is impractical to develop a bounding analysis for use in a PPE-based ESPA, and accordingly, for the purposes of evaluating offsite post-accident doses, the vendor analysis with the highest resultant post-accident dose was selected for use in the CRN Site-specific dose analysis presented here. (Reference 7.1-1)

At this time, the site layout and building configuration for each of the proposed reactor designs for the CRN Site has yet to be determined, making it impractical to model near-field atmospheric dispersion around buildings in order to determine doses in the main control room and other areas where habitability is required post-accident. Thus, these types of detailed accident analyses are more appropriately performed during the combined license application (COLA) stage when a technology is selected and the orientation of the plant on the site is known.

Pressurized water reactor (PWR) designs, as documented in ESPAs to date, have shown that offsite doses due to a postulated loss-of-coolant accident (LOCA) are expected to more closely approach Title 10 of the Code of Federal Regulations (10 CFR) 52.17 limits than other DBAs that may have a greater probability of occurrence but a lesser magnitude of activity release, as evidenced by the following:

- Clinton Site ESPA, Environmental Report (ER) Table 7.1-2 (Reference 7.1-2)
- Grand Gulf Site ESPA, ER Table 7.1-1 (Reference 7.1-3)
- North Anna Site ESPA, ER Table 7.1-2 (Reference 7.1-4)
- Vogtle Site ESPA, ER Table 7.1-12 (Reference 7.1-5)
- PSEG Site ESPA, ER Tables 7.1-39, 7.1-47 and 7.1-56 (Reference 7.1-6)
- Victoria County Station Site ESPA, ER Table 7.1-5 (Reference 7.1-7)

Based on initial design feedback, TVA has reasonably high confidence that the consequences of a LOCA will be shown to be proportionally less than those for large PWR designs, and that no new events of greater consequence will be identified. Each of the four small modular PWR designs under consideration for the CRN Site is expected to include advanced design features that would further minimize accident consequences. In particular, based on initial design feedback, Tennessee Valley Authority (TVA) anticipates that the consequences of a LOCA will be less than those for large PWR designs and that no events of greater consequence will be identified.

Thus, analysis of postulated DBAs other than a LOCA is not necessary for the ESPA, because the maximum potential offsite doses have been evaluated, demonstrating the ability of the site to comply with the dose limits in 10 CFR 52.17. The COLA will verify that the accident doses provided in this ESPA are bounded or will provide an evaluation of accident radiological consequences.

7.1.2 Source Term

The bounding design basis accident (LOCA) source term is provided in Table 7.1-1.

The LOCA source term (radionuclide activity released to the environment) selected for inclusion in the PPE is based upon vendor input and represents the design with the highest resulting doses at the exclusion area boundary (EAB) and the low population zone (LPZ) boundary from the four SMR designs under consideration. Key input parameters associated with the accident source term in the PPE have been evaluated to assess their reasonableness for and representativeness of SMR designs.

The PPE LOCA source term is based on a design that uses standard light water reactor fuel, which is representative of the SMR designs under consideration, and assumes a core power level for a single unit at 800 megawatt thermal (MWt). The methodology and analytical techniques used for development of the source term are similar to those used for large light water reactors, and TVA anticipates that comparable methodologies and techniques will be used in the development of the SMR accident source terms to be presented in the SMR design control documents.

To assess reasonableness, a comparison of the PPE LOCA source term to that of the AP1000 design (as provided in the Vogtle 3 and 4 ESPA) was performed, scaling the source term presented in the Vogtle ESPA by a factor of 0.235 (800 MWt/3400 MWt) to account for the smaller core thermal power of the SMR designs being considered for the CRN Site. The activity release associated with the worst 2-hour time period of the scaled-down AP1000 is approximately 25 percent greater than that of the surrogate plant (as provided in the PPE). This difference is reasonable given that SMR designs contain additional safety features that will result in increased safety margins general improvements over the AP1000 design. The activity release for the 30-day duration of the LOCA is approximately equivalent to that of the surrogate plant and is also considered reasonable.

The source terms developed for the surrogate plant are representative of the potential SMR designs considering core power and average burnup. The surrogate plant assumes a core power that is bounding but representative of the remaining SMR designs being considered. The maximum average burnup assumed for the surrogate plant is 51 gigawatt days per metric tons of uranium (GWD/MTU), while the maximum average burnup for the remaining SMR designs is less than 41 GWD/MTU. Although it is recognized that core power and burnup do not necessarily result in one-to-one ratios to activity releases, it is anticipated the larger core power and burnup would result in larger activity releases than those associated with the remaining SMR designs.

7.1.3 Evaluation Methodology and Conclusions

Doses for a LOCA are evaluated at the EAB and LPZ boundary. The evaluation uses the following parameters, as shown in Table 7.1-2:

- Short-term 50th percentile accident atmospheric dispersion factors (*X*/Qs) for the CRN Site
- Bounding vendor-provided LOCA doses
- *X*/Q values associated with the bounding vendor-provided LOCA doses

Doses are calculated based on the amount of activity released to the environment, the dispersion of activity during transport to the receptor (*X*/Q), the breathing rate at the receptor, and the applicable dose conversion factors. The only parameters that are site-specific are the *X*/Qs. Hence, it is reasonable to adjust the vendor LOCA doses for site-specific *X*/Q values, provided in Table 2.7.5-13

For a given time step, the vendor dose is multiplied by the ratio of the site-specific *X*/Q to the vendor *X*/Q, as shown in the following equation:

$$Dose_{Site} = Dose_{Vendor}\left[\frac{(X/Q)_{Site}}{(X/Q)_{Vendor}}\right]$$

The resulting accident doses are expressed as total effective dose equivalent (TEDE) consistent with 10 CFR 52.17. As shown in Table 7.1-2, all site LOCA doses meet the 25 rem TEDE limit specified in 10 CFR 52.17.

7.1.4 References

Reference 7.1-1. Nuclear Energy Institute, "Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit," May, 2012.

Reference 7.1-2. Exelon Nuclear, Early Site Permit Application for the Clinton ESP Site, Revision 4 (Chapter 7), Website: http://pbadupws.nrc.gov/docs/ML0611/ML061100280.pdf, April 4, 2006.

Reference 7.1-3. Entergy, Grand Gulf Early Site Permit Application, Revision 2 (Chapter 7), Website: http://pbadupws.nrc.gov/docs/ML0527/ML052790350.pdf, October 3, 2005.

Reference 7.1-4. Dominion Nuclear North Anna, LLC, North Anna Early Site Permit Application, Revision 9 (Chapter 3), Website: http://pbadupws.nrc.gov/docs/ML0625/ML062580114.pdf, September 12, 2006.

Reference 7.1-5. Southern Nuclear Operating Company, Inc., Vogtle Early Site Permit Application, Revision 5 (Chapter 7), Website: http://pbadupws.nrc.gov/docs/ML0915/ML091540840.pdf, December 23, 2008.

Reference 7.1-6. PSEG Power, LLC, Application for Early Site Permit for the PSEG Site, Revision 3 (Chapter 7), Website: http://pbadupws.nrc.gov/docs/ML1409/ML14093A939.pdf, March 31, 2014.

Reference 7.1-7. Exelon Generation, Application for Early Site Permit for Victoria County Station, Revision 1 (Chapter 7), Website: http://pbadupws.nrc.gov/docs/ML1213/ML12131A101.pdf, May 30, 2012.

Table 7.1-1 (Sheet 1 of 2)
LOCA Bounding Design Basis Accident
Atmospheric Radioactive Release (in Ci)

Nuclide	Worst 2 hour	0-8 hour	8-24 hour	1-4 days	4-30 days
Kr-85m	3.51E+02	9.28E+02	4.60E+02	2.10E+01	0.00E+00
Kr-85	3.01E+01	1.05E+02	2.50E+02	5.64E+02	4.84E+03
Kr-87	2.66E+02	4.84E+02	1.22E+01	1.00E-02	0.00E+00
Kr-88	7.48E+02	1.74E+03	4.05E+02	4.20E+00	0.00E+00
Xe-131m	1.92E+01	6.69E+01	1.55E+02	3.13E+02	1.27E+03
Xe-133m	1.17E+00	3.98E+00	8.14E+00	1.07E+01	6.72E+00
Xe-133	3.82E+03	1.32E+04	2.95E+04	5.25E+04	1.04E+05
Xe-135m	2.08E+00	8.91E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	8.53E+02	2.57E+03	2.70E+03	5.62E+02	2.30E+00
Xe-138	5.81E+00	2.92E+01	0.00E+00	0.00E+00	0.00E+00
I-130	2.12E+00	4.19E+00	1.55E-01	8.10E-03	2.00E-04
I-131	1.34E+02	2.76E+02	1.52E+01	5.80E+00	1.75E+01
I-132	9.61E+01	1.69E+02	1.11E+00	1.00E-07	0.00E+00
I-133	2.59E+02	5.20E+02	2.28E+01	2.54E+00	2.50E-01
I-134	4.98E+01	9.21E+01	2.10E-02	0.00E+00	0.00E+00
I-135	2.06E+02	3.94E+02	1.02E+01	1.50E-01	0.00E+00
Cs-134	2.35E+01	4.71E+01	2.06E+00	1.11E-02	9.60E-02
Cs-136	6.70E+00	1.20E+01	4.60E-01	3.09E-03	9.00E-03
Cs-137	1.80E+01	3.63E+01	1.59E+00	9.07E-03	7.50E-02
Cs-138	1.14E+01	2.75E+01	1.00E-08	0.00E+00	0.00E+00
Rb-86	2.06E-01	4.15E-01	1.81E-02	9.07E-05	4.80E-04
Te-127m	2.74E-01	5.48E-01	2.62E-02	1.40E-04	1.11E-03
Te-127	1.34E+00	2.52E+00	7.40E-02	0.00E+00	0.00E+00
Te-129m	9.09E-01	1.82E+00	8.65E-02	4.00E-04	3.00E-03
Te-129	1.14E+00	1.80E+00	1.70E-03	0.00E+00	0.00E+00
Te-131m	3.32E+00	6.51E+00	2.67E-01	5.00E-04	2.00E-04
Te-132	2.59E+01	5.14E+01	2.32E+00	8.00E-03	9.00E-03
Sb-127	1.59E+00	3.17E+00	1.44E-01	6.00E-04	7.00E-04
Sb-129	3.38E+00	5.99E+00	9.99E-02	0.00E+00	0.00E+00
Sr-89	7.79E+00	1.56E+01	7.42E-01	4.00E-03	2.80E-02
Sr-90	9.52E-01	1.91E+00	9.12E-02	5.00E-04	4.30E-03
Sr-91	8.01E+00	1.51E+01	4.46E-01	0.00E+00	0.00E+00
Sr-92	5.48E+00	9.27E+00	8.32E-02	0.00E+00	0.00E+00
Ba-139	4.14E+00	6.61E+00	1.17E-02	0.00E+00	0.00E+00

Table 7.1-1 (Sheet 2 of 2)
LOCA Bounding Design Basis Accident
Atmospheric Radioactive Release (in Ci)

Nuclide	Worst 2 hour	0-8 hour	8-24 hour	1-4 days	4-30 days
Ba-140	1.33E+01	2.66E+01	1.25E+00	6.00E-03	2.60E-02
Ru-103	1.40E+00	2.81E+00	1.34E-01	7.00E-04	4.80E-03
Ru-105	6.27E-01	1.12E+00	1.93E-02	0.00E+00	0.00E+00
Ru-106	4.75E-01	9.52E-01	4.55E-02	2.50E-04	2.08E-03
Rh-105	8.37E-01	1.65E+00	6.92E-02	2.00E-04	0.00E+00
Mo-99	1.71E+00	3.38E+00	1.51E-01	5.00E-04	4.00E-04
Tc-99m	1.16E+00	2.12E+00	4.78E-02	0.00E+00	0.00E+00
Ce-141	3.19E-01	6.38E-01	3.03E-02	1.60E-04	1.02E-03
Ce-143	2.81E-01	5.53E-01	2.30E-02	5.00E-05	2.00E-05
Ce-144	2.65E-01	5.31E-01	2.54E-02	1.40E-04	1.15E-03
Pu-238	5.94E-04	1.19E-03	5.68E-05	3.00E-07	2.70E-06
Pu-239	7.10E-05	1.42E-04	6.79E-06	4.00E-08	3.20E-07
Pu-240	1.08E-04	2.16E-04	1.03E-05	6.00E-08	4.80E-07
Pu-241	2.64E-04	5.30E-04	2.53E-05	1.40E-07	1.19E-06
Np-239	3.16E+00	6.26E+00	2.76E-01	8.00E-04	6.00E-04
Y-90	9.55E-03	1.89E-02	8.43E-04	2.00E-06	3.00E-06
Y-91	1.01E-01	2.02E-01	9.62E-03	5.00E-05	3.80E-04
Y-92	6.37E-02	1.11E-01	1.47E-03	0.00E+00	0.00E+00
Y-93	9.72E-02	1.84E-01	5.60E-03	0.00E+00	0.00E+00
Nb-95	1.34E-01	2.69E-01	1.28E-02	7.00E-05	4.40E-04
Zr-95	1.32E-01	2.65E-01	1.26E-02	7.00E-05	5.00E-04
zr-97	1.17E-01	2.25E-01	8.22E-03	1.00E-05	0.00E+00
La-140	1.32E-01	2.61E-01	1.11E-02	3.00E-05	1.00E-05
La-142	4.13E-02	6.62E-02	1.66E-04	0.00E+00	0.00E+00
Nd-147	4.89E-02	9.77E-02	4.60E-03	2.00E-05	8.00E-05
Pr-143	1.17E-01	2.34E-01	1.11E-02	5.00E-05	2.40E-04
Am-241	1.56E-05	3.12E-05	1.49E-06	8.00E-09	7.00E-08
Cm-242	3.16E-03	6.33E-03	3.02E-04	1.70E-06	1.32E-05
Cm-244	1.84E-04	3.69E-04	1.76E-05	1.00E-07	8.30E-07
Total	6.64E+03	2.00E+04	3.31E+04	5.39E+04	1.10E+05

Table 7.1-2
CRN Site LOCA Doses

Location	Time (hr)	*X/Q* (sec/m^3)		*X/Q* Ratio (Site/Vendor)	Dose (rem TEDE)	
		Site (50th %)	Vendor		Vendor	Site
EAB	0-2	5.58E-04	1.0E-03	0.56	4.4	2.4[1]
LPZ	0-8	4.27E-05	5.0E-04	0.085	4.4	0.38
	8-24	3.80E-05	3.0E-04	0.13	0.20	0.025
	24-96	2.94E-05	1.5E-04	0.20	0.05	0.0098
	96-720	2.04E-05	8.0E-05	0.26	0.06	0.015
				LPZ Total	4.8	0.43[1,2]

1 Versus the 25 rem TEDE limit specific in 10 CFR 52.17.
2 Column total dose not equal sum of individual values due to rounding.

7.2 SEVERE ACCIDENTS

This section evaluates the potential environmental impacts of severe accidents at the Clinch River Nuclear (CRN) Site. Four small modular reactor (SMR) designs are included in the plant parameter envelope (PPE): BWXT mPower, Holtec, NuScale, and Westinghouse. The environmental impacts from postulated severe accidents were calculated using site-specific data to demonstrate acceptability.

The current United States nuclear fleet has an exceptional safety record. Each of the four SMR designs included in the PPE includes design features that result in additional enhancements to safety.

Severe accidents are defined as accidents with substantial damage to the reactor core and degradation of containment systems. Subpart B of Title 10 of the Code of Federal Regulations Part 52 requires applications for standard design certification to include information from the probabilistic risk assessment (PRA) of the design. No application for standard design certification has been submitted to the U.S. Nuclear Regulatory Commission (NRC) for the four designs included in the PPE, and the final design and PRA information was not available for these designs at the time of preparation of this early site permit application.

The Tennessee Valley Authority (TVA) requested that each of the four SMR vendors provide information from the PRA for its design to allow the assessment of potential severe accident consequences. While information regarding severe accident release categories, source terms, and release frequencies was provided to TVA from some SMR vendors, information was not received from all vendors. Therefore, TVA made a reasonable, bounding estimate of the severe accident consequences for the PPE by evaluating the SMR design that represents the largest SMR considered for the CRN Site.

This section uses preliminary PRA information for severe accidents for the largest SMR design, along with site-specific characteristics, to estimate the impacts of severe accidents. The purpose of this analysis is to identify the environmental impacts from potential severe accidents.

7.2.1 Methodology

The MACCS2 computer code was developed to model the environmental consequences of the severe accidents (Reference 7.2-1). MACCS2 was developed specifically for the NRC to evaluate severe accidents at nuclear power plants. The NRC has approved MACCS2 analyses of environmental consequences from a new pressurized water reactor (PWR) design with passive safety features. The ratio of the thermal power rating of the previously analyzed PWR to the largest SMR considered for the CRN Site was used to estimate the source terms required for analysis of the impacts of severe accidents. Use of the largest SMR for the severe accident analysis is considered to provide representative accident consequences. The relative frequencies, source term chemical groups, and source term release fractions for the severe accident scenarios were calculated as part of the PRA for the SMR design with the maximum thermal output and are shown in Tables 7.2-1, 7.2-2, and 7.2-3, respectively. This data was used together with the MACCS2 ATMOS module input files and an estimated core damage frequency (CDF) to approximate the consequences of severe accidents for the SMR.

The SMR design used for this analysis differs from the surrogate SMR defined by the PPE. The PPE defines a maximum thermal power rating for the CRN Site based on the range of SMR designs under consideration. The individual reactor considered for this analysis uses the maximum thermal power rating for a single reactor unit [800 megawatt thermal (MWt)] from one of the potential SMR vendors; thus maximizing the severe accident consequences for an accident involving a single unit.

The CDF is a measure of the likelihood of severe accidents associated with reactor core damage. CDF is estimated using PRA modeling, which evaluates how changes to the reactor or auxiliary systems can change the severity of the accident. The vendor of the SMR considered in this analysis estimates the total CDF for the design to be approximately 4.65E-08 per reactor year (Ryr), which is lower than the CDF of 2.41E-07 for a modern full scale reactor (Reference 7.2-2). Table 7.2-1 presents the relative frequency of each release category.

The SMR used in this analysis utilizes six severe accident sequences (release categories) as follows:

- Intact Containment (IC): Containment integrity is maintained throughout the accident. The release of radioactivity to the environment is due to nominal design leakage.
- Containment Bypass (BP): Radioactivity is released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to the onset of core damage. This accident class contributes to the large, early release frequency.
- Containment Isolation Failure (CI): Radioactivity is released through a failure of the valves that close the penetrations between containment and the environment. Containment failure occurs prior to the onset of core damage. This accident class contributes to the large, early release frequency.
- Early Containment Failure (CFE): Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after the onset of core damage but prior to core relocation. Such phenomena could include hydrogen detonation, hydrogen diffusion flame, steam explosions, or vessel failures. This accident class contributes to the large, early release frequency.
- Intermediate Containment Failure (CFI): Radioactivity release occurs through a containment failure caused by some dynamic severe accident phenomenon after core relocation but before 24 hours (hr) have passed since initiation of the accident. Such phenomena could include hydrogen detonation and hydrogen deflagration. This accident class contributes to large releases but does not occur early in the accident life cycle.

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

7.2.2 TVA Methodology

The MACCS2 computer code (Version 3.10.0, with the WinMACCS graphical user interface), was used to evaluate the environmental consequences of the severe accidents (Reference 7.2-3). The exposure pathways modeled include external exposure from the passing plume, external exposure from 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. The code also evaluates the extent of contamination. The analysis used site-specific meteorology and population data and also included the ingestion pathway over the entire life cycle of the accident.

To assess human health impacts, TVA determined the collective dose, risk of early fatalities, and the risk of latent cancer fatalities from a severe accident for the population within a 50-mile (mi) radius. Economic costs were also determined, including the costs associated with short-term relocation of people, decontamination of property and equipment, and interdiction of food supplies.

MACCS2 requires five input files: ATMOS, EARLY, CHRONC, METEOROLOGICAL, 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 reactor and site-specific inputs in this file include the core inventory, release fractions, and geometry of the reactor and associated buildings. These input data are the same as those in the MACCS2 input files used by the vendor of the SMR. EARLY provides inputs to calculations regarding exposure in the time period immediately following the release. Important site-specific information includes emergency response information such as evacuation time. CHRONC provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors. These files access a meteorological file and a site characteristics file. METEOROLOGICAL provides actual site-specific meteorological monitoring data (hourly data that includes wind speed and direction, stability class, and rainfall) for one year (mid-2012 to mid-2013). SITE provides site-specific population data, land usage, watershed index, and economic data for the region.

The MACCS2 calculations and accident frequency information are used to determine risk. The sum of the accident frequencies, the CDF, includes only internally initiated events. Risk is the product of frequency of an accident multiplied by the consequences of the accident. The consequence can be radiation dose, fatalities, economic cost or farmland that needs to be decontaminated. Dose-risk is the product of the collective dose times the accident frequency.

Because the severe accident analysis addressed a suite of accidents (release categories), the individual risks are summed to provide a total risk (person-sievert [Sv] per Ryr). The same process was applied to estimating the risk of fatalities (fatalities per Ryr), the economic cost-risk (dollars per Ryr), and the risk of farmland decontamination (hectares per Ryr).

Chapter 5 of NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, Rev. 0, assesses the impacts of postulated accidents at nuclear power plants on the environment. NUREG-1437 was revised in 2013. Appendix E of NUREG-1437, Rev. 1, provides an update on postulated accident risk. Rev. 1 considers how more recent information on postulated accidents would affect the conclusions of Rev. 0 and provides comparative data where appropriate. However, Rev. 1 does not provide new information necessary for the evaluation of postulated accidents for all dose pathways, and was not used for this evaluation.

7.2.3 Consequences to Population Groups

This subsection evaluates impacts of severe accidents from air, surface water, and groundwater pathways. The MACCS2 code was used to evaluate the doses from the air pathway and from water ingestion with site-specific data. MACCS2 does not model other surface water and groundwater dose pathways. These are analyzed qualitatively based on a comparison of doses from the atmospheric pathway for CRN Site to those of the existing fleet of United States nuclear reactors.

7.2.3.1 Air Pathways

The potential severe accidents for the SMR considered in this analysis were grouped into six accident classes (release categories) based on the similarity of their characteristics. The number and description of release categories is reactor design-specific. Radionuclides that may be released are organized into groups having similar chemical characteristics as shown in Table 7.2-2. Each release category was assigned a set of characteristics representative of the chemical elements for that category as shown in Table 7.2-3. Each release category was analyzed with MACCS2 to calculate population dose, number of early and latent fatalities, economic cost, and the amount of farmland requiring decontamination. The analysis assumed that 99.5 percent of the population within the 2-mi emergency planning zone (EPZ) of the CRN Site would be evacuated following declaration of a general emergency.

For each release category, risk was calculated by multiplying each consequence (population dose, fatalities, cost, and area of contaminated land) by the total CDF and the relative frequency for the release category. The sum of the long-term dose risk to the 50-mi population from atmospheric releases was calculated by MACCS2 for the 2-mi EPZ to be 7.71E-05 person-Sv/Ryr (Table 7.2-4). As shown in Tables 7.2-5 and 7.2-6, this 50-mi population risk is much lower than the risk estimated for (1) the five plants evaluated in NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, (2) the other current operating reactors in the United States, (3) the recently licensed AP1000 reactors at the V.C. Summer and Vogtle sites, and (4) the NRC Safety Goals (51 CFR 30028).

For an additional comparison, as reported in Section 5.4, the calculated collective total body dose based on the PPE source term from normal operations at the CRN Site due to radioactive effluents (liquid and gaseous) is 6.8E+01 person-rem/Ryr (6.8E-01 person-Sv/Ryr) (from Table 5.4-16). As previously described, dose risk is the total population dose rate (in person-Sv/Ryr) multiplied by the frequency, and normal operation has a frequency of one. Therefore, the calculated population dose risk for normal operation is also 6.8E-01person-Sv/Ryr. Comparison of this value to the severe accident dose risk of 7.71E-05 person-Sv/Ryr indicates that the calculated dose risk from severe accidents is far less than the calculated dose risk from normal operation.

The economic costs (in dollars per reactor year) are also provided in Table 7.2-4. The total cost calculation considered consequences, such as evacuation costs, value of crops contaminated and condemned, value of milk contaminated and condemned, cost of property decontamination, and indirect costs resulting from loss of property use and incomes as a result of the accident. The calculated economic risk of a severe accident for the largest potential SMR at the CRN Site is 29.3 dollars/Ryr. The area of farmland requiring decontamination was calculated by MACCS2 for the 2-mi EPZ to be 1.69E-04 hectares/Ryr. These impacts are smaller than the impacts that were estimated for most other new reactor license applications. Again, these impact risks are lower than those presented in the Final Environmental Impact Statements for recently approved reactor license applications such as Vogtle (NUREG-1872, *Final Environmental Impact Statement for an Early Site Permit (ESP) at the Vogtle ESP Electric Generating Plant Site*) and V.C. Summer (NUREG-1939, *Final Environmental Impact Statement for Combined Licenses for Virgil C. Summer Nuclear Station, Units 2 and 3*) and found to be acceptable.

7.2.3.2 Surface Water Pathways

People can be exposed to radiation when airborne radioactivity is deposited onto surface water. The exposure pathways can include drinking the water, aquatic food, swimming, and shoreline pathways. Surface water bodies within the 50 mi of the CRN Site include the Tennessee River, Clinch River, Norris Lake, and other smaller bodies of water.

The NRC examined the aquatic food, swimming, and shoreline pathways in NUREG-0769, *Final Environmental Impact Statement Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2,* and demonstrated that the dose from the aquatic food pathway was more than ten times the dose from the combined swimming and shoreline doses. The examination concluded that the uninterdicted aquatic food pathway was the principal pathway of exposure and the swimming and shoreline pathways were not significant.

The NRC also evaluated doses from the aquatic food pathway for nuclear power plants discharging to various bodies of water in NUREG-1437, Rev. 0. NUREG-1437, Subsection 5.3.3.3.3 concluded that the risk associated with the aquatic food pathway is small relative to the atmospheric pathway for most sites, including small and large river sites. The CRN Site is a good approximation of the generic small river site examined in the NUREG-0440, *Liquid Pathway Generic Study: Impacts of Accidental Radioactive Releases to the Hydrosphere from*

Floating and Land-based Nuclear Power Plants, (the source of the NUREG-1437 analysis). Appendix E of NUREG-1437, Rev. 1, provides no updated information on the evaluation of the aquatic food pathway.

MACCS2 calculated the dose from drinking water pathway for surface water sources. The sum of the severe accident dose risk to the 50-mi population from drinking water was calculated by MACCS2 for the 2-mi EPZ to be 1.19E-06 person-Sv/Ryr) (Table 7.2-4). This value is the sum of the drinking water risk from each of the six release categories. As Table 7.2-4 shows, the total drinking water dose risk is very small in comparison to the total dose risk for the atmospheric pathways. This dose risk is also lower than the dose risk from the drinking water pathway presented in the final environmental impact statements for recently approved reactor license applications such as Vogtle (NUREG-1872) and V.C. Summer (NUREG-1939) and found to be acceptable.

7.2.3.3 Groundwater Pathways

People also could receive a dose from groundwater pathways. Radioactivity released during an accident can enter groundwater that serves as a source of drinking water or move through an aquifer that eventually discharges to surface water. The MACCS2 code does not calculate the dose from groundwater pathways. NUREG-1437, Rev. 0, evaluated the groundwater pathway dose, based on the analysis in NUREG-0440. NUREG-0440 analyzed a core meltdown that contaminated groundwater and subsequently contaminated surface water. NUREG-0440 did not analyze direct groundwater drinking at small river sites because of the limited number of potable groundwater wells. Therefore, Subsection 5.3.3.4.1 of NUREG-1437, Rev. 0, concludes that the dose from the groundwater pathway for small river sites is considered to be "minor or nonexistent." As stated previously, the CRN Site is a good approximation of the generic small river site examined in NUREG-0440. Appendix E of NUREG-1437, Rev. 1, provides no updated information on the evaluation of the groundwater pathway.

7.2.4 Health Risks

Based on the total calculated dose risk from the SMR at the CRN Site considered in this analysis, the risk of early fatalities to the 50-mi population was calculated to be 2.00E-11 fatalities/Ryr and the risk of latent cancer fatalities to the 50-mi population was calculated by MACCS2 for the 2-mi EPZ to be 4.09E-06 fatalities/Ryr. These fatality risks are lower than the fatality risks presented in the Final Environmental Impact Statements (FEIS) for recently approved reactor license applications. In NUREG-1872 fatality risks are reported as1.9E-10 early fatalities/Ryr and 1.9E-05 latent fatalities/Ryr. In NUREG-1939 fatality risks are reported as 2.4E-08 early fatalities/Ryr and 6.4E-05 latent fatalities/Ryr, respectively. While these risks are site-specific and dependent on local meteorology and regional populations, CRN Site risks are considered comparable to other facilities.

In addition, the MACCS2 computer code estimated the average individual fatality risks to be 1.27E-13 per Ryr from early fatalities within about one mi of CRN Site and 9.12E-12 per year from latent cancer fatalities within 10 mi. These risks are well below the safety goals for the average individual early fatality and latent cancer fatality risks set by the NRC in the Safety Goal Policy Statement (51 FR 30028) - less than 0.1 percent of risk resulting from other accidents. As indicated in draft NUREG-2168, *Environmental Impact Statement for an Early Site Permit (ESP) at the PSEG Site, Final Report*, the individual risk of a prompt fatality from all other accidents to which members of the United States population are generally exposed is about 4.1E-04 per year, and the sum of cancer fatality risks resulting from all other causes for an individual is taken to be the cancer fatality rate in the United States, which is about 1 in 500 or 2E-03 per year. The risks estimated for the CRN Site are much less than one-tenth of one percent of these everyday public risks.

7.2.5 Conclusions

These estimates of the environmental impacts of severe accidents are considered to be bounding for each of the four SMR designs; the power levels of the other SMR designs are lower than the power level of the SMR considered in this analysis. Also, as provided in Tables 7.2-5 and 7.2-6, the 50-mi population dose risks and the population fatality risks are less than those calculated for other operating reactors or new reactors currently under construction and the individual fatality risks are several orders of magnitude below the NRC Safety Goals.

Based on the discussions in the subsections above, these environmental impacts are concluded to be SMALL.

7.2.6 References

Reference 7.2-1. Sandia National Laboratories, "Code Manual for MACCS2," 1998.

Reference 7.2-2. U.S. Nuclear Regulatory Commission, Westinghouse AP1000 Design Control Document Rev. 19 (Chapter 19, Section 19.59), Website: http://pbadupws.nrc.gov/docs/ML1117/ML11171A411.pdf, June 21, 2011.

Reference 7.2-3. McFadden, K., Bixler, N. E., Eubanks, Lee, and Haaker, R., "WinMACCS, Calculating Health and Economic Consequences from Radioactive Material Accidents," 2007.

Table 7.2-1
Bounding CRN Site SMR Release Category Relative Frequencies

Release Category	Description	Relative Frequency (%)
IC	Intact Containment	91.9
BP	Containment Bypass	4.37
CFE	Early Containment Failure	3.11
CI	Containment Not Isolated	0.55
CFI	Intermediate Containment Failure	0.08
CFL	Late Containment Failure	0.000001
	Total	100

Table 7.2-2
Representative CRN Site SMR Chemical Group Assignment

Group	Nuclides
1	Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135
2	I-131, I-132, I-133, I-134, I-135
3	Rb-86, Cs-134, Cs-136, Cs-137
4	Sb-127, Sb-129, Te-127, Te-127m, Te-129, Te-129m, Te-131m, Te-132
5	Sr-89, Sr-90, Sr-91, Sr-92
6	Mo-99, Tc-99m, Ru-103, Ru-105, Ru-106, Rh-105
7	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Am-241, Cm-242, Cm-244
8	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu240, Pu-241
9	Ba-139, Ba-140

Table 7.2-3
Representative CRN Site SMR Source Term Release Fractions

Release Category	Plume No.	Group No. 1	Group No. 2	Group No. 3	Group No. 4	Group No. 5	Group No. 6	Group No. 7	Group No. 8	Group No. 9
Intermediate Containment Failure (CFI)	1	5.40E-01	3.19E-03	3.18E-03	4.18E-04	2.11E-02	9.11E-03	3.53E-03	2.64E-05	1.62E-02
	2	2.58E-01	1.35E-04	1.35E-04	1.67E-05	6.50E-04	1.68E-04	4.53E-03	1.68E-05	3.40E-04
	3	8.40E-02	0.00E+00	0.00E+00	4.47E-06	0.00E+00	0.00E+00	6.00E-03	2.17E-05	0.00E+00
	4	3.83E-02	0.00E+00	0.00E+00	1.57E-06	0.00E+00	0.00E+00	5.22E-03	1.89E-05	0.00E+00
Early Containment Failure (CFE)	1	4.16E-01	5.53E-02	5.37E-02	1.23E-03	3.14E-03	1.16E-02	5.57E-05	9.54E-07	4.63E-03
	2	4.05E-01	1.26E-03	1.21E-03	1.61E-04	3.43E-04	2.58E-03	9.66E-06	4.56E-08	6.45E-04
	3	1.08E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	4	3.43E-02	0.00E+00	0.00E+00	6.04E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Intact Containment (IC)	1	9.83E-04	1.20E-05	1.15E-05	8.04E-07	1.07E-05	1.31E-05	1.35E-06	5.85E-09	1.20E-05
	2	4.93E-04	0.00E+00	0.00E+00	4.83E-09	0.00E+00	0.00E+00	6.00E-09	3.20E-11	0.00E+00
	3	3.94E-04	0.00E+00	0.00E+00	1.21E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	4	7.72E-04	0.00E+00	0.00E+00	6.04E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Containment Bypass (BP)	1	1.00E+00	1.69-01	1.62E-01	6.27E-03	3.57E-03	4.48E-02	1.30E-04	3.19E-06	8.93E-03
	2	0.00E+00	4.64E-02	3.38E-02	3.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.00E-06
	3	0.00E+00	2.31E-01	6.60E-02	5.32E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	4	0.00E+00	2.80E-03	9.96E-03	1.57E-03	0.00E+00	0.00E+00	0.00E+00	1.00E-06	0.00E+00
Containment Isolation Failure (CI)	1	5.73E-01	4.56E-02	2.10E-02	1.64E-03	2.03E-02	4.04E-02	2.39E-04	2.97E-06	3.16E-02
	2	1.13E-01	0.00E+00	0.00E+00	1.15E-05	0.00E+00	0.00E+00	1.00E-07	0.00E+00	0.00E+00
	3	5.66E-02	0.00E+00	0.00E+00	8.10E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	4	2.74E-02	0.00E+00	0.00E+00	1.27E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Late Containment Failure (CFL)	1	3.36E-04	1.20E-05	1.15E-05	1.00E-06	1.57E-05	1.68E-05	9.96E-07	7.41E-09	1.61E-05
	2	1.19E-03	5.00E-08	3.23E-08	1.75E-08	1.04E-06	2.90E-07	1.07E-05	4.05E-08	6.60E-07
	3	9.79E-01	2.13E-05	1.16E-05	2.47E-05	2.39E-03	1.26E-03	9.75E-02	3.68E-04	2.25E-03
	4	0.00E+00	0.00E+00	2.56E-07	1.20E-05	4.42E-04	1.55E-04	4.39E-02	1.66E-04	3.46E-04

Table 7.2-4
Environmental Impacts with a 50-Mile Radius for Severe Accidents at CRN Site

Release Category	Population Dose Risk (person-Sv per Ryr)		Risk of Fatalities (fatalities per Ryr)		Economic Cost (dollars per Ryr)	Farmland Decontamination (hectares per Ryr)
	Water Ingestion	Total	Early	Latent		
Containment Bypass (BP)	1.01E-06	6.12E-05	1.77E-11	3.19E-06	2.42E+01	1.35E-04
Early Containment Failure (CFE)	1.55E-07	1.26E-05	0.00E+00	6.57E-07	4.50E+00	3.08E-05
Containment Isolation Failure (CI)	2.18E-08	2.54E-06	2.28E-12	1.97E-07	5.73E-01	3.86E-06
Intact Containment (IC)	1.94E-09	4.79E-07	0.00E+00	2.21E-08	2.53E-02	3.40E-10
Intermediate Containment Failure (CFI)	2.07E-09	3.84E-07	4.06E-15	2.18E-08	4.09E-02	2.81E-07
Late Containment Failure (CFL)	4.50E-13	1.52E-09	0.00E+00	8.25E-11	6.05E-04	3.90E-09
Total	1.19E-06	7.71E-05	2.00E-11	4.09E-06	2.93E+01	1.69E-04

Table 7.2-5
Comparison of Environmental Risks for the PPE with Risks for Current-Generation Reactors at Five Sites Evaluated in NUREG−1150

Reactor Facility	Core Damage Frequency (/Ryr)	50-mi Population Dose Risk (Person-Sv/Ryr)	Fatalities (/Ry)		Average Individual Fatality Risk (/Ryr)	
			Early	Latent	Early	Latent Cancer
Grand Gulf[1]	4.0E-06	5E-01	8E-09	9E-04	3E-11	3E-10
Peach Bottom[1]	4.5E-06	7E+00	2E-08	5E-03	5E-11	4E-10
Sequoyah[1]	5.7E-05	1E+01	3E-05	1E-02	1E-08	1E-08
Surry[1]	4.0E-05	5E+00	2E-06	5E-03	2E-08	2E-09
Zion[1]	3.4E-04	5E+01	4E-05	2E-02	9E-09	1E-08
PPE at the CRN Site[2]	4.7E-08	8E-05	2E-11	4E-06	1E-13	9E-12
NRC Safety Goals[3]	NA	NA	NA	NA	4E-07	2E-06

1 Risks were calculated using the MACCS code and presented in NUREG-1150.
2 Risks were calculated with MACCS2 code using CRN Site site-specific input.
3 Provided by the NRC in the Safety Goal Policy Statement (51 FR 30028).

Note:
NA = Not Applicable

Table 7.2-6
Comparison of Environmental Risks from Severe Accidents for PPE with Risks for Current Nuclear Power Plants Undergoing Operating License Renewal Review

	Core Damage Frequency (per year)	50-mi Population Dose Risk (person-Sv/Ryr)
Current Reactor Maximum[1]	2.4E-04	6.9E-01
Current Reactor Mean[1]	3.1E-05	1.5E-01
Current Reactor Median[1]	2.5E-05	1.3E-01
Current Reactor Minimum[1]	1.9E-06	3.4E-01
AP1000 Reactor at Summer site[2]	2.4E-07	1.0E-03
AP1000 Reactor at Vogtle site[3]	2.4E-07	2.8E-04
PPE at the CRN Site[4]	4.7E-08	7.7E-05

[1] Based on MACCS calculations for over 70 current plants at over 40 sites (NUREG-2168).
[2] NUREG-1939 (FEIS for V.C. Summer COL)
[3] NUREG-1872 (FEIS for Vogtle ESP)
[4] Calculated with MACCS code using CRN Site-specific input.

7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES

This section is not required for an Early Site Permit Application.

7.4 TRANSPORTATION ACCIDENTS

This section describes the environmental impacts of postulated transportation accidents involving the shipment of radioactive materials including unirradiated fuel, irradiated (spent) fuel and radioactive waste to and from the Clinch River Nuclear (CRN) Site and alternative sites discussed in detail in Section 9.3. The evaluations in this section assume that all fuel and radioactive waste shipments are by truck.

Because a small modular reactor (SMR) technology has not been selected, a plant parameter envelope (PPE) has been developed for use in evaluating potential environmental impacts. The PPE is described in Sections 3.1 and 3.2. The SMR technologies being considered for the CRN Site, which are based on a pressurized water reactor (PWR) design, are: BWXT mPower, Holtec, NuScale, and Westinghouse. The PPE is based on the values of fuel-related parameters for the four SMR technologies.

The NRC evaluated the environmental effects of fuel and waste transportation for light water reactors (LWRs) in WASH-1238, “Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Plants,” and NUREG-75/038, *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants*, Supplement 1, and found the impacts to be SMALL (Reference 7.4-1). These documents provide the basis for Table S-4 in Title 10 of the Code of Federal Regulations (10 CFR) 51.52, which summarizes the environmental impacts of fuel and waste transportation to and from one reference LWR. The impacts are provided for normal transport conditions and accidents in transport assuming a 1100 megawatt electric (MWe) LWR with a capacity factor of 0.8, referred to as the “reference reactor.”

As stated in 10 CFR 51.52:

> “Under § 51.50, every environmental report prepared for the construction permit stage or early site permit stage or combined license stage of a light-water-cooled nuclear power reactor, and submitted after February 4, 1975, shall contain a statement concerning transportation of fuel and radioactive wastes to and from the reactor. That statement shall indicate that the reactor and this transportation either meet all of the conditions in paragraph (a) of this section or all of the conditions of paragraph (b) of this section.”

10 CFR 51.52(a)(1) through (5) delineate specific conditions the license applicant’s proposed reactor(s) must meet to use Table S-4 as part of its Environmental Report (ER):

1) The reactor has a core thermal power level not exceeding 3800 megawatt thermal (MWt).
2) Fuel is in the form of sintered uranium oxide pellets having a uranium-235 enrichment not exceeding 4 percent by weight; and the pellets are encapsulated in zircaloy rods.

3) The average level of irradiation of the fuel from the reactor does not exceed 33,000 megawatt days per metric tons uranium (MWD/MTU), and no irradiated fuel assembly is shipped until at least 90 days after it is discharged from the reactor.

4) With the exception of irradiated fuel, all radioactive waste shipped from the reactor is packaged and in solid form.

5) Unirradiated fuel is shipped to the reactor by truck; irradiated (spent) fuel is shipped from the reactor by truck, rail, or barge, and radioactive waste other than irradiated fuel is shipped from the reactor by truck or rail.

For reactors not meeting all of the conditions in 10 CFR 51.52 paragraph (a), paragraph (b) requires a further analysis of the transportation effects.

The PPE differs from conditions 2 and 3 of 10 CFR 51.52(a). As provided in Table 3.1-2, Item 18.1, the fuel enrichment for the PPE could be as high as 5 percent by weight of uranium-235. Also, as provided in Table 3.1-2, Item 18.0.1, the maximum average discharge batch irradiation (burnup) of the irradiated fuel could be as high as 51 gigawatt days per metric tons uranium (GWD/MTU) (51,000 MWD/MTU). Therefore, 10 CFR 51.52 (b) requires "... a full description and detailed analysis of the environmental effects of transportation of fuel and wastes to and from the reactor, including values for the environmental impact under normal conditions of transport and for the environmental risk from accidents in transport. The statement shall indicate that the values determined by the analysis represent the contribution of such effects to the environmental costs of licensing the reactor."

A comparison of fuel and radioactive waste parameters in Table 3.1-2 to the reference reactor parameters in Table S-4, including a discussion of the acceptability of the parameters that differ from Table S-4, is provided in Subsection 5.7.2.1. As discussed in Subsection 3.2.1, the per reactor unit thermal output of the SMR technologies being considered varies from approximately 160 MWt to 805 MWt, with a site total of 1920 MWt to 2420 MWt. Table 3.1-2, Item 16.6 provides the generating output of the SMRs at the CRN Site as 800 MWe.

The reference reactor for Table S-4 is an 1100-MWe LWR with a capacity factor of 80 percent (1100 MWe times 80 percent equals 880 MWe). Table 3.1-2, Item 16.6 shows the generating output of the SMRs at the CRN Site as 800 MWe, and Item 16.4 shows a station capacity factor of 90 percent (800 MWe times 90 percent equals 720 MWe). In each subsection below, the expected number of shipments is multiplied by the ratio, 1.22, to estimate the number of shipments normalized to the reference reactor used in Table S-4.

7.4.1 Radiological Impacts

Accident risks are the product of accident frequency and the consequence of the accident. According to NUREG-1815, *Environmental Impact Statement for an Early Site Permit (ESP) at the Exelon ESP Site*, Appendix G, accident frequencies today are likely to be lower than those used in the WASH-1238 analysis, because traffic accident, injury, and fatality rates have fallen over the past 30 years (yr).

7.4.1.1 Transportation of Unirradiated Fuel

The following assumptions are made in this analysis of the transportation of unirradiated fuel:

- Unirradiated fuel would be transported to the CRN Site via truck in robust packages designed to protect the fuel from damage from dropping or puncture.
- The WASH-1238 analysis of postulated accidents during the transportation of unirradiated fuel found accident impacts to be negligible.
- As noted in NUREG-1815, accident frequencies are likely to be lower in the future than those used in the analysis in WASH-1238 because traffic accident, injury, and fatality rates have fallen since the initial analyses were performed.
- Advanced fuel behaves like fuel evaluated in the analyses provided in WASH-1238.
- Per NUREG-1815, there is no significant difference in the consequences of accidents severe enough to result in a release of unirradiated fuel particles to the environment between advanced LWRs and previous-generation LWRs because the fuel form, cladding, and packaging are similar to those analyzed in WASH-1238.
- The fuel form, cladding, and packaging for the SMR designs considered in the PPE would be similar to the fuel form, cladding, and packaging for advanced LWRs.

Based on this information, the dose impact from nuclides released from postulated accidents involving new fuel is assumed to be negligible when compared to dose from postulated irradiated fuel and radiation waste transportation accidents. Therefore, quantitative analysis of dose from new fuel accidents was not performed.

The radiological impacts from incident free transportation of unirradiated fuel were estimated using the WebTRAGIS 6.0 and RADTRAN 6.5 computer codes (Reference 7.4-3; Reference 7.4-4). The evaluation model assumes that unirradiated fuel is shipped from a fuel fabrication facility located in Richland, Washington, to the CRN Site. The distance from Richland, Washington, to the CRN Site was determined to be 2451 miles (mi; 3944 kilometers [km]) by the WebTRAGIS computer code for a commercial road route. The fuel fabrication facility in Richland is the farthest fabrication facility in the United States from the CRN Site that is currently in operation; therefore, to maximize the transportation distance and potential impacts, it was used as a representative fuel fabrication facility for the purposes of the evaluation. The dose impacts from incident free transportation of unirradiated fuel are summarized in Subsection 5.7.2.2.

7.4.1.2 Transportation of Irradiated Fuel

In accordance with 10 CFR 51.52(a), a full description and detailed analysis of transportation impacts is not required when licensing an LWR (i.e., impacts are assumed to be bound by table S-4) if the reactor meets the following criteria:

- The reactor has a core thermal power level not exceeding 3800 MWt.

- Fuel is in the form of sintered Uranium Dioxide (UO_2) pellets having a uranium-235 enrichment not exceeding 4 percent by weight; and pellets are encapsulated in zircaloy-clad fuel rods.
- The average level of irradiation of the fuel from the reactor does not exceed 33 GWD/MTU, and no irradiated fuel assembly is shipped until at least 90 days after it is discharged from the reactor.
- With the exception of irradiated fuel, all radioactive waste shipped from the reactor is packaged and in solid form.
- Unirradiated fuel is shipped to the reactor by truck; irradiated (spent) fuel is shipped from the reactor by truck, railroad, or barge; and radioactive waste other than irradiated fuel is shipped from the reactor by truck or railroad.

While the SMR design to be deployed at the CRN Site has not been selected, the CRN Site would generate power of up to 800 MWe or 2420 MWt (Table 3.1-2, Item 16.1), well below the power criterion above. Fuel for the reactors would be enriched up to 5 weight percent uranium-235 (Table 3.1-2, Item 18.1), and the expected irradiation level is about 51 GWD/MTU (Table 3.1-2, Item 18.0.1), both exceeding the associated 10 CFR 51.52(a) condition. Therefore, a detailed analysis of transportation impacts was performed.

The radiological impacts from incident free transportation of irradiated fuel and transportation accidents were estimated using the WebTRAGIS and RADTRAN and the accident, injury and fatality rates provided in Table 7.4-1. The irradiated fuel transportation evaluation model assumes that irradiated fuel is shipped by truck to the geological spent fuel repository previously proposed for construction at Yucca Mountain, Nevada. Although the Yucca Mountain repository may no longer be considered a probable shipment location, the impacts of the transportation of spent fuel to Yucca Mountain provide a reasonable estimate of the transportation risks to a monitored retrievable storage facility because of the distances involved. The distance from the CRN Site to the potential repository is 2292 mi (3689 km) as determined by the WebTRAGIS computer code for a highway route controlled quantity (HRCQ) route. Because of the distance from the CRN Site to Yucca Mountain, the impacts of shipments to a regional spent fuel storage facility are considered to be bounded by the transportation analysis for Yucca Mountain. The dose impacts from incident free transportation of irradiated fuel are summarized in Subsection 5.7.2.2.

The initial irradiated fuel activity is decayed five years to account for the minimum decay period prior to shipment to the repository. The NRC has used the five-year decay period in its evaluated the environmental effects of extending fuel burnup in NUREG/CR-6703, *Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU*. The source term in curies per MTU used for the analysis (i.e., with 5 yr decay) is provided in Table 7.4-2. This source term is based on the radionuclide inventory for irradiated AP1000 reactor fuel provided in NUREG-1939, *Final Environmental Impact Statement (EIS) for Combined License for Virgil C.*

Summer Nuclear Station, Units 2 and 3, Table 6-10, Radioactive Inventories Used in Transportation-Accident Risk Calculations for the Westinghouse AP1000.

The AP1000 and the SMR designs considered in the PPE are PWRs with sintered uranium dioxide fuel pellets in Zircaloy cladding. In the absence of a selected design, a surrogate source term (in curies per MTU) is provided in Table 7.4-2. This source term is based on the radionuclide inventory for irradiated AP1000 reactor fuel from NUREG-1939. This inventory was selected as representative based on its similarity in design, i.e., both AP1000 and SMR designs considered in the PPE are PWRs with sintered uranium dioxide fuel pellets in zircaloy cladding. The enrichment for the CRN Site fuel may be as high as 5 percent U-235 by weight, slightly higher than the maximum enrichment of 4.45 U-235 percent by weight for the AP1000 (NUREG-1939). As provided in Table 3.1-2, Item 18.0.1, the maximum assembly average burnup at end of assembly life is 51 GWD/MTU for the CRN Site PPE, approximately the same as the average 50.5 GWD/MTU burnup for the AP1000 fuel (NUREG 1939).

The source term inventory provided in Table 7.4-2 includes cobalt-60, which is used to represent fuel rod surface contamination by corrosion-related unidentified deposits (CRUD). NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, concluded that cobalt-60 is the dominant contributor to the dose from fuel rod surface contamination. The accident severity categories, severity fractions, and release fractions for gas, volatiles, particulates, and CRUD used in the RADTRAN analyses are presented in Table 7.4-3. Table 7.4-3 was obtained from Table 8, “Severity and release fractions for uncanistered truck-transported PWR spent fuel,” in the RADTRAN 6/RadCat 6 User Guide, Rev. 1. This table was adapted from the U.S. Department of Energy (DOE), “Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.”

As provided in Table 3.1-2, Item 18.0.2, each fuel assembly is assumed to contain 0.304 MTU, and each shipment of irradiated fuel is assumed to contain 4 fuel assemblies or 1.22 MTU, assuming a standard GA4 cask as provided in NUREG 2125, Appendix B. As discussed in Subsection 5.7.2.1.11 of the ER and provided in Table 5.7-7, an average of 46 shipments of irradiated fuel per year for a total of 56.1 MTU is expected. Normalized to the reference reactor used to estimate the parameters in 10 CFR Part 51, Table S-4, 56.1 MTU per year is divided by 0.5 MTU per shipment and 0.82 for the ratio of net power of the CRN Site SMRs to the reference reactor. Therefore, the accident doses calculated by RADTRAN were multiplied by 137 normalized shipments per year to obtain a dose risk value. Using this calculation, the population dose risk from transportation accidents involving irradiated fuel is 2.44E-04 person-rem per year, as summarized in Table 7.4-4. This total dose risk is lower than the reference reactor and estimates for new reactors provided in recent EISs published by the NRC, such as NUREG-2176, *Environmental Impact Statement for Combined Licenses (COL) for Turkey Point Nuclear Plant Units 6 and 7*.

7.4.1.3 Transportation of Radioactive Waste

This subsection discusses the environmental effects of transporting radioactive waste other than spent fuel from the CRN Site. The environmental conditions listed in 10 CFR 51.52 that apply to shipments of radioactive waste are as follows:

- Radioactive waste (except spent fuel) would be packaged and in solid form.
- Radioactive waste (except spent fuel) would be shipped from the reactor by truck or railroad.
- The weight limitation of 73,000 pounds (lb) per truck and 100 tons per cask per railcar would be met.
- Traffic density would be less than the one truck shipment per day or three railcars per month condition.

Radioactive waste other than spent fuel from the CRN Site would be shipped in compliance with federal or state weight restrictions. The sum of the daily shipments of unirradiated fuel, spent fuel, and radioactive waste for the CRN Site PPE is less than the one-truck-shipment-per-day condition given in 10 CFR 51.52, Table S–4.

As provided in Table 3.1-2, Item 11.2.3, the annual volume of radioactive waste generated and shipped from the CRN Site would be 5000 cubic feet per year (ft^3/yr). Table 5.7-7 shows the expected number of radioactive waste shipments from the CRN Site is 61 shipments per year. Multiplying by the ratio of 1.22, discussed above, the estimated number of shipments per year is 75, normalized to the reference reactor used to estimate the parameters in 10 CFR Part 51, Table S-4. Table 3.5-5 provides the projected annual radioisotope inventory of the principal radionuclides in solid radioactive waste produced at the CRN Site. Radionuclides in the list with a radiological half-life greater than 2 days were included in the source term for the RADTRAN calculations.

Tennessee Valley Authority (TVA) plans to ship radioactive waste to the Waste Control Specialists disposal facility in Andrews County, Texas. The distance from the CRN Site to the Andrews repository is 1214 mi (1954 km) as determined by the WebTRAGIS computer code for a HRCQ route. The radiological impacts from incident free transportation of radioactive waste and transportation accidents were estimated using the WebTRAGIS and RADTRAN 6.5 computer codes.

The state-specific safety parameter values that were used to estimate the frequencies of accidents involving the trucks carrying radioactive waste were obtained from the same sources discussed in Subsection 7.4.1.2.

The accident severity categories, conditional probabilities, and release fractions for particulates, CRUD, gas, and volatiles used in the analysis of radioactive waste transportation are the same as those used for the irradiated fuel analysis (Table 7.4-3). The accident data in Table 7.4-3 provides a generally representative description of transportation accidents for other radioactive waste shipments that meet U.S. Department of Transportation requirements.

The resulting total dose risk from transportation accidents involving radioactive waste is 3.13E-08 person-rem per reactor year assuming 75 shipments per year normalized to the reference reactor. This result is provided in Table 7.4-4. This population dose risk impact from accidents is small, much lower than the dose to the exposed population along the transportation route provided in Table S-4.

7.4.2 Non-Radiological Impacts

Non-radiological impacts associated with the postulated accidents are calculated for:

- Injuries and fatalities during transportation of unirradiated fuel
- Injuries and fatalities during transportation of irradiated fuel
- Injuries and fatalities during transportation of radioactive waste

The non-radiological impacts from postulated accidents during transportation were evaluated using the WebTRAGIS code to define appropriate routing and the RADTRAN 6 code to calculate the non-radiological impacts (e.g., injuries and fatalities).

The non-radiological impacts were based on round-trip distances because the return of the empty truck is included in the evaluation. Therefore, the frequency (fatalities per reactor-year and injuries per reactor-year) was multiplied by two.

7.4.2.1 Transportation of Unirradiated Fuel

The evaluation model assumes that unirradiated fuel is shipped by truck from Richland, Washington, to the CRN Site. The distance from Richland, Washington, to the CRN Site was determined to be 2451 mi. (3944 km) by the WebTRAGIS computer code for a HRCQ route. The fabrication facility in Richland is the farthest fabrication facility in the United States from the CRN Site that is currently in operation; therefore, it was used as a representative fuel fabrication facility for the purposes of the evaluation.

As discussed in Subsection 5.7.2.1.11 and Table 5.7-6, the total number of lifetime shipments of unirradiated fuel for the CRN Site is 492, and the average is 12.3 shipments per year. Multiplying by the ratio of 1.22, discussed above, the estimated number of shipments per year is 15 (600 total shipments), normalized to the reference reactor used to estimate the parameters in 10 CFR Part 51, Table S-4.

The non-radiological fatality rates and injury rates normalized to the transportation rates for the reference reactor are provided in Tables 7.4-5 and 7.4-6. Subsection 7.4.2.4 discusses the significance of these rates.

7.4.2.2 Transportation of Irradiated Fuel

The routing and accident parameters used to analyze non-radiological impacts of transporting irradiated fuel were the same as those used to analyze the radiological impacts of transporting irradiated fuel described in Subsection 7.4.1.2. As noted above and provided in Table 5.7-7, the number of shipments of irradiated fuel from the CRN Site normalized to the reference reactor would be 137 shipments of irradiated fuel per year.

The non-radiological fatality rates and injury rates normalized to the transportation rates for the reference reactor are provided in Tables 7.4-5 and 7.4-6. Subsection 7.4.2.4 discusses the significance of these rates.

7.4.2.3 Transportation of Radioactive Waste

The routing and accident parameters used to analyze non-radiological impacts of transporting radioactive waste were the same as those used to analyze the radiological impacts of transporting radioactive waste described in Subsection 7.4.1.3.

As shown in Table 3.1-2, Item 11.2.3, the annual volume of radioactive waste generated and shipped from the CRN Site would be 5000 ft^3/yr. Table 5.7-7 shows the number of radioactive waste shipments from the CRN Site to be 61 shipments per year. As noted above and in Table 5.7-7, the number of shipments of radioactive waste (other than spent fuel) normalized to the reference reactor is 75 shipments per year.

The non-radiological fatality rates and injury rates normalized to the transportation rates for the reference reactor are provided in Tables 7.4-5 and 7.4-6. Subsection 7.4.2.4 discusses the significance of these rates.

7.4.2.4 Comparison to 10 CFR 51.52 Table S-4

For an equal comparison to the reference reactor in 10 CFR 51.52 Table S-4, the normalized number of shipments provided in the subsections above were used to determine the non-radiological environmental impacts due to transportation accidents. Tables 7.4-5 and 7.4-6 indicate the fatal and non-fatal injury consequences, respectively, for unirradiated fuel, irradiated fuel, and radioactive waste shipments based on the normalized numbers of shipments. Table 7.4-7 is a comparison of the CRN Site to the summary of “Accidents in Transport” in 10 CFR 51.52 Summary Table S-4. The estimated number of fatal injuries is 2.24E-02 per reactor year for the CRN Site, slightly more than two fatal injuries in 100 reactor years or more than twice the number of fatal injuries assumed for the reference reactor in Table S-4. The estimated number of non-fatal injuries is 3.50E-01 per reactor year (3.5 in 10 reactor years) for the CRN Site, more than three times the value of 1 non-fatal injury in 10 reactor years for the reference reactor in Table S-4. The estimated numbers of fatal injuries and non-fatal injuries for the CRN Site are higher than the values for the reference reactor because the one-way shipping distances for unirradiated fuel, irradiated fuel, and radioactive waste shipments are more than twice the distances assumed in the analyses for Table S-4 (WASH-1238). Considering these differences in the analyses, the impacts are comparable. Therefore, as the Table S-4 values are considered SMALL, the estimated numbers of fatal injuries and non-fatal injuries for the CRN Site are also SMALL.

7.4.3 Summary and Conclusion

A detailed accident analysis of the environmental impacts for the transportation of unirradiated fuel, irradiated fuel, and radioactive waste transported to and from the CRN Site was performed in accordance with 10 CFR 51.52(b).

The results of the radiological accident analysis are summarized in Table 7.4-4, and the results of the non-radiological accident analysis are summarized in Tables 7.4-5 and 7.4-6. The values determined by these analyses represent the environmental impacts of licensing SMRs at the CRN Site.

As discussed in Subsections 7.4.1.2 and 7.4.1.3, because the number of normalized shipments of irradiated fuel and radioactive waste provided in Table 5.7-7 are not significantly different from number of shipments from the reference reactor, the impacts from radiological accidents from the CRN Site are consistent with the “Small” impacts designation provided in Table S-4. The calculated dose risks provided in Table 7.4-4 are also SMALL. As discussed in Subsection 7.4.2.4, the non-radiological accident environmental impacts related to transportation of unirradiated fuel, irradiated fuel, and radioactive waste are also consistent with the Table S-4 fatality and nonfatal injury rates.

Therefore, the overall corresponding impacts from accidents associated with the transportation of fuel and waste to and from the CR SMR Project are SMALL.

7.4.4 References

Reference 7.4-1. U.S. Atomic Energy Commission, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants (WASH-1238)," December, 1972.

Reference 7.4-2. Weiner, Ruth F., Hinojosa, Daniel, Heames, Terence, and Farnum, Cathy O., "RADTRAN 6/RadCat6 User Guide Rev 1," 2015.

Reference 7.4-3. UT-Battelle, LLC, Transportation Routing Analysis Geographic Information System (TRAGIS), Version 6.0. U.S. Department of Energy Contract No. DE-AC05-00OR22725. Website: https://webtragis.ornl.gov/tragis/app/map/view, 2017.

Reference 7.4-4. Sandia National Laboratories, "RADTRAN 6 Technical Manual," SAND2013-0780, January 2014.

**Table 7.4-1
CRN Site Model
Accident, Fatality and Injury Rates**

State	Accident Rate (Accident/km)	Fatality Rate (Fatality/km)	Fatality Rate (Fatality/ Accident)	Injury Rate (Injury/km)	Injury Rate (Injury/Accident)
AR	2.20E-07	9.73E-09	4.43E-02	1.18E-07	5.35E-01
AZ	2.16E-07	1.48E-08	6.82E-02	1.40E-07	6.49E-01
CA	2.62E-07	1.10E-07	4.20E-02	1.44E-07	5.50E-01
CO	7.31E-07	1.79E-08	2.45E-02	3.78E-07	5.17E-01
IA	1.84E-07	1.48E-08	8.03E-02	1.03E-07	5.62E-01
ID	4.84E-07	5.97E-09	1.23E-02	3.68E-07	7.61E-01
IL	3.64E-07	1.30E-08	3.58E-02	1.80E-07	4.94E-01
KS	4.66E-07	8.16E-09	1.75E-02	3.05E-07	6.54E-01
KY	5.08E-07	2.01E-08	3.95E-02	2.65E-07	5.22E-01
MO	7.61E-07	1.95E-08	2.56E-02	3.77E-07	4.95E-01
NE	5.23E-07	2.15E-08	4.11E-02	2.36E-07	4.52E-01
NM	1.85E-07	1.85E-08	1.00E-01	1.38E-07	7.46E-01
NV	3.69E-07	1.04E-08	2.81E-02	1.78E-07	4.81E-01
OK	4.40E-07	2.09E-08	4.75E-02	3.47E-07	7.89E-01
OR	3.54E-07	3.20E-08	9.04E-02	1.63E-07	4.61E-01
TN	2.02E-07	1.57E-08	7.78E-02	1.10E-07	5.47E-01
TX	9.84E-07	2.04E-08	2.07E-02	6.55E-07	6.66E-01
UT	4.76E-07	1.87E-08	3.93E-02	3.04E-07	6.38E-01
WA	4.35E-07	2.83E-09	6.50E-03	2.16E-07	4.97E-01
WY	1.11E-06	1.70E-08	1.53E-02	3.88E-07	3.51E-01

Table 7.4-2
CRN Site Model
Irradiated Fuel Source Term

Nuclide	Activity (Ci/MTU)
Am-211	7.27E+02
Am-242m	1.31E+01
Am-234	3.34E+01
Ce-144	8.87E+03
Pr-144 (D)[1]	-
Pr-144m (D)[1]	-
Cm-242	2.83E+01
Cm-243	3.07E+01
Cm-244	7.75E+03
Cm-245	1.21E+00
Co-60 (CRUD)	4.09E+00
Cs-134	4.80E+04
Cs-137	9.31E+04
Ba-137m (D)[1]	-
Eu-154	9.13E+03
Eu-155	4.62E+03
I-129	4.65E-02
Kr-85	8.9E+03
Pm-147	1.76E+04
Pu-238	6.07E+03
Pu-239	2.55E+02
Pu-240	5.43E+02
Pu-241	6.96E_04
Pu-242	1.82E+00
Ru-106	1.55E+04
Rh-106(D)[1]	-
Sb-125	3.83E+03
Sr-90	6.19E+04
Y-90	6.19E+04

[1] The nuclides labeled with a (D) are daughter products and are included with the parent in the RADCAT/RADTRAN program.

Source: NUREG-1939, Table 6-10

Table 7.4-3
CRN Site Model
Severity and Release Fractions for Uncanistered Truck-Transported Fuel

Severity Category	Severity Fraction	Gas Release Fraction	Volatiles Release Fraction	Particulates Release Fraction	CRUD Release Fraction
0	0.99993	0	0	0	0
1	6.06E-05	9.0E-02	1.5E-06	3.36E-06	1.00E-03
2	5.86E-06	9.0E-02	1.5E-06	3.36E-06	1.00E-03
3	4.95E-07	9.0E-02	1.5E-06	3.36E-06	1.00E-03
4	7.49E-08	9.0E-02	1.5E-06	3.36E-06	1.00E-03
5	3.00E-10	9.0E-02	6.75E-07	1.54E-06	1.00E-03

Table 7.4-4
CRN Site Model
Radiological Accident Analysis Results
(per CRN Site operating year)

Environmental Impact	Unirradiated Fuel	Irradiated Fuel	Radioactive Waste	Total
Annual Dose for CRN Site	Not Calculated	2.44E-04 Person-rem/yr	3.13E-08 Person-rem/yr	2.44E-04 Person-rem/yr

Table 7.4-5
CRN Site Model
Non-Radiological Accident Analysis Results for
Normalized Number of Shipments: Fatalities

	Fatalities per Shipment	Normalized Shipments Per Year	Fatalities per Year[1]	Fatalities per 100 Years
New Fuel	6.08E-05	15	1.82E-03	1.82E-01
Spent Fuel	5.73E-05	137	1.57E-02	1.57E+00
Radioactive Waste	3.24E-05	75	4.86E-03	4.86E-01
Total	-	227	2.24E-02	2.24E+00

[1] The fatalities per year are calculated assuming a round trip for the truck. Therefore the normalized number of shipments was doubled when calculating total route fatalities.

Table 7.4-6
CRN Site Model
Non-Radiological Accident Analysis Results for Normalized Number of Shipments: Injuries

	Injuries per Shipment	Normalized Shipments Per Year	Injuries per Year[1]	Injuries per 10 Years
New Fuel	1.18E-03	15	3.54E-02	3.54E-01
Spent Fuel	7.55E-04	137	2.07E-01	2.07E+00
Radioactive Waste	7.21E-04	75	1.08E-01	1.08E+00
Total	-	227	3.50E-01	3.50E+00

[1] The fatalities per year are calculated assuming a round trip for the truck. Therefore the normalized number of shipments was doubled when calculating total route injuries.

Table 7.4-7
CRN Site Model Comparison to
10 CFR 51.52 Summary Table S-4: “Accidents in Transport”
Bounding Technology Summary

Types of Effects	Environmental Risk	CRN Site Model
	10 CFR 51.52 Table S-4 Reference Reactor	**CRN Site SMRs**
Radiological effects of transportation of unirradiated fuel, irradiated fuel, and radioactive waste Person-rem per reactor-year	Small	2.44E-04
Non-radiological effects of transportation of unirradiated fuel, irradiated fuel, and radioactive waste	1 fatal injury in 100 reactor years 1 nonfatal injury in 10 reactor years	2.24E+00 (fatalities per 100 years) 3.50E+00 (injuries per 10 years)