CHAPTER 7
ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

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ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ABWR | advanced boiling water reactor |
| AP600 | Westinghouse AP600 Reactor |
| AP1000 | Westinghouse AP1000 Reactor |
| ATMOS | Input file used by the MACCS2 code |
| BP | containment bypass |
| BWR | boiling water reactor |
| °C | degrees Celsius |
| CDF | core damage frequency |
| CE | combustion engineering |
| CEDE | Committed Effective Dose Equivalent |
| CFE | early containment failure |
| CFI | intermediate containment failure |
| CFL | late containment failure |
| CFR | Code of Federal Regulations |
| CHRONC | input file used by the MACCS2 code |
| CI | containment isolation failure |
| COL | Combined License |
| COLA | Combined License Application |
| CP&L | Carolina Power and Light Company |
| DBA | design basis accident |
| DCD | Westinghouse Electric Company, LLC, AP1000 Design Control Document |

ACRONYMS AND ABBREVIATIONS (CONTINUED)

| | |
|---|---|
| DOE | U.S. Department of Energy |
| DOT | U.S. Department of Transportation |
| EAB | exclusion area boundary |
| EARLY | Input file used by the MACCS2 code |
| EDE | Effective Dose Equivalent |
| EI | exposure index |
| ER | Environmental Report |
| °F | degrees Fahrenheit |
| FES | Final Environmental Statement |
| ft. | foot/feet |
| $ft^3$/sec | cubic feet per second |
| FHA | fuel-handling accident |
| GEIS | Generic Environmental Impact Statement |
| gpm | gallons per minute |
| HAR | proposed Shearon Harris Nuclear Power Plant Units 2 and 3 |
| HAR 2 | proposed Shearon Harris Nuclear Power Plant Unit 2 |
| HAR 3 | proposed Shearon Harris Nuclear Power Plant Unit 3 |
| HNP | existing Shearon Harris Nuclear Power Plant Unit 1 |
| hr. | hour |
| IC | intact containment |
| IEM | Innovative Emergency Management |
| IRWST | in-contaminant refueling water storage tank |
| km | kilometer |

ACRONYMS AND ABBREVIATIONS (CONTINUED)

| | |
|---|---|
| LPGS | Liquid Pathway Generic Study |
| lpm | liter per minute |
| LOCA | Loss of Coolant Accident |
| LPZ | low-population zone |
| m | meter |
| $m^3$/sec | cubic meter per second |
| MACCS2 | MELCOR Accident Consequence Code System, Revsion 2 |
| MACR | Maximum Averted Cost Risk |
| MET | Input file used by the MACCS2 code |
| mi. | mile |
| NEPA | National Environmental Policy Act |
| NRC | U.S. Nuclear Regulatory Commission |
| PEC | Progress Energy Carolinas, Inc. |
| person-rem/yr. | person-roentgen equivalent man per year |
| person-Sev/yr. | person-Sievert per year |
| PRA | probalistic risk assessment |
| PWR | pressurized water reactor |
| RCS | reactor coolant system |
| rem | roentgen equivalent man |
| SAMA | severe accident mitigation alternative |
| SAMDA | severe accident mitigation design alternative |
| sec/$m^3$ | second per cubic meter |

ACRONYMS AND ABBREVIATIONS (CONTINUED)

| | |
|---|---|
| SECPOP | Sector Population |
| SER | Safety Evaluation Report |
| SG | steam generator |
| SITE | Input file used by the MACCS2 code |
| SRP | Standard Review Plan |
| Sv | Sieverts |
| TEDE | Total Effective Dose Equivalent |
| USEPA | U.S. Environmental Protection Agency |
| Westinghouse | Westinghouse Electric Company, LLC |
| yr. | year |

# 7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter evaluates the environmental impacts of postulated accidents involving radioactive materials related to the operation of the proposed Shearon Harris Nuclear Power Plant Units 2 and 3 (HAR) and several appurtenant facilities. These appurtenant facilities include electric transmission lines; an electric switchyard; a Cape Fear River water intake structure and pumphouse; a makeup water pipeline, discharge structure on Harris Reservoir, and blowdown pipeline from the HAR into Harris Reservoir.

The evaluation of environmental impacts of postulated accidents involving radioactive materials includes the following key components:

- Section 7.1 — Design Basis Accidents
- Section 7.2 — Severe Accidents
- Section 7.3 — Severe Accident Mitigation Measures
- Section 7.4 — Transportation Accidents

Each of these topical areas is discussed in detail in the sections of this chapter that follow.

For the purposes of this discussion and consistent with the information presented in Environmental Report (ER) Chapters 2 and 4, the following terms are used:

- **Plant Site**. The plant site is the area within the fence line (Figure 4.0-2). This area includes the footprint of the HAR, including the reactor buildings and generating facilities.

- **HAR Site**. The HAR site is an irregularly-shaped area comprised of the following site components: the plant site (area within the fence line), the area within the Harris Reservoir perimeter, the dam at Harris Reservoir, the area within the perimeter of the Auxiliary Reservoir, the Auxiliary Reservoir dam, the pipeline corridor, and the intake structure and pumphouse (Figure 2.0-2). The HAR site is located within two counties: Wake and Chatham.

- **Exclusion Zone**. The exclusion zone is the area within the exclusion area boundary (EAB). The exclusion zone is defined as two overlapping areas centered on the reactor building of each unit (Figure 4.0-3). The areas are defined by a circular distance of 1600 meters (m) (5249 feet [ft.]) in the seven southerly sectors beginning with ESE clockwise through WSW and 1245 meters (m) (4085 feet [ft.]) in the nine remaining sectors.

- **Pipeline corridor**. The pipeline corridor includes the Harris Lake makeup water system pipeline and corridor connecting the Harris Reservoir and the Cape Fear River. The pipeline components will transport makeup water from the Cape Fear River to Harris Reservoir. Water from the Cape Fear River will be used to increase the water level of Harris Reservoir approximately 6 m (20 ft.) to provide adequate cooling tower makeup water for the HAR (Figure 4.0-4).

- **Intake Structure and Pumphouse**. The Harris Lake makeup water system intake structure and pumphouse will be constructed on the Cape Fear River (Figure 4.0-5).

- **Harris Reservoir**. The Harris Reservoir is also known as the Main Reservoir. It does not include the affiliated Auxiliary Reservoir.

- **Harris Reservoir Perimeter**. The area impacted by the 6-m (20-ft.) change in the reservoir's water level.

- **Transmission Corridors and Off-Site Areas**. Transmission corridors and off-site areas describe areas outside the site boundary that may fall within the footprint of new or existing transmission lines.

- **Vicinity**. The vicinity is a band or belt 9.7-km (6-mi.) wide surrounding the HAR site. The vicinity includes a much larger tract of land than the HAR site. The vicinity is located within four counties: Wake, Chatham, Harnett, and Lee.

- **Region**. As stated in the introduction section of ER Chapter 5, the region applies to the area between a 9.7-km (6-mi.) radius and an 80-km (50-mi.) radius from the center point of the HAR power block footprint (Figure 4.0-6).

## 7.1 DESIGN BASIS ACCIDENTS

The purpose of this section is to provide a comparison of the off-site dose consequences and resulting health effects for design basis accidents (DBAs), as identified in the Westinghouse Electric Company, LLC, AP1000 Design Control Document (DCD) and those contained in Section 15 of the Safety Evaluation Report (SER). The following sections contain information to meet the requirements specified in Chapter 7 of NUREG-1555. More specifically these include:

- The list of DBAs identified in the AP1000 DCD having a potential for release to the environment and analysis of the dose consequences from these accidents.

- The list of DBAs considered in the staff's safety evaluation and the analysis of the magnitude of the source-term for off-site releases (from Chapter 15 of the SER).

- The 50$^{th}$ percentile normalized concentrations (X/Q) at appropriate distances from the effluent release points for the HAR.

### 7.1.1 SELECTION OF DESIGN BASIS ACCIDENTS

The DBAs considered in this section are from the DCD and are consistent with the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183 and NUREG-1555. Table 7.1-1 lists the DBAs having the potential for releases to the environment and provides an initial evaluation of each accident. The radiological consequences of the DBAs listed in Table 7.1-1 are assessed to demonstrate that two new AP1000 units can be sited at the HAR site without undue risk to the health and safety of the public.

### 7.1.2 EVALUATION METHODOLOGY

Doses for the selected DBAs were evaluated at the HAR EAB and low population zone (LPZ). The AP1000 DCD presents the radiological consequences for the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with HAR site data to demonstrate that the DCD analyses are bounding for the HAR site. The basic scenario for each accident is that some quantity of activity is released at the accident location inside a building, and this activity is eventually released to the environment. The transport of activity within the plant is independent of the site and specific to the AP1000 design. Details about the methodologies and assumptions pertaining to each of the accidents are provided in the DCD. These doses must meet the site acceptance criteria in 10 Code of Federal Regulations (CFR) 50.34 given as follows:

(1) An individual, located at any point on the boundary of the exclusion area for any 2-hour (hr.) period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25-roentgen equivalent man (rem) Total Effective Dose Equivalent (TEDE).

(2) An individual, located at any point on the outer boundary of the low-population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25-rem TEDE.

These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation (for example, a large-break Loss of Coolant Accident [LOCA]). For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the additional acceptance criteria provided in Regulatory Guide 1.183. The dose acceptance criteria from Regulatory Guide 1.183, with one exception, are listed in Table 7.1-2. No dose limit is listed in Regulatory Guide 1.183 for the small line break outside containment. Therefore, the criterion was adopted from Section 15.6.2 of the DCD consistent with Section 15.6.2 of NUREG-0800. The dose limits ensure that the consequences of each DBA are acceptable from an overall risk perspective.

The dose to an individual located at the EAB or the 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 off-site location, the breathing rate of the individual at the off-site location, the time of exposure and activity-to-dose conversion factors. The only site-specific parameter is atmospheric dispersion. The DCD doses are determined using time-dependent X/Q values corresponding to the top 5$^{th}$ percentile meteorology during the first 2 hours (hr.) of the accident, meaning that conditions would be more favorable for dispersion 95 percent of the time. The doses evaluated herein are calculated based on the 50$^{th}$ percentile site-specific X/Q values during the first 2 hr. of the accident, reflecting more realistic meteorological conditions. The 50$^{th}$ percentile values were calculated using 5 years of on-site data. Site-specific doses are obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (X/Q values). Because the site-specific X/Q values are bounded by the DCD X/Q values, this approach demonstrates that the site-specific doses are within those calculated in the DCD.

The HAR short-term X/Qs are calculated using Regulatory Guide 1.145 methods with site-specific meteorological data. The Regulatory Guide 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 16 downwind direction sectors and then calculates overall (non direction-specific) X/Q values. For a given location, either the EAB or the LPZ, the 0- to 2-hr. X/Q value is the 50$^{th}$ percentile overall value calculated by PAVAN. For the LPZ, the X/Q values for all subsequent times are calculated by logarithmic interpolation between the 50$^{th}$ percentile X/Q value and the annual average X/Q value. Releases of activity are assumed to be at ground level.

The accident doses are expressed as TEDE doses. The TEDE dose is the summation of the Committed Effective Dose Equivalent (CEDE) from inhalation of radioactive particles and the Effective Dose Equivalent (EDE) from external exposure. The CEDE is determined using the dose conversion factors in Federal Guidance Report 11 (Reference 7.1-001), while the EDE is based on the dose conversion factors in Federal Guidance Report 12 (Reference 7.1-002). As indicated in Regulatory Guide 1.183, the dose conversion factors in Federal Guidance Reports 11 and 12 are acceptable to the NRC staff. Appendix 15A of the AP1000 DCD provides information of this methodology.

## 7.1.3 RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS

This subsection identifies the postulated accidents and provides a brief description of each accident used in the HAR dose consequence assessments. A more detailed description of each accident is provided in Chapter 15 of the AP1000 DCD. An overall summary of the results of the HAR site-specific evaluated accident doses appears in Table 7.1-2. Table 7.1-2 shows that the evaluated dose consequences are well below the regulatory acceptance criteria.

The analysis approach for evaluating the AP1000 DBAs discussed in the following subsections is based upon the EAB and LPZ doses provided by Westinghouse and given in Chapter 15 of the AP1000 DCD. The ratio of the HAR site X/Q value to the AP1000 site X/Q value for each post-accident time period is given in Table 7.1-3. Note that the X/Q value for 1.4 to 3.4 hours at the HAR site was not calculated. To calculate the EAB dose for the LOCA accident, the X/Q value for the period between 0 and 2 hr. was used instead.

### 7.1.3.1 Main Steam Line Break Outside Containment

The bounding AP1000 steam line break for the radiological consequence evaluation occurs outside containment. The facility is designed so that only one SG experiences an uncontrolled blowdown even if one of the main steam isolation valves fails to close. Feedwater is isolated after the rupture and the faulted SG dries out. The secondary side inventory of the faulted SG is released to the environs along with the entire amount of iodine and alkali metals contained in the secondary side coolant.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. The TEDE doses for the pre-existing iodine spike are shown in Table 7.1-4. The doses at the EAB and the LPZ are a small fraction of the 25-rem TEDE identified in 10 CFR 50.34. A “small fraction” is defined as 10 percent or less in Regulatory Guide 1.183. The doses for the accident-initiated iodine spike are shown in Table 7.1-5. These doses meet the TEDE dose guidelines of 10 CFR 50.34.

### 7.1.3.2 Locked Rotor

The AP1000 locked rotor event is the most severe of several possible decreased reactor coolant flow events. This accident is postulated as an instantaneous seizure of the pump rotor in one of four reactor coolant pumps. The rapid reduction in flow in the faulted loop causes a reactor trip. Heat transfer of the stored energy in the fuel rods to the reactor coolant causes the reactor coolant temperature to increase. The reduced flow also degrades heat transfer between the primary and secondary sides of the SGs. The event can lead to fuel cladding failure, which results in an increase of activity in the coolant. The rapid expansion of coolant in the core, combined with decreased heat transfer in the SG, causes the reactor coolant system (RCS) pressure to increase dramatically.

Cooling down of the plant by steaming off the SGs provides a pathway for the release of radioactivity to the environment. In addition, primary side activity, carried over because of leakage in the SGs, mixes in the secondary side and becomes available for release. The primary side coolant activity inventory increases because of the postulated failure of some of the fuel cladding with the consequential release of the gap fission product inventory to the coolant. The significant releases from this event are the iodines, alkali metals, and noble gases. No fuel melting occurs.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. The TEDE doses for the locked rotor accident, both with and without feedwater available, are shown in Table 7.1-6. The doses at the EAB and the LPZ are a small fraction of the TEDE limits identified in 10 CFR 50.34.

### 7.1.3.3 Control Rod Ejection

This accident is postulated as the gross failure of one control rod mechanism pressure housing resulting in ejection of the control rod cluster assembly and drive shaft. The failure leads to a rapid positive reactivity insertion, potentially leading to localized fuel rod damage and significant releases of radioactivity to the reactor coolant.

Two activity release paths contribute to this event. First, the equilibrium activity in the reactor coolant and the activity from the damaged fuel are blown down through the failed pressure housing to the containment atmosphere. The activity can leak to the environment over a relatively long period because of the containment’s design basis leakage. Decay of radioactivity occurs during hold-up inside containment before release to the environs.

The second release path is from the release of steam from the steam generators (SGs) following the reactor trip. With a coincident loss of off-site power, additional steam must be released to cool down the reactor. The SG activity consists of the secondary side equilibrium inventory plus the additional contributions from reactor coolant leaks in the steam generators. The reactor coolant activity levels are increased for this accident, because the activity released from the damaged fuel mixes into the coolant before being leaked to the SGs. The iodines, alkali metals, and noble gases are the significant activity sources for this event. Noble gases entering the secondary side are quickly released to the atmosphere by way of the steam releases through the atmospheric relief valves. A small fraction of iodines and alkali metals in the flashed part of the leak flow are available for immediate release without benefit of partitioning. The unflashed portion mixes with secondary side fluids where partitioning occurs before the release as steam.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. The doses at the EAB and the LPZ shown in

Table 7.1-7 are well within the TEDE limits identified in 10 CFR 50.34. “Well within” is given as 25 percent or less in NUREG-0800.

#### 7.1.3.4 Steam Generator Tube Rupture

The AP1000 SG tube rupture accident assumes the complete severance of one SG tube. The accident causes an increase in the secondary side activity because of reactor coolant flow through the ruptured tube. With the loss of off-site power, contaminated steam is released from the secondary system because of the turbine trip and dumping of steam by way of the atmospheric relief valves. Steam dump (and retention of activity) to the condenser is precluded because of the assumption of loss of off-site power. The release of radioactivity depends on the primary to secondary leakage rate, the flow to the faulted SG from the ruptured tube, the percentage of defective fuel in the core, and the duration/amount of steam released from the SGs.

The radioiodines, alkali metals, and noble gases are the significant nuclide groups released during a SG tube rupture accident. Multiple release pathways are analyzed for the tube rupture accident. The noble gases in the reactor coolant enter the ruptured SG and are available for immediate release to the environment. In the intact loops, iodines and alkali metals, which leaked to the secondary side during the accident are partitioned as the intact SG is steamed down until switchover to the residual heat removal system occurs. In the ruptured SG, some of the reactor coolant flowing through the tube break flashes to steam while the unflashed portion mixes with the secondary side inventory. Iodines and alkali metals in the flashed fluid are not partitioned during steam releases while activity in the secondary side of the faulted generator is partitioned before being released as steam.

The AP1000 DCD doses were re-evaluated using the HAR short-term, accident-dispersion characteristics. The TEDE doses for the SG tube rupture accident with the accident-initiated iodine spike are shown in Table 7.1-8. The doses at the EAB and the LPZ are a small fraction of the TEDE limits identified in 10 CFR 50.34. The pre-existing iodine spike doses are shown in Table 7.1-9. These doses meet the TEDE dose guidelines of 10 CFR 50.34.

#### 7.1.3.5 Failure of Small Lines Carrying Primary Coolant Outside of Containment

Small lines carrying reactor coolant outside the AP1000 containment include the RCS sample line and the chemical and volume control system discharge line to the radwaste system. These lines are not continuously used. The failure of the discharge line is neither significant nor analyzed. The assumed flow is approximately 0.0063 cubic meters per second ($m^3$/sec) (0.22 cubic feet per second ($ft^3$/sec) or 100 gallons per minute (gpm) and when leaving containment, is assumed to cool below 60 degrees Celsius ($^oC$) (140 degrees Fahrenheit [$^oF$]) and has been cleaned by the mixed-bed demineralizer. The reduced iodine concentration, low flow, and temperature make this break nonlimiting with respect to off-site dose consequences.

The RCS sample line break is the more limiting break. This line is postulated to break between the outboard isolation valve and the reactor coolant sample panel. Off-site doses are calculated assuming a break flow limited to 0.0082 $m^3$/sec (0.29 $ft^3$/sec) or 130 gpm by flow restrictors with isolation occurring at 30 minutes. Radioiodines and noble gases are the only significant activities released. The source term is based on an assumed accident-initiated iodine spike that increases the iodine release rate from the fuel by a factor of 500 throughout the event. The activity is assumed to be released to the environment without decay or holdup in the auxiliary building.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. The results are shown in Table 7.1-10. The resulting dose at the EAB and the LPZ is a small fraction of the TEDE limits identified in 10 CFR 50.34.

7.1.3.6 Large Break Loss of Coolant Accident

The core response analysis for the AP1000 demonstrates that the reactor core maintains its integrity for the large break LOCA. However, significant core degradation and melting is assumed in this DBA. The assumption of major core damage is intended to challenge various accident mitigation features and provide a conservative basis for calculating site radiological consequences. The source term used in the analysis is adopted from NUREG-1465 and Regulatory Guide 1.183 with the nuclide inventory determined for a three-region equilibrium cycle core at the end of life.

The activity released consists of the equilibrium activity in the reactor coolant and the activity released from the damaged core. The AP1000 is a leak before break design; therefore, the coolant is assumed to blow down to the containment for 10 minutes. One-half of the iodine and the noble gases in the blowdown stream are released to the containment atmosphere.

The event assumes that a core release starts after the 10-minute blowdown of reactor coolant. It is futher assumed that the fuel rod gap activity is released over the next half hour, followed by an in-vessel core melt that lasts 1.3 hr. Iodines, alkali metals, and noble gases are released during the gap activity release. During the core melt phase, five additional nuclide groups are released: the tellurium group, the noble metals group, the lanthanides group, the cerium group, and the barium and strontium group.

Activity is released from the containment by way of the containment purge line at the beginning of the accident. After isolation of the purge line, activity continues to leak from the containment at its design basis leak rate. There is no emergency core cooling leakage activity because the passive core cooling system does not pass coolant outside the containment. A coincidental loss of off-site power has no impact on the activity released to the environment because of the passive designs for the core cooling and fission product control systems.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. Table 7.1-11 provides the EAB and the LPZ doses. Both doses meet the TEDE dose guideline in 10 CFR 50.34. The activity released from the core melt phase of the accident is the greatest contributor to off-site doses. The EAB dose in Table 7.1-11 is given for the 2-hr. period during which the dose is greatest at this location. The initial 2 hr. of the accident is not the worst 2-hr. period because of the delays associated with cladding failure and fuel damage.

#### 7.1.3.7 Fuel-Handling Accidents

The AP1000 fuel-handling accident (FHA) can occur inside containment or in the fuel-handling area of the auxiliary building. The accident postulates the dropping of a fuel assembly over the core or in the spent fuel pool. The cladding of the fuel rods is assumed breached and the fission products in the fuel rod gaps are released to the reactor refueling cavity water or spent fuel pool.

The AP1000 DCD doses were re-evaluated using the HAR site short-term, accident-dispersion characteristics. The resulting doses at the EAB and the LPZ are summarized in Table 7.1-12. The doses are applicable to fuel-handling accidents inside containment and in the spent fuel pool in the auxiliary building. The doses are well within the TEDE guidelines in 10 CFR 50.34.

### 7.1.4 REFERENCES

7.1-001 U.S. Environmental Protection Agency, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion”, Federal Guidance Report 11, USEPA-520/1-88-020, September 1988.

7.1-002 U.S. Environmental Protection Agency, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report 12, USEPA-402-R-93-081, September 1993.

**TABLE 7.1-1 (Sheet 1 of 2)**
**Selection of Accidents**

| SRP SECTION (A) | SRP DESCRIPTION | DCD DESCRIPTION | IDENTIFIED IN NUREG-1555 APPENDIX A(A) | COMMENT |
|---|---|---|---|---|
| 15.1.5A | RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURES OUTSIDE CONTAINMENT OF A PRESSURIZED WATER REACTOR (PWR) | STEAM SYSTEM PIPING FAILURE | YES | DCD SECTION 15.1.5 |
| 15.2.8 | FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR) | FEEDWATER SYSTEM PIPE BREAK | YES | IN ACCORDANCE WITH DCD, BOUNDED BY SECTION 15.1.5 ACCIDENT |
| 15.3.3 | REACTOR COOLANT PUMP ROTOR SEIZURE | REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR) | YES | |
| 15.3.4 | REACTOR COOLANT PUMP SHAFT BREAK | REACTOR COOLANT PUMP SHAFT BREAK | YES | IN ACCORDANCE WITH DCD, BOUNDED BY SECTION 15.3.3 ACCIDENT |
| 15.4.8 | SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) | SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS | NO | INCLUDED FOR COMPLETENESS |
| 15.6.2 | RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING COOLANT OUTSIDE CONTAINMENT | FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE PRIMARY CONTAINMENT | YES | |
| 15.6.3 | RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR (SG) TUBE RUPTURE (PWR) | SG TUBE FAILURE | YES | |
| 15.6.5A | RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS OF COOLANT ACCIDENT (LOCA) INCLUDING CONTAINMENT LEAKAGE CONTRIBUTION | LOCA RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY | YES | DCD SECTION 15.6.5 |

**TABLE 7.1-1 (Sheet 2 of 2)**
**Selection of Accidents**

| SRP SECTION (A) | SRP DESCRIPTION | DCD DESCRIPTION | IDENTIFIED IN NUREG-1555 APPENDIX A(A) | COMMENT |
|---|---|---|---|---|
| 15.6.5B | RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOCA: LEAKAGE FROM ENGINEERED SAFETY FEATURE COMPONENTS OUTSIDE CONTAINMENT | LOCA RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS SAFETY FEATURE COMPONENTS OUTSIDE CONTAINMENT WITHIN THE REACTOR COOLANT CONTAINMENT PRESSURE BOUNDARY | YES | DCD SECTION 15.6.5 |
| 15.7.4 | RADIOLOGICAL CONSEQUENCES OF FUEL-HANDLING ACCIDENTS | FUEL-HANDLING ACCIDENT | YES | |

NOTES:
A) 15.4.9A AND 15.6.5D WERE NOT INCLUDED IN THE TABLES AS THEY ARE ONLY APPLICABLE TO BOILING WATER REACTORS (BWRS).

**Table 7.1-2**
**Summary of HAR Site-Specific Off-Site Doses Consequences**

| Accident | EAB Dose TEDE Rem | LPZ Dose TEDE Rem | Guideline Limit TEDE Rem |
|---|---|---|---|
| **Main Steam Line Break** | | | |
| Pre-existing Iodine Spike | 5.6E-02 | 1.6E-02 | 25 |
| Accident-initiated Iodine Spike | 6.2E-02 | 4.9E-02 | 2.5 |
| **Reactor Coolant Pump Locked Rotor** | | | |
| No Feedwater | 4.5E-02 | 6.9E-03 | 2.5 |
| Feedwater Available | 3.4E-02 | 1.4E-02 | 2.5 |
| Control Rod Ejection Accident | 2.0E-01 | 1.0E-01 | 6.3 |
| **Steam Generator (SG) Tube Rupture** | | | |
| Pre-existing Iodine Spike | 1.2E-01 | 2.2E-02 | 25 |
| Accident-initiated Iodine Spike | 6.2E-02 | 1.5E-02 | 2.5 |
| Small Line Break | 1.2E-01 | 1.8E-02 | 2.5 |
| Design Basis LOCA | 2.7E+00 | 9.5E-01 | 25 |
| Fuel Handling Accident | 2.9E-01 | 4.6E-02 | 6.3 |

Notes:
Doses are based on FGR 11 (Reference 7.1-001) and FGR 12 (Reference 7.1-002) dose conversion.

TEDE guidelines from Regulatory Guide 1.183. Small line break criteria based on SRP 15.6.2

## Table 7.1-3
## Ratio of HAR 50-Percent Accident Site X/Q Values to AP1000 DCD X/Q Values

| | | | χ/Q Ratio |
|---|---|---|---|
| **Post-Accident Time Period (hr.)** | **HAR Site X/Q Values ($sec/m^3$)** | **AP1000 X/Q Values ($sec/m^3$)** | **HAR Site / AP1000 DCD** |
| **LOCA** | | | |
| **EAB** | | | |
| 1.4 to 3.4 hr. (1) | 5.64E-05 | 5.10E-04 | 1.11E-01 |
| **LPZ** | | | |
| 0 to 8 hr. | 8.80E-06 | 2.20E-04 | 4.00E-02 |
| 8 to 24 hr. | 7.70E-06 | 1.60E-04 | 4.81E-02 |
| 24 to 96 hr. | 5.84E-06 | 1.00E-04 | 5.84E-02 |
| 96 to 720 hr. | 3.84E-06 | 8.00E-05 | 4.80E-02 |
| **All Other Accidents** | | | |
| **EAB** | | | |
| 0--2 hr | 5.64E-05 | 1.00E-03 | 5.64E-02 |
| LPZ | | | |
| 0--8 hr | 8.80E-06 | 5.00E-04 | 1.76E-02 |
| 8--24 hr | 7.70E-06 | 3.00E-04 | 2.57E-02 |
| 24--96 hr | 5.84E-06 | 1.50E-04 | 3.89E-02 |
| 96--720 hr | 3.84E-06 | 8.00E-05 | 4.80E-02 |

Notes:

1) The EAB X/Q value for the period 0 to 2 hours was used for the 1.4 to 3.4 hour period for the HAR site. The 1.4 to 3.4 hour period represents the worst two-hour period for the EAB dose.

Definitions:

EAB = exclusion area boundary

LPZ = low population zone

$sec/m^3$ = seconds per cubed meter

X/Q = atmospheric dispersion coefficient

**Table 7.1-4**
**Main Steam Line Break, 0 to 96 Hours, Pre-Existing Iodine Spike**

| Time | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 1.00E+00 | |
| 0 to 8 hr. | - | 5.81E-01 |
| 8 to 24 hr. | - | 7.18E-02 |
| 24 to 96 hr. | - | 1.08E-01 |
| Total | 1.00E+00 | 7.61E-01 |
| **HAR COLA** | | |
| 0 to 2 hr. | 5.64E-02 | |
| 0 to 8 hr. | - | 1.02E-02 |
| 8 to 24 hr. | - | 1.84E-03 |
| 24 to 96 hr. | - | 4.20E-03 |
| Total | 5.64E-02 | 1.63E-02 |

**Table 7.1-5**
**Main Steam Line Break, 0 to 96 Hours, Accident-Initiated Iodine Spike**

| Time | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 1.10E+00 | |
| 0 to 8 hr. | - | 1.02E+00 |
| 8 to 24 hr. | - | 3.77E-01 |
| 24 to 96 hr. | - | 5.36E-01 |
| Total | 1.10E+00 | 1.93E+00 |
| **HAR COLA** | | |
| 0 to 2 hr. | 6.20E-02 | |
| 0 to 8 hr. | - | 1.80E-02 |
| 8 to 24 hr. | - | 9.68E-03 |
| 24 to 96 hr. | - | 2.09E-02 |
| Total | 6.20E-02 | 4.85E-02 |

**Table 7.1-6**
**Locked Rotor Accident, 0 to 1.5 Hours, Pre-Existing Iodine Spike**

| | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **No Feedwater** | | |
| **AP1000 Tier 2** | | |
| 0 to 1.5 hr. | 8.00E-01 | 3.89E-01 |
| Total | 8.00E-01 | 3.89E-01 |
| | | |
| **HAR COLA** | | |
| 0 to 1.5 hr. | 4.51E-02 | 6.85E-03 |
| Total | 4.51E-02 | 6.85E-03 |
| Locked Rotor Accident, 0 to 8 Hours, Pre-Existing Iodine Spike | | |
| **FW Available** | | |
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 6.00E-01 | |
| 0 to 8 hr. | - | 7.94E-01 |
| Total | 6.00E-01 | 7.94E-01 |
| | | |
| **HAR COLA** | | |
| 0 to 2 hr. | 3.38E-02 | |
| 0 to 8 hr. | - | 1.40E-02 |
| Total | 3.38E-02 | 1.40E-02 |

**Table 7.1-7**
**Control Rod Ejection Accident, 0 to 720 Hours, Pre-Existing Iodine Spike**

| Time | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 3.60E+00 | |
| 0 to 8 hr. | - | 4.58E+00 |
| 8 to 24 hr. | - | 7.84E-01 |
| 24 to 96 hr. | - | 6.32E-02 |
| 96 to 720 hr. | - | 2.06E-02 |
| Total | 3.60E+00 | 5.45E+00 |
| | | |
| **HAR COLA** | | |
| 0 to 2 hr. | 2.03E-01 | |
| 0 to 8 hr. | - | 8.06E-02 |
| 8 to 24 hr. | - | 2.01E-02 |
| 24 to 96 hr. | - | 2.46E-03 |
| 96 to 720 hr. | - | 9.89E-04 |
| Total | 2.03E-01 | 1.04E-01 |

**Table 7.1-8**
**Steam Generator Tube Rupture, 0 to 24 Hours, Accident-Initiated Iodine Spike**

| Time | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 1.10E+00 | |
| 0 to 8 hr. | - | 6.27E-01 |
| 8 to 24 hr. | - | 1.69E-01 |
| Total | 1.10E+00 | 7.96E-01 |
| | | |
| **HAR COLA** | | |
| 0 to 2 hr. | 6.20E-02 | |
| 0 to 8 hr. | - | 1.10E-02 |
| 8 to 24 hr. | - | 4.34E-03 |
| Total | 6.20E-02 | 1.54E-02 |

**Table 7.1-9**
**Steam Generator Tube Rupture, 0 to 24 Hours, Pre-Existing Iodine Spike**

| | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 2.20E+00 | |
| 0 to 8 hr. | - | 1.16E+00 |
| 8 to 24 hr. | - | 7.24E-02 |
| Total | 2.20E+00 | 1.23E+00 |
| **HAR COLA** | | |
| 0 to 2 hr. | 1.24E-01 | |
| 0 to 8 hr. | - | 2.04E-02 |
| 8 to 24 hr. | - | 1.86E-03 |
| Total | 1.24E-01 | 2.23E-02 |

**Table 7.1-10**
**Small Line Break Accident, 0 to 0.5 Hour, Accident-Initiated Iodine Spike**

| | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 0.5 hr. | 2.10E+00 | 1.02E+00 |
| Total | 2.10E+00 | 1.02E+00 |
| **HAR COLA** | | |
| 0 to 0.5 hr. | 1.18E-01 | 1.80E-02 |
| Total | 1.18E-01 | 1.80E-02 |

## Table 7.1-11
## AP1000 Design Basis LOCA, 0 to 720 Hours

| | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 1.4 to 3.4 hr. | 2.46E+01 | - |
| 0 to 8 hr. | - | 2.17E+01 |
| 8 to 24 hr. | - | 7.50E-01 |
| 24 to 96 hr. | - | 2.93E-01 |
| 96 to 720 hr. | - | 5.49E-01 |
| Total | 2.46E+01 | 2.33E+01 |
| | | |
| **HAR COLA** | | |
| 1.4 to 3.4 hr. | 2.70E+00 | - |
| 0 to 8 hr. | - | 8.68E-01 |
| 8 to 24 hr. | - | 3.61E-02 |
| 24 to 96 hr. | - | 1.71E-02 |
| 96 to 720 hr. | - | 2.64E-02 |
| Total | 2.70E+00 | 9.48E-01 |

## Table 7.1-12
## Fuel-Handling Accidents, 0 to 2 Hours

| | EAB Dose TEDE Rem | LPZ Dose TEDE Rem |
|---|---|---|
| **AP1000 Tier 2** | | |
| 0 to 2 hr. | 5.20E+00 | 2.59E+00 |
| Total | 5.20E+00 | 2.59E+00 |
| | | |
| **HAR COLA** | | |
| 0 to 2 hr. | 2.93E-01 | 4.56E-02 |
| Total | 2.93E-01 | 4.56E-02 |

## 7.2 SEVERE ACCIDENTS

Section 7.1 provides a comparison of the off-site dose consequences and resulting health effects for DBAs, as identified in the AP1000 DCD and those contained in Section 15 of the SER. A direct comparison of the off-site dose consequences and health effects, as required by NUREG-1555, is difficult. Section 7.1 provides quantitative results, whereas the results reported in this section are mostly expressed probabilistically. However, doses calculated at the EAB and LPZ in Section 7.1 from DBAs compare favorably to those calculated from severe accidents at a 0 to 80-km (50-mi.) radius (internal events only).

### 7.2.1 INTRODUCTION

This section evaluates the potential environmental impacts of severe accidents at the HAR site. This section, and the section that follows (Section 7.3), relies on information obtained from NUREG-1437 in order to meet the requirements specified in NUREG-1555. Both documents are referenced throughout Sections 7.2 and 7.3. In addition, severe accidents were evaluated as part of the NRC's Final Safety Evaluation Report (FSER) for the AP1000, where the NRC concluded that the approach used in the DCD was acceptable.

As a class, severe accidents are considered less likely to occur and are not part of the design basis for the AP1000; however, because the consequences could be more severe, severe accidents are considered important both in terms of impact to the environment and off-site costs. Severe accidents can be distinguished from DBAs in two primary respects: (1) they involve substantial physical deterioration of the fuel in the reactor core, including overheating to the point of melting; and (2) they involve deterioration of the containment system capability to perform its intended function of limiting the release of radioactive materials to the environment.

In NUREG-1437, the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, the NRC generically assessed the impacts of severe accidents during license renewal periods using the results of existing analyses and site-specific information to conservatively predict the environmental impacts of severe accidents for each plant during the renewal period. The results of this report are used as a basis for evaluating the severe accident environmental impacts of a new nuclear power plant that may be built on the HAR site.

In addition, Westinghouse completed a probabilistic risk assessment (PRA) for the AP1000 design as documented in the AP1000 DCD as part of the design certification process. The PRA included the development of a Level 3 PRA model. The Westinghouse Level 3 PRA model used generic characteristics to represent site-specific attributes. This section also presents an update of the generic PRA analysis of severe accidents to include Level 3 modeling of the site-specific characteristics of the HAR site.

The results of NUREG-1437 and the HAR site-specific Level 3 analysis demonstrate that the potential impacts of a severe accident for the AP1000 design on the HAR site are of small significance, as defined by the NRC. The potential impacts are equivalent to or less than the potential impacts of a severe accident with the HNP. These results are also used to support the severe accident mitigation alternative (SAMA) analyses in Section 7.3.

## 7.2.2 APPLICABILITY OF EXISTING GENERIC SEVERE ACCIDENT STUDIES

Section 5.3.3 of NUREG-1437 presents an assessment of impacts of severe accidents from existing reactor plants during the license renewal period. This study was conducted by the NRC staff. The NUREG-1437 evaluations and conclusions are based on existing assessments of severe accident impacts presented in numerous Final Environmental Statements (FES) published after 1980 for a representative set of United States plants and sites (HNP included) in the NUREG-1150 series of documents. Methodologies were developed to evaluate each of the dose pathways by which a severe accident may result in adverse environmental impacts and to estimate off-site costs of severe accidents. Three pathways for release of radioactive material to the environment were evaluated (i.e., atmospheric, air to surface water, and groundwater to surface water).

The NUREG-1437 assessment methodology and the resulting conclusions are considered broadly applicable beyond the license renewal context, including evaluation of severe accident impacts associated with determining site suitability for a nuclear power plant. The NRC later confirmed, in 61 FR 28467-28497 that “the analyses performed for the Generic Environmental Impact Statement (GEIS) represent adequate, plant-specific estimates of the impacts from severe accidents…”

As described in the NUREG-1437, the purpose of the evaluation of severe accidents was “to use, to the extent possible, the available severe accident results, in conjunction with those factors that are important to risk and that change with time to estimate the consequences of nuclear plant accidents for all plants for a time period that exceeds the time frame of existing analyses.” The NUREG-1437 estimation process was completed by predicting increases or decreases in consequences because the plant lifetime was extended past the normal license period by considering the projected changes in the risk factors. The primary assumption in the NUREG-1437 analysis was that regulatory controls ensure that the physical plant condition (i.e., the predicted probability of, and radioactive releases from, an accident) is maintained at a constant level during the renewal period; therefore, the frequency and magnitude of a release remains relatively constant. In other words, significant changes in consequences would result only from changes in the plant's external environment.

The use of severe accident risk per reactor-year of operation as the principal metric for evaluating severe accident environmental impacts, and the assumption that this risk remains constant over the life of the plant, are equally applicable and appropriate in both the license renewal and combined license (COL) context. When applied to new advanced reactor designs, such as the AP1000, the NUREG-1437 approach introduces additional conservatism because advanced reactor designs have lower severe accident frequencies compared to the existing fleet of reactors.

Therefore, the generic analysis of severe accident impacts presented in the NUREG-1437 also provides an appropriate basis and method for evaluating severe accident impacts for a COL Application (COLA).

### 7.2.3 SIGNIFICANCE CRITERIA FOR POTENTIAL SEVERE ACCIDENT RELEASES

The significance of the impacts associated with severe accident releases may be categorized as either SMALL, MODERATE, or LARGE, consistent with the criteria that the NRC established in 10 CFR 51, Appendix B, Table B-1, Footnote 3 as follows:

- SMALL—Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the NRC has concluded that those impacts that do not exceed permissible levels in the NRC's regulations are considered SMALL.

- MODERATE—Environmental effects are sufficient to alter noticeably, but not to destabilize, any important attribute of the resource.

- LARGE—Environmental effects are clearly noticeable and are sufficient to destabilize any important attributes of the resource.

In accordance with National Environmental Policy Act (NEPA) practice, potential additional mitigation is considered in proportion to the significance of the impact to be addressed (i.e., impacts that are SMALL receive less mitigative consideration than impacts that are LARGE).

### 7.2.4 NUREG-1437 BASIS EVALUATION

NUREG-1437 evaluated the HNP for severe accident impacts for license renewal considerations. This section evaluates the analysis presented in NUREG-1437 as it applies to a new advanced reactor (i.e., AP1000) at the HAR site.

#### 7.2.4.1 Evaluation of Potential Releases by Way of Atmospheric Pathway

Detailed severe accident consequence (early and latent fatalities and total dose) evaluations were not available for all plants considered in the NUREG-1437. Therefore, a predictor for these consequences was developed using correlations based upon the calculated results from the existing FES severe accident analyses. The developed predictor, termed the exposure index (EI), was then used to infer the future consequence level of all individual nuclear plants. Correlations were developed using two environmental parameters that are available for all plants (i.e., population distribution and wind direction frequency).

NUREG-1437 provides the following discussion of EI:

> Population, which changes over time, defines the number of people within a given distance from the plant. Wind direction, which is assumed not to change from year to year, helps determine what proportion of the population is at risk in a given direction, because radionuclides are carried by the wind. Therefore, an EI relationship was developed by multiplying the wind direction frequency (fraction of the time per year) for each of 16 (22.5°) compass sectors times the population in that sector for a given distance from the plant and summing all products.…Population varies with population growth and movement, and with the distance from any given plant. As the population changes for that plant, the EI also changes (the larger the EI, the larger the number of people at risk). Thus, EI is proportional to risk and an EI for a site for a future year can be used to predict the risk to the population around that site in that future year.

Thus, the EI is a function of population surrounding the plant, weighted by the site-specific wind direction frequency, and is, therefore, a site-specific parameter. Because meteorological patterns, including wind direction frequency, tend to remain constant over time, the site meteorology should not be significantly different for the HAR site than the meteorology considered in NUREG-1437 for the HNP and only population should significantly affect the resulting risk in any given year of reactor operation.

Two EI values were evaluated in NUREG-1437. A 16-km (10-mi.) EI was found to best correlate with early fatalities and a 241-km (150-mi.) EI was found to best correlate with latent fatalities and total dose. For both measures, the HNP was found to be well within the range of all plant sites, as demonstrated in Table 7.2-1 where the EI values from NUREG-1437 for three sites are presented.

Using these indices, NUREG-1437 (Section 5.5.2.1) determined that the risk of early and latent fatalities from individual nuclear power plants (including the HNP) is SMALL and represents only a small fraction of the risk to which the public is exposed from other sources.

The NUREG-1437 conclusions are judged to remain valid when applied to the advanced reactor design of the AP1000 at the HAR site. The region around the HAR site has experienced population growth since the time of the NUREG-1437 study, which would result in higher calculated EI values. The severe accident frequencies of the AP1000 (which are not explicitly reflected in the EI value methodology), however, are lower than those of the current designs evaluated by NUREG-1437. Thus, the HAR site risks for the atmospheric exposure pathway will be within the range of those considered as SMALL significance in

NUREG-1437. This is demonstrated quantitatively in Section 7.2.5 through the development of the HAR site MACCS2 model.

#### 7.2.4.2 Evaluation of Potential Releases by Way of Atmospheric Fallout onto Open Bodies of Water

Following a severe accident, a radiation hazard may exist from the deposition of airborne, radioactive fallout onto open bodies of water. Depending on the type of water body, this hazard may lead to internal exposure from the ingestion of contaminated water or from consuming contaminated aquatic fauna. External exposure may result from swimming in the contaminated water or from recreational activities on the shoreline. The extent of the hazard is largely determined by the proximity of individuals to the reactor, the areal extent of contamination, and the ability for interdiction to reduce the exposure hazard. The risk from this exposure at plants sited on all types of water bodies was evaluated in NUREG-1437 and compared with that of the Fermi plant, located on Erie Reservoir, for which an analysis has been performed for an uninterdicted dose. For the Fermi plant, NUREG-1437 estimates that the uninterdicted dose from fallout onto open bodies of water is less than 2 percent of that from the atmospheric pathway total.

In NUREG-1437, the HNP is described as a “small river site” for surface water pathway purposes. In Table 5.15 of NUREG-1437, the site is listed as one that may not be bounded by the Fermi 2 surface water analysis. The HNP (and 12 other sites) may not be bounded by the Fermi analysis because of the following combined characteristics:

- Low on-site average annual flow rates.
- Comparatively long residence times.
- Comparatively large surface-area-to-volume ratios.

NUREG-1437 notes that because the combined residence time and surface-area-to-volume ratios for the 13 small river sites in Table 5.15 exceed values at the Fermi plant by less than a factor of 3, and these sites have populations lower than the Fermi plant by at least a factor of 2 (HNP population is smaller by a factor of 3.1 per NUREG-1437, Table 5.14b), the population dose at these sites is expected to remain a small fraction of the value estimated for the atmospheric pathway. Additionally, NUREG-1437 notes that the HNP is considered to be at least as amenable to interdictive measures as the Fermi plant, which would further reduce population dose. Therefore, NUREG-1437 concludes that for both drinking water and aquatic food pathways, the probability weighted consequences caused by severe accidents is of SMALL significance.

Site population projections for the HAR site show a moderate population increase over the projected license period. This population increase, however, would not be expected to change the conclusions of NUREG-1437 for the HAR site because the population increase would also increase the total atmospheric pathway dose to which fallout onto open bodies of water is being compared.

The conclusions of NUREG-1437, that the consequences of atmospheric fallout onto open bodies of water caused by severe accidents at the HNP is of SMALL significance and is judged to remain valid for the HAR site.

#### 7.2.4.3 Evaluation of Potential Releases to Groundwater

The potential for radiation exposure from the groundwater pathway, as the result of postulated severe accidents at the HNP, is also evaluated in NUREG-1437.

For this pathway, the core is postulated to melt down, breach the reactor vessel, and fall onto the reactor building floor. As a result of chemical energy and decay heat, the melted fuel reacts with the concrete floor. Without cooling water addition to the core debris, the basemat of the containment building may eventually breach; molten core debris and radioactive water penetrate strata beneath the plant. The soluble radionuclides in the debris can be leached and transported with groundwater and contaminated reactor water to downgradient domestic wells used for drinking water, or to surface water bodies used for drinking water, aquatic food, and recreation.

As identified in NUREG-1437, groundwater contamination caused by severe accidents has been evaluated generically in NUREG-0440, the Liquid Pathway Generic Study (LPGS). The LPGS assumes that core melt with subsequent basemat melt-through occurs, and evaluates the consequences. The LPGS examines six generic sites using typical or comparative assumptions on geology and adsorption factors. Relevant site-specific features include the following:

- Groundwater travel time.
- Retention-adsorption coefficients.
- Distance to surface water.
- Soil, sediment, and rock characteristics.

In accordance with NUREG-1437, the LPGS results are believed to provide generally conservative uninterdicted population dose estimates in the six generic plant-site categories. According to NUREG-0440, the generic liquid pathway uninterdicted dose estimates are one or more orders of magnitude lower than those attributed to the atmospheric pathway. The six generic sites typify those adjacent to the following:

- Small rivers
- Large rivers

- The Great Reservoirs
- Oceans
- Estuaries
- “Dry” site

Twenty-seven sites (including the HNP) of the 73 nuclear power plant sites that performed groundwater pathways analyses for their FESs are compared with one another and the results of the generic site. For individual sites that do not significantly exceed those of the generic counterpart, the liquid pathway may be considered an insignificant contributor to the population dose that could result from a severe accident for the plants.

NUREG-1437 concludes that the risk from the groundwater exposure pathway, generally contributes only a small fraction of that risk attributable to the population from the atmospheric pathway, but in a few cases, may contribute a comparable risk.

The HNP liquid pathway “realistic” dose estimates are presented in Table 5.18 of NUREG-1437 as a dose ratio (i.e., HNP dose divided by the generic “small river” site dose). For the HNP, the combined dose ratio for drinking water dose, ingestive dose, and direct contact is specified as “<<1” indicating that, based on this comparison to the generic small river site dose, the HNP dose is significantly less than the small river generic site liquid pathway dose.

It is also noted that the AP1000 design has intentionally included design elements to minimize the potential for a severe accident to lead to core concrete interactions and an eventual breach of the containment building basemat. These design elements include in-vessel retention of core debris by external reactor vessel cooling (i.e., submerging the reactor vessel in water to facilitate cooling and thereby prevent vessel failure) and ex-vessel core debris cooling in the reactor cavity (i.e., providing a water-filled reactor cavity to receive core debris upon vessel failure). These design elements are discussed in more detail in the AP1000 DCD.

The conclusions summarized in NUREG-1437 are that the release consequences to the groundwater caused by severe accidents at the HNP are of SMALL significance and are judged to remain valid for the HAR site.

#### 7.2.4.4 NUREG-1437 Evaluation Conclusions

In NUREG-1437, the NRC evaluated the HNP for severe accident impacts for license renewal considerations and concluded that the environmental impacts were of SMALL significance. The impacts could be severe; however, because of the low likelihood of occurrence, the impacts are classified as SMALL significance. These conclusions are found to remain valid for an advanced reactor design (i.e., AP1000) located at the HAR site, specifically, as follows:

- The HAR site risks for the atmospheric exposure pathway will be within the range of those considered as SMALL significance in NUREG-1437.

- The consequences of atmospheric fallout onto open bodies of water are of SMALL significance for the HAR site.

- The consequences of releases to the groundwater are of SMALL significance for the HAR site.

### 7.2.5 HAR SITE-SPECIFIC LEVEL 3 PRA ANALYSIS

This subsection updates the Westinghouse generic PRA analysis of severe accidents to include HAR site-specific attributes in the Level 3 modeling. The Level 3 PRA model uses the NRC-endorsed MACCS2 computer code, the same code used by Westinghouse. The MACCS2 dose pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground and skin, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water. The MACCS2 code primarily addresses dose from the air pathway, but also calculates dose from surface runoff and deposition on surface water. The code also evaluates the extent of contamination to the surrounding area.

To assess human health impacts, the collective dose to the 80-km (50-mi.) population, number of latent cancer fatalities, and number of early fatalities associated with a severe accident were determined. Economic costs were also determined, including the costs associated with relocation of people, decontamination of property and equipment, and interdiction of food supplies.

#### 7.2.5.1 HAR MACCS2 Input

The AP1000 PRA formed the foundation for the HAR MACCS2 analysis and is described in Section 19 of the AP1000 DCD. The PRA identified six source term categories that may be used to represent the suite of potential severe accidents, and the internal events accident frequency associated with each (i.e., core damage frequency [CDF]). The six source terms categories or accident classes are fully described in the AP1000 DCD, and are titled as follows:

- Early Containment Failure (CFE)

- Intermediate Containment Failure (CFI)

- Late Containment Failure (CFL)

- Containment Bypass (BP)

- Containment Isolation Failure (CI)

- Intact Containment (IC)

MACCS2 uses five input files to process numerous user specified parameters. The input files include: ATMOS, MET, SITE, EARLY, and CHRONC. AP1000 design-specific and HAR site-specific parameters are used where appropriate. Otherwise, input parameters are consistent with the MACCS2 User's Guide, those provided in Sample Problem A (distributed with the MACCS2 code), or other recognized sources.

The ATMOS file includes inputs specific to the reactor and plume release and dispersion after an accident. AP1000-specific input includes the core inventory, reactor and associated building dimensions, and source terms including release fractions, developed based on data provided by Westinghouse (Reference 7.2-001). Consistent with the Westinghouse modeling, releases were assumed to occur at the top of the containment building, and plume heat energy was neglected.

The meteorological data used in the MACCS2 model MET file consisted of 5 years of hourly observations of wind speed, wind direction, stability class (derived from vertical temperature gradient), and precipitation. HNP/HAR site-specific meteorology data was obtained from the existing HNP on-site meteorological monitoring station that is located east of the existing and proposed reactor sites as described in Sections 2.7 and 6.4. The period of record for the MACCS2 MET file data is 2001 through 2005. Based on an analysis of all 5 years of meteorological data, the worst year (i.e., the year that resulted in the highest predicted off-site impacts) was determined to be 2003 and was subsequently used as the base case for additional analysis. The meteorological data used in these analyses are identical to what was used as input to the MACCS2 model for the HNP license renewal application environmental report. While the meteorological data period of record used in the MACCS2 analysis differs from the period of record used in the X/Q analysis provided in Section 2.7 (March 1, 1994 through February 28, 1999), it is noted that the MACCS2 modeling analysis and associated results are focused on 50-mi. impacts (cost and dose). At these distances, MACCS2 calculations tend to be driven by higher wind speeds and precipitation related deposition impacts, whereas X/Q impacts are typically driven by low wind speeds and at closer downwind distances.

The SITE file includes inputs specific to the region surrounding the reactor site. HAR site-specific parameters are used in the SITE file, which include year 2060 projected population, land fraction, watershed indices and ingestion factors, and economic data. HAR site-specific economic parameters for the 80-km (50-mi.) region are developed based on the SECPOP2000 methodology using the 2002 Census of Agriculture, Bureau of Labor Services, and Bureau of Economic Analysis data (Reference 7.2-002, Reference 7.2-003, and Reference 7.2-004).

The EARLY file includes input specific to the early time phase (i.e., one week) after an accident, which is used to calculate early dose exposure and health effects. Protective action considerations are included in the input file using the HAR site-specific inputs. Protective action considerations include the evacuation time estimates for the 16 km (10 mi.) emergency planning zone (Reference 7.2-005). Shielding and exposure factors are those used for Surry (provided in Sample Problem A). Ninety-five percent of the population was assumed to evacuate following the declaration of a general emergency.

The CHRONC file includes input specific to the long-term consequences of an accident. Input parameters in the CHRONC file are used to calculate long-term dose and health effect estimates, as well as off-site economic cost estimates associated with interdiction, decontamination, and land condemnation. HAR site-specific input includes updating generic economic cost input to the 2007 value using the Consumer Price Index, as well as calculating HAR site-specific farm and nonfarm wealth values based on the 2002 Census of Agriculture, Bureau of Labor Services, and Bureau of Economic Analysis data (Reference 7.2-002, Reference 7.2-003, and Reference 7.2-004).

#### 7.2.5.2 HAR MACCS2 Results

The results of the HAR MACCS2 calculation and AP1000 internal event accident frequencies are used to calculate the risk from a severe accident for the region surrounding the HAR site. The risk is calculated as the product of the individual accident class frequency multiplied by the MACCS2 consequence associated with that accident class, such that the overall result represents the frequency weighted risk for the metric of interest (for example, population dose risk, early fatality risk, latent cancer fatality risk, cost risk) caused by internal events.

The HAR MACCS2 summary results are provided in Table 7.2-2. The results associated with each accident category are provided in Table 7.2-3 and Table 7.2-4. The results presented incorporate a variety of contributors such as evacuation costs, value of crops contaminated and condemned, value of milk contaminated and condemned, cost of decontamination of property, and indirect costs resulting from loss of use of the property and incomes derived as a result of the accident. Discussion of the results is presented in the following subsections.

In addition, the following quantitative health objectives are used in determining achievement of the safety goals for the operation of a reactor in the United States:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the United States population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

These quantitative health objectives are translated into two numerical objectives as follows:

- The individual risk of a prompt fatality from all "other accidents to which members of the United States population are generally exposed," such as fatal automobile accidents, is about $5 \times 10^{-4}$ per year. One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5 \times 10^{-7}$ per reactor year.

- "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 $2 \times 10^{-3}$ per year. One-tenth of 1 percent of this implies that the risk of cancer to the population in the area near a nuclear power plant because of its operation should be limited to $2 \times 10^{-6}$ per reactor year.

Table 7.2-5 presents the the average individual risk for early fatalities and latent cancer fatalities from severe accidents associated with the operation of the HAR. Table 7.2-6 presents that average individual risk from early fatalities and latent cancer as compared to the safety goal.

7.2.5.3 MACCS2 Analysis Results for Atmospheric Pathway

Table 7.2-7 presents the population dose risk 2.20E-03 person-Sv/yr. (2.20E-01 person-rem/yr.) calculated by MACCS2 for all pathways considered in MACCS2. The atmospheric pathway dose, however, is a large portion of the total population dose, so the total population dose is used here to represent the atmospheric dose risk.

The HAR MACCS2 population dose result is compared to the total population dose risk results of other studies in Table 7.2-7 (based on internal events). As can be seen, the population dose risk for the AP1000 at the HAR site is lower then current design reactors and is less than one percent of that associated with the HNP. It is noted that the HAR population dose risk is slightly larger than that listed in the AP1000 DCD, for a generic site. This is attributed to the fact that the AP1000 generic analysis is based on the 24-hr. dose while the HAR MACCS2 analysis (as well as the other studies) includes long-term dose contributors.

7.2.5.4 MACCS2 Analysis Results for Fallout onto Open Bodies of Water

Following a severe accident, a radiation hazard may exist from the deposition of airborne, radioactive fallout onto open bodies of water. Depending on the type of water body, this hazard may lead to internal exposure from the ingestion of contaminated water or from consuming contaminated aquatic fauna. External exposure may result from swimming in the contaminated water or from recreational activities on the shoreline. The extent of the hazard is largely determined by the proximity of individuals to the reactor, the extent of contamination, and the ability for interdiction to reduce the exposure hazard. Of these various water-related pathways, MACCS2 calculates only the dose from drinking water.

As presented in Table 7.2-2, the HAR MACCS2 total population dose risk is 2.20E-03 person-Sv/yr. (2.20E-01 person-rem/yr.). The MACCS2 portion derived from drinking water is 3.15E-05 person-Sv/yr (3.15E-03 person-rem/yr.), which is less than 2 percent of the total population dose. This is judged to represent a very SMALL impact.

Although the other surface water pathways (for example, consuming aquatic fauna, swimming) are not modeled by MACCS2, they have been evaluated previously NUREG-1437 and shown to be of SMALL significance for most sites, especially if interdiction is considered. Therefore, consideration of atmospheric fallout onto open bodies of water can be concluded to be a SMALL impact.

#### 7.2.5.5 MACCS2 Analysis Results for Groundwater Pathways

People can also receive a radiation exposure from groundwater pathways. For this pathway, the core is postulated to melt down, breach the reactor vessel, and fall onto the reactor building floor. As a result of chemical energy and decay heat, the melted fuel reacts with the concrete floor. Without the cooling water addition to the core debris, the basemat of the containment building may eventually breach, and molten core debris and radioactive water penetrate strata beneath the plant. The soluble radionuclides in the debris can be leached and transported with groundwater and contaminated reactor water to downgradient domestic wells used for drinking water or to surface water bodies used for drinking water, aquatic food, and recreation.

Groundwater pathways are not modeled by MACCS2. The HAR site has the same groundwater characteristics as the existing HNP, which has been evaluated in NUREG-1437, and shown to be acceptable. Because the severe accident frequency associated with the HAR units is lower than that of the HNP, the dose risk attributed to groundwater pathway is less than that of the HNP. The AP1000 design has intentionally included design elements to minimize the potential for a severe accident to lead to core concrete interactions and an eventual breach of the containment building basemat. These design elements include in-vessel retention of core debris by external reactor vessel cooling (i.e., submerging the reactor vessel in water to facilitate cooling and thereby prevent vessel failure) and ex-vessel core debris cooling in the reactor cavity (i.e., providing a water-filled reactor cavity to receive core debris upon vessel failure). These design elements are discussed in more detail in the AP1000 DCD.

Based on the previous discussion, the consideration of groundwater pathways can be concluded to be a SMALL impact.

#### 7.2.5.6 External Event Risk

The HAR MACCS2 results previously presented are based on internal events, consistent with the Level 3 risk results presentation in the AP1000 DCD. The AP1000 DCD, however, does present the AP1000 core damage frequency contributions associated with external events and internal flooding, as summarized in Table 7.2-8.

The internal flood and internal fire CDF contributions combined are only approximately 24 percent of the internal events CDF. Because the seismic CDF is not quantified for the AP1000, it cannot be evaluated quantitatively as a contributor.

To generically evaluate the potential risk impacts associated with these additional events, the internal events core damage frequency may be multiplied by a factor of two, and the assumption made that the release category frequency proportions remain the same. Using these assumptions, the population dose risk for all at-power events would be 4.40E-03 person-Sv/yr. (4.40E-01 person-rem/yr.), that is, twice that calculated for internal events alone. This value is still very small and is significantly less than the risk associated with only internal events of current plant designs (presented in Table 7.2-7). Therefore, external event risk is judged to be acceptable.

#### 7.2.5.7 Cumulative Risk

The HAR MACCS2 analysis examines the risk caused by internal events associated with a single AP1000 plant. It is noted that Progress Energy Carolinas, Inc. (PEC) proposes constructing two AP1000 plants at the site. The two new units would be colocated with the HNP. In consideration of the multiple units located on the HNP, the cumulative population dose risk may be estimated by summing the individual dose risk associated with each unit, as provided in Table 7.2-9.

Table 7.2-9 demonstrates that the cumulative risk of constructing two new advanced AP1000 reactors at the HNP increases a negligible amount over that associated with the HNP.

#### 7.2.5.8 Impacts to Biota

The impact of radiological releases caused by severe accidents on biota (for example, plants, animals, and endangered species) is a special consideration. ER Section 2.4 discusses ecological considerations of the HNP, vicinity, and region, including the presence of threatened and endangered species. Off-site dose consequences and health effects for normal and anticipated releases are included in Section 7.1 and will not be repeated here.

Because of the spectrum of potential biota surrounding a plant and the lack of specific data regarding impacts of severe accident dose exposures to biota, dose criteria applicable to humans may be applied. Such application of human-based criteria to biota, even when adjusted for body mass and size, are judged to be conservative. Human dose conversion factors are based on 30- to 50-yr. life expectancy exposure predictions. Life expectancy for biota is generally considerably shorter, thereby limiting the cumulative radiological impacts. These considerations support the general conclusion that impacts to biota can be reasonably approximated or bounded by impact estimates to humans.

The impacts to the human population (i.e., population dose risk) surrounding the HNP caused by severe accident radiological releases, as evaluated using MACCS2 for the AP1000 plant, have been shown to be significantly less than the current generation of operating plants. The severe accident radiological release impacts for the AP1000 plant are approximately two orders of magnitude less than that estimated for HNP. The significantly lower population dose risk of the AP1000 plant at the HAR site supports the conclusion that the radiological impacts to biota caused by severe accidents is of SMALL significance.

#### 7.2.5.9 MACCS2 Analysis Conclusions

The HAR MACCS2 analysis of severe accidents for the AP1000 reactor design shows that the 80-km (50-mi.) population dose risk of 2.20E-03 person-Sv/yr. (2.20E-01 person-rem/yr.) is significantly lower than that for current reactor designs and is less than 1 percent of that associated with the current HNP.

This population dose is primarily attributable to the atmospheric pathway. MACCS2 does not specifically calculate population dose resulting from radioactive fallout onto open bodies of water except for doses associated with drinking water (i.e., external exposure from recreational activities like swimming in contaminated water, or consuming contaminated aquatic fauna is not calculated). The MACCS2 population dose derived from drinking water is less than 2 percent of the total population dose.

Based on the metric of the 80-km (50-mi.) population dose, the cumulative population dose risk associated with constructing two AP1000 plants at the site will increase a negligible amount over that associated with the current HNP. Thus, the environmental impacts are found to be of SMALL significance.

Other metrics of interest, including early fatality risk, latent cancer fatality risk, affected land, and cost risk are presented. The calculated cost risk value of $2010/yr. is used in Section 7.3 for the SAMA analysis.

### 7.2.6 CONCLUSIONS

In NUREG-1437, the NRC specifically evaluated the HNP for severe accident impacts for license renewal considerations and concluded that the environmental impacts were of SMALL significance. The consequences could be severe, but because of their low likelihood of occurrence, the risk impact is classified as SMALL significance. Specifically, the following conclusions are found to remain valid for an advanced reactor design (i.e., AP1000) located at the HAR site:

- The HAR site risks for the atmospheric exposure pathway will be within the range of those considered as SMALL significance in NUREG-1437.
- The consequences of atmospheric fallout onto open bodies of water are of SMALL significance for the HAR site.
- The consequences of releases to the groundwater are of SMALL significance for the HAR site.

The HAR site-specific MACCS2 analysis of severe accidents for the AP1000 reactor design shows that the 80-km (50-mi.) population dose risk of 2.20E-03 person-Sv/yr. (2.20E-01 person-rem/yr.) is significantly lower than that for current reactor designs and is less than 1 percent of that associated with the current HNP.

Based on the metric of the 80-km (50-mi.) population dose, the cumulative population dose risk associated with constructing two AP1000 plants at the site will increase a negligible amount over that associated with the current HNP. Thus the environmental impacts are found to be of SMALL significance.

## 7.2.7 REFERENCES

7.2-001 Westinghouse Electric Company, LLC, Response to RFI# 205, “Request for Fission Product Release Fractions as a Function of Time for the Six Release Categories,” June 2007.

7.2-002 U.S. Department of Agriculture, “2002 Census of Agriculture, North Carolina State and County Data,” AC-02-A-33, Vol. 1, Part 33, June 2004.

7.2-003 U.S. Department of Labor, Bureau of Labor Statistics, www.bls.gov/data/, Accessed May 7, 2007.

7.2-004 Bureau of Economic Analysis, Regional Economic Accounts, “North Carolina Population [CA1-3 – personal income summary],” Website, www.bea.gov/regional/reis/drill.cfm, accessed 2007.

7.2-005 Innovative Emergency Management, “Evacuation Time Estimates for the Harris Nuclear Plant,” IEM/TEC02-065, October 29, 2002.

**Table 7.2-1**
**NUREG-1437 Exposure Index (EI) Values**

| Plant | 16 km (10 mi.) EI | | 241 km (150 mi.) EI | |
|---|---|---|---|---|
| | Yr. 2010[(a)] | Yr. 2050[(b)] | Yr. 2010[(a)] | Yr. 2050[(c)] |
| Limerick | 10,307 | 10,709 | 2,455,497 | 2,647,224 |
| HNP | 1415 | 1773 | 550,951 | 688,554 |
| Vogtle | 117 | 141 | 469,641 | 590,283 |

Notes:
a) NUREG-1437—Table 5.5

b) NUREG-1437—Table 5.7

c) NUREG-1437—Table 5.8

**Table 7.2-2**
**HAR MACCS2 Results**
**(0 to 80-km [50-mi.] Radius, Internal Events Only)**

| Plant Design | 80-km (50-mi.) Dose Risk (person-Sv/yr./ rem/yr.) | Cost Risk ($/yr.) | Fatality Risk (per yr.) | |
|---|---|---|---|---|
| | | | Early | Latent Cancer |
| AP1000 | 2.2E-03/2.2E-01 | 2010 | 2.55E-9 | 1.25E-04 |

**Table 7.2-3**
**HAR MACCS2 Consequence Results by Source Term**
**(0 to 80-km [50-mi.] Radius, Internal Events Only)**

| Source Term | Frequency (per yr.) | Dose (person-Sv/ person-rem) | Dose Risk (person-Sv/yr./ person –rem/yr.) | Early Fatalities | Early Fatality Risk (per yr.) | Latent Cancer Fatalities | Latent Cancer Fatality Risk (per yr.) | Total Cost ($) | Cost Risk ($/yr.) |
|---|---|---|---|---|---|---|---|---|---|
| ST1 - CFI | 1.89E-10 | 6.27E+04/6.27E+06 | 1.19E-05/1.19E-03 | 6.32E-03 | 1.19E-12 | 2.89E+03 | 5.46E-07 | 5.48E+10 | 1.04E+01 |
| ST2 - CFE | 7.47E-09 | 6.70E+04/6.70E+06 | 5.00E-04/5.00E-02 | 6.51E-02 | 4.86E-10 | 3.65E+03 | 2.73E-05 | 5.28E+10 | 3.94E+02 |
| ST3 - IC | 2.21E-07 | 2.44E+02/2.44E+04 | 5.39E-05/5.39E-03 | 0.00E+00 | 0.00E+00 | 1.09E+01 | 2.41E-06 | 2.48E+06 | 5.48E-01 |
| ST4 - BP | 1.05E-08 | 1.50E+05/1.50E+07 | 1.58E-03/1.58E-01 | 1.96E-01 | 2.06E-09 | 8.45E+03 | 8.87E-05 | 1.47E+11 | 1.54E+03 |
| ST5 - CI | 1.33E-09 | 6.27E+04/6.27E+06 | 8.34E-05/8.34E-03 | 5.91E-03 | 7.86E-12 | 4.56E+03 | 6.06E-06 | 4.31E+10 | 5.73E+01 |
| ST6 - CFL | 3.45E-13 | 2.94E+04/2.94E+06 | 1.01E-08/1.01E-06 | 0.00E+00 | 0.00E+00 | 9.72E+02 | 3.35E-10 | 6.77E+10 | 2.34E-02 |
| Total | 2.41E-07 | -- | 2.22E-03/2.22E-01 | -- | 2.55E-09 | -- | 1.25E-04 | -- | 2.01E+03 |

**Table 7.2-4
Affected Land Results by Source Term
(0 to 80-km [50-mi.] Radius)**

| Source Term | Decontaminated Land (hectares) | Condemned Land (hectares) |
|---|---|---|
| ST1 - CFI | 27,100 | 2430 |
| ST2 - CFE | 30,600 | 869 |
| ST3 - IC | 3 | 0 |
| ST4 - BP | 82,800 | 3950 |
| ST5 - CI | 23,800 | 2040 |
| ST6 - CFL | 53,600 | 250 |
| **Worst Case** | **82,800** | **3950** |

**Table 7.2-5
HAR AP1000
Average Individual Risk from Early Fatalities and Latent Cancers**

| Source | Frequency | Population Weighted | Frequency Weighted | Early Fatalities Contribution | Population Weighted | Frequency Weighted | Latent Cancer Fatalities Contribution |
|---|---|---|---|---|---|---|---|
| **Term** | **(per y.r)** | **Early Fatalities (1.6 km [1 mi.])** | **Early Fatalities Risk** | **%** | **Latent Cancer Fatalities (16 km [10 mi.])** | **Latent Cancer Fatalities Risk** | **%** |
| ST1 - CFI | 1.89E-10 | 1.53E-04 | 2.89E-14 | 0.21 | 1.38E-03 | 2.61E-13 | 0.83 |
| ST2 - CFE | 7.47E-09 | 8.37E-04 | 6.25E-12 | 45.45 | 1.91E-03 | 1.43E-11 | 45.20 |
| ST3 - IC | 2.21E-07 | 0.00E+00 | 0.00E+00 | 0.00 | 1.04E-05 | 2.30E-12 | 7.28 |
| ST4 - BP | 1.05E-08 | 6.88E-04 | 7.22E-12 | 52.52 | 1.04E-03 | 1.09E-11 | 34.60 |
| ST5 - CI | 1.33E-09 | 1.88E-04 | 2.50E-13 | 1.82 | 2.87E-03 | 3.82E-12 | 12.09 |
| ST6 - CFL | 3.45E-13 | 0.00E+00 | 0.00E+00 | 0.00 | 1.34E-04 | 4.62E-17 | 0.00 |
| Total | 2.40E-07 | -- | **1.38E-11** | 100.00 | -- | **3.16E-11** | 100.00 |

## Table 7.2-6[a]
## Comparison of the Average Individual Risk versus the Safety Goal

| Consequence Metric | HAR MACCS2 Results | Safety Goal |
|---|---|---|
| Early Fatalities[b] | 1.4E-11 | < 5E-07[d] |
| Latent Cancer Fatalities[c] | 3.2E-11 | < 2E-06[e] |

Notes:
a) Frequency weighted for each source term (based on internal events only).

b) Population weighted early fatality risk within 1.6 km (1 mi.), includes evacuation.

c) Population weighted latent cancer fatality risk within 16 km (10 mi.), includes evacuation.

d) Individual risk consequence goal is based on the NRC safety goal policy statement and developed into a numerical goal by the NRC staff in NUREG-1811, volume 1 (North Anna EIS), December 2006.

e) Societal risk consequence goal is based on the NRC safety goal policy statement and developed into a numerical goal by the NRC staff in NUREG-1811, volume 1 (North Anna EIS), December 2006.

## Table 7.2-7
## Mean Annual Dose Risk for Several Sites
## (Internal Events Only)

| Plant | Population Dose Risk (80-km [50-mi.]) (person-Sv/yr. [person-rem/yr.]) |
|---|---|
| HAR AP1000[a] | 2.20E-03/ 2.20E-01 |
| Zion | 5.47EE-01/5.47E+01[b] |
| Grand Gulf | 5.20E-03/5.20E-01[c] |
| Surry | 5.80E-02/5.80E+00[d] |
| HNP | 2.89E-01/2.89E+01[e] |
| DCD AP1000 | 4.30E-04/4.32E-02[f] |

Notes:
a) Located at the HNP.

b) Table 5.1-1 in NUREG/CR-4551, Vol. 7, Rev. 1.

c) Table 5.1-1 in NUREG/CR-4551, Vol. 6, Rev. 1.

d) Table 5.1-1 in NUREG/CR-4551, Vol. 3, Rev. 1.

e) Table E.3-3.

f) Table 1B-1 in AP1000 DCD, located at a Generic Site, 24-hr.emergency phase dose only.

## Table 7.2-8
## AP1000 PRA CDF Results [a]

| Events | Core Damage Frequency (/yr.) (At-power) |
|---|---|
| Internal Events | 2.41E-7 |
| Internal Flood | 8.82E-10 |
| Internal Fire | 5.61E-8 |
| Seismic | NA[b] |
| Total | 2.97E-7 |

Notes:
a) Based on Table 1B-2 of the AP1000 DCD.

b) Seismic risk CDF is not quantified for the AP1000. The seismic margin method was used.

## Table 7.2-9
## Mean Annual Cumulative Dose Risk
## (Due To Internal Events Only)

| Plant | Population Dose Risk (50-mi.) (person-Sv/yr. [person-rem/yr.]) |
|---|---|
| HNP | 2.89E-01/2.89E+01[a] |
| Proposed HAR 2 (AP1000)[b] | 2.20E-03/ 2.20E-01 |
| Proposed HAR 3 (AP1000)[b] | 2.20E-03/ 2.20E-01 |
| Total | 2.94E-01/2.94E+01 |

Notes:
a) Table E.3-3.

b) Located at the HAR site.

## 7.3 SEVERE ACCIDENT MITIGATION MEASURES

A Severe Accident Mitigation Design Alternative (SAMDA) evaluation was performed for the AP1000 plant design and is presented in the DCD, Appendix 1B. The evaluation was performed to identify potential safety beneficial design alternatives and to evaluate whether the safety benefit of the alternative design candidates outweighs the costs associated with implementation. Because the AP1000 is an advanced reactor design that incorporates many safety features, the SAMDA analysis did not find any additional design alternatives to be cost beneficial. The AP1000 SAMDA analysis was based on data representing a generic site.

This section updates the Westinghouse SAMDA analysis based upon the HAR site specific MACCS2 model results presented in Section 7.2 (Severe Accidents) to determine if the DCD conclusions remain valid (i.e., none of the identified design alternatives are cost beneficial).

### 7.3.1 THE SAMA ANALYSIS PROCESS

Design or procedural modifications that could mitigate the consequences of a severe accident are known as SAMAs. In the past SAMAs were known as SAMDAs, which primarily focused on design changes and did not consider procedural changes. The Westinghouse DCD analysis is a SAMDA analysis.

For an existing plant with a well-defined design and established procedural controls, the normal evaluation process for identifying potential SAMAs includes the following four steps:

1. **Define the Baseline** — The plant's PRA results are used to calculate the population dose risk and cost risk associated with severe accidents in the baseline plant configuration (i.e., before implementation of any SAMAs). The NRC-approved methodologies are used to calculate the monetary value of unmitigated severe accident risk. This monetary value, sometimes termed the Maximum Averted Cost Risk (MACR), reflects the monetary value of eliminating all severe accident risk, and therefore, provides a conservative baseline screening value for the SAMA candidates.

2. **Identify and Screen Potential SAMAs** — Potential SAMA candidates are identified from the plant's Individual Plant Examination, insights from the plant's PRA, and the results of other plants' SAMA analyses. A conservatively low implementation cost for each SAMA candidate is estimated based on historical costs, similar design changes, and/or engineering judgment. The estimated implementation costs are then compared against the baseline screening value (MACR). SAMA candidates whose implementation cost exceeds the MACR can be screened and not evaluated further.

3. **Develop Detailed Cost Estimates** — For each SAMA remaining following the screening process, a detailed engineering cost estimate is developed using current plant engineering processes. If a SAMA candidate-detailed cost estimate is below the MACR, the candidate is retained for further detailed benefit estimation.

4. **Develop Detailed Benefit Estimates** — For each SAMA remaining unscreened, the PRA model is used to determine the risk reduction associated with implementation of the proposed SAMA. The benefit risk reduction is then monetized, and the cost benefit is evaluated. Cost beneficial SAMA candidates are further evaluated for implementation.

The scope of the plant PRA available is often limited to internal events. However, external events (e.g., seismic events, fire events) have been identified by the nuclear industry as small, but non-negligible contributors to plant risk. SAMA assessments generally address the potential impact of external events through either their inclusion quantitatively (where frequency data is available), through quasiquantitative inclusion (for example, using a common multiplier factor on the internal event inputs or the MACR result), through sensitivity studies, qualitative assessment, or a combination of all of these.

### 7.3.2 AP1000 DCD SAMDA ANALYSIS

The AP1000 SAMDA evaluation is presented in Appendix 1B of the DCD. A list of SAMDA candidates was developed based on a review of SAMDAs evaluated for other plant designs, including the AP600, and probabilistic risk assessment results. Fifteen candidate design alternatives were selected for further evaluation for the AP1000 design. Table 7.3-1 identifies the 15 candidate design alternatives considered for the AP1000 and the estimated implementation costs for each. Additional discussion of each design alternative is presented in the AP1000 DCD.

An evaluation of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative was conservatively maximized. As part of this process, it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the implementation cost estimate for each alternative was intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

Using the cost benefit calculation methodology of NUREG/BR-0184, the MACR was calculated using the dose risk and cost risk values developed for a generic site. The calculated MACR value was $21,000.

A comparison of the implementation costs for each SAMDA to the MACR value of $21,000 found that none of the SAMDAs would be cost effective. The least costly SAMDA, self-actuating containment isolation valves, had an implementation cost of approximately $33,000, with the others having costs at least an order of magnitude greater. The self-actuating containment isolation valve SAMDA candidate was further evaluated and found to result in minimal risk reduction achievement, thereby confirming its status as not cost beneficial.

7.3.3 HAR SAMA ANALYSIS

For the HAR site, the DCD SAMDA evaluation is reperformed incorporating the HAR MACCS2 analysis results to determine if the DCD conclusions remain valid.

The principal inputs to the baseline calculation are the internal events core damage frequency (reported in Section 7.2), population dose risk and cost risk (reported in Table 7.2-2), exposure cost value ($2,000/person-rem/year, as provided in NUREG/BR-0184, licensing period (40 years), and economic discount rate (7 percent).

For the HAR analysis, the MACR value based on internal events was calculated to be approximately $22,000. To account for external events, this MACR value was multiplied by a factor of two to achieve an MACR value of $44,000. As discussed in Section 7.2, and presented in Table 7.2-7, the internal flood and internal fire CDF contributions combined are only approximately 24 percent of the internal events CDF. The seismic CDF is not quantified for the AP1000, it cannot be evaluated quantitatively as a contributor. To generically evaluate the potential impacts associated with internal flooding and external events, a factor of two is applied to the MACR result, which is equivalent to applying a factor of two to the MACCS2 population dose risk and cost risk results. The MACR results are presented in Table 7.3-2, showing the various contributors.

The 15 SAMDA candidates identified in the AP1000 DCD form an initial list of potential cost beneficial plant modifications. In consideration of additional potential candidates for the HAR SAMA analysis, it is noted that the NRC previously evaluated additional potential design candidates for the AP1000 SAMDA, as documented in NUREG-1793, including those candidates evaluated for the AP600 which might have applicability to the AP1000. NUREG-1793 indicates that “the staff’s review of more than 120 candidate design alternatives considered for the AP600 did not identify any new alternatives more likely to be cost beneficial than those included in the AP1000 design evaluations.” Regarding the NRC review of the AP1000 candidates, NUREG-1793 states that “the staff’s review did not reveal any additional design alternatives that obviously should have been given consideration by the applicant.” Based on the previous extensive review for additional design candidates, no new design candidates are identified.

In the absence of a completed plant with established procedural and administrative controls, the HAR analysis can only evaluate physical plant modifications. Evaluation of administrative SAMAs would not be appropriate until a plant design is finalized, and plant administrative processes and procedures are being developed. At that time, appropriate administrative controls on plant operations will be incorporated into the plant's management systems as part of its baseline.

The implementation cost estimates developed by Westinghouse for the AP1000 SAMA candidates have been reviewed by the NRC for reasonableness, including comparisons with cost estimates developed for other plant designs, such as the ABWR and combustion engineering (CE) System 80+, as documented in NUREG-1793. The NRC concluded that the approximate cost estimates developed by Westinghouse are adequate for the purposes of the cost benefit evaluation. Therefore, no implementation cost estimate revisions are judged required for the HAR SAMA analysis.

When the HAR site MACR is compared against the implementation costs of the AP1000 SAMDA candidate design alternatives presented in Table 7.3-1, only one alternative has the potential of being cost effective. Alternative 3 (self-actuating containment isolation valves) has a cost below the MACR value of $44,000. The remaining alternatives are nearly an order of magnitude more costly (i.e., the next lowest cost alternative being alternative 14 (a more reliable diverse actuation system) with an estimated implementation cost of $470,000. Thus, only design alternative 3 needs to be further evaluated.

The AP1000 DCD further examines this design alternative and notes that this alternative provides almost no benefit in reducing the plant CDF, and the benefit related to release can be estimated by assuming the modification eliminated all the CI release category. Using these assumptions, the AP1000 DCD finds that the benefit is of the order of a few thousand dollars, and therefore not cost beneficial. The HAR MACCS2 analysis (Table 7.2-3) shows that the CI release category contributes only approximately 3 percent to the total population dose risk and cost risk, such that there would be a negligible quantified benefit. The HAR MACCS2 analysis thus confirms the AP1000 DCD conclusions that this SAMA candidate is not cost beneficial.

A number of SAMA sensitivity cases were examined to assess the impact of key inputs and assumptions. The results of the sensitivity cases are presented in Table 7.3-3. The sensitivity cases examined are similar to those conducted in the AP1000 SAMDA. The results indicate that there is significant margin in the conclusions of the SAMA analysis, and that none of the SAMA candidates are cost beneficial for the AP1000 plant located at the HAR site.

### 7.3.4 CONCLUSIONS

For the HAR site, the AP1000 DCD SAMDA evaluation has been reperformed incorporating the HAR MACCS2 analysis results and found that the DCD conclusions remain valid. No SAMA candidates are found to be cost beneficial.

This conclusion is consistent with the NRC AP1000 SAMDA review conclusions presented in NUREG-1793, which states the following:

The staff concurs with the applicant's conclusion that none of the potential design modifications evaluated are justified on the basis of cost benefit considerations. It is further concluded that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations because the estimated CDFs would remain very low on an absolute scale.

**Table 7.3-1[(a)]**
**AP1000 SAMDA Candidate Design Alternatives**

| No. | Design Alternative | Implementation Cost ($) |
|---|---|---|
| 1 | Upgrade Chemical, Volume, and Control System for Small Loss of Coolant Accident (LOCA) | 1,500,000 |
| 2 | Containment-Filtered Vent | 5,000,000 |
| 3 | Self-Actuating Containment Isolation Valves | 33,000 |
| 4 | Safety Grade Passive Containment Spray | 3,900,000 |
| 5 | Active High Pressure Safety Injection System | NA - (Not consistent with passive system design objectives) |
| 6 | Steam Generator (SG) Shellside Heat Removal | 1,300,000 |
| 7 | SG Relief Flow to In-Containment Refueling Water Storage Tank (IRWST) | 620,000 |
| 8 | Increased SG Pressure Capability | 8,200,000 |
| 9 | Secondary Containment Ventilation with Filtration | 2,200,000 |
| 10 | Diverse IRWST Injection Valves | 570,000 |
| 11 | Diverse Containment Recirculation Valves | NA - (Already implemented in the AP1000 design) |
| 12 | Ex-Vessel Core Catcher | 1,660,000 |
| 13 | High-Pressure Containment Design | 50,000,000 |
| 14 | More Reliable Diverse Actuation System | 470,000 |
| 15 | Locate Residual Heat Removal System Inside Containment | NA - (Negligible achievable risk reduction) |

Notes:

a) Based on Table 1B-5 of the AP1000 DCD.

## Table 7.3-2
## HAR SAMA Baseline Costs

| | |
|---|---|
| Off-Site Exposure Cost | $2398 |
| Off-Site Economic Cost | $10,855 |
| On-Site Exposure Cost | $88 |
| On-Site Cleanup Cost | $3557 |
| Replacement Power Cost | $5046 |
| Summed Cost<br>(Based on Internal Events) | $21,944 |
| Total Cost<br>(Summed Cost X 2 to Account For External Events and Rounded Up) | $44,000 |

## Table 7.3-3
## Cost Benefit Sensitivity Results

| | Case Studied | Cost ($) |
|---|---|---|
| Base Case | 7-percent Discount Rate | 44,000 |
| S-1 | 3-percent Discount Rate | 109,000 |
| S-2 | High Dose (10 times the base case) | 88,000 |
| S-3 | 50-percent core damage frequency (CDF) | 22,000 |
| S-4 | Twice the base CDF | 88,000 |
| S-5 | 10 times the benefit (10x MACR) | 439,000 |

## 7.4 TRANSPORTATION ACCIDENTS

The advanced light water reactor (ALWR) technology being considered for the HAR and alternative sites (Brunswick Nuclear Power Plant (BNP), H.B. Robinson Nuclear Power Plant (RNP) and Marion County [refer to ER Subsection 9.3.2]) is the AP1000. The configuration for this new nuclear power generating facility is two units. A single AP1000 unit was used to evaluate transportation impacts in ER Section 3.8 and the accidents from transportation in this section relative to the reference light water reactor (LWR) in WASH-1238.

Subparagraphs 10 CFR 51.52(a)(1) through (5) delineate specific conditions the reactor licensee must meet to use Table S-4 (reproduced in this ER as Table 3.8-1) as part of its ER. For reactors not meeting all of the conditions in paragraph (a) of 10 CFR 51.52, paragraph (b) of 10 CFR 51.52 requires a further analysis of the transportation effects.

The conditions in paragraph (a) of 10 CFR 51.52 establishing the applicability of Table S-4 are reactor core thermal power, fuel form, fuel enrichment, fuel encapsulation, average fuel irradiation, time after discharge of irradiated fuel before shipment, mode of transport for unirradiated fuel, mode of transport for irradiated fuel, radioactive waste form and packaging, and mode of transport for radioactive waste other than irradiated fuel.

Based on comparison of the AP1000 characteristics to the criteria listed in 10 CFR 51.52(a), the AP1000 does not meet the following two evaluation criteria (as discussed in ER Subsections 3.8.1.3 and 3.8.1.5, respectively):

- Subparagraph 10 CFR 51.52(a)(2) requires that the reactor fuel have a uranium-235 (U-235) enrichment not exceeding 4 percent by weight. As noted in DCD Table 4.1-1, for the AP1000, the enrichment of the initial core varies by region from 2.35 to 4.45 percent, and the average for reloads is 4.51 percent. The AP1000 fuel exceeds the 4 percent U-235 condition.

- Subparagraph 10 CFR 51.52(a)(3) requires that the average burnup not exceed 33,000 megawatt days per metric ton of uranium (MWd/MTU). According to the DCD, the AP1000 has an average maximum burnup of 60,000 MWd/MTU for the peak rod. The extended burnup is 62,000 MWd/MTU. Therefore, the AP1000 does not meet this subsequent evaluation condition.

Because the AP1000 does not meet all criteria set forth in Table S-4, a subsequent analysis was performed for the HAR and the alternative sites that is used as the supporting basis for ER Section 3.8 and this section.

ER Section 3.8 addresses issues associated with the transportation of radioactive materials from the HAR and alternative sites. This section addresses accidents associated with the shipment of unirradiated and spent fuel.

## 7.4.1 TRANSPORTATION OF UNIRRADIATED FUEL

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 CFR 51.52(a) (see Table 3.8-1). The consequences of accidents that are severe enough to result in a release of unirradiated particles to the environment from ALWR fuels are not significantly different from those for current generation LWRs. The fuel form, cladding, and packaging are similar to those LWRs analyzed in WASH-1238. Consequently, as described in the NRC's assessment of environmental impacts at the North Anna, Clinton, and Grand Gulf Early Site Permit (ESP) sites (NUREG-1811, NUREG-1815, and NUREG-1817, respectively), the NRC concluded that the overall transportation accident risks associated with advanced reactor spent fuel shipments are likely to be SMALL and are consistent with the risks associated with transportation of spent fuel from current generation reactor.

## 7.4.2 TRANSPORTATION OF SPENT FUEL

In its assessments of the proposed ESP sites, the NRC used the radioactive material transportation (RADTRAN) 5 computer code to estimate impacts of transportation accidents involving spent fuel shipments (Reference 7.4-001). As provided in Draft NUREG-1872, "RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences (e.g., "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 NRC conducted a screening analysis on the inventories reported in an Idaho National Engineering and Environmental Laboratory document entitled, "Early Site Permit ER Sections and Supporting Documentation," to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations (Reference 7.4-002). The screening identified the radionuclides that would contribute more than 99.999 percent of the dose from inhalation, and the results are reported in NUREG-1811, NUREG-1815, and NUREG-1817.

Radionuclide inventories are important parameters in the calculation of accident risks. The radionuclide inventories used in this analysis were taken directly from NUREG-1811, NUREG-1815, and NUREG-1817, with the exception of Cobalt-60 (Co-60), which is discussed below.

Co-60 inventories were taken directly from NUREG/CR-6672. The following discussion is from Section 7.2.3.5 of NUREG/CR-6672 and provides a discussion regarding the importance of including Co-60 in the overall source term:

During reactor operation, corrosion products formed in the reactor's primary cooling system deposit on fuel assembly surfaces where elements in these deposits are activated by neutron bombardment. The resulting radioactive deposits are called CRUD. Due to vibratory loads during incident free transportation, impact loads during collision accidents, and thermal loads during accidents that lead to fires, portions of these radioactive deposits may spall from the rods. Then, if some of these spalled materials become airborne during an accident, their release to the atmosphere could contribute to the radiation exposures caused by the accident. Although CRUD contains a number of radionuclides, only Co-60 would contribute significantly to these radiation exposures. Since the CRUD deposits on typical [pressurized water reactor] PWR spent fuel rods typically contain 0.2 [Curies] Ci of Co-60 per rod and the generic PWR assemblies for which ORIGEN inventories were calculated contain respectively 289 spent fuel rods, the amounts of Co-60 produced by activation of deposits on assembly surfaces is 57.8 Ci for the generic PWR assembly (115.6 [Curies per metric ton of uranium] Ci/MTU based on 0.5 MTU/assembly).

The spent fuel inventory used in this analysis for the AP1000 is presented in Table 7.4-1.

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71. Spent fuel shipping casks must be certified Type B packaging systems, meaning they must withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability. As noted in Draft NUREG-1872, "the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). The staff assumed that shipping casks for Westinghouse AP1000 reactor spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo."

The NRC performed the RADTRAN 5 accident risk calculations using unit radionuclide inventories (Ci/MTU) for the spent fuel shipments from the ALWRs. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (metric tons of uranium per year [MTU/yr]) to derive estimates of the annual accident risks associated with spent fuel shipments from each potential ALWR. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 24 MTU/yr for the AP1000. This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2. Information on how these values were calculated is presented in ER Section 3.8.

In the NRC's assessment of the proposed ESP sites, the NRC used the release fractions for current generation LWR fuels to approximate the impacts from the ALWR spent fuel shipments. This assumed that the fuel materials and containment systems (cladding and fuel coatings) behave similarly to current

LWR fuel under applied mechanical and thermal conditions. For this analysis, the same release fractions were used to approximate the impacts from the AP1000 spent fuel shipments.

The shipping distances and population distribution information for the routes from the HAR and alternative sites were the same as those used for the "incident-free" transportation impacts analysis (described in ER Subsection 3.8.2).

Table 7.4-2 presents unit accident risks associated with transportation of spent fuel from the HAR and alternative sites to the proposed Yucca Mountain repository. The accident risks are provided in the form of a unit collective population dose (person-roentgen equivalent man [person-rem]). The table also presents estimates of accident risk per reference reactor year (RRY) normalized to the reference LWR analyzed in WASH-1238.

The estimated shipping distances from the HAR and alternative sites to the spent fuel disposal facility are presented in ER Section 3.8.

## 7.4.3 NONRADIOLOGICAL IMPACTS

Nonradiological impacts are calculated using accident, injury, and fatality rates from published sources. The rates (that is, impacts per vehicle-km traveled) are then multiplied by estimated travel distances for workers and materials. The general formula for calculating nonradiological impacts is as follows:

Impacts = (unit rate) x (round-trip shipping distance) x (annual number of shipments)

In this formula, impacts are presented in units of the number of accidents, number of injuries, and number of fatalities per year. Corresponding unit rates (impacts per vehicle-km traveled) are used in the calculations.

The general approach used in this analysis to calculate nonradiological impacts of unirradiated and spent fuel shipments is based on the approach used in the Yucca Mountain Supplemental Environmental Impact Statement, which used adjusted state-level accident, injury, and fatality statistics, as shown in Table 7.4-3 (References 7.4-003 and 7.4-004). The round-trip distances between the proposed ALWR sites and the fuel fabrication facility (assumed to be located in Columbia, South Carolina, and Lynchburg, Virginia) and Yucca Mountain, Nevada (Table 7.4-4) provided the data for the last part of the equation. State-by-state shipping distances were obtained from the Web-TRAGIS output file and combined with the annual number of shipments and accident, injury, and fatality rates by state (References 7.4-003 and 7.4-004), to calculate nonradiological impacts. The results are shown in Table 7.4-4. The values presented in Table 7.4-5 were calculated from the values reported in Table 7.4-4 multiplied by the applicable number of shipments for unirradiated and spent fuel. Table 7.4-5 values were then compared to those reported in Table S-4 of 10 CFR 51.52 (see

Table 3.8-1). It should be noted that because of the larger round trip distances and greater number of shipments, 95 percent of the total nonradiological impacts (fresh fuel and spent nuclear fuel), are from the shipment of spent nuclear fuel. Also it should be noted that the fatalities/RRY calculated for the shipment of fresh and spent nuclear fuel are slightly smaller than those reported in Table S-4. This is primarily due to the longer shipping distances and adjusted accident, injury, and fatality rate data that were used for the shipment of fresh fuel to and spent fuel from HAR and the alternative sites versus what was used for the basis to support Table S-4.

### 7.4.4 CONCLUSION

Considering the uncertainties in the data and computational methods, the NRC concluded that the overall transportation accident risks associated with ALWR unirradiated and spent fuel shipments are considered to be SMALL and are consistent with the transportation risks from current generation reactors presented in Table S-4 of 10 CFR 51.52. The same conclusion is true of the transportation accident risks associated with the spent fuel from the proposed new reactors at the HAR site and the alternative sites.

### 7.4.5 REFERENCES

7.4-001 Neuhauser, K. S. and F. L. Kanipe, *RADTRAN 5 User Guide*, Sandia National Laboratories, SAND2003-2354, July 2003.

7.4-002 Idaho National Engineering and Environmental Laboratory, “Early Site Permit ER Sections and Supporting Documentation,” Engineering Design File Number 3747, July 2003.

7.4-003 Saricks, C.L. and M.M. Tompkins, *State-Level Accident Rates of Surface Freight Transportation: A Reexamination*, Argonne National Laboratory, ANL/ESD/TM-150, April 1999.

7.4-004 Blower, Daniel and Anne Matteson, Center for National Truck Statistics, “Evaluation of the Motor Carrier Management Information System Crash File, Phase 1,” UMTRI 2003-6, prepared for Federal Motor Carrier Safety Administration, March 2003.

## Table 7.4-1
## Radionuclide Inventory Used in Transportation Accident Risk Calculations for the AP1000

| Radionuclide | AP1000 Inventory (Ci/MTU) |
|---|---|
| Am-241 | 7.27E+02 |
| Am-242m | 1.31E+01 |
| Am-243 | 3.34E+01 |
| Ce-144 | 8.87E+03 |
| Cm-242 | 2.83E+01 |
| Cm-243 | 3.07E+01 |
| Cm-244 | 7.75E+03 |
| Cm-245 | 1.21E+00 |
| Cs-134 | 4.80E+04 |
| Cs-137 | 9.31E+04 |
| Co-60 [a] | 1.20E+02 |
| Eu-154 | 9.13E+03 |
| Eu-155 | 4.62E+03 |
| Pm-147 | 1.76E+04 |
| Pu-238 | 6.07E+03 |
| Pu-239 | 2.55E+02 |
| Pu-240 | 5.43E+02 |
| Pu-241 | 6.96E+04 |
| Pu-242 | 1.82E+00 |
| Ru-106 | 1.55E+04 |
| Sb-125 | 3.83E+03 |
| Sr-90 | 6.19E+04 |
| Y-90 | 6.19E+04 |

Notes:

The “m” next to an isotope indicates a metastable state.

a) Co-60 is the key radionuclide constituent of fuel assembly crud.

Ci/MTU = Curies per metric ton uranium

**Table 7.4-2**
**Spent Fuel Transportation Accident Risks for the AP1000**

| Site | Unit Population Dose (person-rem) [(a)] | Shipments per Year [(b)] | Population Dose (person-rem per RRY) [(c)] |
|---|---|---|---|
| HAR | 1.43E-06 | 39 | 5.58E-05 |
| BNP | 1.55E-06 | 39 | 6.05E-05 |
| RNP | 1.29E-06 | 39 | 5.03E-05 |
| Marion County | 1.30E-06 | 39 | 5.07E-05 |
| Table S-4 | -- | -- | SMALL |

Notes:

a) The inventory in RADTRAN calculations was adjusted for the 0.5 MTU per shipment.

b) Calculations are based on 39 normalized shipments per year.

c) Values are the product of unit population dose multiplied by normalized shipments per year.

person-rem = person-roentgen equivalent man
RRY = reference reactor year

## Table 7.4-3 (Sheet 1 of 2)
## Adjusted Accident, Injury, and Fatality Rates for the United States

| | Accidents/Trucks (km) | | Fatalities/Trucks (km) | | Injuries/Trucks (km) | |
|---|---|---|---|---|---|---|
| **State/Parameter** | **Interstate** | **Total** | **Interstate** | **Total** | **Interstate** | **Total** |
| Alabama | 4.63E-07 | 6.19E-07 | 1.35E-08 | 3.45E-08 | 1.78E-07 | 2.56E-07 |
| Arizona | 2.17E-07 | 1.76E-07 | 1.48E-08 | 1.48E-08 | 1.4E-07 | 1.1E-07 |
| Arkansas | 2.2E-07 | 2.43E-07 | 9.76E-09 | 3.5E-08 | 1.18E-07 | 1.49E-07 |
| California | 2.63E-07 | 1.36E-07 | 1.1E-08 | 5.67E-09 | 1.49E-07 | 7.68E-08 |
| Colorado | 7.32E-07 | 7.12E-07 | 1.8E-08 | 2.76E-08 | 3.78E-07 | 3.64E-07 |
| Connecticut | 1.48E-06 | 1.45E-06 | 2.28E-08 | 3.01E-08 | 7.36E-07 | 7.39E-07 |
| Delaware | 8.5E-07 | 1.19E-06 | 8.82E-09 | 3.7E-08 | 4.1E-07 | 6.13E-07 |
| Florida | 1.13E-07 | 1.46E-07 | 1.21E-08 | 1.69E-08 | 6.6E-08 | 8.52E-08 |
| Georgia | N/A | 1.1E-06 | N/A | 3.07E-08 | N/A | 5.51E-07 |
| Idaho | 4.84E-07 | 6.48E-07 | 5.98E-09 | 3.92E-08 | 3.68E-07 | 4.73E-07 |
| Illinois | 3.64E-07 | 4.86E-07 | 1.31E-08 | 1.73E-08 | 1.8E-07 | 1.97E-07 |
| Indiana | 3.69E-07 | 2.77E-07 | 1.06E-08 | 1.35E-08 | 1.68E-07 | 1.38E-07 |
| Iowa | 1.84E-07 | 2.43E-07 | 1.48E-08 | 2.11E-08 | 1.03E-07 | 1.36E-07 |
| Kansas | 4.66E-07 | 6.29E-07 | 8.19E-09 | 3.61E-08 | 3.05E-07 | 4.14E-07 |
| Kentucky | 5.09E-07 | 8.5E-07 | 2.02E-08 | 3.61E-08 | 2.65E-07 | 4.33E-07 |
| Louisiana | N/A | 3.63E-07 | N/A | 1.45E-08 | N/A | 2.21E-07 |
| Maine | 7.2E-07 | 6.76E-07 | 1.43E-08 | 1.23E-08 | 3.74E-07 | 4E-07 |
| Maryland | 8.86E-07 | 1.22E-06 | 1.02E-08 | 3.13E-08 | 5.51E-07 | 7.27E-07 |
| Massachusetts | 1.41E-07 | 2.54E-07 | 1.26E-09 | 5.98E-09 | 6.12E-08 | 1.25E-07 |
| Michigan | 4.64E-07 | 3.53E-07 | 1.69E-08 | 1.69E-08 | 3.13E-07 | 2.64E-07 |
| Minnesota | 2.81E-07 | 2.89E-07 | 4.72E-09 | 1.89E-08 | 1.01E-07 | 1.45E-07 |
| Mississippi | 7.88E-08 | 1.03E-07 | 3.94E-09 | 5.35E-09 | 4.68E-08 | 6.84E-08 |
| Missouri | 7.62E-07 | 8.8E-07 | 1.95E-08 | 3.1E-08 | 3.77E-07 | 4.38E-07 |
| Montana | 1.02E-06 | 9.54E-07 | 2.14E-08 | 3.2E-08 | 3.07E-07 | 3.1E-07 |
| Nebraska | 5.24E-07 | 7.12E-07 | 2.16E-08 | 2.95E-08 | 2.36E-07 | 3.11E-07 |
| Nevada | 3.69E-07 | 4.02E-07 | 1.04E-08 | 1.4E-08 | 1.78E-07 | 1.94E-07 |

## Table 7.4-3 (Sheet 2 of 2)
## Adjusted Accident, Injury, and Fatality Rates for the United States

| | Accidents/Trucks (km) | | Fatalities/Trucks (km) | | Injuries/Trucks (km) | |
|---|---|---|---|---|---|---|
| State/Parameter | Interstate | Total | Interstate | Total | Interstate | Total |
| New Hampshire | 4.32E-07 | 6.25E-07 | N/A | 1.86E-08 | 1.96E-07 | 2.81E-07 |
| New Jersey | 9.27E-07 | 8.09E-07 | 1.91E-08 | 1.12E-08 | 4.69E-07 | 4.55E-07 |
| New Mexico | 1.85E-07 | 1.77E-07 | 1.86E-08 | 1.73E-08 | 1.38E-07 | 1.3E-07 |
| New York | N/A | 5.66E-07 | N/A | 1.95E-08 | N/A | 2.22E-07 |
| North Carolina | 5.68E-07 | 5.48E-07 | 2.35E-08 | 2.55E-08 | 3.8E-07 | 3.79E-07 |
| North Dakota | 4.96E-07 | 5.61E-07 | 1.61E-08 | 1.75E-08 | 2.27E-07 | 3.04E-07 |
| Ohio | 2.69E-07 | 1.9E-07 | 6.14E-09 | 6.14E-09 | 1.68E-07 | 1.28E-07 |
| Oklahoma | 4.4E-07 | 4.53E-07 | 2.09E-08 | 2.32E-08 | 3.47E-07 | 3.42E-07 |
| Oregon | N/A | 3.54E-07 | N/A | 3.21E-08 | N/A | 1.63E-07 |
| Pennsylvania | 8.44E-07 | 1.11E-06 | 2.13E-08 | 3.83E-08 | 4.6E-07 | 6.4E-07 |
| Rhode Island | N/A | N/A | N/A | N/A | N/A | N/A |
| South Carolina | N/A | 7.7E-07 | N/A | 4.09E-08 | N/A | 3.96E-07 |
| South Dakota | 3.82E-07 | 3.76E-07 | 9.61E-09 | 2E-08 | 2.06E-07 | 1.91E-07 |
| Tennessee | 2.02E-07 | 2.61E-07 | 1.57E-08 | 2.05E-08 | 1.1E-07 | 1.52E-07 |
| Texas | 9.85E-07 | 1.08E-06 | 2.05E-08 | 4.25E-08 | 6.57E-07 | 6.45E-07 |
| Utah | 4.76E-07 | 5.58E-07 | 1.87E-08 | 2.19E-08 | 3.04E-07 | 3.41E-07 |
| Vermont | 3.09E-07 | 4.89E-07 | N/A | 1.53E-08 | 1.82E-07 | 2.64E-07 |
| Virginia | 6.45E-07 | 4.35E-07 | 2.54E-08 | 1.83E-08 | 3.72E-07 | 2.59E-07 |
| Washington | 4.35E-07 | 3.36E-07 | 2.83E-09 | 8.35E-09 | 2.16E-07 | 1.68E-07 |
| West Virginia | 2.82E-07 | 3.53E-07 | 2.65E-08 | 4.38E-08 | 1.34E-07 | 1.68E-07 |
| Wisconsin | 7.37E-07 | 9.04E-07 | 1.43E-08 | 3.5E-08 | 4E-07 | 4.92E-07 |
| Wyoming | 1.11E-06 | 1.11E-06 | 1.7E-08 | 1.95E-08 | 3.88E-07 | 3.88E-07 |

Notes:

km = kilometer
N/A = not available

Sources: References 7.4-003 and 7.4-004

**Table 7.4-4**
**Nonradiological Impacts, Per Shipment, Resulting from Shipment of Unirradiated and Spent Nuclear Fuel**

| | Unirradiated Fuel | | | | Spent Nuclear Fuel | | | |
|---|---|---|---|---|---|---|---|---|
| | **Round-trip distance, km** | **Accidents** | **Injuries** | **Fatalities** | **Round-trip distance, km** | **Accidents** | **Injuries** | **Fatalities** |
| HAR | **306** | 4.11E-04 | 2.38E-04 | 1.98E-05 | 4294.0 | 2.91E-03 | 1.91E-03 | 1.42E-04 |
| BNP | **525.5** | 6.22E-04 | 3.97E-04 | 2.53E-05 | 4526.7 | 3.17E-03 | 2.08E-03 | 1.53E-04 |
| RNP | **408.2** | 5.10E-04 | 3.10E-04 | 2.20E-05 | 4234.3 | 2.96E-03 | 1.87E-03 | 1.49E-04 |
| Marion County | **434.2** | 5.29E-04 | 3.29E-04 | 2.23E-05 | 4272.2 | 3.02E-03 | 1.90E-03 | 1.53E-04 |

Notes:

km = kilometer

**Table 7.4-5**
**Nonradiological Impacts Resulting from the Total Amount of Shipments of Unirradiated and Spent Nuclear Fuel for a RRY, Normalized to Reference LWR**

| Site | Accidents per RRY[(a)] | Injuries per RRY[(a)] | Fatalities per RRY[(a)] |
|---|---|---|---|
| HAR | 1.16E-01 | 7.57E-02 | 5.64E-03 |
| BNP | 1.27E-01 | 8.31E-02 | 6.09E-03 |
| RNP | 1.18E-01 | 7.44E-02 | 5.92E-03 |
| Marion Co. | 1.20E-01 | 7.57E-02 | 6.08E-03 |
| Table S-4 | -- | 1.00E-01 | 1.00E-02 |

Notes:

a) The values in the table have been calculated from the values presented in Table 7.4-4 based on 4.9 shipments per year of unirradiated fuel and 39 shipments per year of spent fuel ([(unirradiated fuel accidents – 4.11E-04) x (4.9 shipments)] + [(spent fuel accidents – 2.91E-03) x (39 shipments)] = Accidents per RRY – 1.16E-01).

km = kilometer
RRY = reference reactor year