7.0 Environmental Impacts of Postulated Accidents Involving Radioactive Materials

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

7.1 Design Basis Accidents

This section evaluates the radiological consequences of design basis accidents at STP 3 & 4.

7.1.1 Selection of Accidents

Consistent with the Advanced Boiling Water Reactor (ABWR) Design Control Document (DCD), Revision 4 (Reference 7.1-1), the following design basis accidents are evaluated as those having potential for radioactivity releases to the environment:

- Failure of small lines carrying primary coolant outside containment
- Main steam line break
- Loss of coolant accident (LOCA)
- Cleanup water line break outside containment
- Fuel handling accident

The radiological consequences of these accidents are assessed to demonstrate that STP 3 & 4 could be sited at the STP site without undue risk to the health and safety of the public.

The above accidents are identified in NUREG-1555, Section 7.1, Appendix A, for consideration in an environmental report. The following additional accidents identified in NUREG-1555, Section 7.1, Appendix A, are not evaluated for the reasons provided below:

- Radiological consequences of main steam line failures outside containment of a PWR - Not applicable to the ABWR
- Feedwater system pipe breaks inside and outside containment (PWR) - Not applicable to the ABWR
- Reactor coolant pump rotor seizure - As indicated in the ABWR DCD, because this accident does not result in any fuel failures, it has no radiological consequences (Reference 7.1-1, Subsection 15.3.3.5)
- Reactor coolant pump shaft break - As indicated in the ABWR DCD, because this accident does not result in any fuel failures, it has no radiological consequences (Reference 7.1-1, Subsection 15.3.4.5)

- Radiological consequences of control rod drop accident (BWR) - As indicated in the ABWR DCD, there is no basis for this accident to occur (Reference 7.1-1, Subsection 15.4.10.3)
- Radiological consequences of steam generator tube failure (PWR) - Not applicable to the ABWR

7.1.2 Evaluation Methodology

The ABWR DCD presents the radiological consequences of the accidents identified in Subsection 7.1.1. The basic scenario for each accident is that some quantity of activity is released at the accident location inside a building and this activity is eventually released to the environment. The transport of activity within the plant is independent of the site and is specific to the ABWR design. Details about the methodologies and assumptions pertaining to each of the accidents, such as activity release pathways and credited mitigation features, are provided in the reference ABWR DCD (Reference 7.1-1, Chapter 15).

The dose to an individual located at the exclusion area boundary (EAB) or in the low population zone (LPZ) is calculated based on the amount of activity released to the environment, the atmospheric dispersion of the activity during the transport from the release point to the off-site location, the breathing rate of the individual at the off-site location, and activity-to-dose conversion factors. The only site-specific parameter is atmospheric dispersion. Site-specific doses are obtained by adjusting the reference ABWR DCD doses to reflect site-specific atmospheric dispersion factors (χ/Q values).

NUREG-1555 provides two options for calculating χ/Q values: either 50th percentile χ/Q values based on onsite meteorological data or 10% of the levels given in Regulatory Guide 1.3 (Reference 7.1-2) to represent more realistic dispersion conditions than assumed in the safety evaluation. The option to use 50th percentile site-specific χ/Q values was selected. Short-term accident χ/Q values are calculated using the methodology of Regulatory Guide 1.145, Revision 1 (Reference 7.1-3) with site-specific meteorological data. As indicated in Section 2.7, the Regulatory Guide 1.145 methodology is implemented in the NRC-sponsored PAVAN computer program. This program computes χ/Q values at the EAB and the LPZ for each combination of wind speed and atmospheric stability for each of the 16 downwind direction sectors. Releases are assumed to be at ground level, and the shortest distances between the power block and the off-site locations are selected to conservatively maximize the χ/Q values.

Consistent with the reference ABWR DCD, the accident doses are presented for the whole body and the thyroid. Furthermore, the whole body and thyroid doses are converted into total effective dose equivalent (TEDE) to demonstrate compliance with 10 CFR 50.34. The conversion to TEDE is performed by multiplying the thyroid dose by a weighting factor of 0.03 and adding the result to the whole body dose, in accordance with ICRP 30 (Reference 7.1-4).

7.1.3 Source Terms

The design basis accident source terms in the reference ABWR DCD are presented as time-dependent isotopic activity releases to the environment in the unit of megabecquerel (MBq) in Tables 7.1-1 to 7.1-6.

7.1.4 Radiological Consequences

As indicated in NUREG-1555, environmental report design basis accident doses are evaluated based on more realistic meteorological conditions than those used in the safety analysis report. For each of the accidents identified in Subsection 7.1.1, the site-specific dose for a given time interval is calculated by multiplying the ABWR DCD dose by the ratio of the site χ/Q value provided in Section 2.7 to the DCD χ/Q value. The time-dependent DCD and site χ/Q values and their ratios are shown in Table 7.1-7. Since all site χ/Q values are bounded by DCD χ/Q values, site-specific doses for all accidents would also be bounded by DCD doses. However, site-specific doses are presented for completeness. All accident doses are presented in Tables 7.1-8 to 7.1-13.

The 10 CFR 50.34 dose limit is 25 rem TEDE at the EAB and the LPZ, as specified in 50.34(a)(1)(ii). The ABWR is certified to the 10 CFR 100.11 dose limits of 25 rem to the whole body and 300 rem to the thyroid. The limit of 10 CFR 50.34(a)(1)(ii) applies to extremely low probability accidents that could result in the release of significant quantities of radioactive fission products. Similarly, the limits of 10 CFR 100.11 apply to a major accident with the release of appreciable quantities of fission products. For accidents with smaller releases, more restrictive dose limits are specified in Subsections 15.6.2, 15.6.4, and 15.7.4 of NUREG-0800. Where applied, the more restrictive dose limits are either 10% or 25% of the 10 CFR 100.11 limits. Although conformance to these more restrictive dose limits is not required for an environmental report, they are shown in Tables 7.1-8 to 7.1-13 for comparison purposes.

The doses shown in Tables 7.1-8 to 7.1-13 are summarized in Tables 7.1-14 and 7.1-15. Tables 7.1-14 and 7.1-15 also show the equivalent TEDE doses. The summary tables demonstrate that, in addition to meeting the limits of 10 CFR 100.11 and NUREG-0800 as indicated above, all accident doses also meet the 25 rem TEDE acceptance criteria of 10 CFR 50.34(a)(1)(ii). Because the dose criterion of 10 CFR 50.34 is intended to provide assurance of low risk to the public under postulated accidents, any health effects resulting from the design basis accidents are considered to be negligible.

7.1.5 References

7.1-1 ABWR Design Control Document, Tier 2, GE Nuclear Energy, Revision 4.

7.1-2 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, Revision 2, June 1974.

7.1-3 “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Regulatory Guide 1.145, Revision 1, November 1982.

7.1-4 “Limits for Intakes of Radionuclides by Workers,” Part 1, ICRP Publication 30, International Commission on Radiological Protection, Pergamon Press, 1979.

Table 7.1-1 Activity Releases for Failure of Small Lines Carrying Primary Coolant Outside Containment

Isotope	Activity Release (MBq)	
	0-2 Hours	**0-8 Hours**
I-131	6.81E+04	1.41E+05
I-132	5.96E+05	1.19E+06
I-133	4.59E+05	9.44E+05
I-134	9.92E+05	1.90E+06
I-135	6.59E+05	1.34E+06
Total	**2.77E+06**	**5.51E+06**

Source: ABWR DCD (Reference 7.1-1, Table 15.6-2)

Table 7.1-2 Activity Releases for Main Steam Line Break

Isotope	Activity Release (MBq)	
	Preexisting Iodine Spike	Equilibrium Iodine Activity
I-131	1.46E+06	7.29E+04
I-132	1.42E+07	7.10E+05
I-133	9.99E+06	5.00E+05
I-134	2.79E+07	1.40E+06
I-135	1.46E+07	7.29E+05
Kr-83m	2.44E+03	4.07E+02
Kr-85m	4.29E+03	7.18E+02
Kr-85	1.36E+01	2.26E+00
Kr-87	1.47E+04	2.44E+03
Kr-88	1.48E+04	2.46E+03
Kr-89	5.92E+04	9.88E+03
Kr-90	1.55E+04	2.55E+03
Xe-131m	1.06E+01	1.76E+00
Xe-133m	2.04E+02	3.39E+01
Xe-133	5.70E+03	9.47E+02
Xe-135m	1.74E+04	2.89E+03
Xe-135	1.62E+04	2.70E+03
Xe-137	7.40E+04	1.23E+04
Xe-138	5.66E+04	9.44E+03
Xe-139	2.59E+04	4.33E+03
Total Iodine	6.81E+07	3.41E+06
Total Noble Gases	3.07E+05	5.11E+04

Source: ABWR DCD (Reference 7.1-1, Table 15.6-6)

Table 7.1-3 Activity Releases from Reactor Building for Loss-of-Coolant Accident

Isotope	Activity Release (MBq)				
	0-2 Hours	0-8 Hours	0-24 Hours	0-96 Hours	0-720 Hours
I-131	9.6E+06	1.3E+07	3.6E+07	1.9E+08	6.7E+08
I-132	1.3E+07	1.4E+07	1.5E+07	1.5E+07	1.5E+07
I-133	2.0E+07	2.6E+07	5.6E+07	1.2E+08	1.3E+08
I-134	1.9E+07	1.9E+07	1.9E+07	1.9E+07	1.9E+07
I-135	1.9E+07	2.3E+07	3.1E+07	3.5E+07	3.5E+07
Kr-83m	1.2E+07	2.8E+07	3.3E+07	3.3E+07	3.3E+07
Kr-85	1.5E+06	1.2E+07	8.1E+07	6.7E+08	5.6E+09
Kr-85m	3.1E+07	1.3E+08	2.7E+08	2.9E+08	2.9E+08
Kr-87	4.4E+07	7.8E+07	8.1E+07	8.1E+07	8.1E+07
Kr-88	7.8E+07	2.5E+08	3.6E+08	3.7E+08	3.7E+08
Kr-89	6.7E+06	6.7E+06	6.7E+06	6.7E+06	6.7E+06
Xe-131m	7.8E+05	5.9E+06	4.1E+07	3.0E+08	1.4E+09
Xe-133	2.8E+08	2.1E+09	1.4E+10	8.9E+10	2.5E+11
Xe-133m	1.1E+07	8.5E+07	5.2E+08	2.6E+09	4.1E+09
Xe-135	3.4E+07	1.9E+08	6.7E+08	1.0E+09	1.0E+09
Xe-135m	1.8E+07	1.8E+07	1.8E+07	1.8E+07	1.8E+07
Xe-137	1.9E+07	1.9E+07	1.9E+07	1.9E+07	1.9E+07
Xe-138	7.4E+07	7.4E+07	7.4E+07	7.4E+07	7.4E+07
Total Iodine	8.1E+07	9.5E+07	1.6E+08	3.8E+08	8.6E+08
Total Noble Gases	6.1E+08	3.0E+09	1.6E+10	9.4E+10	2.6E+11

Source: ABWR DCD (Reference 7.1-1, Tables 15.6-10 and 15.6-12)

Table 7.1-4 Activity Releases from Condenser for Loss-of-Coolant Accident

Isotope	Activity Release (MBq)				
	0-2 Hours	0-8 Hours	0-24 Hours	0-96 Hours	0-720 Hours
I-131	1.2E+04	8.5E+05	1.2E+07	1.8E+08	2.0E+09
I-132	1.1E+04	2.4E+05	4.4E+05	4.4E+05	4.4E+05
I-133	2.4E+04	1.5E+06	1.5E+07	7.4E+07	8.9E+07
I-134	8.5E+03	4.8E+04	4.8E+04	4.8E+04	4.8E+04
I-135	2.1E+04	9.3E+05	5.2E+06	7.4E+06	7.4E+06
Kr-83m	7.8E+04	1.3E+06	1.9E+06	1.9E+06	1.9E+06
Kr-85	1.3E+04	9.3E+05	1.3E+07	2.3E+08	5.9E+09
Kr-85m	2.3E+05	8.5E+06	3.0E+07	3.6E+07	3.6E+07
Kr-87	2.4E+05	2.4E+06	2.8E+06	2.8E+06	2.8E+06
Kr-88	5.6E+05	1.4E+07	3.1E+07	3.2E+07	3.2E+07
Kr-89	4.1E+00	4.1E+00	4.1E+00	4.1E+00	4.1E+00
Xe-131m	6.7E+03	4.8E+05	6.7E+06	1.0E+08	1.3E+09
Xe-133	2.4E+06	1.6E+08	2.2E+09	3.0E+10	1.8E+11
Xe-133m	1.0E+05	6.7E+06	8.1E+07	8.1E+08	2.0E+09
Xe-135	2.7E+05	1.4E+07	9.6E+07	2.0E+08	2.0E+08
Xe-135m	1.0E+04	1.3E+04	1.3E+04	1.3E+04	1.3E+04
Xe-137	3.5E+01	3.5E+01	3.5E+01	3.5E+01	3.5E+01
Xe-138	3.2E+04	3.7E+04	3.7E+04	3.7E+04	3.7E+04
Total Iodine	7.7E+04	3.5E+06	3.3E+07	2.6E+08	2.1E+09
Total Noble Gases	3.9E+06	2.1E+08	2.5E+09	3.2E+10	1.9E+11

Source: ABWR DCD (Reference 7.1-1, Tables 15.6-10, and 15.6-12)

Table 7.1-5 Activity Releases for Cleanup Water Line Break Outside Containment

Isotope	Activity Release (MBq) 0-2 Hours
I-131	8.1E+04
I-132	1.9E+05
I-133	2.3E+05
I-134	3.2E+05
I-135	2.5E+05
Total	1.1E+06

Source: ABWR DCD (Reference 7.1-1, Table 15.6-17)

Table 7.1-6 Activity Releases for Fuel-Handling Accident

Isotope	Activity Release (MBq) 0-2 Hours
I-131	4.55E+06
I-132	5.62E+06
I-133	4.70E+06
I-134	2.28E-01
I-135	7.62E+05
Kr-83m	2.38E+05
Kr-85m	3.16E+06
Kr-85	1.77E+07
Kr-87	4.55E+02
Kr-88	8.99E+05
Kr-89	3.01E-06
Xe-131m	3.09E+06
Xe-133m	4.07E+07
Xe-133	1.04E+09
Xe-135m	8.18E+06
Xe-135	2.36E+08
Xe-137	7.66E-06
Xe-138	1.59E-05
Total Iodine	1.56E+07
Total Noble Gases	1.35E+09

Source: ABWR DCD (Reference 7.1-1, Table 15.7-10)

Table 7.1-7 Atmospheric Dispersion Factors

Location	Time (hr)	χ/Q (sec/m^3)		Ratio (Site/DCD)
		Site	DCD	
EAB – Cleanup Water Line Break	0 – 2	8.18E-05	2.29E-02	3.57E-03
EAB – Other	0 – 2	8.18E-05	1.37E-03	5.97E-02
LPZ – LOCA	0 – 8	6.30E-06	1.56E-04	4.04E-02
	8 – 24	5.07E-06	9.61E-05	5.28E-02
	24 – 96	3.17E-06	3.36E-05	9.43E-02
	96 – 720	1.62E-06	7.42E-06	2.18E-01
LPZ – Cleanup Water Line Break	0 – 8	6.30E-06	2.29E-02	2.75E-04
LPZ –Other	0 – 8	6.30E-06	1.37E-03	4.60E-03

Notes:

The site χ/Q values are from Section 2.7.

The DCD χ/Q values are from the ABWR DCD (Reference 7.1-1, Tables 15.6-3, 15.6-7, 15.6-13, 15.6-18, and 15.7-11), based on "Chp 2" distance, with the exception of Table 15.6-18, which is based on "max" distance.

The DCD does not show LPZ doses for accidents other than LOCA. Site LPZ doses for these non-LOCA accidents, all of which have their activity releases terminated within 8 hr, are estimated by multiplying the DCD EAB dose by the ratio of site LPZ χ/Q to DCD EAB χ/Q shown in the last row above.

Table 7.1-8 Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0-8	9.4E-04	4.8E-02	5.97E-02	5.6E-03	2.9E-01
LPZ	0-8	9.4E-04	4.8E-02	4.60E-03	4.3E-04	2.2E-02
	8-24					
	24-96					
	96-720					
	Total				**4.3E-04**	**2.2E-02**
Regulatory Limit (NUREG-0800, Subsection 15.6.2)					2.5	30

Note:

DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.6-3).

The DCD does not provide 0-2 hr doses. The site EAB doses are obtained by multiplying the DCD 0-8 hr doses by the ratio of the site EAB χ/Q to DCD EAB χ/Q.

Table 7.1-9 Doses for Main Steam Line Break, Preexisting Iodine Spike

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0 - 2	1.3E-02	5.1E-01	5.97E-02	7.8E-02	3.0E+00
LPZ	0 - 8			4.60E-03	6.0E-03	2.3E-01
	8 - 24					
	24 - 96					
	96 - 720					
	Total				**6.0E-03**	**2.3E-01**
Regulatory Limit (10 CFR 100.11)					25	300

Note:

DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.6-7).

The ABWR DCD does not provide LPZ doses. The site LPZ doses are obtained by multiplying the DCD EAB doses by the ratio of LPZ χ/Q to DCD EAB χ/Q.

Table 7.1-10 Doses for Main Steam Line Break, Equilibrium Iodine Activity

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0-2	6.2E-04	2.6E-02	5.97E-02	3.7E-03	1.6E-01
LPZ	0-8			4.60E-03	2.9E-04	1.2E-02
	8-24					
	24-96					
	96-720					
	Total				**2.9E-04**	**1.2E-02**
Regulatory Limit (NUREG-0800, Subsection 15.6.4)					2.5	30

Note:

DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.6-7).

The ABWR DCD does not provide LPZ doses. The site LPZ doses are obtained by multiplying the DCD EAB doses by the ratio of LPZ χ/Q to DCD EAB χ/Q.

Table 7.1-11 Doses for Loss-of-Coolant Accident

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0-2	4.1E-02	1.9E+00	5.97E-02	2.4E-01	1.1E+01
LPZ	0-8	1.0E-02	3.1E-01	4.04E-02	4.0E-02	1.3E+00
	8-24	8.0E-03	2.0E-01	5.28E-02	4.2E-02	1.1E+00
	24-96	1.1E-02	7.9E-01	9.43E-02	1.0E-01	7.5E+00
	96-720	9.0E-03	1.1E+00	2.18E-01	2.0E-01	2.4E+01
	Total	**3.8E-02**	**2.4E+00**		**3.8E-01**	**3.4E+01**
Regulatory Limit (10 CFR 100.11)					25	300

Note: DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.6-13)

Table 7.1-12 Doses for Cleanup Water Line Break Outside Containment

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0-2	2.8E-03	3.0E-01	3.57E-03	1.0E-03	1.1E-01
LPZ	0-8			2.75E-04	7.7E-05	8.3E-03
	8-24					
	24-96					
	96-720					
	Total				**7.7E-05**	**8.3E-03**
Regulatory Limit (10 CFR 100.11)					25	300

Notes:

DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.6-18).

The DCD does not provide LPZ doses. The site LPZ doses are obtained by multiplying the DCD EAB doses by the ratio of LPZ χ/Q to DCD EAB χ/Q.

Table 7.1-13 Doses for Fuel Handling Accident

Location	Time (hr)	DCD Dose (Sv)		χ/Q Ratio (Site/DCD)	Site Dose (rem)	
		Whole Body	Thyroid		Whole Body	Thyroid
EAB	0-2	1.2E-02	7.5E-01	5.97E-02	7.2E-02	4.5E+00
LPZ	0-8			4.60E-03	5.5E-03	3.4E-01
	8 -24					
	24-96					
	96-720					
	Total				**5.5E-03**	**3.4E-01**
Regulatory Limit (NUREG-0800, Subsection 15.7.4)					6	75

Note:

DCD doses are from the ABWR DCD (Reference 7.1-1, Table 15.7-11).

The DCD does not provide LPZ doses. The site LPZ doses are obtained by multiplying the DCD EAB doses by the ratio of LPZ χ/Q to DCD EAB χ/Q.

Table 7.1-14 Summary of Design Basis Accident EAB Doses

DCD Section	Accident	Site Dose (rem)			Dose Limit (rem)	
		Whole Body	Thyroid	TEDE	Whole Body	Thyroid
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	5.6E-03	2.9E-01	1.4E-02	2.5	30
15.6.4	Main Steam Line Break	-	-	-	-	-
	Preexisting Iodine Spike	7.8E-02	3.0E+00	1.7E-01	25	300
	Equilibrium Iodine Activity	3.7E-03	1.6E-01	8.4E-03	2.5	30
15.6.5	Loss-of-Coolant Accident	2.4E-01	1.1E+01	5.9E-01	25	300
15.6.6	Cleanup Water Line Break Outside Containment	1.0E-03	1.1E-01	4.2E-03	25	300
15.7.4	Fuel-Handling Accident	7.2E-02	4.5E+00	2.1E-01	6	75

Notes:

The site doses and dose limits are taken from Tables 7.1-8 to 7.1-13.

The dose limits are from either NUREG-0800 or 10 CFR 100.11, as indicated in Tables 7.1-8 to 7.1-13.

Preexisting Iodine Spike and Equilibrium Iodine Activity are subsets of Main Steam Line Break.

All accidents meet the 10 CFR 50.34(a)(1)(ii) dose limit of 25 rem TEDE.

Table 7.1-15 Summary of Design Basis Accident LPZ Doses

DCD Section	Accident	Site Dose (rem)			Dose Limit (rem)	
		Whole Body	Thyroid	TEDE	Whole Body	Thyroid
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	4.3E-04	2.2E-02	1.1E-03	2.5	30
15.6.4	Main Steam Line Break	-	-	-	-	-
	Preexisting Iodine Spike	6.0E-03	2.3E-01	1.3E-02	25	300
	Equilibrium Iodine Activity	2.9E-04	1.2E-02	6.4E-04	2.5	30
15.6.5	Loss-of-Coolant Accident	3.8E-01	3.4E+01	1.4E+00	25	300
15.6.6	Cleanup Water Line Break Outside Containment	7.7E-05	8.3E-03	3.2E-04	25	300
15.7.4	Fuel Handling Accident	5.5E-03	3.4E-01	1.6E-02	6	75

Notes:

The site doses and dose limits are taken from Tables 7.1-8 to 7.1-13.

The dose limits are from either NUREG-0800 or 10 CFR 100.11, as indicated in Tables 7.1-8 to 7.1-13.

Preexisting Iodine Spike and Equilibrium Iodine Activity are subsets of Main Steam Line Break.

All accidents meet the 10 CFR 50.34(a)(1)(ii) dose limit of 25 rem TEDE.

7.2 Severe Accidents

As stated in NUREG-1555, “Severe accidents are those involving multiple failures of equipment or function and, therefore, the likelihood of occurrence is lower for severe accidents than for Design Bases Accidents, but the consequences of such accidents may be higher.” Because the probability of a severe accident is very low for the ABWR, severe accidents are not part of the design basis for the plant. However, NRC requires, in its Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50 FR 32138), the completion of a probabilistic risk assessment (PRA) for severe accidents for new reactor designs. This requirement is codified under 10 CFR 52.47.

GE completed a PRA as part of the Standard Safety Analysis Report (SSAR), Amendment 35 for the ABWR design (Reference 7.2-1). The ABWR design was prepared as part of GE’s application for design certification. NRC reviewed the ABWR design and the review was documented in NUREG-1503 (Reference 7.2-2). NRC has certified the ABWR design, concluding that the ABWR is of a robust design, and that the design meets NRC’s safety goals.

The GE PRA for the ABWR established a containment event tree that defined the possible end states of the containment following a severe accident. These end states can logically be grouped to produce 10 source term categories that represent the entire suite of potential severe accidents. An accident frequency was assigned to each of the 10 source term categories.

GE then used the CRAC-2 (Calculations of Reactor Accident Consequences, Reference 7.2-3) code to model the environmental consequences of the severe accidents using the generic meteorology, population, and evacuation characteristics as described in Section 19E.3 of the Standard Safety Analysis Report (Reference 7.2-1). CRAC-2 is a revision of the CRAC program developed in support of NRC’s Reactor Safety Study (often referred to as WASH-1400) to assess the risk from potential accidents at nuclear power plants.

7.2.1 STP-Specific Analysis

STPNOC has updated the generic analysis by including site-specific characteristics of the STP site. The purpose of this STP-specific analysis is two fold: 1) to disclose the impacts of severe accidents, and 2) to support the severe accident mitigation alternatives (SAMA) analyses presented in Section 7.3.

To evaluate site-specific consequences of severe accidents, STPNOC used the MACCS2 (MELCOR Accident Consequence Code System, Version 2) computer code, which was developed by NRC for this purpose (Reference 7.2-4). The pathways modeled include external exposure to the passing plume, external exposure to material deposited on the ground, inhalation of material in the passing plume or resuspended from the ground, and ingestion of contaminated food and surface water. The MACCS2 code primarily addresses the dose from the air pathway, but also calculates the dose from surface runoff and deposition on surface water. The MACCS2 code also evaluates the extent of contamination.

To assess human health impacts from severe accidents, STPNOC determined the collective dose to the 50-mile population, the number of late cancer fatalities, and the number of early fatalities associated with a severe accident. A 50-mile circular area is the standard range used in modeling consequences to the offsite population from an airborne release. Economic costs were also determined, including the cost of emergency response actions and long-term protective actions. Emergency response costs include compensation for evacuees and relocated people who are removed from their homes as a result of radiation exposure during the course of the accident (food, housing, transportation, and lost income). Longer-term protective action costs within a 50-mile radius include:

- Costs of interdiction of farms, residences, and food
- Decontamination of farm and residential land
- Permanent condemnation of residential and farm land, milk, and crops

Five files provide input to a MACCS2 analysis. The first three are as follows:

- ATMOS provides data to track the material released to the atmosphere as it is dispersed and deposited. The calculation uses a Gaussian plume model. Important inputs in this file include the core inventory, release fractions, and geometry of the reactor and associated buildings. To the extent possible, the input data are the same as those in the GE CRAC-2 input files in the generic probabilistic risk assessment.
- EARLY provides inputs to calculations regarding exposure in the time period immediately following the release. Important site-specific information includes emergency response information such as evacuation time.
- CHRONC provides data for calculating long-term impacts and economic costs and includes region-specific data on agriculture and economic factors.

The following MACCS2 files access:

- A meteorological file, which uses actual STP meteorological monitoring data.
- A site characteristics file that is built using SECPOP2000 (Reference 7.2-5). SECPOP2000 incorporates 2000 census data for the 50-mile region around the STP site. For this analysis, the census data were modified to include transient populations and were projected to the year 2060. All releases are modeled as occurring at ground level.

The results of the MACCS2 analysis and accident frequency information from GE (Reference 7.2-6) were used to determine risk, which is the product of the frequency of an accident and the consequences of the accident. The sum of the accident frequencies is known as the core damage frequency and includes only internally initiated events. Externally initiated events, and their associated small contribution to risk, are described in FSAR Section 19.4. Core damage frequencies and nuclide release fractions for each release category are given in Table 7.2-1. The consequence can be either collective radiation dose or economic cost. Dose-risk is the product of the radiation dose and the accident frequency. Because the ABWR's severe accident analysis addressed a suite of accidents, the individual risks are added to provide a total risk. The same process was applied to estimating cost-risk. Therefore, risk can be reported as person-rem per reactor year or dollars per reactor year.

7.2.2 Consequences to Population Groups

The pathway consequences to population groups including air pathways, surface water, and groundwater pathways are discussed in the following sections. The presence of threatened and endangered species and federally designated critical habitat are discussed in Subsections 2.4.1 and 2.4.2. The impacts on threatened and endangered species due to the previously calculated radiation exposure levels are discussed in Subsection 5.4.4. The all pathways (water + air), water ingestion, and air pathways dose-risks for each of the three years of meteorology are presented by source term category in Table 7.2-2.

7.2.2.1 Air Pathways

Each of the accident categories was analyzed with MACCS2 to estimate population dose, number of early and latent fatalities, cost, and farm land requiring decontamination. The analysis assumed that 95% of the 10-mile population was evacuated following declaration of a general emergency. People that do not evacuate remain at their residences and are exposed as if they are going about their normal activities. For each accident category, the risk for each analytical endpoint was calculated by multiplying the analytical endpoint by the accident category frequency and adding across all accident categories. The results are provided in Table 7.2-3.

7.2.2.2 Surface Water Pathways

People can be exposed to radiation when airborne radioactivity is deposited onto the ground and runs off into surface water or is deposited directly onto surface water. The exposure pathway can be from drinking the water, submersion in the water (swimming), undertaking activities near the shoreline (fishing and boating), or ingestion of fish or shellfish. For the surface water pathway, MACCS2 only calculates the dose from drinking the water. The maximum MACCS2 severe accident dose-risk to the 50-mile population from drinking the water is 3.4×10^{-4} person-rem per year of ABWR operation. This value is included in the air pathways dose (Table 7.2-1) and is the sum of all accident category risks.

Surface water bodies within the 50-mile region of the STP site that are accessible to the public include the Colorado River, the Gulf of Mexico, and other smaller water bodies. In NUREG-1437, the NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet of reactors (Reference 7.2-7). For sites discharging to small rivers, the NRC evaluation estimated the uninterdicted aquatic food pathway dose risk as 0.4 person-rem per reactor year. For sites near large water bodies, values ranged from 270 person-rem per reactor year (Hope Creek on Delaware Bay) to 5500 person-rem per reactor year (Calvert Cliffs on Chesapeake Bay). Although the STP site is not specifically identified in this NUREG-1437 analysis, it would more likely fall between the small river analysis and the least impactive large water body analysis (Hope Creek), given the STP site's distance from nearby major water bodies (6 to 10 miles). Actual dose-risk values would be expected to be much less (by a factor of 2 to 10) due to interdiction of contaminated foods (Reference 7.2-7). Furthermore, because the ABWR atmospheric pathway doses are significantly lower than those of the current nuclear fleet, it is reasonable to conclude that the doses from surface water sources would be consistently lower than those reported above for the surface water pathway.

Surface water pathways involving submersion in the water and undertaking activities near the shoreline are not modeled by MACCS2. Neither does NUREG-1437 provide specific data on submersion and shoreline activities. However, it does indicate that these contributors to dose are much less than for drinking water and consuming aquatic foods, especially at estuary sites.

7.2.2.3 Groundwater Pathways

Radioactivity released during a severe accident could enter the groundwater. ER section 2.3.1.2 (and Figures 2.3.1-23 and 24) describes the groundwater flow path from the STP site. Shallow aquifer water flows towards discharge points at a livestock well and towards the Colorado River. The deep aquifer is separated from the shallow aquifer by a 100-150 foot thick clay and silt layer. Groundwater in the deep aquifer flows toward site production wells, thus precluding the potential for offsite migration. Due to the separation of shallow and deep layers and the deep aquifer flow directions, the latter layer does not affect potential offsite migration impacts.

NUREG-1437 evaluated the groundwater pathway dose, based on the analysis in NUREG-0440, the Liquid Pathway Generic Study (LPGS) (Reference 7.2-8). NUREG-0440 analyzed a core meltdown that is assumed to contaminate groundwater that subsequently contaminates surface water. NUREG-1437 compares STP 1 & 2 groundwater pathway severe accident doses to the results of NUREG-0440; the STP 1 & 2 results are shown to be very much less than the LPGS value. NUREG-1437 concludes that the risk from groundwater releases is a small fraction of that from atmospheric releases for sites such as STP.

The proposed location for STP 3 & 4 has the same groundwater characteristics as the location for STP 1 & 2. The severe accident frequency for the ABWR ($1.5x10^{-7}$ per reactor year) is lower than that of STP 1 & 2 ($1x10^{-5}$ per reactor year). Furthermore, the ABWR has containment features which would mitigate and even prevent a core meltdown from escaping the primary containment. These features include: the containment having inert gas, a high density basaltic concrete below the vessel (corium shield), and fusible plug valves that would allow the lower drywell to flood from the suppression pool. Therefore, the risks from the STP 3 & 4 groundwater pathway would not only be less than from the air pathways, but would be less than from the existing units.

7.2.3 Comparison to NRC Safety Goals

The ABWR SSAR (Reference 7.2-1) evaluates performance of the ABWR under generic conditions to three safety goals: (1) individual risk goal, (2) societal risk goal, and (3) radiation risk goal. These goals are defined in the following subsections.

7.2.3.1 Individual Risk Goal

The risk to an average individual in the vicinity of a nuclear power plant of experiencing a prompt fatality resulting from a severe reactor accident should not exceed one-tenth of one percent (0.1%) of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. population are generally exposed. As noted in the Safety Goals Policy statement (51 FR 30028), "vicinity" is defined as the area within one mile of the plant site boundary. "Prompt Fatality Risks" are defined as the sum of risks which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities (driving, household, chores, occupational activities, etc.). For this evaluation, the sum of prompt fatality risks was taken as the U.S. accidental death risk value of 36.9 deaths per 100,000 people per year (Reference 7.2-9).

7.2.3.2 Societal Risk Goal

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 one percent (0.1%) of the sum of the cancer fatality risks resulting from all other causes. As noted in the Safety Goal Policy Statement (51 FR 30028), "near" is defined as within 10 miles of the plant. The cancer fatality risk was taken as 187.5 deaths per 100,000 people per year based upon National Center for Health Statistics and U.S. Census Bureau data for 2002-2004 (Reference 7.2-9, Reference 7.2-10, Reference 7.2-11, and Reference 7.2-12).

7.2.3.3 Radiation Dose Goal

The probability of an individual exceeding a whole body dose of 0.25 sievert at a distance of one-half mile from the reactor shall be less than one in a million per reactor year.

Table 7.2-4 provides the quantitative evaluation of these three safety goals, the generic ABWR calculation of these risk values, and the STP-specific calculation of these risk values.

7.2.4 Conclusions

The total calculated dose-risk to the 50-mile population from airborne releases from an ABWR reactor at the STP site would be 0.0043 person-rem per reactor year (Table 7.2-1). This value is less than the value calculated by GE in the Technical Support Document for the ABWR (Reference 7.2-6); 0.269 person-rem for 60 years plant life, which is equal to 4.48×10^{-3} person-rem per reactor year), less than the population risk for all current reactors that have undergone license renewal, and less than that for the five reactors analyzed in NUREG-1150 (Reference 7.2-13), which range from 50 to 5,000 person-rem per reactor year for dose-risk. As reported in NUREG-1811

(Reference 7.2-14), the lowest dose-risk for reactors currently undergoing license renewal is 0.55 person-rem per reactor year.

Comparisons with the existing nuclear reactor fleet (Subsection 7.2.2.2) indicate that risk from the surface water pathway is small. Under the severe accident scenarios, surface water is primarily contaminated by atmospheric deposition. The ABWR atmospheric pathway doses are significantly lower than those of the current nuclear fleet. Therefore, it is reasonable to conclude that the doses from the surface water pathway at STP 3 & 4 would be consistently lower than those reported in Subsection 7.2.2.2 for the current fleet.

The risks of groundwater contamination from a severe ABWR accident (see Subsection 7.2.2.3) would be much less than the risk from currently licensed reactors. Additionally, interdiction could substantially reduce the groundwater pathway risks.

For comparison, as reported in Section 5.4, the total collective dose from STP 3 & 4 normal airborne releases is expected to be 0.5 person-rem annually. As previously described, dose-risk is dose times frequency. Normal operations have a frequency of one. Therefore, the dose-risk for normal operations is 0.5 person-rem per reactor year. Comparing this value to the severe accident dose-risk of 0.0043 person-rem per reactor year indicates that the dose risk from severe accidents is approximately 0.9 percent of the dose risk from normal operations.

The probability-weighted risk of cancer fatalities (early and late) from a severe accident for STP 3 or 4 is reported in Table 7.2-1 as 2.6×10^{-6} fatalities per reactor year. The probability of an individual dying from any cancer from any cause is approximately 0.23 over a lifetime. Comparing this value to the 2.6×10^{-6} fatalities per reactor year indicates that individual risk is 0.0011% of the background risk, which is less than 0.1% of the background risk.

7.2.5 References

7.2-1 "Deterministic Evaluations," Chapter 19E, ABWR Standard Safety Analysis Report, Amendment 35, General Electric.

7.2-2 "Final Safety Evaluation Report Related to Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, July 1994.

7.2-3 "Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code User's Guide," NUREG/CR-2326, February 1983.

7.2-4 "Code Manual for MACCS2: User's Guide," NUREG/CR-6613, SAND97-0594, Volume 1, Chanin, D. I. and M. L. Young, Sandia National Laboratories, Albuquerque, New Mexico, May 1998.

7.2-5 "SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program," NUREG/CR-6525, Rev. 1, August 2003.

7.2-6 "Technical Support Document for the ABWR," Revision 1, MPL No. A90-3230, General Electric, San Jose, California, November 18, 1994.

7.2-7 “Generic Environmental Impact Statement for License Renewal of Nuclear Plants,” NUREG-1437, May 1996.

7.2-8 “Liquid Pathway Generic Study: Impacts of Accidental Radioactive Releases to the Hydrosphere from Floating and Land-Based Nuclear Power Plants,” NUREG-0440, February 1978.

7.2-9 “Table A. Deaths percentage of total deaths, death rates, and age-adjusted death rates for the 15 leading causes of death in 2002,” National Vital Statistics Report, CDC, United States, Volume 54, Number 10, January 31, 2006. Available at http://www.cdc.gov/nchs/data/nvsr/nvsr54/nvsr54_10.pdf, accessed May 7, 2007.

7.2-10 GCT-T1-R. Population Estimates, 2006 Population Estimates, USCB (U.S. Census Bureau). Available at http://factfinder.census.gov/servlet/GCTTable?_bm=y&-geo_id=01000US&-_box_head_nbr=GCT-T1-R&-ds_name=PEP_2006_EST&-_lang=en&-format=US-9S&-_sse=on, accessed May 7, 2007.

7.2-11 “Table 10. Number of Deaths from 113 Selected Causes, by Age: United States 2003,” National Vital Statistics Report, Volume 54, Number 13, April 19, 2006, CDC. Available at http://www.cdc.gov/nchs/data/nvsr/nvsr54/nvsr54_13.pdf, accessed May 7, 2007.

7.2-12 “Table 1. Deaths, Age-Adjusted Death Rate, and Life Expectancy at Birth, by Race and Sex, and Infant Death Mortality Rates, by Race: United States 2003 and 2004,” National Center for Health Statistics, CDC. Available at http://www.cdc.gov/nchs/products/pubs/pubd/hestats/finaldeaths04/finaldeaths04_tables.pdf#2, accessed May 7, 2007.

7.2-13 “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, June 1989.

7.2-14 “Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site,” NUREG 1811, December 2006.

Table 7.2-1 Core Damage Frequency and Release Fractions by Release Category

Release Category	Core Damage Frequency	Nuclide Group Release Fraction		
		Noble	I	Cs
NCL	1.34E-7	0.044	2.30E-05	2.30E-05
CASE1	2.08E-8	1	1.50E-07	1.30E-05
CASE2	<1.E-10 (taken as 1.E-10)	1	5.00E-06	5.00E-06
CASE3	<1.E-10 (taken as 1.E-10)	1	2.80E-04	2.20E-03
CASE4	<1.E-10 (taken as 1.E-10)	1	1.60E-03	1.60E-03
CASE5	<1.E-10 (taken as 1.E-10)	1	6.00E-03	5.30E-04
CASE6	<1.E-10 (taken as 1.E-10)	1	3.10E-02	7.70E-02
CASE7	3.91E-10	1	8.90E-02	9.90E-02
CASE8	4.05E-10	1	1.90E-01	2.50E-01
CASE9	1.70E-10	1	3.70E-01	3.60E-01

NOTE: Releases for the nuclide groups not included above are negligible.

Table 7.2-2 All Pathways and Water Ingestion Severe Accident Dose-Consequences

Year of Meteorological Data/ Source Term Category	All Pathways Dose-Consequence (person-rem)	Water Ingestion Dose-Consequence (person-rem)	Air Pathway Dose-Consequence (person-rem)
1997			
NCL	1.89E+04	3.59E+01	1.89E+04
Case 1	1.06E+04	2.03E+01	1.06E+04
Case 2	4.84E+03	7.80E+00	4.83E+03
Case 3	3.60E+05	3.43E+03	3.57E+05
Case 4	2.36E+05	2.50E+03	2.34E+05
Case 5	1.13E+05	8.27E+02	1.12E+05
Case 6	9.28E+05	1.20E+05	8.08E+05
Case 7	1.01E+06	1.54E+05	8.56E+05
Case 8	1.46E+06	4.07E+05	1.05E+06
Case 9	1.63E+06	5.62E+05	1.07E+06
1999			
NCL	2.01E+04	3.05E+01	2.01E+04
Case 1	1.14E+04	1.73E+01	1.14E+04
Case 2	5.20E+03	6.66E+00	5.19E+03
Case 3	3.33E+05	2.91E+03	3.30E+05
Case 4	2.00E+05	2.13E+03	1.98E+05
Case 5	1.02E+05	7.06E+02	1.01E+05
Case 6	8.16E+05	1.02E+05	7.14E+05
Case 7	8.98E+05	1.31E+05	7.67E+05
Case 8	1.31E+06	3.55E+05	9.55E+05
Case 9	1.44E+06	4.77E+05	9.63E+05
2000			
NCL	1.98E+04	2.81E+01	1.98E+04
Case 1	1.11E+04	1.59E+01	1.11E+04
Case 2	5.00E+03	6.13E+00	4.99E+03
Case 3	3.77E+05	2.69E+03	3.74E+05
Case 4	2.53E+05	1.96E+03	2.51E+05
Case 5	1.20E+05	6.50E+02	1.19E+05
Case 6	9.59E+05	9.42E+04	8.65E+05
Case 7	1.04E+06	1.21E+05	9.19E+05
Case 8	1.48E+06	3.18E+05	1.16E+06
Case 9	1.59E+06	4.40E+05	1.15E+06

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

Environmental Risk from Airborne Releases					
		Number of Fatalities (per reactor year)			
Year of Meteorological Data	Population Dose-Risk (person-rem per reactor year)	Early	Late	Cost in Dollars (per reactor year)	Land Requiring Decontamination (acres per reactor year)
1997	4.2×10^{-3}	**4.8×10^{-13}**	2.5×10^{-6}	2.5	1.6×10^{-4}
1999	4.2×10^{-3}	4.4×10^{-13}	2.5×10^{-6}	2.2	1.6×10^{-4}
2000	**4.3×10^{-3}**	3.7×10^{-13}	**2.6×10^{-6}**	**2.6**	**1.8×10^{-4}**

NOTE: Bold print represents the highest value among the three years considered

Table 7.2-4 Comparison to NRC Safety Goals

Safety Risk			
Year of Meteorological Data	Prompt Fatality Risk (individual 0-1 mile) (deaths per reactor year)	Cancer Fatality Risk (0-10 mile cancers) (deaths per year per reactor year)	Probability of Exceeding 0.025 Sv (25 Rem) at 0.5 mile (per reactor-year)
1997	**2.1×10^{-14}**	7.7×10^{-13}	6.6×10^{-10}
1999	1.9×10^{-14}	**8.5×10^{-13}**	**6.9×10^{-10}**
2000	1.6×10^{-14}	7.2×10^{-13}	6.5×10^{-10}
Safety Goal [1]	$<3.7 \times 10^{-7}$	$<1.9 \times 10^{-6}$	$<1.0 \times 10^{-6}$
Generic ABWR Analysis [1]	2.2×10^{-13}	1.3×10^{-12}	$<1.0 \times 10^{-9}$

[1] Reference 7.2-1

NOTE: Bold print represents the highest value among the three years considered

7.3 Severe Accident Mitigation Alternatives

As described in ER Section 7.2, GE performed a generic severe accident analysis for the ABWR as part of the design certification process (Reference 7.3-1). ER Section 7.2 extends the GE generic severe accident analysis to examine STP 3 & 4 and determined that the generic conclusions remain valid for STP 3 & 4. GE also submitted an analysis of severe accident mitigation design alternatives (SAMDAs) and determined that no potential mitigating design alternatives are cost-effective, that is, appropriate mitigating measures are already incorporated into the ABWR design (Reference 7.3-2). This section addresses whether there are cost-beneficial severe accident procedural modifications that would need to be implemented for STP 3 & 4 to mitigate the impacts from severe accidents.

7.3.1 SAMA Analysis Process

Design or procedural modifications that could mitigate the consequences of a severe accident are known as severe accident mitigation alternatives (SAMAs). SAMAs are somewhat broader than SAMDAs, which primarily focus on design changes and do not consider procedural modifications. The GE analysis is a SAMDA analysis in which one of the stated purposes is to support a conclusion that:

> No further evaluation of severe accidents for the ABWR design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement, or other environmental analysis prepared in connection with issuance of a combined license for a nuclear power plant referencing a certified ABWR design (Reference 7.3-2).

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

(1) Define the base case —The base case is the dose-risk and cost-risk of a severe accident before implementation of any SAMAs. A plant's probabilistic risk assessment is the primary source of data in calculating the base case. The base case risks are converted to a monetary value to use for screening SAMAs. ER Section 7.2 presents the base case for a single ABWR unit at the STP site, without the monetary valuation step(which is discussed in Section 7.3.3 below).

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

(3) Determine the cost and net value of each SAMA — Each SAMA remaining after Step 2 has a detailed engineering cost estimate developed using current plant engineering processes. If the SAMA does not exceed the base case screening value, Step 4 is performed.

(4) Determine the benefit associated with each remaining SAMA — Each SAMA that passes the screening in Step 3 is evaluated using the probabilistic risk assessment model to determine the reduction in risk associated with implementation of the proposed SAMA. The reduction in risk benefit is then converted to a monetary value and compared to the detailed cost estimate developed in Step 3. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

Since the GE analysis has shown that there are no additional cost-beneficial design modifications, no further assessment of design modifications is required. In the absence of an existing plant with established procedural controls, the STP SAMA analysis thus is limited to determining the magnitude of plant-specific procedural modifications that would be cost-effective. Determining the magnitude of cost-effective procedural modifications is the same as “1. Define base case” for existing nuclear units. The monetary value of the base case benefit is calculated by assuming the current dose-risk of the unit could be reduced to zero and assigning a defined dollar value for this reduction in risk. Any procedural change with a cost that exceeds the benefit value would not be considered cost-effective.

The dose-risk and cost-risk results from the ER Section 7.2 analyses are converted to monetary value in accordance with methods established in NUREG/BR-0184 (Reference 7.3-3). NUREG/BR-0184 presents methods for determining the value of decreases in risk using four types of attributes: public health, occupational health, offsite property, and onsite property. Any SAMAs in which the conservatively low implementation cost exceeds the base case monetary value would not pass the screening in Step 2. If the STP baseline analysis produces a monetary value of the benefit that is below the cost expected for implementation of any SAMA, the remaining steps of the SAMA analysis are not necessary.

7.3.2 ABWR SAMA Analysis

In the certification process, only design alternatives were of interest. The GE SAMDA analysis is presented in the Technical Support Document for the ABWR (Reference 7.3-2). The monetary valuation of the averted cost-risk (defined as the monetary valuation of reducing the base case core damage frequency to zero) was based solely on the cumulative dose-risk over the 60-year life of the plant, assuming the NRC-generated value of $1000 per person-rem. The resulting dose-risk was determined to be 0.269 person-rem (4.48 × 10-3 person-rem per reactor year), so the averted cost-risk was calculated to be $269. GE determined that no design change would be cost-effective with this low value of averted cost-risk.

7.3.3 Monetary Valuation of the STP 3 & 4 Cost-Risk

The principal inputs to the base case calculations are as follows:

Dose-risk	4.3×10^{-3} person-rem per reactor year (reported in Table 7.2-1)
Cost-risk	2.6 dollars per reactor year (reported in Table 7.2-1)
Dollars per person-rem	$2000 (provided in NUREG/BR-0184)
Licensing period	40 years
Economic discount rate	7% and 3% (recommended in NUREG/BR-0184)

With these inputs, the monetary valuation of reducing the base case core damage frequency to zero is presented in Table 7.3-1. The monetary valuation, known as the maximum averted cost-risk, is conservative because no SAMA can reduce the core damage frequency to zero.

The maximum averted cost-risk for a single ABWR at the proposed STP site is $6,900. Even with a conservative 3% discount rate, the valuation of the averted risk is only approximately $12,500.

These values are higher than the GE generic analysis result of $269. However, the GE analysis (Reference 7.3-2) used a different methodology that did not calculate a cost-risk for each accident sequence, did not calculate net present value, and used $1000 per person-rem instead of $2000. If STPNOC were to perform the analysis described in ER Section 7.2 using the GE methodology (Reference 7.3-2), the resulting dose-risk value would be $258. This $258 value is approximately the same as the GE value. Even using the STPNOC values, the results of the SAMDA analysis performed by GE for the ABWR would not be affected; i.e., there still would be no cost-effective design alternatives.

Due to the costs associated with processing administrative changes (including training costs), administrative changes are likely to cost more than the maximum averted cost-risk of $6,900 (or even $12,500). Furthermore, since administrative changes would likely have a small impact on risk, the reduction in risk benefit of administrative changes will likely be substantially less than the cost of the administrative changes. Therefore, it may be concluded that administrative changes are not reasonable SAMAs.

Evaluation of specific administrative controls will occur when the STP 3 & 4 design is finalized and plant administrative processes and procedures are being developed. At that time, appropriate administrative controls on plant operations would be incorporated into the management systems for STP 3 & 4.

7.3.4 References

7.3-1 "Probabilistic Evaluations," Chapter 19D, ABWR Standard Safety Analysis Report, Amendment 35, General Electric.

7.3-2 "Technical Support Document for the ABWR," Revision 1, MPL No. A90-3230, General Electric, San Jose, California, November 18, 1994.

7.3-3 "Regulatory Analysis Technical Evaluation Handbook," NUREG/BR-0184, January 1997.

Table 7.3-1 STP Maximum Averted Cost-Risk for one ABWR Net Present Value (2007 dollars)

	7% Discount Rate	3% Discount Rate
Offsite exposure cost	$66	$158
Offsite economic cost	$20	$48
Onsite exposure cost	$68	$140
Onsite cleanup cost	$2,300	$4,700
Replacement power cost	$4,400	$7,400
Approximate Total	**$6,900**	**$12,500**

7.4 Transportation Accidents

A description of the methodology used to analyze the environmental impacts associated with the transportation of radioactive materials, including accidents, is provided in Subsection 5.11.2. The NRC has analyzed the transportation of radioactive materials in its assessments of environmental impacts for the proposed ESP sites at North Anna, Clinton, and Grand Gulf (References 7.4-1, 7.4-2, and 7.4-3). STPNOC has reviewed the NRC analyses for guidance in assessing transportation impacts for the two proposed ABWR units at the STP site (Reference 7.4-1, 7.4-2, 7.4-3). As discussed in Subsection 5.11.1, transportation of radioactive waste is enveloped by the analyses in [WASH 1238 and NUREG-75/038] and no further analysis is required. STPNOC has prepared additional analyses of fuel transportation effects.

7.4.1 Transportation of Unirradiated Fuel

Accident risks are calculated as a product of frequency and consequence. Accident frequencies for transportation of fuel to future reactors are expected to be lower than those used in the analysis in WASH-1238 (Reference 7.4-4), which forms the basis for Table S-4 of 10 CFR 51.52, because of improvements in highway safety and security since that document was written. Traffic accident, injury, and fatality rates have fallen over the past 30 years. The consequences of accidents that are severe enough to result in a release of unirradiated particles to the environment from advanced light water reactors (LWR) fuels are projected to not be significantly different from those for current generation LWRs. The fuel form, cladding, and packaging are similar to those used in the LWRs analyzed in WASH-1238 (Reference 7.4-4). Consequently, as described in NUREG-1811 (Reference 7.4-1), NUREG-1815 (Reference 7.4-2), and NUREG-1817 (Reference 7.4-3), the risks of accidents during transportation of unirradiated fuel to STP 3 & 4 would be expected to be smaller than the reference LWR results listed in Table S-4.

7.4.2 Transportation of Spent Fuel

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

The radionuclide inventory for the ABWR spent fuel after five years of decay has been developed by GE. Previous NRC evaluations (References 7.4-1, 7.4-2, and 7.4-3) have identified the radionuclides that are the dominant contributors to transportation accident risks. The dominant radionuclides are similar regardless of the fuel type. Based on a review of the NRC analyses, it was determined that the screening results were appropriate to apply to the ABWR fuel inventory to simplify the RADTRAN 5 calculations. The spent fuel inventory used in this analysis for the ABWR is presented in Table 7.4-1.

Engineered shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71, "Packaging and Transportation of Radioactive Material." Spent fuel shipping casks must be certified as Type B packaging systems, meaning they must withstand a series of severe hypothetical accidents with essentially no loss of containment or shielding capability. According to NUREG/CR-6672, Volume 1 (Reference 7.4-5), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01% (i.e., more than 99.99% of all accidents would result in no release of radioactive material from the shipping cask). This analysis assumed that shipping casks for advanced LWR spent fuels would provide mechanical and thermal protection of the spent fuel cargo that is equivalent to that for current generation spent fuel.

The RADTRAN 5 accident risk calculations were performed using radionuclide inventories per shipment for the spent fuel from the ABWR assuming 0.5 MTU per shipment. The resulting risk estimates were multiplied by the expected annual spent fuel shipments (MTU per year) to derive estimates of the annual accident risks associated with spent fuel shipments from the ABWR. The amount of spent fuel shipped per year was assumed to be equivalent to the annual discharge quantity: 42 MTU per year for an ABWR. (This discharge quantity has not been normalized to the reference LWR. The normalized value is presented in Table 7.4-2.)

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

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

(1) External dose from exposure to the passing cloud of radioactive material.

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

(3) Internal dose from inhalation of airborne radioactive contaminants.

(4) Internal dose from resuspension of radioactive materials that were deposited on the ground (the radiation exposures from this pathway were included even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures).

(5) Internal dose from ingestion of contaminated food (the analysis assumed interdiction of foodstuffs and evacuation after an accident; no internal dose due to ingestion of contaminated foods was calculated).

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

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

Table 7.4-2 presents unit (per MTU) accident risks associated with transportation of spent fuel from the STP site to the proposed Yucca Mountain repository. The accident risks are provided in the form of a collective population dose (i.e., person-rem over the shipping campaign). The table also presents estimates of accident risk per reactor year normalized to the reference reactor analyzed in WASH-1238 (Reference 7.4-4). The transportation accident impacts were also calculated for the alternative sites (Allens Creek, Limestone, Malakoff, and Parish) within the region of interest.

The risk to the public from radiation exposure was estimated using the nominal probability coefficient for total detriment [730 fatal cancers, nonfatal cancers, and severe hereditary effects per 1×10^6 person-rem from ICRP Publication 60 (Reference 7.4-6)] health effects per reference reactor year. These values are presented in Table 7.4-2. These estimated risks are quite small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation using the same linear, no-threshold dose response model. Therefore, there will be negligible increases in environmental risk effects as a result of accidents that may result from shipping spent fuel from the STP site to a spent fuel disposal repository.

7.4.3 Conclusion

Based on these analyses, STPNOC concludes that the overall transportation accident risks associated with spent fuel shipments from the proposed ABWR units at the STP site are SMALL. This is consistent with the NRC conclusion regarding the risks associated with transportation of spent fuel from current generation reactors presented in WASH-1238 (Reference 7.4-4) and in Table S-4 of 10 CFR 51.52.

7.4.4 References

7.4-1 "Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site, NUREG-1811," December 2006.

7.4-2 "Environmental Impact Statement for an Early Site Permit (ESP) at the Exelon ESP Site, NUREG-1815," July 2006.

7.4-3 "Environmental Impact Statement for an Early Site Permit (ESP) at the Grand Gulf ESP Site, NUREG-1817," April 2006.

7.4-4 "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238," December 1972.

7.4-5 "Reexamination of Spent Fuel Shipment Risk Estimates," NUREG/CR-6672, Volume 1, March 2000.

7.4-6 "Recommendations of the International Commission on Radiological Protection, ICRP Publication 60," ICRP (International Commission on Radiological Protection), 1991.

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

Radionuclide	ABWR Inventory Ci/MTU
Co-60	3630
Am-241	1440
Am-242m	33.2
Am-243	59.5
Ce-144	1.32×10^4
Cm-242	62.2
Cm-243	61.7
Cm-244	1.35×10^4
Cm-245	2.25
Cs-134	7.76×10^4
Cs-137	1.58×10^5
Eu-154	1.56×10^4
Eu-155	8270
Pm-147	3.13×10^4
Pu-238	1.09×10^4
Pu-239	427
Pu-240	852
Pu-241	1.35×10^5
Pu-242	3.19
Ru-106	2.29×10^4
Sb-125	7,170
Sr-90	1.06×10^5
Y-90	1.06×10^5

Ci/MTU = curies per metric ton uranium

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

Site	Unit Population Dose (Person-Rem per MTU) [1]	MTU per Reference Reactor Year	Population Dose (Person-Rem per Reference Reactor Year) [1]	Total Detrimental Health Effects per Reference Reactor Year
STP	5.15×10^{-8}	29	1.50×10^{-6}	1.09×10^{-9}
Allens Creek	5.05×10^{-8}	29	1.47×10^{-6}	1.07×10^{-9}
Limestone	5.32×10^{-8}	29	1.54×10^{-6}	1.13×10^{-9}
Malakoff	5.21×10^{-8}	29	1.51×10^{-6}	1.10×10^{-9}
Parish	5.75×10^{-8}	29	1.67×10^{-6}	1.22×10^{-9}

[1] Value presented is the product of probability and collective dose

7.5S Design Basis Accident or Severe Accident Impact on Other STP Units

Section 7.1 describes the impacts that a design basis accident would have at one of the ABWR units (STP 3 or 4). Sections 7.2 and 7.3 describe the impacts and costs that a severe accident would have at one of the ABWR units. This section describes (1) the impacts that a design basis accident or severe accident at one of the ABWR units would have on the other three onsite units (the other ABWR unit and STP 1 & 2); and (2) the impacts that a design basis accident or severe accident at either STP 1 or 2 would have on the ABWR units. With a few exceptions, this section does not evaluate the impacts of an accident at STP 1 on STP 2, or vice versa, because such an evaluation is unrelated to STP 3 & 4.

There is no mechanism for fire or explosion by which one unit could affect other units. FSAR Section 2.2S.3 evaluates potential accidents that could impact STP 3 & 4, including those resulting in fires or explosions. That section demonstrates that STP 3 & 4 are located at a safe distance from chemical storage facilities for STP 1 & 2 and therefore are not at risk to impacts from explosions, fires, or release of toxic chemicals from STP 1 & 2. As further discussed in FSAR Section 2.2S.3, the chemicals used at STP 3 & 4 are similar to the chemicals used in STP 1 & 2 and would not be stored any closer than the determined safe distances from explosions, fires, or release of toxic chemicals to STP 1 & 2 and STP 3 & 4. Furthermore, each of the ABWR units is designed to withstand or achieve safe shutdown in the event of fires, explosions, and toxic gases originating at that unit; therefore, each would be able to withstand such events originating at the other ABWR unit. Additionally, design features of the ABWR and STP 1 & 2 would mitigate other types of indirect impacts. For example, in the event of a power disruption caused by the accident at one unit, the emergency diesel generators at the other units can be started to ensure that the other units have sufficient electrical power to provide for and maintain safe shutdown of the other units.

Therefore, this section evaluates a scenario in which airborne radioactivity released from a design basis accident or severe accident at an affected unit (the unit at which the initiating accident occurs) may result in an accident or service disruption at an unaffected unit (a unit other than that at which the initiating accident occurs). A service disruption would entail a delay in returning the unaffected units to service as a result of repair, refurbishment, decontamination, or corrective action. The evaluation considers whether exposures could interrupt safe shutdown of an unaffected unit by either interfering with operator actions or by interfering with or damaging equipment with a safety function. Additionally, this evaluation discusses the environmental impacts and quantifies the potential cost of the temporary loss of use of one of the unaffected units, analogously to that of the affected unit, as described in Section 7.3.

7.5S.1 Background on Impact Mitigation and Prevention

As discussed below, various factors at an unaffected unit mitigate or prevent the impacts from a radiological accident at the affected unit. These include the warning time that the unaffected unit receives of an accident, the ability to place the unaffected unit into a safe shutdown condition, control room habitability, and shielding to personnel and equipment from the plant design.

Plant design and procedures provide protection for operators. Control room habitability systems are designed to protect the control room during an accident and include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, and fire protection. 10CFR50, Appendix A, General Design Criterion (GDC) 19 specifies a 5 rem control room operator dose limit for releases from a design basis accident at that unit. If this dose were to be exceeded within the control room during a severe accident, operators could be protected with additional measures, such as donning a SCBA (self contained breathing apparatus) and limiting exposure time. Additionally, once a plant is shut down, stable, and in long term decay heat removal, operator action is not continually necessary to maintain the plant in a safe shutdown condition. Once a stable cool down rate has been established, operator adjustments are no longer required to maintain the plant in a stable safe shutdown condition. Therefore, at that time, the operators could be evacuated from the control room if necessary.

An important factor in mitigating and preventing major impacts at an unaffected unit is the warning time that an operator of that unit receives of an accident at the affected unit. With sufficient warning time prior to radioactive releases from the affected unit, the unaffected units can be put into a safe shutdown mode. Additionally, in the event of a design basis accident or severe accident, non-essential site personnel could be evacuated in keeping with site emergency procedures.

A unit can be put into a hot shutdown condition, in which the reactor is completely shutdown, within minutes. Cooling operations can be commenced shortly after hot shutdown. After about 3 hours, the reactor will be in a stable long-term decay heat removal condition. Once long term decay heat removal is established the operations staff will adjust the cooling systems to establish a stable cool down rate. After that time, operator action is not necessary to maintain the plant in a safe shutdown condition. If the time increment between the onset of the accident and the airborne radioactivity release at the affected unit is longer than the time it takes to place the reactor into a stable long-term decay heat removal condition (approximately 3 hours), then there would be no impact on safe shutdown of an unaffected unit. Once the unaffected units are in safe shutdown, a release from the affected unit would not adversely impact maintenance of that safe shutdown condition because, as described below, the equipment can withstand the doses associated with the release without loss of safety function.

Equipment can inherently withstand large radiation doses from an accident, and plant design features, such as shielding, provide additional protection. For example, the concrete of the containment structure provides substantial shielding and the containment is sealed, thus preventing the intrusion of airborne radioactivity to equipment within containment. The concrete in other concrete buildings also would provide some shielding to equipment. Equipment can also withstand the release of radioactive material from accidents. For example, FSAR Section 2.2S.3.1.7 explains that safety-related structures, systems, and components for the ABWR are designed to withstand the effects of radiological events and the consequential releases from design basis accidents.

Section 7.5S.4 below discusses the expected doses at the exterior of the unaffected units resulting from the evaluated accidents, and demonstrates that the structures and equipment in the unaffected units would still be able to perform their safety function given such doses.

7.5S.2 Evaluation of Impacts of Design Basis Accidents on Safe Shutdown of Other Units

Design Basis Accidents Originating in STP 3 or 4

Section 7.1 provides STP 3 & 4 Exclusion Area Boundary doses that would result from design basis accidents at either STP 3 or 4. FSAR Section 15.6 demonstrates that doses in the control room would be within the regulatory limit of 5 rem during a design basis accident at that same unit. The dose at the control room of the unaffected ABWR would also be within the regulatory limit of 5 rem during a design basis accident at the affected ABWR unit because the design and protection of the control rooms are identical. The control room dose at STP 1 & 2 would be similar in magnitude to the doses experienced at the unaffected ABWR unit control room because of the protection of the control room habitability systems at STP 1 & 2, which are required to satisfy the requirements of GDC 19 similar to STP 3 & 4. In fact, the doses would likely be less due to the larger distance between the affected unit and STP 1 & 2 than the affected unit and the other ABWR. Therefore, the doses experienced at the control rooms of STP 1 & 2 or the unaffected ABWR unit from a design basis accident at either STP 3 or 4 would not prevent the operators from completing safe shutdown of the unaffected units.

As discussed below in Section 7.5S.4, equipment can withstand severe accident doses without loss of function. Therefore, a design basis accident at STP 3 or 4 would not have an impact on the safe shutdown of STP 1 & 2 and the other ABWR unit.

Design Basis Accidents Originating in STP 1 or 2

As described in FSAR Section 2.2S.3.1.7, radiological releases from a design basis accident at STP 1 & 2 would not threaten the safety of STP 3 & 4. As noted previously, control room habitability systems can detect and protect control room personnel from airborne radioactivity. Therefore, a design basis accident at STP 1 or 2 would not cause the operators at STP 3 & 4 to exceed the 5 rem limit in GDC 19 and would not prevent the operators from completing safe shutdown of the ABWR units.

FSAR Section 2.2S.3.1.7 also states that safety related structures, systems, and components for the ABWR have been designed to withstand the effects of airborne releases from a design basis accident at the ABWR that would bound the airborne release from either STP 1 or 2. Therefore, a design basis accident at STP 1 or 2 would not have an impact on the safe shutdown of STP 3 & 4.

In summary, a design basis accident at any affected unit would not impact the safe shutdown of an evaluated unaffected unit.

7.5S.3 Evaluation of Impacts of Severe Accidents on Safe Shutdown of Other Units

Severe Accidents Originating in STP 3 or 4

Section 7.2 describes the offsite dose and cost risks that could accompany a severe accident at either STP 3 or 4. A number of accident sequences, each of which represents a broader family of accidents, are analyzed for such an accident class. For the ABWR, ten accident sequences are analyzed for internally initiated events. The sum of the frequencies of occurrence for each of the ten accident sequences, which are shown in Table 7.2-1, is the core damage frequency (CDF). The CDF of an ABWR for internal events is 1.6 $x10^{-7}$ per year.

However, not all core damage events result in a large release (i.e., the containment is able to prevent the release of significant amounts of radioactivity to the environment). Absent a large release, there would be no impact on the safe shutdown of the unaffected units. In general, the large release frequency (LRF) for the ABWR for internal events is about 2.2×10^{-8} per year. Furthermore, most of the LRF consists of a release which has been scrubbed by the suppression pool water and passes through the Containment Overpressure Protection System. The LRF for a release that has not been scrubbed is 1.0×10^{-9} per year.

Externally initiated events, and their associated small contribution to risk, are described in FSAR Section 19.4 and 19.6, which in turn incorporate by reference the associated sections of the ABWR DCD. As stated in the Final Safety Evaluation Report for the ABWR (NUREG-1503):

> Although direct comparison of external-event results to [the Commission's safety goals] is not possible, the ABWR design has significant margins above the design bases for seismic, fire, and internal flood-initiating events and, where computed, has low estimated core damage frequencies from these bounding analyses. The staff believes that the ABWR design meets the Commission's safety goals.

As discussed previously, operators with sufficient warning of an accident at an affected unit can safely shutdown an unaffected unit. The time increment from general emergency warning time until the first release of radioactivity to the environment for all ten accident sequences is greater than the time required to put an unaffected unit into a stable long-term decay heat removal condition. Therefore, any doses experienced at the control rooms of STP 1 & 2 or the unaffected ABWR unit from a severe accident at either STP 3 or 4 would not prevent the operators from completing safe shutdown of the unaffected units. Additionally, as discussed below in Section 7.5S.4, equipment can withstand the bounding radiation dose from a severe accident without loss of function.

Severe Accidents Originating in STP 1 or 2

A similar analysis of representative accident sequences was performed for STP 1 & 2. Nine representative sequences were analyzed, with a total CDF of $1.0x10^{-5}$ per year and a Large Early Release Frequency (LERF) of $6.1x10^{-7}$ per year for internal and external events. The time increment from general emergency warning time until the first release of radioactivity to the environment for all nine representative sequences is greater than the time required to put an unaffected unit into a stable long-term decay heat removal condition. Therefore, any doses experienced at the control rooms of STP 3 or 4 from a severe accident at either STP 1 or 2 would not prevent the operators from completing safe shutdown of the unaffected ABWR units. Additionally, as discussed below in Section 7.5S.4, equipment can withstand the bounding radiation dose from a severe accident without loss of function.

In summary, a severe accident at any affected unit would not impact the safe shutdown of an evaluated unaffected unit.

7.5S.4 Evaluation of Impacts of Design Basis Accidents and Severe Accidents on Equipment Function

Equipment at the unaffected units would continue to function under the accident scenario considered above in Sections 7.5S.2 and 7.5S.3 that results in the highest radiation dose. The accident with the highest radiation dose is an Induced Steam Generator Tube Rupture (ISGTR) at STP 1 or 2, which is characterized by a low CDF of 6.1×10^{-7} per year.

The ISGTR is initiated by a loss of offsite power (LOOP) that is not recoverable prior to fuel damage in the affected unit. All emergency diesel generators are assumed to also fail and are not able to be recovered. The turbine driven auxiliary feedwater pump likewise is assumed to fail and is also not able to be recovered.

Exposures at an unaffected ABWR unit resulting from severe accident releases from this accident sequence were calculated using the MACCS2 code. That code is described in Section 7.2, and was used there to calculate offsite dose and cost risks from severe accidents. Only the code's early release phase was exercised for this analysis, i.e., short-term doses to personnel and equipment during the release plume passage were calculated. This is the period of concern relative to safe shutdown of the unaffected units and onsite contamination.

The STP 1 & 2 ISGTR accident sequence results in an estimated worst case dose at STP 3 & 4 of up to 2,500 rad to air. This calculated exposure is without any shielding, and is cumulative over the entire duration of the early airborne releases. The meteorology is also conservatively assumed to be at the 95% level, meaning that 95% of the time actual weather conditions would lead to less exposure at that location.

Equipment at the unaffected units can withstand this bounding radiation dose and continue to properly function. As discussed in the STP 1 & 2 Design Criteria for Equipment Qualification Program (Reference 7.5S-1), an environment with exposures of less than 1×10^5 rad is considered to be a mild environment and does not require any special qualification requirements to ensure equipment function. Thus, equipment installed at the STP units can withstand 1×10^5 rad of radiation exposure without any impact on equipment functionality. As discussed above, the bounding radiation dose for the evaluated accidents is 2,500 rad, which is significantly less than the radiation level that would impact equipment functionality. Therefore, all equipment necessary to complete safe shutdown of the unaffected units would be able to operate as designed without any degradation to its functional capabilities for the exposure levels associated with the airborne release from the accidents evaluated above.

7.5S.5 Economic Impacts of a Temporary Shutdown of the Unaffected Units

The potential economic impacts from an accident at an affected unit on an unaffected unit are quantified by monetizing the onsite exposure and cleanup costs at the unaffected units together with replacement power costs from an outage at the unaffected units. The calculations are analogous to those in Section 7.3 (using the same methodology as described there unless otherwise noted) for an accident and attendant impacts from a single affected unit.

The principal inputs to the analysis are the severe accident CDF, the outage period of the other site units, and the economic discount rate (7 percent and 3 percent are NRC precedents established by NUREG/BR-0184). With these inputs, monetized impacts per unaffected unit are presented in Tables 7.5S-1 and 7.5S-2 for an event at one of the existing units (STP 1 & 2) and at one of the ABWR units (STP 3 & 4), respectively. A design basis accident would have much lower releases associated with the accident compared to a severe accident, resulting in much lower contamination levels that would be bounded by the evaluation for a severe accident. Therefore, only the severe accident CDFs were considered to produce a long term outage period from the associated cleanup and refurbishment of equipment.

Unlike in Section 7.3, where an accident at one unit could result in offsite impacts due to radiation releases from that unit, no releases or offsite impacts would result from the unaffected units. In order to determine the economic impact of a temporary shutdown in the unaffected units, the methodology used for the affected unit is conservatively applied here to all unaffected units.

The onsite cleanup cost includes cleanup and decontamination of the unaffected units. The cleanup of these units, unlike for the affected unit (as analyzed in Section 7.3), is based on recovering the units for restart. Recoverable cleanup costs have been estimated as 30% of the non-recoverable cleanup costs (as included in the Section 7.3 initiating unit costs) for BWR units (STP 3 & 4) and 26% for PWR units (STP 1 & 2) (Reference 7.5S-2); 30% is conservatively used here for all units. Those costs are based on cleanup of a small LOCA (loss of cooling accident) which results in a moderately contaminated containment building; applying these costs to units which would not have internal releases but instead can be contaminated by external releases from the initiating unit is conservative.

It is expected that the unaffected units could be restarted within months. However, it took 6 years to restart Three Mile Island Unit 1 (TMI-1) after the accident at TMI-2. It should be noted that cleanup and refurbishment were not the driving actions for the restart delay. Instead, the restart awaited application and approval of lessons learned from the TMI-2 accident. This analysis assumes that the unaffected unit with the same design of the affected unit would be shutdown for 6 years for cleanup, refurbishment, and application and approval of lessons learned. The analysis assumes that the other two unaffected units would be shutdown for 2 years for cleanup and refurbishment.

The cost of repairs and refurbishment at the unaffected units was estimated at $1,400 per hour of outage duration, which is $1,000 per hour escalated to the calculation basis year (Reference 7.5S-2), and is included in the onsite cleanup cost. If Tables 7.5S-1 and 2 were based on 6 months of outage rather than 6 years, the total costs would be 40-45% (depending on discount rate) of those indicated in the tables. Almost all of that decrease would be due to the decrease in replacement power cost.

NRC suggests that a typical short-term replacement power cost (i.e., power plant that will be restarted) for a 910 MWe power plant is $310,000 per day (Reference 7.5S-2). That value, scaled to the STP power levels, replaced the present value in the replacement power cost calculation of Section 7.3 for this analysis.

The monetized impacts for an accident at STP 3 or 4 affecting another unit are very low, as shown in Table 7.5S-2. The cost at a 7 percent discount rate at the other ABWR unit would be approximately $3,000 and the cost at STP 1 or 2 would be approximately $1,800 per unit. Even at a 3 percent discount rate, the cost at the other ABWR unit would be approximately $4,500 and the cost at STP 1 or 2 would be approximately $3,000 per unit. These costs are less than half of the costs of an accident at the affected unit. The Section 7.3 conclusion that there is no cost-effective ABWR operation design change holds for the mitigation of impacts at other site units.

The monetized risk-based impacts from an accident at STP 1 or 2 on the ABWR units are larger than the impacts from an ABWR initiated accident case, due to the larger CDF of the existing units. The monetized impact cost to an ABWR unit from a large severe accident release at one of the existing units is shown in Table 7.5S-1. The cost at a 7 percent discount rate at the ABWR units would be approximately $110,000 per unit, and at a 3 percent discount rate would be approximately $170,000 per unit. None of the severe accident mitigation design alternatives considered for the ABWR would be cost effective and mitigate the potential impacts (contamination and down time) from a large release severe accident at the existing units (Reference 7.5S-3).

7.5S.6 Conclusions

As demonstrated above, a design basis accident or a severe accident at the affected unit would not prevent the unaffected units from safely shutting down. Additionally, all equipment necessary to complete safe shutdown of the unaffected units would be able to operate as designed without any degradation to its functional capabilities for the exposure levels associated with the airborne release from the accidents evaluated. Therefore, the accident scenarios would not result in any incremental environmental impacts attributable to the unaffected units beyond those evaluated in Sections 7.1 and 7.2.

Furthermore, even if it is arbitrarily postulated that a severe accident in the affected unit could cause a simultaneous severe accident in each of the unaffected units, the cumulative environmental impacts would still be SMALL. In such a scenario, the releases of radioactivity from all four units would be approximately four times the release from an individual unit. However, even if the environmental impacts (risks) discussed in Section 7.2.4 for an accident originating in one of the ABWR units were to be multiplied by a factor of four, the environmental risks would still be insignificant.

For example, the cumulative risk from all four units would be about 0.017 person-rem/year (i.e., 4 x 0.0043 person-rem per reactor year), which is more than a factor of ten less than the cumulative dose risk from normal operation (about 0.5 person-rem per reactor year). Furthermore, the risk of cancer from such an accident scenario would be about 0.0044% of the background risk (i.e., four times 0.0011% of the background risk). This value is well below the 0.1% value specified in the Commission's Safety Goal Policy Statement.

As discussed in Section 7.5S.3, the LERF for Units 1 and 2 is approximately 30 times greater than the LRF for Units 3 and 4. However, even if the risk-based values in the previous paragraph were to be multiplied by a factor of 30, the resulting dose risk would be equivalent to the cumulative dose risk from normal operation and the resulting cancer risk would be equivalent to the Commission’s Safety Goal. Therefore, the environmental impact from such a scenario would be SMALL.

7.5S.7 References

7.5S-1 Design Criteria for Equipment Qualification Program 4EQ19NQ1009 for South Texas Nuclear Operating Company (Units 1&2), January 26, 2000.

7.5S-2 NRC (U.S. Nuclear Regulatory Commission). 1997. Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184. Office of Nuclear Reactor Regulation. Washington, D.C. January.

7.5S-3 GE (General Electric Company). 1995. Technical Support Document for the ABWR, Revision 1, MPL Number A90-3230, 25A5680, January.

Table 7.5S-1 Monetized Impacts for an Accident at STP 1 or 2 per Unaffected Unit.

	STP 3 or 4 Impacts	
	7 Percent Discount Rate	3 Percent Discount Rate
Onsite exposure cost	$5,569	$10,355
Onsite cleanup cost	$61,900	$108,526
Replacement power cost	$42,727	$55,194
Total	$110,196	$174,075

Table 7.5S-2 Monetized Impacts for an Accident at STP 3 or 4 per Unaffected Unit.

	STP 3 or 4 Impacts		STP 1 or 2 Impacts	
	7 Percent Discount Rate	3 Percent Discount Rate	7 Percent Discount Rate	3 Percent Discount Rate
Onsite exposure cost	$76	$152	$137	$219
Onsite cleanup cost	$949	$1,764	$934	$1,676
Replacement power cost	$1,980	$2,557	$688	$1,153
Total	$3,005	$4,473	$1,759	$3,049