Nuclear Development
244 Chestnut Street, Salem, NJ 08079

PSEG
Power LLC

ND-2012-0072
October 19, 2012

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **PSEG Early Site Permit Application**
Docket No. 52-043
Response to Request for Additional Information, No. Env-11, ESP EIS 5.11 - Environmental Impacts of Postulated Accidents

References: 1) PSEG Power, LLC Letter No. ND-2012-0031 to USNRC, Submittal of Revision 1 of the Early Site Permit Application for the PSEG Site, dated May 21, 2012

2) Env-11, Review Section: ESP EIS 5.11 - Environmental Impacts of Postulated Accidents, dated September 20, 2012 (eRAI 6739)

The purpose of this letter is to respond to the request for additional information (RAI) identified in Reference 2 above. This RAI addresses Question Nos. ESP EIS 5.11-1 through ESP EIS 5.11-6 for the Environmental Report (ER), as submitted in Part 3 of the PSEG Site Early Site Permit Application, Revision 1.

Enclosure 1 provides our response for RAI No. Env-11, Question Nos. ESP EIS 5.11-1 and ESP EIS 5.11-3 through ESP EIS 5.11-6 (rACC-01, rACC-01b, rACC-01c, rACC-02, and rACC-03). Question No. ESP EIS 5.11-6 requests the MACCS2 input and output files and Calculation 2009-11222, Revision 1, "Environmental Consequence Analysis for PSEG ESPA". The MACCS2 files and the requested calculation contain information proprietary to individual reactor vendors in the form of severe accident scenarios and inventories. PSEG is working with the reactor vendors to determine what portions of this information can be released publicly. A supplemental response to Question No. ESP EIS 5.11-6 will be provided by November 2, 2012. The response to RAI No. Env-11, Question No. ESP EIS 5.11-2 (rACC-01a) will be provided by November 2, 2012, as provided for in the issuance of the final RAI.

D079
NRO

Enclosure 2 includes the revisions to the ER resulting from our response to RAI No. Env-11, Question Nos. ESP EIS 5.11-1 and 5.11-5 (rACC-01 and rACC-02).

If any additional information is needed, please contact David Robillard, PSEG Nuclear Development Licensing Engineer, at (856) 339-7914.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 19th day of October, 2012.

Sincerely,

Christine T. Neely
Director Regulatory Affairs
PSEG Nuclear, LLC

Enclosure 1: Response to NRC Request for Additional Information, RAI No. Env-11, Question Nos. ESP EIS 5.11-1, and ESP EIS 5.11-3 through ESP EIS 5.11-6 (rACC-01, rACC-01b, rACC-01c, rACC-02, and rACC-03), Review Section: ESP EIS 5.11 - Environmental Impacts of Postulated Accidents

Enclosure 2: Proposed Revisions, Part 3 – Environmental Report (ER), Chapter 7 – Environmental Impacts of Postulated Accidents Involving Radioactive Materials

cc: USNRC Project Manager, Division of New Reactor Licensing, PSEG Site (w/enclosures)
USNRC Environmental Project Manager, Division of New Reactor Licensing (w/enclosures)
USNRC Region I, Regional Administrator (w/enclosures)
Oak Ridge National Laboratory

PSEG Letter ND-2012-0072, dated October 19, 2012

ENCLOSURE 1

RESPONSE to RAI No. Env-11

QUESTION Nos.
ESP EIS 5.11-1 (rACC-01)
ESP EIS 5.11-3 (rACC-01b)
ESP EIS 5.11-4 (rACC-01c)
ESP EIS 5.11-5 (rACC-02)
ESP EIS 5.11-6 (rACC-03)

Response to RAI No. Env-11, Question ESP EIS 5.11-1

In Reference 2, the NRC staff asked PSEG for information regarding Environmental Impacts of Postulated Accidents, as described in Chapter 7 of the Environmental Report. The specific request was:

> *rACC-01: Provide the proper meteorological data for the 50th percentile (χ/Q values) and revise the associated tables in the ER*
>
> *As per ESRP 7.1, the NRC staff must ensure that the applicant used a 50th percentile χ/Q value that was based on onsite meteorological data, or 10% of the levels given in Regulatory Guide 1.3 or Regulatory Guide 1.4, to represent more realistic dispersion conditions than assumed in the safety evaluation.*
>
> *During the Environmental Site Audit, it was found that some of the meteorological data used to determine the 50th percentile χ/Q values were incorrect. ER Tables 7.1-38, 7.1-40, 7.1-46, and 7.1-55 need to be revised.*

PSEG Response to NRC RAI:

The 50^{th} percentile χ/Q values are calculated and summarized in Table ESP EIS 5.11-1-1. The new sector dependent 50^{th} percentile χ/Q values replace the sector dependent annual average χ/Q values for determining the LPZ time dependent χ/Q values. These values are used to determine the doses at the PSEG Site by multiplying the reactor-specific doses by the ratio of the associated site χ/Q value and the reactor-specific χ/Q value. The following tables summarize the χ/Q ratios and the doses at the PSEG Site for the reactor technologies addressed in the Early Site Permit application:

- Tables ESP EIS 5.11-1-2 and ESP EIS 5.11-1-3 - US-APWR
- Tables ESP EIS 5.11-1-4 through ESP EIS 5.11-1-9 – ABWR
- Tables ESP EIS 5.11-1-10 through ESP EIS 5.11-1-18 - AP1000
- Tables ESP EIS 5.11-1-19 and ESP EIS 5.11-1-20 - U.S.-EPR

The content of these tables is used to update the affected Environmental Report (ER) tables. The affected ER tables, including ER Tables 7.1-38, 7.1-40, 7.1-46, and 7.1-55, are revised as specified in Enclosure 2 of this document.

Associated PSEG Site ESP Application Revisions:

ER Tables 7.1-38 through 7.1-56 will be updated and associated text changes made to ER Subsections 7.1.2 and 7.1.4 as specified in Enclosure 2 of this document. In addition, the title for Regulatory Guide 1.183 is corrected in Subsection 7.1.2 and a typographical error is corrected in the title to Table 7.1-30A in Chapter 7 List of Tables, as specified in Enclosure 2 of this document.

Table ESP EIS 5.11-1-1: PSEG Site Atmospheric Dispersion Factors (50th Percentile)

Location	Time (hr.)	Site χ/Q (sec/m^3)
EAB	0 to 2	1.41E-04
LPZ	0 to 2	4.72E-06
	0 to 8	2.30E-06
	8 to 24	1.61E-06
	24 to 96	7.51E-07
	96 to 720	3.05E-07

Table ESP EIS 5.11-1-2: US-APWR Radiological Consequences Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	5.00E-04	1.41E-04	0.282
LPZ	0 to 8	2.10E-04	2.30E-06	0.011
	8 to 24	1.30E-04	1.61E-06	0.012
	24 to 96	6.90E-05	7.51E-07	0.011
	96 to 720	2.80E-05	3.05E-07	0.011

Table ESP EIS 5.11-1-3: US-APWR Radiological Consequences Dose Summary

Accident	DCD Dose (rem TEDE)		χ/Q ratio (Site/DCD)		Site Dose (rem TEDE)		Limit
	EAB	LPZ	EAB	LPZ[a]	EAB	LPZ	
Steam System Piping Failure - Pre-Existing Iodine Spike	0.19	0.11	0.282	0.012	5.36E-02	1.32E-03	25
Steam System Piping Failure - Accident-Initiated Iodine Spike	0.32	0.28	0.282	0.012	9.02E-02	3.36E-03	2.5
Reactor Coolant Pump Rotor Seizure	0.49	0.7	0.282	0.012	1.38E-01	8.40E-03	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.1	4.5	0.282	0.012	1.44E+00	5.40E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.5	0.6	0.282	0.012	4.23E-01	7.20E-03	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	3.6	1.5	0.282	0.012	1.02E+00	1.80E-02	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.96	0.43	0.282	0.012	2.71E-01	5.16E-03	2.5
LOCA	13	13	0.282	0.012	3.67E+00	1.56E-01	25
Fuel Handling Accident	3.3	1.4	0.282	0.012	9.31E-01	1.68E-02	6.3

a) LPZ doses are not given in time-dependent form; therefore, the most conservative Site/DCD χ/Q ratio (from the 8 to 24 hour interval) is used.

Table ESP EIS 5.11-1-4: ABWR Radiological Consequences Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
All Accidents	EAB	0 to 2	1.37E-03	1.41E-04	0.103
	LPZ	0 to 2	4.11E-04	4.72E-06	0.011
LOCA Only		0 to 8	1.56E-04	2.30E-06	0.015
		8 to 24	9.61E-05	1.61E-06	0.017
		24 to 96	3.36E-05	7.51E-07	0.022
		96 to 720	7.42E-06	3.05E-07	0.041

Table ESP EIS 5.11-1-5: ABWR Radiological Consequences PSEG Site-Specific Dose Summary

Accident	Thyroid Dose (Sv)	Whole Body Dose (Sv)	Thyroid Limit (Sv)	Whole Body Limit (Sv)
Failure of Small Lines Carrying Primary Coolant Outside Containment[(a)]	4.94E-03	9.68E-05	3.00E-01	2.50E-02
LOCA - EAB	2.14E-01	4.62E-03	3.00E+00	2.50E-01
LOCA - LPZ	7.72E-02	9.82E-04	3.00E+00	2.50E-01
Fuel Handling Accident[(a)]	8.46E-02	1.35E-03	7.50E-01	6.25E-02
Main Steamline Break Case 1[(a)(b)]	2.68E-03	6.39E-05	3.00E-01	2.50E-02
Main Steamline Break Case 2[(a)(b)]	5.25E-02	1.34E-03	3.00E+00	2.50E-01

a) The dose is calculated for the maximum two hour EAB meteorology, only, based on the DCD.

b) The level of activity is consistent with an off-gas release rate of 3.7 GBq/s for Case 1 and 14.8 GBq/s for Case 2, referenced to a 30 minute decay. The iodine concentrations in the reactor coolant are tabulated below for each case.

	MBq/g	
Isotope	Case 1	Case 2
I-131	0.001739	0.03515
I-132	0.01536	0.30747
I-133	0.01206	0.24161
I-134	0.02634	0.52688
I-135	0.01647	0.3293

Table ESP EIS 5.11-1-6: ABWR Radiological Consequences Dose for an Instrument Line Break Accident

DCD			Site	
Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
4.80E-02	9.40E-04	0.103	4.94E-03	9.68E-05

Table ESP EIS 5.11-1-7: ABWR Radiological Consequences Dose for a Fuel Handling Accident

DCD				Site	
Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Uprate Ratio	Thyroid Dose (Sv)	Whole Body Dose (Sv)
7.50E-01	1.20E-02	0.103	1.095	8.46E-02	1.35E-03

Table ESP EIS 5.11-1-8: ABWR Radiological Consequences Dose for a LOCA

		DCD				Site	
Location	Time (hr.)	Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Uprate Ratio	Thyroid Dose (Sv)	Whole Body Dose (Sv)
EAB	0 to 2	1.90E+00	4.10E-02	0.103	1.095	2.14E-01	4.62E-03
LPZ	0 to 8	3.10E-01	1.00E-02	0.015	1.095	5.09E-03	1.64E-04
	0 to 24	5.10E-01	1.80E-02	0.017	1.095	8.81E-03	3.13E-04
	0 to 96	1.30E+00	2.90E-02	0.022	1.095	2.78E-02	5.78E-04
	0 to 720	2.40E+00	3.80E-02	0.041	1.095	7.72E-02	9.82E-04

Table ESP EIS 5.11-1-9: ABWR Radiological Consequences Dose for a Main Steamline Break

	DCD			Site	
	Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
Case 1	2.60E-02	6.20E-04	0.103	2.68E-03	6.39E-05
Case 2	5.10E-01	1.30E-02	0.103	5.25E-02	1.34E-03

Table ESP EIS 5.11-1-10: AP1000 Radiological Consequences Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
LOCA	EAB	0 to 2	5.10E-04	1.41E-04	0.276
	LPZ	0 to 8	2.20E-04	2.30E-06	0.010
		8 to 24	1.60E-04	1.61E-06	0.010
		24 to 96	1.00E-04	7.51E-07	0.008
		96 to 720	8.00E-05	3.05E-07	0.004
Other Accidents	EAB	0 to 2	8.00E-04	1.41E-04	0.176
	LPZ	0 to 8	5.00E-04	2.30E-06	0.005
		8 to 24	3.00E-04	1.61E-06	0.005
		24 to 96	1.50E-04	7.51E-07	0.005
		96 to 720	8.00E-05	3.05E-07	0.004

Table ESP EIS 5.11-1-11: AP1000 Radiological Consequences
PSEG Site-Specific Dose Summary

Accident	Site Dose (rem TEDE)		
	EAB	**LPZ**	**Limit**
Steam System Piping Failure – Pre-Existing Iodine Spike	1.76E-01	3.81E-03	25
Steam System Piping Failure – Accident-Initiated Iodine Spike	1.94E-01	9.67E-03	2.5
Reactor Coolant Pump Shaft Seizure – No Feedwater	1.41E-01	1.95E-03	2.5
Reactor Coolant Pump Shaft Seizure – Feedwater Available	1.06E-01	3.97E-03	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	6.34E-01	2.72E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	3.70E-01	5.10E-03	2.5
Steam Generator Tube Rupture – Pre-Existing Iodine Spike	3.87E-01	6.16E-03	25
Steam Generator Tube Rupture – Accident-Initiated Iodine Spike	1.94E-01	3.99E-03	2.5
LOCA	6.71E+00	2.31E-01	25
Fuel Handling Accident	9.15E-01	1.72E-02	6.3

Table ESP EIS 5.11-1-12: AP1000 Radiological Consequences Dose for a Steam System Piping Failure

Doses for Steam System Piping Failure with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.00E+00		0.176	1.76E-01	
0 to 8		5.81E-01	0.005		2.91E-03
8 to 24		7.18E-02	0.005		3.59E-04
24 to 96		1.08E-01	0.005		5.40E-04
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.00E+00	7.61E-01		1.76E-01	3.81E-03
Limit				25	25

Doses for Steam System Piping Failure with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.10E+00		0.176	1.94E-01	
0 to 8		1.02E+00	0.005		5.10E-03
8 to 24		3.77E-01	0.005		1.89E-03
24 to 96		5.36E-01	0.005		2.68E-03
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.10E+00	1.93E+00		1.94E-01	9.67E-03
Limit				2.5	2.5

Table ESP EIS 5.11-1-13: AP1000 Radiological Consequences Dose for a Reactor Coolant Pump Shaft Seizure Accident

Doses for Reactor Coolant Pump Shaft Seizure with No Feedwater

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	8.00E-01		0.176	1.41E-01	
0 to 8		3.89E-01	0.005		1.95E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	8.00E-01	3.89E-01		1.41E-01	1.95E-03
Limit				2.5	2.5

Doses for Reactor Coolant Pump Shaft Seizure with Feedwater Available

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	6.00E-01		0.176	1.06E-01	
0 to 8		7.94E-01	0.005		3.97E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	6.00E-01	7.94E-01		1.06E-01	3.97E-03
Limit				2.5	2.5

Table ESP EIS 5.11-1-14: AP1000 Radiological Consequences Dose for Spectrum of Rod Cluster Control Assembly Ejection Accidents

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	3.60E+00		0.176	6.34E-01	
0 to 8		4.58E+00	0.005		2.29E-02
8 to 24		7.84E-01	0.005		3.92E-03
24 to 96		6.32E-02	0.005		3.16E-04
96 to 720		2.06E-02	0.004		8.24E-05
Total	3.60E+00	5.45E+00		6.34E-01	2.72E-02
Limit				6.3	6.3

Table ESP EIS 5.11-1-15: AP1000 Radiological Consequences Dose for Failure of Small Lines Carrying Primary Coolant Outside Containment

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.10E+00		0.176	3.70E-01	
0 to 8		1.02E+00	0.005		5.10E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	2.10E+00	1.02E+00		3.70E-01	5.10E-03
Limit				2.5	2.5

Table ESP EIS 5.11-1-16: AP1000 Radiological Consequences Dose for Steam Generator Tube Rupture

Doses for Steam Generator Tube Rupture with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			**Site Dose (rem TEDE)**	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.20E+00		0.176	3.87E-01	
0 to 8		1.16E+00	0.005		5.80E-03
8 to 24		7.24E-02	0.005		3.62E-04
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	2.20E+00	1.23E+00		3.87E-01	6.16E-03
Limit				25	25

Doses for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			**Site Dose (rem TEDE)**	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.10E+00		0.176	1.94E-01	
0 to 8		6.27E-01	0.005		3.14E-03
8 to 24		1.69E-01	0.005		8.45E-04
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.10E+00	7.96E-01		1.94E-01	3.99E-03
Limit				2.5	2.5

Table ESP EIS 5.11-1-17: AP1000 Radiological Consequences Dose for LOCA

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	2.43E+01		0.276	6.71E+00	
0 to 8		2.17E+01	0.010		2.17E-01
8 to 24		7.69E-01	0.010		7.69E-03
24 to 96		3.71E-01	0.008		2.97E-03
96 to 720		8.70E-01	0.004		3.48E-03
Total	2.43E+01	2.37E+01		6.71E+00	2.31E-01
Limit				25	25

Table ESP EIS 5.11-1-18: AP1000 Radiological Consequences Dose for a Fuel Handling Accident

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	5.20E+00		0.176	9.15E-01	
0 to 8		3.44E+00	0.005		1.72E-02
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	5.20E+00	3.44E+00		9.15E-01	1.72E-02
Limit				6.3	6.3

Table ESP EIS 5.11-1-19: U.S. EPR Radiological Consequences Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	1.00E-03	1.41E-04	0.141
LPZ	0 to 8	1.35E-04	2.30E-06	0.017
	8 to 24	1.00E-04	1.61E-06	0.016
	24 to 96	5.40E-05	7.51E-07	0.014
	96 to 720	2.20E-05	3.05E-07	0.014

Table ESP EIS 5.11-1-20: U.S. EPR Radiological Consequences Dose Summary

	DCD Dose (rem TEDE)		χ/Q ratio (Site/DCD)		Site Dose (rem TEDE)		
Accident	**EAB**	**LPZ**	**EAB**	**LPZ[a]**	**EAB**	**LPZ**	**Limit**
Main Steam Line Break - Pre-Existing Iodine Spike	0.2	0.1	0.141	0.017	2.82E-02	1.70E-03	25
Main Steam Line Break - Accident-Initiated Iodine Spike	0.3	0.2	0.141	0.017	4.23E-02	3.40E-03	2.5
Main Steam Line Break - Fuel Rod Clad Failure	5.3	2.6	0.141	0.017	7.47E-01	4.42E-02	25
Main Steam Line Break - Fuel Overheat	5.8	2.8	0.141	0.017	8.18E-01	4.76E-02	25
Reactor Coolant Pump Shaft Seizure	2.3	0.9	0.141	0.017	3.24E-01	1.53E-02	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.7	3.5	0.141	0.017	8.04E-01	5.95E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.8	0.3	0.141	0.017	2.54E-01	5.10E-03	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	1.1	0.3	0.141	0.017	1.55E-01	5.10E-03	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.7	0.5	0.141	0.017	9.87E-02	8.50E-03	2.5
LOCA	12.2	11.1	0.141	0.017	1.72E+00	1.89E-01	25
Fuel Handling Accident	5.6	1	0.141	0.017	7.90E-01	1.70E-02	6.3

a) LPZ doses are not given in time-dependent form; therefore, the most conservative Site/DCD χ/Q ratio (from the 0 to 8 hour interval) was used.

Response to RAI No. Env-11, Question ESP EIS 5.11-3

In Reference 2, the NRC staff asked PSEG for information regarding Environmental Impacts of Postulated Accidents, as described in Chapter 7 of the Environmental Report. The specific request was:

> *rACC-01b: For ER Table 7.1-39, confirm that the DCD Dose calculated for the worst 2-hour release is correct. If it is not correct, provide dose calculations for the worst 2-hour release and provide justification for the calculations.*
>
> *As per ERSP 7.1, the NRC staff needs to complete dose calculations for the worst 2-hour release; thus, staff needs the 2-hour source term releases. Particularly, the current siting regulations require an exclusion area of such a size that an individual located for any 2-hour period at the exclusion area boundary would receive a dose that would not be in excess of 0.25 sievert (25 rem) total effective dose equivalent (TEDE).*

PSEG Response to NRC RAI:

Environmental Report (ER) Table 7.1-39 provides the radiological consequence dose summary for the US-APWR. The EAB doses in ER Table 7.1-39 are the worst 2-hour doses. The worst 2-hour dose is due to a Loss of Coolant Accident (LOCA). The exclusion area boundary (EAB) doses in ER Table 7.1-39 are based on input from Table 15.0-17 of the US-APWR Design Control Document (DCD, Revision 1). Per Table 15.0-17 of the US-APWR DCD, the dose associated with a LOCA is 13 rem TEDE at the Exclusion Area Boundary (EAB). The 2-hour release data associated with the worst 2-hour dose is provided in response to RAI No. Env-11, Question No. 5.11-2.

Associated PSEG Site ESP Application Revisions:

None.

Response to RAI No. Env-11, Question ESP EIS 5.11-4

In Reference 2, the NRC staff asked PSEG for information regarding Environmental Impacts of Postulated Accidents, as described in Chapter 7 of the Environmental Report. The specific request was:

> *rACC-01c: Provide a cross-reference table that links the tables in the PSEG ER with the appropriate tables in the DCD.*
>
> *Under 10 CFR 51.41, information that may be useful in aiding the NRC in complying with section 102(2) of NEPA may be requested of the applicant.*
>
> *As discussed during the Environmental Site Audit, the requested table would assist the NRC staff to confirm the proper DCD tables were used to prepare the ER tables.*

PSEG Response to NRC RAI:

Table ESP EIS 5.11-4-1 provides cross-references between the tables in Environmental Report (ER) Section 7.1 and the corresponding input documents. The input documents are publicly available and include the associated Design Control Documents (DCDs).

The following document revisions were used for the information in Chapter 7 of the PSEG Site Environmental Report. DCD revision numbers are documented in the reference section, i.e., Section 7.1.5 of the PSEG Site Environmental Report.

- US-APWR DCD, Revision 1
- ABWR DCD, Revision 4
- AP1000 DCD, Revision 17
- CCNPP COLA, Revision 6
- Vogtle ESPA SSAR, Revision 5

Associated PSEG Site ESP Application Revisions:

None.

Table ESP EIS 5.11-4-1-1
Cross-Reference Table for ER Section 7.1

ER Table Number	ER Table Title	Basis for Input Data in the ER Table
7.1-1	US-APWR Design Basis Accident List	N/A
7.1-2	ABWR Design Basis Accident List	N/A
7.1-3	AP1000 Design Basis Accident List	N/A
7.1-4	U.S. EPR Design Basis Accident List	N/A
7.1-5	US-APWR Source Terms – Time Dependent Released Activity during LOCA	US-APWR DCD Table 15A-24
7.1-6	US-APWR Source Terms – Time Dependent Released Activity during Steam System Piping Failure (Transient-Initiated Iodine Spike)	US-APWR DCD Table 15A-25
7.1-7	US-APWR Source Terms – Time Dependent Released Activity during Steam System Piping Failure (Pre-Transient Iodine Spike)	US-APWR DCD Table 15A-26
7.1-8	US-APWR Source Terms – Time Dependent Released Activity during Steam Generator Tube Rupture (Transient-Initiated Iodine Spike)	US-APWR DCD Table 15A-27
7.1-9	US-APWR Source Terms – Time Dependent Released Activity during Steam Generator Tube Rupture (Pre-Transient Iodine Spike)	US-APWR DCD Table 15A-28
7.1-10	US-APWR Source Terms – Time Dependent Released Activity during RCP Rotor Seizure	US-APWR DCD Table 15A-29
7.1-11	US-APWR Source Terms – Time Dependent Released Activity during Rod Ejection Accident	US-APWR DCD Table 15A-30
7.1-12	US-APWR Source Terms – Time Dependent Released Activity during Fuel Handling Accident	US-APWR DCD Table 15A-31
7.1-13	US-APWR Source Terms – Time Dependent Released Activity during Failure of Small Lines Carrying Primary Coolant Outside Containment	US-APWR DCD Table 15A-32
7.1-14	ABWR Source Terms – Iodine Activity Release to the Environment during a LOCA	ABWR DCD Table 15.6-10[a]
7.1-15	ABWR Source Terms – Noble Gas Activity Release to the Environment during a LOCA	ABWR DCD Table 15.6-12[a]
7.1-16	ABWR Source Terms – Activity Released to the Environment during a Main Steamline Break Accident	ABWR DCD Table 15.6-6

Table ESP EIS 5.11-4-1-1
Cross-Reference Table for ER Section 7.1

ER Table Number	ER Table Title	Basis for Input Data in the ER Table
7.1-17	ABWR Source Terms – Isotopic Releases during an Instrument Line Break Accident	ABWR DCD Table 15.6-2
7.1-18	ABWR Source Terms – Isotopic Release to Environment during a Fuel Handling Accident	ABWR DCD Table 15.7-10[(a)]
7.1-19	AP1000 Source Terms – Activity Releases for Steam System Piping Failure with Pre-Existing Iodine Spike	Vogtle ESPA SSAR Table 15-2
7.1-20	AP1000 Source Terms – Activity Releases for Steam System Piping Failure with Accident-Initiated Iodine Spike	Vogtle ESPA SSAR Table 15-3
7.1-21	AP1000 Source Terms – Activity Releases for Reactor Coolant Pump Shaft Seizure	Vogtle ESPA SSAR Table 15-4
7.1-22	AP1000 Source Terms – Activity Releases for Spectrum of Rod Cluster Control Assembly Ejection Accidents	Vogtle ESPA SSAR Table 15-5
7.1-23	AP1000 Source Terms – Activity Releases for Failure of Small Lines Carrying Primary Coolant Outside Containment	Vogtle ESPA SSAR Table 15-6
7.1-24	AP1000 Source Terms – Activity Releases for Steam Generator Tube Rupture with Pre-Existing Iodine Spike	Vogtle ESPA SSAR Table 15-7
7.1-25	AP1000 Source Terms – Activity Releases for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike	Vogtle ESPA SSAR Table 15-8
7.1-26	AP1000 Source Terms – Activity Releases for LOCA Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Vogtle ESPA SSAR Table 15-9
7.1-27	AP1000 Source Terms – Activity Releases for Fuel Handling Accident	Vogtle ESPA SSAR Table 15-10
7.1-28	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Main Steamline Break with Pre-Accident Iodine Spike	CCNPP COLA ER Table 7.1-14
7.1-29	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Main Steamline Break with Accident-Induced (Coincident) Iodine Spike	CCNPP COLA ER Table 7.1-15

Table ESP EIS 5.11-4-1-1
Cross-Reference Table for ER Section 7.1

ER Table Number	ER Table Title	Basis for Input Data in the ER Table
7.1-30A	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Main Steamline Break with Accident-Induced 3.3% Clad Failure	CCNPP COLA ER Table 7.1-16 (Pages 1 and 2 of 4)
7.1-30B	U.S. EPR Source Terms - Radionuclide Releases to Atmosphere for Main Steam Line Break with Accident-Induced 0.58% Fuel Overheat	CCNPP COLA ER Table 7.1-16 (Pages 3 and 4 of 4)
7.1-31	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Pump Locked Rotor Accident (LRA) with Accident-Induced 9.5% Clad Failure	CCNPP COLA ER Table 7.1-17
7.1-32	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Design Basis Small Line Break	CCNPP COLA ER Table 7.1-18
7.1-33	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Steam Generator Tube Rupture with Pre-Accident Spike	CCNPP COLA ER Table 7.1-19
7.1-34	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Steam Generator Tube Rupture with Accident-Induced (Coincident) Iodine Spike	CCNPP COLA ER Table 7.1-20
7.1-35	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Design Basis LOCA	CCNPP COLA ER Table 7.1-21
7.1-36	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Fuel Handling Accident	CCNPP COLA ER Table 7.1-22
7.1-37	U.S. EPR Source Terms – Radionuclide Releases to Atmosphere for Rod Ejection Accident (REA) with Accident-Induced 36.7% Clad Failure	CCNPP COLA ER Table 7.1-23
7.1-38	US-APWR Radiological Consequences – Atmospheric Dispersion Factors	US-APWR DCD Table 2.0-1
7.1-39	US-APWR Radiological Consequences – Dose Summary	US-APWR DCD Table 15.0-17
7.1-40	ABWR Radiological Consequences – Atmospheric Dispersion Factors	ABWR DCD Tables 2.0-1 and 15.6-13
7.1-41	ABWR Radiological Consequences – PSEG Site-Specific Dose Summary	N/A, however, note (b) in Table 7.1-41 is from ABWR DCD Subsection 15.6.4.5.1.1
7.1-42	ABWR Radiological Consequences – Doses for an Instrument Line Break Accident	ABWR DCD Table 15.6-3

Table ESP EIS 5.11-4-1-1
Cross-Reference Table for ER Section 7.1

ER Table Number	ER Table Title	Basis for Input Data in the ER Table
7.1-43	ABWR Radiological Consequences – Doses for a Fuel Handling Accident	ABWR DCD Table 15.7-11
7.1-44	ABWR Radiological Consequences – Doses for a LOCA	ABWR DCD Table 15.6-13
7.1-45	ABWR Radiological Consequences – Doses for a Main Steamline Break	ABWR DCD Table 15.6-7
7.1-46	AP1000 Radiological Consequences – Atmospheric Dispersion Factors	Vendor letter (ADAMS Accession Number ML070850489)
7.1-47	AP1000 Radiological Consequences – PSEG Site-Specific Dose Summary	N/A
7.1-48	AP1000 Radiological Consequences – Doses for a Steam System Piping Failure	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.1.5.4.6
7.1-49	AP1000 Radiological Consequences – Doses for a Reactor Coolant Pump Shaft Seizure Accident	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.3.3.3.6
7.1-50	AP1000 Radiological Consequences – Doses for Spectrum of Rod Cluster Control Assembly Ejection Accidents	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.4.8.3.6
7.1-51	AP1000 Radiological Consequences – Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.6.2.6
7.1-52	AP1000 Radiological Consequences – Doses for Steam Generator Tube Rupture	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.6.3.3.6
7.1-53	AP1000 Radiological Consequences – Doses for LOCA	Vendor letter (ADAMS Accession Number ML070850489)
7.1-54	AP1000 Radiological Consequences – Doses for a Fuel Handling Accident	Vendor letter (ADAMS Accession Number ML070850489) and AP1000 DCD Subsection 15.7.4.5
7.1-55	U.S. EPR Radiological Consequences – Atmospheric Dispersion Factors	U.S. EPR FSAR Table 2.1-1
7.1-56	U.S. EPR Radiological Consequences – Dose Summary	U.S. EPR FSAR Table 15.0-12
a) The ABWR DCD source terms correspond to 4005 MWt. For the ABWR at the PSEG Site the analysis is done for 4386 MWt. The values from the DCD are multiplied by the factor of 1.095 (4386/4005=1.095) to reflect the increased power.		

Response to RAI No. Env-11, Question ESP EIS 5.11-5

In Reference 2, the NRC staff asked PSEG for information regarding Environmental Impacts of Postulated Accidents, as described in Chapter 7 of the Environmental Report. The specific request was:

> *rACC-02: Revise the list of reservoirs to include two reservoirs in Salem County (Laurel and Elkinton Pond) that do not appear to be considered in the list.*
>
> *As per ESRP 7.2 the NRC staff must confirm the potential consequences of a liquid-pathway release as presented in NUREG-0440 (NRC 1978) and NUREG-1437 (1996).*
>
> *A significant portion (about 1/3) of Salem County's drinking water is obtained from water reservoirs.*

PSEG Response to NRC RAI:

The list of surface waters in Environmental Report (ER) Subsection 7.2.2.2 will be updated to include the Laurel Lake and the Elkinton Millpond. The MACCS2 analysis discussed in the subsection includes these two surface waters even though they are not listed in the ER subsection.

Associated PSEG Site ESP Application Revisions:

ER Subsection 7.2.2.2 will be updated as specified in Enclosure 2 of this document.

Response to RAI No. Env-11, Question ESP EIS 5.11-6

In Reference 2, the NRC staff asked PSEG for information regarding Environmental Impacts of Postulated Accidents, as described in Chapter 7 of the Environmental Report. The specific request was:

> *rACC-03: Provide the input and output MACCS2 files used for the severe accident calculations and the calculation package 2009-11222 that describes the input to the calculations.*
>
> *As per ESRP 7.2 the NRC staff must check the MACCS calculations input and output results and the calculation package. The environmental consequences of severe accidents are estimated using acceptable methodology (such as the MACCS code package; Chanin et al. [1990]).*

PSEG Response to NRC RAI:

Question No. ESP EIS 5.11-6 requests the MACCS2 input and output files and Calculation 2009-11222, Revision 1, "Environmental Consequence Analysis for PSEG ESPA". The MACCS2 files and the requested calculation contain information proprietary to individual reactor vendors in the form of severe accident scenarios and inventories. PSEG is working with the reactor vendors to determine what portions of this information can be released publicly. A supplemental response to Question No. ESP EIS 5.11-6 will be provided by November 2, 2012.

Associated PSEG Site ESP Application Revisions:

None.

PSEG Letter ND-2012-0072, dated October 19, 2012

ENCLOSURE 2

Proposed Revisions
Part 3 – Environmental Report (ER)
Chapter 7 – Environmental Impacts of Postulated Accidents Involving Radioactive Materials

<u>Marked-up Pages</u>

LIST OF TABLES (CONTINUED)

RAI No. ENV-11,
Question 5.11-1

LIST OF TABLES (CONTINUED)

RAI No. ENV-11, Question 5.11-1

Accident

PSEG Site-Specific

PSEG Site-Specific

CHAPTER 7

RAI No. ENV-11,
Question 5.11-1

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter evaluates the environmental impacts of postulated accidents involving radioactive materials. Section 7.1 discusses design basis accidents (DBAs). Section 7.2 discusses the impacts of severe accidents. Section 7.3 discusses severe accident mitigation alternatives. Section 7.4 discusses transportation accidents.

7.1 DESIGN BASIS ACCIDENTS

Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

PSEG is considering constructing a new plant at the PSEG Site. The designs under consideration include an Advanced Boiling Water Reactor (ABWR), an Advanced Passive 1000 Reactor (AP1000) (dual unit), a U.S. Evolutionary Power Reactor (U.S. EPR), or a U.S. Advanced Pressurized Water Reactor (US-APWR). All of these designs are light water reactors (LWR). This section evaluates the radiological consequences of DBAs for the four reactor technologies.

7.1.1 SELECTION OF DESIGN BASIS ACCIDENTS

NUREG-1555, *Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan,* Section 7.1 Appendix A states that all DBAs having the potential to release activity to the environment must be identified. Due to differences in reactor technologies, not all accidents identified in NUREG-1555 apply to each reactor design. Tables 7.1-1 through 7.1-4 provide lists of applicable accidents corresponding to the different reactor technologies.

7.1.2 EVALUATION METHODOLOGY

Doses for selected accidents involving possible fission product release are evaluated at the exclusion area boundary (EAB) and at the outer boundary of the low population zone (LPZ) to demonstrate the new plant's capabilities to mitigate the radiological consequences of an accident. Although the emergency safeguard features are expected to prevent core damage and mitigate the radioactivity release, the bounding Loss of Coolant Accident (LOCA) analysis presumes substantial core damage with fission product release. Other DBAs of lesser magnitude, but greater frequencies of occurrence, are not expected to approach the 10 CFR 50.34, *Contents of Applications; Technical Information,* or 10 CFR 100, *Reactor Site Criteria,* limits as closely as a LOCA. For these accidents, the more restrictive dose limits in Regulatory Guide (RG) 1.183, ~~*Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes*~~ Revision 0, 2000, and NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,* are invoked to determine if the accidents are acceptable from an overall risk perspective. Accident doses to an individual are evaluated at any point on the EAB and at any point on the outer boundary of the LPZ to meet limits specified in 10 CFR 50.34 and 10 CFR 100. Radiological consequences related to control room personnel are evaluated as part of the combined license (COL) review.

The dose to an individual located on the EAB or the outer boundary of the LPZ is calculated based on the amount of activity released to the environment through multiple pathways, the

RAI No. ENV-11,
Question 5.11-1

atmospheric dispersion of the activity during transport from the release point to the dose point, the breathing rate of the individual at the dose point location and the activity-to-dose conversion factors. The atmospheric dispersion factor (χ/Q) is the only site-specific parameter required for determining the dose to an individual. The Design Certification Documents (DCDs) have developed χ/Q values that are not expected to be exceeded at most reactor sites. For this evaluation, the accident doses at the EAB and the outer boundary of the LPZ are calculated using the ratio of the site-specific and design certified χ/Q values for each respective reactor technology and then compared to the acceptance criteria in RG 1.183 and NUREG-0800. Site-specific χ/Q values are based on on-site meteorology and described in Site Safety Analysis Report (SSAR) Section 2.3. Site-specific short-term directional dependent χ/Q values are calculated for the PSEG Site using on-site meteorological data and the RG 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,* Revision 1, 1983, methodology.

50th percentile

The accident dose evaluations are performed using χ/Qs and activity releases for the following intervals.

EAB	LPZ
0 to 2 hr.	0 to 8 hr.
	8 to 24 hr.
	24 to 96 hr.
	96 to 720 hr.

The zero to two hour χ/Q value is used for the two hour release duration with the greatest dose consequence at the EAB. Accident doses for the ABWR are expressed as whole body and thyroid doses consistent with 10 CFR 100. Accident doses for the other reactor technologies evaluated are expressed in total effective dose equivalent (TEDE) consistent with 10 CFR 50.34.

Note that SSAR Chapter 15 uses conservative assumptions to perform bounding safety analyses. One such assumption is the use of the 95th percentile χ/Q values. These analyses overstate the environmental impact of the DBAs. Consistent with NUREG-1555, this section uses 50th percentile χ/Q values that ~~correspond to the annual average meteorology, and~~ better reflect probable accident conditions.

7.1.3 SOURCE TERMS

Dose estimates are calculated using time-dependent activities released to the environment for each DBA. The activities are based on the analyses used to support the reactor standard safety analysis reports submitted with the DCD. Each reactor technology uses different source terms and approaches in defining the activity releases.

The US-APWR source terms are calculated using the guidance in NUREG-0800 and RG 1.183. US-APWR source terms are listed in Tables 7.1-5 through 7.1-13, and are obtained from the US-APWR DCD (Reference 7.1-3). LOCA activity releases are calculated for a reactor power level of 4555 megawatts thermal (MWt) (102 percent of rated NSSS power of 4466 MWt). Activity releases for other accidents are calculated for a reactor power level equal to or less than that of the LOCA.

RAI No. ENV-11, Question 5.11-1

The ABWR source terms are calculated using the guidance in RG 1.3, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*, Revision 2, 1974; RG 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, Revision 0, 1972; and TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites*, 1962. The ABWR DCD source terms are given for a reactor power level of 4005 MWt. An uprated, 4300 MWt version of the ABWR is being considered for the PSEG Site. Source terms are calculated for a reactor power level of 4386 MWt (102 percent of the uprated 4300 MWt) by multiplying the source terms in the ABWR DCD (Reference 7.1-2) by a factor of 1.095 (4386/4005), because activity releases scale directly with power. This approach is used for accidents that involve postulated fuel damage (LOCA and fuel handling accidents). The source terms for the ABWR are listed in Tables 7.1-14 through 7.1-18, and are obtained from the ABWR DCD (Reference 7.1-2).

The AP1000 source terms and approaches to assessing accidents are based on the Alternative Source Term (AST) methods as described in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants,* 1995, and are in accordance with RG 1.183. Activity releases are calculated at a power level of 3468 MWt (102 percent of rated core power of 3400 MWt). The source terms for the AP1000 are listed in Tables 7.1-19 through 7.1-27.

The U.S. EPR source terms and approaches to assessing accidents are calculated in accordance with NUREG-0800 and RG 1.183. Activity releases are calculated for a reactor power level of 4612 MWt (4590 MWt rated core power + 22 MWt heat balance measurement uncertainty). The source terms for the U.S. EPR are listed in Tables 7.1-28 through 7.1-37.

7.1.4 DOSE CONSEQUENCES

PSEG Site-specific radiation doses at EAB and LPZ are calculated for the applicable postulated DBAs for the four reactor technologies. These PSEG Site-specific doses are calculated by multiplying the reactor DCD dose by the ratio of the site ~~annual average~~ χ/Q value to the DCD χ/Q value. All PSEG Site-specific ~~annual average~~ χ/Q values are bounded by the DCD χ/Q values, therefore all site-specific doses are bounded by DCD doses. The site-specific analysis results demonstrate that all US-APWR, AP1000, and U.S. EPR accident doses meet the site acceptance criteria of 10 CFR 50.34. The results also demonstrate that all ABWR accident doses meet the site acceptance criteria of 10 CFR 100.

50th percentile

The ABWR DCD doses are calculated for a reactor power level of 4005 MWt. An uprated, 4300 MWt version of the ABWR is being considered at the PSEG Site. The power uprate only affects doses of accidents that involve fuel damage (LOCA and fuel handling accidents). Doses for these two accidents are calculated for a reactor power level of 4386 MWt (102 percent of the uprated 4300 MWt) by multiplying the site-specific doses by a factor of 1.095 (4386/4005), since activity releases and thus doses are proportional to power. Reactor technology data table locations are listed below:

Table 7.1-38
US-APWR Radiological Consequences
Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	5.00E-04	~~6.71E-06~~	~~0.013~~
LPZ	0 to 8	2.10E-04	~~1.08E-07~~	~~0.001~~
	8 to 24	1.30E-04	~~1.08E-07~~	~~0.001~~
	24 to 96	6.90E-05	~~1.08E-07~~	~~0.002~~
	96 to 720	2.80E-05	~~1.08E-07~~	~~0.004~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 1

Table 7.1-39
US-APWR Radiological Consequences
Dose Summary

Accident	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q ratio (Site/DCD) EAB	χ/Q ratio (Site/DCD) LPZ[(a)]	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ	Limit
Steam System Piping Failure - Pre-Existing Iodine Spike	0.19	0.11	~~0.013~~	~~0.004~~	~~0.00~~	~~0.00~~	25
Steam System Piping Failure - Accident-Initiated Iodine Spike	0.32	0.28	~~0.013~~	~~0.004~~	~~0.00~~	~~0.00~~	2.5
Reactor Coolant Pump Rotor Seizure	0.49	0.7	~~0.013~~	~~0.004~~	~~0.01~~	~~0.00~~	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.1	4.5	~~0.013~~	~~0.004~~	~~0.07~~	~~0.02~~	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.5	0.6	~~0.013~~	~~0.004~~	~~0.02~~	~~0.00~~	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	3.6	1.5	~~0.013~~	~~0.004~~	~~0.05~~	~~0.01~~	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.96	0.43	~~0.013~~	~~0.004~~	~~0.01~~	~~0.00~~	2.5
LOCA	13	13	~~0.013~~	~~0.004~~	~~0.17~~	~~0.05~~	25
Fuel Handling Accident	3.3	1.4	~~0.013~~	~~0.004~~	~~0.04~~	~~0.01~~	6.3

a) ~~As~~ LPZ doses are not given in time-dependent form, the most conservative Site/DCD χ/Q ratio (from the ~~96 to 720~~ hour interval) is used.

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 2

Rev. 1

Table 7.1-40
ABWR Radiological Consequences
Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
All Accidents	EAB	0 to 2	1.37E-03	~~6.71E-06~~	~~0.005~~
	LPZ	0 to 2	4.11E-04	~~1.08E-07~~	~~0.000~~
LOCA Only		0 to 8	1.56E-04	~~1.08E-07~~	~~0.001~~
		8 to 24	9.61E-05	~~1.08E-07~~	~~0.001~~
		24 to 96	3.36E-05	~~1.08E-07~~	~~0.003~~
		96 to 720	7.42E-06	~~1.08E-07~~	~~0.015~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 3

Table 7.1-41
ABWR Radiological Consequences
PSEG Site-Specific Dose Summary

Accident	Thyroid Dose (Sv)	Whole Body Dose (Sv)	Thyroid Limit (Sv)	Whole Body Limit (Sv)
Failure of Small Lines Carrying Primary Coolant Outside Containment[(a)]	~~2.35E-04~~	~~4.60E-06~~	3.00E-01	2.50E-02
LOCA - EAB	~~1.02E-02~~	~~2.20E-04~~	3.00E+00	2.50E-01
LOCA - LPZ	~~2.08E-02~~	~~2.00E-04~~	3.00E+00	2.50E-01
Fuel Handling Accident[(a)]	~~4.02E-03~~	~~6.44E-05~~	7.50E-01	6.25E-02
Main Steamline Break Case 1[(a)(b)]	~~1.27E-04~~	~~3.04E-06~~	3.00E-01	2.50E-02
Main Steamline Break Case 2[(a)(b)]	~~2.50E-03~~	~~6.37E-05~~	3.00E+00	2.50E-01

a) The dose is calculated for the maximum two hour EAB meteorology, only, based on the DCD.

b) The level of activity is consistent with an offgas release rate of 3.7 GBq/s for Case 1 and 14.8 GBq/s for Case 2 referenced to a 30 minute decay. The iodine concentrations in the reactor coolant are tabulated below for each case.

Isotope	MBq/g Case 1	MBq/g Case 2
I-131	0.001739	0.03515
I-132	0.01536	0.30747
I-133	0.01206	0.24161
I-134	0.02634	0.52688
I-135	0.01647	0.3293

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 4

Table 7.1-42
ABWR Radiological Consequences
Doses for an Instrument Line Break Accident

DCD			Site	
Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
4.80E-02	9.40E-04	~~0.005~~	~~2.35E-04~~	~~4.60E-06~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 5

Table 7.1-43
ABWR Radiological Consequences
Doses for a Fuel Handling Accident

DCD				Site	
Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Uprate Ratio	Thyroid Dose (Sv)	Whole Body Dose (Sv)
7.50E-01	1.20E-02	~~0.005~~	1.095	~~4.02E-03~~	~~6.44E-05~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 6

Table 7.1-44
ABWR Radiological Consequences
Doses for a LOCA

		DCD				Site	
Location	**Time (hr.)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**	**χ/Q Ratio (Site/DCD)**	**Uprate Ratio**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
EAB	0 to 2	1.90E+00	4.10E-02	~~0.005~~	1.095	~~1.02E-02~~	~~2.20E-04~~
LPZ	0 to 8	3.10E-01	1.00E-02	~~0.001~~	1.095	~~2.35E-04~~	~~7.58E-06~~
	0 to 24	5.10E-01	1.80E-02	~~0.001~~	1.095	~~4.81E-04~~	~~1.74E-05~~
	0 to 96	1.30E+00	2.90E-02	~~0.003~~	1.095	~~3.26E-03~~	~~5.61E-05~~
	0 to 720	2.40E+00	3.80E-02	~~0.015~~	1.095	~~2.08E-02~~	~~2.00E-04~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 7

Table 7.1-45
ABWR Radiological Consequences
Doses for a Main Steamline Break

	DCD			Site	
	Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
Case 1	2.60E-02	6.20E-04	~~0.005~~	~~1.27E-04~~	~~3.04E-06~~
Case 2	5.10E-01	1.30E-02	~~0.005~~	~~2.50E-03~~	~~6.37E-05~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 8

Table 7.1-46
AP1000 Radiological Consequences
Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
LOCA	EAB	0 to 2	5.10E-04	~~6.71E-06~~	~~0.013~~
	LPZ	0 to 8	2.20E-04	~~1.08E-07~~	~~0.000~~
		8 to 24	1.60E-04	~~1.08E-07~~	~~0.001~~
		24 to 96	1.00E-04	~~1.08E-07~~	~~0.001~~
		96 to 720	8.00E-05	~~1.08E-07~~	~~0.001~~
Other Accidents	EAB	0 to 2	8.00E-04	~~6.71E-06~~	~~0.008~~
	LPZ	0 to 8	5.00E-04	~~1.08E-07~~	~~0.000~~
		8 to 24	3.00E-04	~~1.08E-07~~	~~0.000~~
		24 to 96	1.50E-04	~~1.08E-07~~	~~0.001~~
		96 to 720	8.00E-05	~~1.08E-07~~	~~0.001~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 9

Table 7.1-47
AP1000 Radiological Consequences
PSEG Site-Specific Dose Summary

	Site Dose (rem TEDE)		
Accident	**EAB**	**LPZ**	**Limit**
Steam System Piping Failure – Pre-Existing Iodine Spike	~~0.01~~	~~0.00~~	25
Steam System Piping Failure – Accident-Initiated Iodine Spike	~~0.01~~	~~0.00~~	2.5
Reactor Coolant Pump Shaft Seizure – No Feedwater	~~0.01~~	~~0.00~~	2.5
Reactor Coolant Pump Shaft Seizure – Feedwater Available	~~0.01~~	~~0.00~~	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	~~0.03~~	~~0.00~~	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	~~0.02~~	~~0.00~~	2.5
Steam Generator Tube Rupture – Pre-Existing Iodine Spike	~~0.02~~	~~0.00~~	25
Steam Generator Tube Rupture – Accident-Initiated Iodine Spike	~~0.01~~	~~0.00~~	2.5
LOCA	~~0.32~~	~~0.01~~	25
Fuel Handling Accident	~~0.04~~	~~0.00~~	6.3

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 10

Table 7.1-48
AP1000 Radiological Consequences
Doses for a Steam System Piping Failure

Doses for Steam System Piping Failure with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.00E+00		~~0.008~~	~~8.39E-03~~	
0 to 8		5.81E-01	~~0.000~~		~~1.25E-04~~
8 to 24		7.18E-02	~~0.000~~		~~2.58E-05~~
24 to 96		1.08E-01	~~0.001~~		~~7.78E-05~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	1.00E+00	7.61E-01		~~8.39E-03~~	~~2.20E-04~~
Limit				25	25

Doses for Steam System Piping Failure with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.10E+00		~~0.008~~	~~9.23E-03~~	
0 to 8		1.02E+00	~~0.000~~		~~2.20E-04~~
8 to 24		3.77E-01	~~0.000~~		~~1.36E-04~~
24 to 96		5.36E-01	~~0.001~~		~~3.86E-04~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	1.10E+00	1.93E+00		~~9.23E-03~~	~~7.42E-04~~
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 11

Table 7.1-49
AP1000 Radiological Consequences
Doses for a Reactor Coolant Pump Shaft Seizure Accident

Doses for Reactor Coolant Pump Shaft Seizure with No Feedwater

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	8.00E-01		~~0.008~~	~~6.71E-03~~	
0 to 8		3.89E-01	~~0.000~~		~~8.40E-05~~
8 to 24		0.00E+00	~~0.000~~		~~0.00E+00~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	8.00E-01	3.89E-01		~~6.71E-03~~	~~8.40E-05~~
Limit				2.5	2.5

Doses for Reactor Coolant Pump Shaft Seizure with Feedwater Available

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	6.00E-01		~~0.008~~	~~5.03E-03~~	
0 to 8		7.94E-01	~~0.000~~		~~1.72E-04~~
8 to 24		0.00E+00	~~0.000~~		~~0.00E+00~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	6.00E-01	7.94E-01		~~5.03E-03~~	~~1.72E-04~~
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 12

Table 7.1-50
AP1000 Radiological Consequences
Doses for Spectrum of Rod Cluster Control Assembly Ejection Accidents

Time (hr.)	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0 to 2	3.60E+00		~~0.008~~	~~3.02E-02~~	
0 to 8		4.58E+00	~~0.000~~		~~9.89E-04~~
8 to 24		7.84E-01	~~0.000~~		~~2.82E-04~~
24 to 96		6.32E-02	~~0.001~~		~~4.55E-05~~
96 to 720		2.06E-02	~~0.001~~		~~2.78E-05~~
Total	3.60E+00	5.45E+00		~~3.02E-02~~	~~1.34E-03~~
Limit				6.3	6.3

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 13

Table 7.1-51
AP1000 Radiological Consequences
Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.10E+00		~~0.008~~	~~1.76E-02~~	
0 to 8		1.02E+00	~~0.000~~		~~2.20E-04~~
8 to 24		0.00E+00	~~0.000~~		~~0.00E+00~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	2.10E+00	1.02E+00		~~1.76E-02~~	~~2.20E-04~~
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 14

Table 7.1-52
AP1000 Radiological Consequences
Doses for Steam Generator Tube Rupture

Doses for Steam Generator Tube Rupture with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.20E+00		~~0.008~~	~~1.85E-02~~	
0 to 8		1.16E+00	~~0.000~~		~~2.51E-04~~
8 to 24		7.24E-02	~~0.000~~		~~2.61E-05~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	2.20E+00	1.23E+00		~~1.85E-02~~	~~2.77E-04~~
Limit				25	25

Doses for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.10E+00		~~0.008~~	~~9.23E-03~~	
0 to 8		6.27E-01	~~0.000~~		~~1.35E-04~~
8 to 24		1.69E-01	~~0.000~~		~~6.08E-05~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	1.10E+00	7.96E-01		~~9.23E-03~~	~~1.96E-04~~
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 15

Table 7.1-53
AP1000 Radiological Consequences
Doses for LOCA

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.43E+01		~~0.013~~	~~3.20E-01~~	
0 to 8		2.17E+01	~~0.000~~		~~1.07E-02~~
8 to 24		7.69E-01	~~0.001~~		~~5.19E-04~~
24 to 96		3.71E-01	~~0.001~~		~~4.01E-04~~
96 to 720		8.70E-01	~~0.001~~		~~1.17E-03~~
Total	2.43E+01	2.37E+01		~~3.20E-01~~	~~1.27E-02~~
Limit				25	25

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 16

Table 7.1-54
AP1000 Radiological Consequences
Doses for a Fuel Handling Accident

Time (hr.)	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0 to 2	5.20E+00		~~0.008~~	~~4.36E-02~~	
0 to 8		3.44E+00	~~0.000~~		~~7.43E-04~~
8 to 24		0.00E+00	~~0.000~~		~~0.00E+00~~
24 to 96		0.00E+00	~~0.001~~		~~0.00E+00~~
96 to 720		0.00E+00	~~0.001~~		~~0.00E+00~~
Total	5.20E+00	3.44E+00		~~4.36E-02~~	~~7.43E-04~~
Limit				~~25~~	~~25~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 17

Table 7.1-55
U.S. EPR Radiological Consequences
Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m³)	Site χ/Q (sec/m³)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	1.00E-03	~~6.71E-06~~	~~0.007~~
LPZ	0 to 8	1.35E-04	~~1.08E-07~~	~~0.001~~
	8 to 24	1.00E-04	~~1.08E-07~~	~~0.001~~
	24 to 96	5.40E-05	~~1.08E-07~~	~~0.002~~
	96 to 720	2.20E-05	~~1.08E-07~~	~~0.005~~

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 18

Table 7.1-56
U.S. EPR Radiological Consequences
Dose Summary

Accident	DCD Dose (rem TEDE)		χ/Q ratio (Site/DCD)		Site Dose (rem TEDE)		
	EAB	LPZ	EAB	LPZ[(a)]	EAB	LPZ	~~Max~~
Main Steam Line Break - Pre-Existing Iodine Spike	0.2	0.1	~~0.007~~	~~0.005~~	~~1.40E-03~~	~~5.00E-04~~	25
Main Steam Line Break - Accident-Initiated Iodine Spike	0.3	0.2	~~0.007~~	~~0.005~~	~~2.10E-03~~	~~1.00E-03~~	2.5
Main Steam Line Break - Fuel Rod Clad Failure	5.3	2.6	~~0.007~~	~~0.005~~	~~3.71E-02~~	~~1.30E-02~~	25
Main Steam Line Break - Fuel Overheat	5.8	2.8	~~0.007~~	~~0.005~~	~~4.06E-02~~	~~1.40E-02~~	25
Reactor Coolant Pump Shaft Seizure	2.3	0.9	~~0.007~~	~~0.005~~	~~1.61E-02~~	~~4.50E-03~~	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.7	3.5	~~0.007~~	~~0.005~~	~~3.99E-02~~	~~1.75E-02~~	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.8	0.3	~~0.007~~	~~0.005~~	~~1.26E-02~~	~~1.50E-03~~	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	1.1	0.3	~~0.007~~	~~0.005~~	~~7.70E-03~~	~~1.50E-03~~	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.7	0.5	~~0.007~~	~~0.005~~	~~4.90E-03~~	~~2.50E-03~~	2.5
LOCA	12.2	11.1	~~0.007~~	~~0.005~~	~~8.54E-02~~	~~5.55E-02~~	25
Fuel Handling Accident	5.6	1	~~0.007~~	~~0.005~~	~~3.92E-02~~	~~5.00E-03~~	6.3

a) ~~As~~ LPZ doses are not given in time-dependent form~~,~~ the most conservative Site/DCD χ/Q ratio (from the ~~96 to 720~~ hour interval) was used.

RAI No. ENV-11,
Question 5.11-1
Replace with Insert No. 19

Table 7.1-38
US-APWR Radiological Consequences
Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	5.00E-04	1.41E-04	0.282
LPZ	0 to 8	2.10E-04	2.30E-06	0.011
	8 to 24	1.30E-04	1.61E-06	0.012
	24 to 96	6.90E-05	7.51E-07	0.011
	96 to 720	2.80E-05	3.05E-07	0.011

RAI No. ENV-11,
Question 5.11-1
Insert No. 1

Table 7.1-39
US-APWR Radiological Consequences
Dose Summary

Accident	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q ratio (Site/DCD) EAB	χ/Q ratio (Site/DCD) LPZ[(a)]	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ	Limit
Steam System Piping Failure - Pre-Existing Iodine Spike	0.19	0.11	0.282	0.012	5.36E-02	1.32E-03	25
Steam System Piping Failure - Accident-Initiated Iodine Spike	0.32	0.28	0.282	0.012	9.02E-02	3.36E-03	2.5
Reactor Coolant Pump Rotor Seizure	0.49	0.7	0.282	0.012	1.38E-01	8.40E-03	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.1	4.5	0.282	0.012	1.44E+00	5.40E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.5	0.6	0.282	0.012	4.23E-01	7.20E-03	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	3.6	1.5	0.282	0.012	1.02E+00	1.80E-02	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.96	0.43	0.282	0.012	2.71E-01	5.16E-03	2.5
LOCA	13	13	0.282	0.012	3.67E+00	1.56E-01	25
Fuel Handling Accident	3.3	1.4	0.282	0.012	9.31E-01	1.68E-02	6.3

a) LPZ doses are not given in time-dependent form; therefore, the most conservative Site/DCD χ/Q ratio (from the 8 to 24 hour interval) is used.

RAI No. ENV-11,
Question 5.11-1
Insert No. 2

Table 7.1-40
ABWR Radiological Consequences
Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
All Accidents	EAB	0 to 2	1.37E-03	1.41E-04	0.103
	LPZ	0 to 2	4.11E-04	4.72E-06	0.011
LOCA Only		0 to 8	1.56E-04	2.30E-06	0.015
		8 to 24	9.61E-05	1.61E-06	0.017
		24 to 96	3.36E-05	7.51E-07	0.022
		96 to 720	7.42E-06	3.05E-07	0.041

RAI No. ENV-11,
Question 5.11-1
Insert No. 3

Table 7.1-41
ABWR Radiological Consequences
PSEG Site-Specific Dose Summary

Accident	Thyroid Dose (Sv)	Whole Body Dose (Sv)	Thyroid Limit (Sv)	Whole Body Limit (Sv)
Failure of Small Lines Carrying Primary Coolant Outside Containment[(a)]	4.94E-03	9.68E-05	3.00E-01	2.50E-02
LOCA - EAB	2.14E-01	4.62E-03	3.00E+00	2.50E-01
LOCA - LPZ	7.72E-02	9.82E-04	3.00E+00	2.50E-01
Fuel Handling Accident[(a)]	8.46E-02	1.35E-03	7.50E-01	6.25E-02
Main Steamline Break Case 1[(a)(b)]	2.68E-03	6.39E-05	3.00E-01	2.50E-02
Main Steamline Break Case 2[(a)(b)]	5.25E-02	1.34E-03	3.00E+00	2.50E-01

a) The dose is calculated for the maximum two hour EAB meteorology, only, based on the DCD.

b) The level of activity is consistent with an offgas release rate of 3.7 GBq/s for Case 1 and 14.8 GBq/s for Case 2 referenced to a 30 minute decay. The iodine concentrations in the reactor coolant are tabulated below for each case.

Isotope	MBq/g Case 1	MBq/g Case 2
I-131	0.001739	0.03515
I-132	0.01536	0.30747
I-133	0.01206	0.24161
I-134	0.02634	0.52688
I-135	0.01647	0.3293

RAI No. ENV-11,
Question 5.11-1
Insert No. 4

Table 7.1-42
ABWR Radiological Consequences
Doses for an Instrument Line Break Accident

DCD			Site	
Thyroid Dose (Sv)	**Whole Body Dose (Sv)**	**χ/Q Ratio (Site/DCD)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
4.80E-02	9.40E-04	0.103	4.94E-03	9.68E-05

RAI No. ENV-11,
Question 5.11-1
Insert No. 5

Table 7.1-43
ABWR Radiological Consequences
Doses for a Fuel Handling Accident

DCD				Site	
Thyroid Dose (Sv)	Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Uprate Ratio	Thyroid Dose (Sv)	Whole Body Dose (Sv)
7.50E-01	1.20E-02	0.103	1.095	8.46E-02	1.35E-03

RAI No. ENV-11,
Question 5.11-1
Insert No. 6

Table 7.1-44
ABWR Radiological Consequences
Doses for a LOCA

Location	Time (hr.)	DCD Thyroid Dose (Sv)	DCD Whole Body Dose (Sv)	χ/Q Ratio (Site/DCD)	Uprate Ratio	Site Thyroid Dose (Sv)	Site Whole Body Dose (Sv)
EAB	0 to 2	1.90E+00	4.10E-02	0.103	1.095	2.14E-01	4.62E-03
LPZ	0 to 8	3.10E-01	1.00E-02	0.015	1.095	5.09E-03	1.64E-04
	0 to 24	5.10E-01	1.80E-02	0.017	1.095	8.81E-03	3.13E-04
	0 to 96	1.30E+00	2.90E-02	0.022	1.095	2.78E-02	5.78E-04
	0 to 720	2.40E+00	3.80E-02	0.041	1.095	7.72E-02	9.82E-04

RAI No. ENV-11,
Question 5.11-1
Insert No. 7

Table 7.1-45
ABWR Radiological Consequences
Doses for a Main Steamline Break

	DCD			Site	
	Thyroid Dose (Sv)	**Whole Body Dose (Sv)**	**χ/Q Ratio (Site/DCD)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
Case 1	2.60E-02	6.20E-04	0.103	2.68E-03	6.39E-05
Case 2	5.10E-01	1.30E-02	0.103	5.25E-02	1.34E-03

RAI No. ENV-11,
Question 5.11-1
Insert No. 8

Table 7.1-46
AP1000 Radiological Consequences
Atmospheric Dispersion Factors

Accident	Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
LOCA	EAB	0 to 2	5.10E-04	1.41E-04	0.276
	LPZ	0 to 8	2.20E-04	2.30E-06	0.010
		8 to 24	1.60E-04	1.61E-06	0.010
		24 to 96	1.00E-04	7.51E-07	0.008
		96 to 720	8.00E-05	3.05E-07	0.004
Other Accidents	EAB	0 to 2	8.00E-04	1.41E-04	0.176
	LPZ	0 to 8	5.00E-04	2.30E-06	0.005
		8 to 24	3.00E-04	1.61E-06	0.005
		24 to 96	1.50E-04	7.51E-07	0.005
		96 to 720	8.00E-05	3.05E-07	0.004

RAI No. ENV-11,
Question 5.11-1
Insert No. 9

Table 7.1-47
AP1000 Radiological Consequences
PSEG Site-Specific Dose Summary

Accident	Site Dose (rem TEDE)		
	EAB	LPZ	Limit
Steam System Piping Failure – Pre-Existing Iodine Spike	1.76E-01	3.81E-03	25
Steam System Piping Failure – Accident-Initiated Iodine Spike	1.94E-01	9.67E-03	2.5
Reactor Coolant Pump Shaft Seizure – No Feedwater	1.41E-01	1.95E-03	2.5
Reactor Coolant Pump Shaft Seizure – Feedwater Available	1.06E-01	3.97E-03	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	6.34E-01	2.72E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	3.70E-01	5.10E-03	2.5
Steam Generator Tube Rupture – Pre-Existing Iodine Spike	3.87E-01	6.16E-03	25
Steam Generator Tube Rupture – Accident-Initiated Iodine Spike	1.94E-01	3.99E-03	2.5
LOCA	6.71E+00	2.31E-01	25
Fuel Handling Accident	9.15E-01	1.72E-02	6.3

RAI No. ENV-11,
Question 5.11-1
Insert No. 10

Table 7.1-48
AP1000 Radiological Consequences
Doses for a Steam System Piping Failure

Doses for Steam System Piping Failure with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	1.00E+00		0.176	1.76E-01	
0 to 8		5.81E-01	0.005		2.91E-03
8 to 24		7.18E-02	0.005		3.59E-04
24 to 96		1.08E-01	0.005		5.40E-04
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.00E+00	7.61E-01		1.76E-01	3.81E-03
Limit				25	25

Doses for Steam System Piping Failure with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	EAB	LPZ	χ/Q Ratio (Site/DCD)	EAB	LPZ
0 to 2	1.10E+00		0.176	1.94E-01	
0 to 8		1.02E+00	0.005		5.10E-03
8 to 24		3.77E-01	0.005		1.89E-03
24 to 96		5.36E-01	0.005		2.68E-03
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.10E+00	1.93E+00		1.94E-01	9.67E-03
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Insert No. 11

Table 7.1-49
AP1000 Radiological Consequences
Doses for a Reactor Coolant Pump Shaft Seizure Accident

Doses for Reactor Coolant Pump Shaft Seizure with No Feedwater

Time (hr.)	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0 to 2	8.00E-01		0.176	1.41E-01	
0 to 8		3.89E-01	0.005		1.95E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	8.00E-01	3.89E-01		1.41E-01	1.95E-03
Limit				2.5	2.5

Doses for Reactor Coolant Pump Shaft Seizure with Feedwater Available

Time (hr.)	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0 to 2	6.00E-01		0.176	1.06E-01	
0 to 8		7.94E-01	0.005		3.97E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	6.00E-01	7.94E-01		1.06E-01	3.97E-03
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Insert No. 12

Table 7.1-50
AP1000 Radiological Consequences
Doses for Spectrum of Rod Cluster Control Assembly Ejection Accidents

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	3.60E+00		0.176	6.34E-01	
0 to 8		4.58E+00	0.005		2.29E-02
8 to 24		7.84E-01	0.005		3.92E-03
24 to 96		6.32E-02	0.005		3.16E-04
96 to 720		2.06E-02	0.004		8.24E-05
Total	3.60E+00	5.45E+00		6.34E-01	2.72E-02
Limit				6.3	6.3

RAI No. ENV-11,
Question 5.11-1
Insert No. 13

Table 7.1-51
AP1000 Radiological Consequences
Doses for Failure of Small Lines Carrying Primary Coolant Outside Containment

Time (hr.)	DCD Dose (rem TEDE) EAB	DCD Dose (rem TEDE) LPZ	χ/Q Ratio (Site/DCD)	Site Dose (rem TEDE) EAB	Site Dose (rem TEDE) LPZ
0 to 2	2.10E+00		0.176	3.70E-01	
0 to 8		1.02E+00	0.005		5.10E-03
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	2.10E+00	1.02E+00		3.70E-01	5.10E-03
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Insert No. 14

Table 7.1-52
AP1000 Radiological Consequences
Doses for Steam Generator Tube Rupture

Doses for Steam Generator Tube Rupture with Pre-Existing Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.20E+00		0.176	3.87E-01	
0 to 8		1.16E+00	0.005		5.80E-03
8 to 24		7.24E-02	0.005		3.62E-04
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	2.20E+00	1.23E+00		3.87E-01	6.16E-03
Limit				25	25

Doses for Steam Generator Tube Rupture with Accident-Initiated Iodine Spike

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	1.10E+00		0.176	1.94E-01	
0 to 8		6.27E-01	0.005		3.14E-03
8 to 24		1.69E-01	0.005		8.45E-04
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	1.10E+00	7.96E-01		1.94E-01	3.99E-03
Limit				2.5	2.5

RAI No. ENV-11,
Question 5.11-1
Insert No. 15

Table 7.1-53
AP1000 Radiological Consequences
Doses for LOCA

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	2.43E+01		0.276	6.71E+00	
0 to 8		2.17E+01	0.010		2.17E-01
8 to 24		7.69E-01	0.010		7.69E-03
24 to 96		3.71E-01	0.008		2.97E-03
96 to 720		8.70E-01	0.004		3.48E-03
Total	2.43E+01	2.37E+01		6.71E+00	2.31E-01
Limit				25	25

RAI No. ENV-11,
Question 5.11-1
Insert No. 16

Table 7.1-54
AP1000 Radiological Consequences
Doses for a Fuel Handling Accident

	DCD Dose (rem TEDE)			Site Dose (rem TEDE)	
Time (hr.)	**EAB**	**LPZ**	**χ/Q Ratio (Site/DCD)**	**EAB**	**LPZ**
0 to 2	5.20E+00		0.176	9.15E-01	
0 to 8		3.44E+00	0.005		1.72E-02
8 to 24		0.00E+00	0.005		0.00E+00
24 to 96		0.00E+00	0.005		0.00E+00
96 to 720		0.00E+00	0.004		0.00E+00
Total	5.20E+00	3.44E+00		9.15E-01	1.72E-02
Limit				6.3	6.3

RAI No. ENV-11,
Question 5.11-1
Insert No. 17

Table 7.1-55
U.S. EPR Radiological Consequences
Atmospheric Dispersion Factors

Location	Time (hr.)	DCD χ/Q (sec/m^3)	Site χ/Q (sec/m^3)	χ/Q Ratio (Site/DCD)
EAB	0 to 2	1.00E-03	1.41E-04	0.141
LPZ	0 to 8	1.35E-04	2.30E-06	0.017
	8 to 24	1.00E-04	1.61E-06	0.016
	24 to 96	5.40E-05	7.51E-07	0.014
	96 to 720	2.20E-05	3.05E-07	0.014

RAI No. ENV-11,
Question 5.11-1
Insert No. 18

Table 7.1-56
U.S. EPR Radiological Consequences
Dose Summary

Accident	DCD Dose (rem TEDE)		χ/Q ratio (Site/DCD)		Site Dose (rem TEDE)		
	EAB	LPZ	EAB	LPZ[a]	EAB	LPZ	Max
Main Steam Line Break - Pre-Existing Iodine Spike	0.2	0.1	0.141	0.017	2.82E-02	1.70E-03	25
Main Steam Line Break - Accident-Initiated Iodine Spike	0.3	0.2	0.141	0.017	4.23E-02	3.40E-03	2.5
Main Steam Line Break - Fuel Rod Clad Failure	5.3	2.6	0.141	0.017	7.47E-01	4.42E-02	25
Main Steam Line Break - Fuel Overheat	5.8	2.8	0.141	0.017	8.18E-01	4.76E-02	25
Reactor Coolant Pump Shaft Seizure	2.3	0.9	0.141	0.017	3.24E-01	1.53E-02	2.5
Spectrum of Rod Cluster Control Assembly Ejection Accidents	5.7	3.5	0.141	0.017	8.04E-01	5.95E-02	6.3
Failure of Small Lines Carrying Primary Coolant Outside Containment	1.8	0.3	0.141	0.017	2.54E-01	5.10E-03	2.5
Steam Generator Tube Rupture - Pre-Existing Iodine Spike	1.1	0.3	0.141	0.017	1.55E-01	5.10E-03	25
Steam Generator Tube Rupture - Accident-Initiated Iodine Spike	0.7	0.5	0.141	0.017	9.87E-02	8.50E-03	2.5
LOCA	12.2	11.1	0.141	0.017	1.72E+00	1.89E-01	25
Fuel Handling Accident	5.6	1	0.141	0.017	7.90E-01	1.70E-02	6.3

a) LPZ doses are not given in time-dependent form; therefore, the most conservative Site/DCD χ/Q ratio (from the 0 to 8 hour interval) was used.

RAI No. ENV-11,
Question 5.11-1
Insert No. 19

RAI No. ENV-11, Question 5.11-5

PRA modeling to evaluate how changes to the reactor or auxiliary systems change the severity of the accident. The CDFs for the ABWR, AP1000, US-APWR, and U.S. EPR are typically one to three orders of magnitude lower than the CDFs for the current nuclear fleet.

7.2.2.1 Air Pathways

For each reactor technology, the potential severe accidents are grouped into release categories based on their similarity. The number of release categories is reactor-specific. Each release category has a set of characteristics representative of that categories chemical elements. Radionuclides that may be released are organized into groups having similar chemical characteristics. Table 7.2-4 provides the groupings. Release categories for each reactor technology are analyzed with MACCS2 to calculate population dose, number of early and latent fatalities, cost, and farm land requiring decontamination. The analysis assumes that 95 percent of the population is evacuated following declaration of a general emergency.

For each release category, risk is calculated by multiplying each consequence (population dose, fatalities, cost, and area of contaminated land) with its corresponding frequency. A summary of the results is provided in Table 7.2-3. The total cost calculation considers other consequences, such as evacuation costs, value of crops contaminated and condemned, value of milk contaminated and condemned, cost of property decontamination, and indirect costs resulting from loss of property use and incomes as a result of the accident.

7.2.2.2 Surface Water Pathways

A population is exposed to radiation when airborne radioactivity is deposited onto surface water. The exposure pathway is from drinking the water, external radiation from submersion in the water, external radiation from activities near the shoreline, or ingestion of fish or shellfish. MACCS2 only calculates the dose from drinking water. The MACCS2 severe accident dose risk to the 50-mi. population from drinking water is 8.74E-03 person-rem/reactor-year (8.74E-05 person-Sv/reactor-year) (Table 7.2-3) for the US-APWR, which is bounding for the four reactor technologies. This value is the sum of all water ingestion doses for the associated US-APWR release categories.

Laurel Lake, Elkinton Millpond,

Surface water pathways involving swimming, fishing, and boating are not modeled by MACCS2. Surface water bodies within the 50-mi. region of PSEG Site include the Chesapeake Bay, Delaware Bay, Delaware River, Susquehanna River, Smyrna River, Schuylkill River, Cooper River and the reservoirs listed on Table 2.3-3. The tributary streams in the vicinity of the PSEG Site are listed in Table 2.3-4. The NRC evaluated doses from the aquatic food pathway (fishing) for the current nuclear fleet discharging to various bodies of water in NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants.* The NRC evaluation concluded that with interdiction, the risk associated with the aquatic food pathway is SMALL relative to the atmospheric pathway for most sites and essentially the same as the atmospheric pathway for the few sites with large annual aquatic food harvests. The new plant atmospheric pathway doses are lower than those of the current U.S. nuclear fleet, therefore, the doses from surface water sources are consistently lower for the new plant as well.