May 10, 2007

Mr. Britt T. McKinney
Sr. Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - CORRECTION TO AMENDMENT NOS. 239 AND 216 (TAC NOS. MC8730 AND MC8731)

Dear Mr. McKinney:

On January 31, 2007, the Nuclear Regulatory Commission (NRC) issued Amendment No. 239 to Facility Operating License No. NPF-14 and Amendment No. 216 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), respectively. These amendments revised the SSES 1 and 2 Technical Specifications (TSs) to incorporate a full-scope application of an alternate source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

Subsequent to the issuance, Mr. Michael Crowthers of your staff pointed out a number of errors in the safety evaluation (SE) supporting the amendment. We agree that editorial errors had been inadvertently made, resulting in several inaccurate statements in the SE. Enclosed please find the corrected pages 2, 5, 15, 17, 20, 26, 27, 28, 29, and 34 of the SE, with side bars highlighting the areas of correction.

Additionally, in the issued Amendment No. 239 for SSES 1, TS page 1.1-3 contained an editorial pagination error. Enclosed is the corrected page for Amendment No. 239.

The NRC regrets any inconvenience that these editorial errors may have caused. If there are any questions regarding this matter, please contact me at 301-415-1030.

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure:
As stated

cc w/encl: See next page

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This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, Standard Review Plan (SRP) 15.0.1, and General Design Criterion (GDC)-19. PPL has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

The following NRC requirements and guidance documents are applicable to the NRC staff's review of PPL's amendment request.

- 10 CFR Section 50.67, "Accident source term"
- 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants"
- GDC 19, "Control Room"
- RG 1.23, "Onsite Meteorological Programs," Rev. 0, February 1972
- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 2, March 1978
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0, June 2003
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Rev. 0, May 2003
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Rev. 2, July 1981
- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Rev. 1, July 1981
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000

3.0 TECHNICAL EVALUATION

3. 1 Radiological Consequences of DBAs

As stated in RG 1.183, Section 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address

dose significant isotopes at end of fuel cycle curie levels, formed the input for the RADTRAD dose evaluation code. PPL performed a calculation (EC-RADN-1135) to evaluate the 60 isotope RADTRAD source term for direct shine evaluations. The calculation includes a series of correction factors to correct for the lack of certain short-lived isotopes in the RADTRAD 60 isotope library. The evaluation concluded that corrections are necessary when calculating early post-accident shine dose rates (decay time less than 8 hours). In addition, the evaluation concluded that the degree of shielding used in the analysis can have a significant impact on the need for correction to avoid non-conservative results.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU. PPL referenced Siemens Power Corporation EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," February 1988, including Supplement, 1(P)(A) and Supplement 2(P)(A) as the basis for the NRC staff approved licensing limit for ATRIUM-10 fuel of 62,000 MWD/MTU for extended burnup design - peak fuel rod exposure and 54,000 MWD/MTU for extended burnup design - peak bundle exposure.

PPL used committed effective dose equivalent (CEDE) and effective dose equivalent (EDE) dose conversion factors (DCFs) from Federal Guidance Reports (FGR) 11 and 12 to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is therefore acceptable to the NRC staff.

3.1.1 Loss-of-Coolant Accident

The radiological consequence design basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. The accident scenario assumes the deterministic failure of the emergency core cooling system (ECCS) to provide adequate core cooling which results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analyses.

When using the AST for the evaluation of a design basis LOCA, it is assumed that the initial fission product release to the containment will last 2 minutes and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 2 minutes, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.5 hours. PPL used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

- Control Structure Elevation 729'-0" (All areas)
- Control Structure Elevation 741'-1" (All areas)
- Computer Room (Room C-202) at Elevation 698'-0"

PPL evaluated the LOCA doses for these three areas assuming that continuous occupancy as defined in RG 1.183 is required. RG 1.183, Section 4.2.6 defines continuous occupancy as follows: The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5E-04 cubic meters per second.

PPL evaluated the LOCA CR and offsite doses from the radioactive plume in PPL Calculation No. EC-RADN-1125. PPL calculated the CRHE dose due to contamination of the atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume to be approximately 3.9 rem TEDE. PPL also calculated the CRHE dose due to direct shine from the DBA-LOCA external radioactive plume released from the facility in PPL Calculation No. EC-RADN-1125 by applying shielding factors to the RADTRAD unprotected CR whole body doses. PPL conservatively assumed a constant average gamma energy of 1.0 million electron volts for the direct dose analyses. Conservatively, PPL did not take credit for the RG 1.183 occupancy factors in the calculation of the direct dose from the airborne activity in the plume outside of the CRHE. PPL modeled the control structure with a minimum concrete side wall thickness of 2.5 feet for shielding from the outside effluent cloud. The licensee calculated the magnitude of the CRHE dose from the external cloud shine as shielded by 2.5 feet of concrete as approximately 0.05 rem EDE. In addition, PPL evaluated the direct shine from the inside cloud directly over the CRHE above elevation 806'. This component is dependent on the amount of shielding between elevation 806' and the receptor within the CRHE. The maximum dose from this component is approximately 50 mrem EDE at the highest elevations of the CRHE and diminishes as subsequent floor slabs are factored into the calculation. PPL has determined that this component is negligible on all CRHE elevations below 783' - 0". Since the highest elevation evaluated for continuous occupancy is at elevation 741', this latter component of direct dose does not impact the CRHE evaluation. Therefore, the shine from the external radioactive effluent plume is calculated to be approximately 0.05 rem EDE assuming 2.5 feet of concrete shielding.

In a letter dated November 14, 2006, PPL stated that due to an input error in the units of wind speed, calculation EC-ENVR-1058, CRHE Accident Dispersion Factors (χ/Q), was revised. The results of the revised calculation (Revision 1) show that except for the time interval from 8 to 24 hours the revised dispersion factors are lower than in the original calculation. PPL has concluded that as a result of the correction in wind speed units, the calculated DBA-LOCA CRHE dose from the external cloud is reduced slightly from 0.0515 rem EDE to 0.048 rem EDE and therefore remains approximately 0.05 rem EDE.

PPL evaluated the post-accident doses for areas not continuously occupied in the control structure. These results can be used to evaluate post-accident exposures to these areas on a case-by-case basis for frequent or infrequent occupancy or access, if required post-accident. PPL has determined that, based on a conservative analysis of the direct shine from core spray piping located in the adjacent RB, certain areas along the east wall of the CRHE may require access control to maintain post LOCA doses below 5 rem TEDE. The access control would

pressure increase or change to the size of the main steam lines. Therefore, PPL asserts and the NRC staff agrees that the use of a 20% increase in mass released for the AST MSLB accident evaluation is a conservative assumption in the calculation.

3.1.2.1 Source Term

RG 1.183, Appendix D, Section 2, states that if no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by TS for the equilibrium case and for the pre-accident iodine spike case. PPL's evaluation indicates that no fuel damage is predicted as a result of an MSLB accident. Therefore, consistent with the current licensing analysis basis and RG 1.183, PPL evaluated the MSLB based on the maximum equilibrium reactor coolant dose equivalent I-131 (DEI) concentration of 0.2 uCi/gm and, in a separate analysis, evaluated the MSLB assuming a pre-accident iodine spike DEI concentration of 4.0 uCi/gm as specified in TS 3.4.7.

PPL assumed that all the activity in the liquid and steam is released to the environment as an instantaneous ground-level puff release with no credit for plateout, hold-up or dilution within any facility structures. PPL evaluated the dose to operators in the CR, using a puff release CR atmospheric dispersion factor. The NRC staff finds the use of the puff release CR atmospheric dispersion factor acceptable because of the very short duration of the MSLB release (5.5 seconds).

3.1.2.2 Transport

PPL followed the guidance as described in RG 1.183, Appendix D, Section 4 in all aspects of the transport analysis by making the following assumptions: the MSIVs close in the maximum time allowed by TS, the total mass of coolant released is the amount in the steam line and connecting lines at the time of break plus the amount that passes through the valves prior to closure and the release to atmosphere is assumed to be a ground level instantaneous release with no credit for plateout, holdup or dilution within the facility structures. As specified in RG 1.183, Appendix D, Section 4.4, PPL assumed that the iodine species released from the main steam line consists of 95% CsI as an aerosol, 4.85% elemental and 0.15% organic. Because PPL's approach follows RG 1.183, the NRC staff finds this approach acceptable.

3.1.2.3 CR ventilation assumptions for the MSLB

The CR dose for the MSLB is calculated based on the assumption that the normal ventilation configuration would persist throughout the accident sequence. In the normal mode of operation, the CR intake air is unfiltered with a variable flow rate of between 5,229 cfm and 6,391 cfm. PPL evaluated the CR dose for a range of intake air flow rates and determined that for the MSLB accident the maximum intake flow rate of 6,391 cfm resulted in the highest dose consequence. Therefore, PPL chose to use the maximum unfiltered intake flow of 6,391cfm. The analysis further assumes that an additional 510 cfm of unfiltered inleakage exists which bounds the tracer gas testing results and includes 10 cfm for ingress/egress leakage considerations.

In a letter dated November 14, 2006, PPL provided a revised atmospheric dispersion calculation for the MSLB (EC-RADN-1128, Revision 1). The atmospheric dispersion factors were revised to conservatively assume a zero differential between the release height and the height of the

analysis assuming that the CRDA occurs at low power without tripping the MVP, which results in an unfiltered release of fission products to the environment from the TB vent stack. In this release scenario there are limited opportunities for fission product removal and hold up processes and therefore this postulated accident sequence results in larger CRHE, EAB, and LPZ doses than for the full-power case.

In accordance with RG 1.183, Appendix C, Section 3.3, PPL assumed that of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers. In accordance with RG 1.183, Appendix C, Section 3.4, PPL assumed that of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment.

For the full-power operation case, PPL assumed that the turbine and condensers leak to the atmosphere as a ground level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is terminated. For the low power case, PPL assumed that the MVP continues to run for the 24-hour duration of the accident releasing activity from the condenser at a rate of 1,212% per day.

3.1.3.3 CR Ventilation Assumptions for the CRDA

The CR dose was calculated based on the assumption that the normal ventilation configuration would persist throughout the accident sequence. In the normal mode of operation, the CR intake air is unfiltered with a variable flow rate of between 5,229 cfm and 6,391 cfm. PPL evaluated the CR dose for a range of intake air flow rates and determined that for the CRDA the maximum intake flow rate of 6,391 cfm resulted in the highest dose consequence. Therefore, PPL chose to use the maximum unfiltered intake flow of 6,391cfm. The analysis further assumes that an additional 510 cfm of unfiltered inleakage exists which bounds the tracer gas testing results and includes 10 cfm for ingress/egress leakage considerations.

The CRHE calculations for the CRDA, as provided in the AST submittal dated October 13, 2005, use atmospheric dispersion factors based on the CRHE outside air intake located at the southeast corner of the RB roof. In a letter dated November 14, 2006, PPL provided an additional set of atmospheric dispersion factors based on a new CRHE outside air intake located at an elevation of 810' 3" along the south wall of the RB. A comparison of the CRHE atmospheric dispersion factors for both intake locations, shows that the values from the original submittal bound the values for the new location. Therefore, PPL did not modify the atmospheric dispersion factors and the associated dose consequence analyses for the CRDA since the results, as originally submitted, are bounding for the new CRHE intake location.

PPL evaluated the radiological consequences resulting from the postulated CRDA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP 15.0.1. The NRC staff's review has found that PPL used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 7 and PPL's calculated dose results are given in Table 1 of Section 3.4 of this SE. The NRC staff performed independent confirmatory dose evaluations to ensure a complete understanding of PPL's methods. The EAB, LPZ, and CR doses estimated by PPL for the CRDA were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.3 Offsite Atmospheric Dispersion Factors

PPL used RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” and the WINDOW computer program which is similar to PAVAN to generate EAB and LPZ χ/Q values. In addition, PPL made comparison calculations with the PAVAN code to demonstrate the acceptability of the χ/Q values which they used in the EAB and LPZ dose assessments. Since all of the postulated release locations are less than 2½ times the height of adjacent structures, PPL used a ground-level release mode, inputting a circular EAB distance of 549 meters and LPZ distance of 4827 meters, reactor building height of 60 meters and reactor building cross-sectional area of 2685 square meters. PPL’s meteorological input consisted of a joint frequency distribution of wind speed, wind direction, and atmospheric stability data for the 1999-2003 period. Wind speed and direction data from the meteorological tower’s 10-meter level were used. Stability class was based on the temperature difference data between the 60-meter and 10-meter levels on the onsite meteorological tower. PPL generated χ/Q values for each of the years and the 5 years combined based upon 12 wind speed categories, with the calm category distributed separately from other wind speed categories. PPL performed calculations as recommended in RG 1.145 and selected the highest resultant 5-year χ/Q values for use in the dose assessment. The NRC staff qualitatively reviewed the inputs to PPL’s computer runs and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff also reran the PAVAN code and obtained similar results.

3.2.4 Atmospheric Dispersion Conclusions

For the reasons cited above, the NRC staff has concluded that the 1999-2003 meteorological data measured at the Susquehanna site provide an acceptable basis for making atmospheric dispersion estimates for use in the DBA dose assessments performed in support of this LAR. The NRC staff has reviewed PPL’s assessments of control room, EAB, and LPZ post-accident dispersion conditions generated from PPL’s meteorological data and atmospheric dispersion modeling. On the basis of this review, the NRC staff concludes that the resulting control room χ/Q values referenced in Table 2 and EAB and LPZ χ/Q values referenced in Table 3 of Section 3.4 of this SE are acceptable for use in the dose assessment described above.

3.3 Technical Specification Changes

3.3.1 TS Definitions Section 1.1, “Dose Equivalent I-131"

The intent of the TS on RCS specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. PPL currently calculates DEI using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole body dose and thyroid dose as done previously. Therefore, PPL proposed to use the inhalation CEDE DCFs from Federal Guidance Report No. 11 (FGR No. 11), “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submission, and Ingestion,” EPA, 1998, and the EDE DCFs from FGR No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” EPA, 1993, to calculate DEI. The NRC staff has evaluated the proposed definition of DEI and has determined that the incorporation of both the CEDE and EDE DCFs in the DEI definition results in a

conservative DEI value when compared to a DEI definition based only on the CEDE DCFs. Therefore, the NRC staff finds the proposed use of FGR No. 11 and 12 in the definition of DEI to be acceptable.

3.4 Data Tables

Table 1
SSES 1 and 2 Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR
Loss of Coolant Accident	8.4E+00	3.8E+00	4.8E+00
Dose Criteria	2.5E+01	2.5E+01	5.0E+00
Main steamline break accident [4]	1.0E-01	6E-03	5E-02 [8]
Dose criteria	2.5E+00	2.5E+00	5.0E+00
Main steamline break accident [5]	2.0E+00	1.2E-01	9.3E-01 [8]
Dose criteria	2.5E+01	2.5E+01	5.0E+00
Control Rod Drop Accident [6]	1.9E-01	5E-02	4.9E-01
Dose criteria	6.3E+00	6.3E+00	5.0E+00
Control Rod Drop Accident [7]	2.3E+00	1.8E-01	1.8E+00
Dose criteria	6.3E+00	6.3E+00	5.0E+00
Fuel Handling Accident	9.6E-01	6E-02	7E-02
Equipment Handling Accident	1.7E+00	1E-01	1.3E-01
Dose Criteria	6.3E+00	6.3E+00	5.0E+00

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Maximum RCS equilibrium iodine activity
[5] Pre-accident iodine spike
[6] 2000 failed rods at full power - mechanical vacuum pump not operational
[7] 30 failed rods at low power - mechanical vacuum pump operational
[8] Results from EC-RADN-1128, Revision 1

Note: Licensee results are expressed to a limit of two significant figures

Table 2
Susquehanna Control Room Atmospheric Dispersion Factors

Source Location / Duration	χ/Q (sec/m^3)	
TB Unit 1 Exhaust Vent (For CRHE evaluation)	AST intake [1]	New intake [2]
0 - 2 hours	1.24E-03	1.09E-03
2 - 8 hours	9.55E-04	8.01E-04
8 - 24 hours	3.14E-04	2.89E-04
24 - 96 hours	1.99E-04	1.72E-04
96 - 720 hours	1.73E-04	1.50E-04
TB Unit 2 Exhaust Vent (For CRHE evaluation)	AST intake [1]	New intake [2]
0 - 2 hours	1.36E-03	1.21E-03
2 - 8 hours	1.03E-03	8.76E-04
8 - 24 hours	3.36E-04	3.16E-04
24 - 96 hours	2.20E-04	1.92E-04
96 - 720 hours	1.85E-04	1.61E-04
SGTS Exhaust Vent (For CRHE evaluation)	AST intake [1]	New intake [2]
0 - 2 hours	1.45E-03	1.16E-03
2 - 8 hours	1.12E-03	8.64E-04
8 - 24 hours	3.35E-04	3.09E-04
24 - 96 hours	2.29E-04	1.87E-04
96 - 720 hours	2.01E-04	1.60E-04
	EC-ENVR-1058	
TB Unit 1 Exhaust Vent (Outside CRHE)	Revision 0	Revision 1
0 - 2 hours	5.09E-03	4.03E-03
2 - 8 hours	4.15E-03	3.61E-03
8 - 24 hours	1.20E-03	1.56E-03
24 - 96 hours	1.16E-03	1.12E-03
96 - 720 hours	1.01E-03	8.71E-04
	EC-ENVR-1058	
TB Unit 2 Exhaust Vent (Outside CRHE)	Revision 0	Revision 1
0 - 2 hours	6.00E-03	4.72E-03
2 - 8 hours	4.93E-03	4.25E-03
8 - 24 hours	1.44E-03	1.84E-03
24 - 96 hours	1.38E-03	1.32E-03
96 - 720 hours	1.21E-03	1.03E-03

[1] CRHE intake on roof of Unit 2 reactor building (values used in the AST dose analyses)
[2] CRHE intake along south wall (elevation 810' 3") of Unit 2 reactor building (values are bounded by the values used in the AST dose analyses)

Corrected by letter dated May 10, 2007

Table 2 (Continued)
Susquehanna Control Room Atmospheric Dispersion Factors

Receptor/ Source Location / Duration		χ/Q (sec/m^3)	
		EC-ENVR-1058	
SGTS Exhaust Vent (Outside CRHE)		Revision 0	Revision 1
	0 - 2 hours	5.15E-03	4.15E-03
	2 - 8 hours	4.22E-03	3.61E-03
	8 - 24 hours	1.23E-03	1.57E-03
	24 - 96 hours	1.19E-03	1.12E-03
	96 - 720 hours	1.04E-03	8.86E-04
CRHE MSLB analysis		EC-ENVR-1128	
		Revision 0	Revision 1
	Puff	5.2E-04 (3)	6.3E-04 (4)

(3) Worst case atmospheric dispersion factor chosen among various effluent release scenarios
(4) Atmospheric dispersion factor associated with the maximum mass and activity release scenario (maximum dose consequence scenario)

Table 3
Offsite Atmospheric Dispersion Factors

Receptor/ Source Location / Duration		χ/Q (sec/m^3)
EAB	0 - 2 hours	8.30E-04
LPZ	0 - 8 hours	4.90E-05
	8 - 24 hours	3.50E-05
	24 - 96 hours	1.70E-05
	96 - 720 hours	6.10E-06

Table 7
SSES Data and Assumptions for the CRD Accident

Core thermal power level	4032 MWt
Radial peaking factor	1.6
Number of fuel assemblies in core	764
Number of equivalent fuel rods per assembly - Atrium 10	87.8
Number of fuel rods damaged in full power CRDA	2000
Number of fuel rods damaged in low power CRDA w/MVP	30
Fraction of fission product inventory in gap	
Noble gases	0.10
Iodines	0.10
Alkali metals (Cs and Rb)	0.12
Fraction of damaged rods experiencing fuel melt	0.77%
Fraction of activity in melted regions released to RCS	
Noble gas	100%
Iodines	50%
Others	
Condenser free air volume	195,000 ft^3
Fraction of activity release in RCS reaching condenser	
Noble gas	100%
Iodines	10%
Others	1%
Fraction of activity reaching condenser that is available for release to environment	
Noble gas	100%
Iodines	10%
Others	1%
Release rate from condenser:	
To turbine building (used in full power case)	1% per day for 24 hours
To environment with MVP running	1212% per day
CR isolation	Not credited
CRE intake flow	6,391 cfm
CRE unfiltered inleakage	510 cfm
CREOASS filter removal efficiency	Not credited