January 31, 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 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NOS. MC8730 AND MC8731)

Dear Mr. McKinney:

The Commission has issued the enclosed 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). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 13, 2005, as supplemented by letters dated May 18, September 15 (PLA-6112 and PLA-6114), September 29, October 20, November 14, December 13, and December 14, 2006.

These amendments revise the SSES 1 and 2 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.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular Biweekly *Federal Register* Notice.

Sincerely,

/RA/

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

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 239 to License No. NPF-14
2. Amendment No. 216 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

Susquehanna Steam Electric Station, Unit Nos. 1 and 2

cc:

Robert A. Saccone
General Manager - Nuclear Operations
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

Terry L. Harpster
General Manager - Plant Support
PPL Susquehanna, LLC
769 Salem Blvd., NUCSA4
Berwick, PA 18603-0467

Rocco R. Sgarro
Manager - Nuclear Regulatory Affairs
PPL Susquehanna, LLC
Two North Ninth Street, GENPL4
Allentown, PA 18101-1179

Walter E. Morrissey
Supervising Engineer
Nuclear Regulatory Affairs
PPL Susquehanna, LLC
769 Salem Blvd., NUCSA4
Berwick, PA 18603-0467

Michael H. Crowthers
Supervising Engineer
Nuclear Regulatory Affairs
PPL Susquehanna, LLC
Two North Ninth Street, GENPL4
Allentown, PA 18101-1179

Steven M. Cook
Manager - Quality Assurance
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB2
Berwick, PA 18603-0467

Luis A. Ramos
Community Relations Manager,
Susquehanna
PPL Susquehanna, LLC
634 Salem Blvd., SSO
Berwick, PA 18603-0467

Bryan A. Snapp, Esq
Assoc. General Counsel
PPL Services Corporation
Two North Ninth Street, GENTW3
Allentown, PA 18101-1179

Supervisor - Document Control Services
PPL Susquehanna, LLC
Two North Ninth Street, GENPL4
Allentown, PA 18101-1179

Richard W. Osborne
Allegheny Electric Cooperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, PA 17108-1266

Director - Bureau of Radiation Protection
Pennsylvania Department of
Environmental Protection
P.O. Box 8469
Harrisburg, PA 17105-8469

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 35, NUCSA4
Berwick, PA 18603-0035

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Board of Supervisors
Salem Township
P.O. Box 405
Berwick, PA 18603-0035

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
443 Orlando Avenue
State College, PA 16803

January 31, 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 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NOS. MC8730 AND MC8731)

Dear Mr. McKinney:

The Commission has issued the enclosed 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). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 13, 2005, as supplemented by letters dated May 18, September 15 (PLA-6112 and PLA-6114), September 29, October 20, November 14, December 13, and December 14, 2006.

These amendments revise the SSES 1 and 2 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.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular Biweekly *Federal Register* Notice.

Sincerely,

/RA/

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

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 239 to License No. NPF-14
2. Amendment No. 216 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:
PUBLIC RidsNrrDorlLpl1-1 RidsNrrPMRGuzman RidsNrrDraAadb RidsOGC Rp
LPLI-1/R/F RidsNrrDciCsgb RidsNrrLASLittle GHill (4)
LBrown JParillo RidsAcrsAcnwMailCenter

ADAMS Accession Number: ML070080301 * SE inputs provided by memo. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	ITSB/BC	CSGB/BC	AADB/BC	OGC	LPLI-1/BC(A)
NAME	RGuzman	SLittle	TKobetz	EMurphy*	MKotzales*	JMartin	DPickett
DATE	1/18/07	1/18/07	1/23/07	4/8/06	12/15/06	1/30/07	1/31/07

OFFICIAL RECORD COPY

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 239
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PPL Susquehanna, LLC, dated October 13, 2005, as supplemented on May 18, September 15 (PLA-6112 and PLA-6114), September 29, October 20, November 14, December 13, and December 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 239 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented no later than October 30, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Douglas V. Pickett, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: January 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 239

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-2	1.1-2
1.1-3	1.1-3
3.1-20	3.1-20
3.3-61	3.3-61

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PPL Susquehanna, LLC, dated October 13, 2005, as supplemented on May 18, September 15 (PLA-6112 and PLA-6114), September 29, October 20, November 14, December 13, and December 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 216 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented no later than October 30, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Douglas V. Pickett, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: January 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-2	1.1-2
1.1-3	1.1-3
3.1-20	3.1-20
3.3-61	3.3-61

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. NPF-22

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated October 13, 2005 Agencywide Documents Access and Management System (ADAMS) Accession No. ML060120353, as supplemented by letters dated May 18, 2006 (ML061520457), September 15, 2006 (ML062710318 and ML062710360), September 29 (ML062850276), October 20, 2006 (ML063040598), November 14, 2006 (ML063310433), December 13, 2006 (ML063540439), and December 14, 2006 (ML063610311), PPL Susquehanna, LLC (PPL, the licensee), requested a license amendment to fully implement an alternative source term (AST) methodology at Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2).

The supplemental letters dated September 15, September 29, October 20, November 14, December 13, and December 14, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 29, 2006 (71 FR 51231).

2.0 REGULATORY EVALUATION

The NRC staff evaluated the radiological consequences of affected design-basis accidents (DBAs) for implementation of the AST methodology at SSES 1 and 2 as proposed by PPL against the dose criteria specified in Section 50.67(b)(2) of Title 10 of the *Code of Federal Regulations* (10 CFR); these criteria are 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE for access and occupancy of the control room (CR) for the duration of the postulated fission product release.

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 1987
- 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

DBAs with radiological consequences based on TS reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. PPL performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. PPL performed AST analyses for the four boiling-water reactor (BWR) DBAs identified in RG 1.183 that could potentially result in significant CR and offsite doses. These include the loss-of-coolant accident (LOCA), the main steam line break accident (MSLB), the fuel-handing accident (FHA), and the control rod drop accident (CRDA).

In PPL's application dated October 13, 2005, PPL indicated that an analysis of the recirculation pump seizure event would be submitted in a separate document at a later date. Subsequently, in a letter dated May 18, 2006, PPL stated that a 10 CFR 50.59 evaluation of the revised recirculation pump seizure analysis concluded that the recirculation pump seizure analysis does not require NRC approval. On June 30, 2006, the NRC completed an inspection at SSES 1 and 2. The inspection report dated July 28, 2006 (ML062090570), documents the acceptability of an evaluation performed by PPL entitled, "E-01-46, LDCN 4313 – Revision to Recirculation Pump Seizure Analysis in Susquehanna Final Safety Analysis Report (FSAR) Section 15.3.3, Rev. 0." This SE, performed by PPL, was one of seven reviewed in the Initiating Event, Mitigating Systems, and Barrier Integrity cornerstones. The inspection was conducted to verify that changes to the facility or procedures as described in the FSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The reviews also included the verification that PPL had appropriately concluded that the changes and tests could be accomplished without obtaining license amendments. Therefore, the NRC staff concludes that NRC approval of the recirculation pump seizure event is not required for the full implementation of the AST at SSES 1 and 2.

The current operating license allows SSES 1 and 2 to operate at a maximum steady-state power level of 3489 megawatts thermal (MWt). PPL is currently engaged in a Constant Pressure Power Uprate (CPPU) project to increase the maximum licensed thermal power to 3952 MWt. Therefore, PPL performed the AST radiological analyses with the core isotopic values at the bounding CPPU power level. For the DBA radiological analyses, the power level is increased by 2% to 4032 MWt to account for measurement uncertainties.

PPL has performed a full implementation of the AST as defined in RG 1.183. PPL has determined that the current Technical Information Document (TID) 14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," accident source term will remain the licensing basis for equipment qualification (EQ), NUREG-0737 evaluations other than CR habitability envelope (CRHE) doses and the radiological consequence analyses for FSAR accidents not included in RG 1.183.

Section 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memo dated April 30, 2001 (ML011210348) and in NUREG-0933, Supplement 25, June 2001 (ML012190402). As stated in the conclusion to generic issue 187, "The NRC staff concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary." Therefore, in consideration of the cited references, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ.

PPL performed a calculation, EC-RADN-1134, Rev. 0, "Impact of AST on Current NUREG-0737 Radiological Evaluations that use TID-14844 DBA-LOCA Releases," to document the impact of the AST on the current NUREG-0737 radiological evaluations that are based on TID-14844 DBA-LOCA releases. The calculation concluded that the current NUREG-0737 radiological evaluations performed with TID-14844 releases are bounding for the DBA-LOCA AST for a reactor core power of 3616 MWt. To account for CPPU changes in power level and to bracket core inventory source term changes, PPL applied a conservative scaling factor of 1.5 to the vital area current licensed thermal power doses. Table 8-1 of the Safety Analysis Report for SSES 1 and 2 - Constant Pressure Extended Power Uprate, entitled Post Accident Vital Occupancy/Mission Dose Summary, indicates that the doses calculated for compliance with NUREG-0737 II.B.2, Design Review of Plant Shielding, are well within the acceptance criteria and are therefore acceptable to the NRC staff.

A full implementation of the AST is proposed for SSES 1 and 2. Therefore, to support the licensing and plant operation changes discussed in the LAR, PPL analyzed the following accidents employing the AST as described in RG 1.183.

1. Loss of Coolant Accident (LOCA)

2. Main Steam Line Break (MSLB) Accident

3. Control Rod Drop Accident (CRDA)

4. Fuel Handling Accident/Equipment Handling Accident (FHA/EHA)

PPL performed dose calculations at the EAB for the worst 2-hour period following the onset of the accident. The integrated doses at the outer boundary of the LPZ and the integrated dose to an SSES 1 and 2 CR operator were evaluated for the duration of the accident. PPL performed all the radiological consequence calculations for the AST with the RADTRAD computer code. NRC sponsored the development of the radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performed independent confirmatory dose evaluations using the same RADTRAD computer code. The results of the evaluations performed by PPL, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 of Section 3.4 of this SE.

PPL generated the core radionuclide inventory for use in determining source term releases using an advanced version of the ORIGEN code (SAS2H/ORIGEN-S). The inventory, consisting of 60 dose significant isotopes at end of fuel cycle curie levels, formed the input for the RADTRAD dose evaluation code. PPL performed a calculation (NEPM-QA-0221-1) 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.

In the evaluation of the LOCA design basis radiological analysis, PPL included dose contributions from the following activity release pathways:

- Primary containment leakage to the reactor building (RB)
- Primary containment bypass leakage directly to the environment
- Engineered Safety Feature (ESF) leakage to the RB
- Main Steam Isolation Valve (MSIV) leakage to the environment via the condenser

PPL included the following DBA LOCA dose contributors to the CRHE analysis:

- Contamination of the CR atmosphere by released activity
- Shine from containment, RB and Turbine Building (TB)
- Shine from piping, components and CR filter loading

3.1.1.1 Assumptions on transport in the primary containment

3.1.1.1.1 Containment leak rate

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of one of the recirculation loops, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. The pipe break results in a blow-down of the reactor pressure vessel (RPV) liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases through a vent system into the suppression pool water, thereby condensing the steam and reducing the drywell pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

SSES 1 and 2 are BWR/4s with a Mark II containment. The SSES 1 and 2 Mark II primary containment consists of two compartments. The two compartments are connected by a vent system that allows steam released from the reactor vessel (located in the drywell) to flow into the suppression pool. The primary containment leakage is limited by TS to 1.0% by weight of containment air per day at the calculated peak accident pressure of 45 pounds per square inch gage (psig). Because of post-accident containment depressurization, this leakage rate will decrease with time. As described in RG 1.183, Appendix A, Section 3.7, for BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the TS leak rate. In the DBA LOCA analysis, PPL assumes that the TS primary containment leak rate is reduced by a factor of two after 24 hours. In response to a request for additional information on this assumption, PPL stated that from FSAR Figures 6.2-6-1, 6.2-6-2 and 6.2-6-3, the maximum containment pressure is 45 psig, and the containment pressure at 24 hours post-LOCA is approximately 9.4 psig. PPL further stated that for extended power uprate (EPU), the maximum calculated containment pressure is 48.6 psig, and the calculated containment pressure at 24 hours is 16.33 psig. PPL asserts and the NRC staff agrees that since the pressure at 24 hours is less than 50% of the peak pressure, a 50% reduction in primary containment leakage after 24 hours is justified.

3.1.1.1.2 Containment mixing

In the submittal dated October 13, 2005 (ML060120353), PPL modeled the primary containment leakage pathway treating the drywell and primary containment as a single, well-mixed volume from the start of the event. This assumption may not be supportable during the early stages of the event. The initial blowdown of the reactor coolant system would have occurred prior to the onset of the in-vessel release phase. Thus, the driving force for mixing between the two volumes will be less at the time when substantial core damage is occurring. Since the LOCA break communicates with the drywell volume only, the use of the drywell and wetwell free volume has the effect of reducing the concentration of the fission products available for release from containment leakage, a non-conservative situation. Because of this uncertainty, the NRC staff has deterministically assumed that complete mixing does not occur until 2 hours post-LOCA, when core reflood is projected as PPL assumed for the MSIV leakage pathway and the secondary containment bypass pathway. In response to a request for additional information (RAI), PPL updated the LOCA analysis assuming that complete mixing within the primary containment does not occur until 2 hours post-LOCA for all pathways. The results of the updated analysis indicate that the CR and the LPZ doses are not changed from the values provided in the AST submittal. The time period for the worst case 2-hour EAB dose changed and the EAB dose from this pathway increased from 3.07 rem to 3.66 rem TEDE. PPL asserts and the NRC staff agrees that the conclusion that the EAB dose is well within the regulatory limit of 25 rem remains valid with the incorporation of the delay in complete containment mixing to 2 hours post-LOCA.

As characterized in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Reactor Plants," the fission product releases (gap and early-in-vessel releases) is assumed to terminate 2 hours after the onset of the postulated LOCA. This would require reflooding of the RPV. PPL asserts, and the NRC staff agrees, that reflooding and core quenching would result in substantial mixing between the drywell and the wetwell. In response to an RAI dated August 17, 2006 (ML062290377), PPL provided specific information describing the mechanisms that would ensure mixing between the drywell and the wetwell upon core reflood. The drywell interacts with the suppression pool and wetwell air space through a total of 87 downcomers, which are vertical pipes with one end open to the drywell atmosphere and the other end submerged in the suppression pool. The purpose of the downcomers is to transport steam from the drywell to the suppression pool for quenching during a large-break LOCA. The suppression chamber airspace communicates with the drywell with five pairs of vacuum breakers, each of which consists of two valves in series. The vacuum breakers are attached to five of the 87 downcomers to allow air and steam flow from the suppression chamber vapor region to the drywell when the drywell is at a negative pressure with respect to the suppression chamber.

The following excerpt from FSAR Section 6.2.1.1.3.3.1.5, "Short Term Accident Response," discusses the interaction between the drywell and the wetwell in response to a recirculation line break LOCA.

> The suppression chamber is pressurized by the carryover of noncondensables from the drywell and by heatup of the suppression pool. As the vapor formed in the drywell is condensed in the suppression pool, the temperature of the suppression pool water peaks and the suppression chamber pressure stabilizes. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence. Drywell pressure decreases as the rate

of energy dumped to the suppression pool via the downcomers exceeds the rate of energy released into the drywell from the primary system. During the RPV depressurization phase, most of the noncondensable gases initially in the drywell are forced into the suppression chamber. However, following the depressurization, the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system. This redistribution takes place as steam in the drywell is condensed by the relatively cool ECCS water which is beginning to cascade from the break causing the drywell pressure to decrease.

The DBA LOCA for dose consequence analyses assumes that the ECCS is ineffective initially, resulting in core damage for the first 2 hours of the accident. At 2 hours, the core damage is assumed to end due to core reflood. Since the ECCS is assumed to be delayed for 2 hours, the subsequent redistribution between the drywell and the suppression chamber via the vacuum breakers as described above is also delayed for 2 hours.

The NRC staff believes that the mass and energy (steaming and steam condensation) created by reflooding (arresting reactor pressure vessel failure) and core quenching will provide sufficient energy to mix the drywell and wetwell air when vacuum breaker cycling occurs during this pressure transient. PPL assumed that the radioactivity release is diluted into the larger volume of the wetwell plus drywell air spaces after 2 hours. Before this time, the radioactivity is only assumed to be released into the drywell net free volume. The NRC staff expects that a significant percentage of the fission products (other than noble gases and iodine in organic form) in the drywell air transferred to the wetwell air space after 2 hours will be scrubbed by the suppression pool water. Conservatively, PPL did not credit any reduction in fission products transferred to the suppression pool air space from the drywell by suppression pool scrubbing. Instead, PPL assumed a well-mixed suppression pool air space and drywell after 2 hours. PPL’s assumption is consistent with the timing of the alternative source term as described in RG 1.183 and therefore, the NRC staff finds this approach acceptable.

3.1.1.1.3 Natural deposition

Consistent with the guidelines provided in RG 1.183, Appendix A, Section 3.2, PPL credited the reduction in airborne radioactivity in the containment by natural deposition. Acceptable models for removal of iodine and aerosols are described in NUREG-0800, SRP Chapter 6.5.2, “Containment Spray as a Fission Product Cleanup System," Rev. 2, December 1988 and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments.” The NUREG/CR-6189 model is incorporated into the RADTRAD code. This simplified model in NUREG/CR-6189 was derived by correlation of the results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. PPL conservatively used the 10th percentile Power's Aerosol Decontamination Model in RADTRAD to account for the reduction in airborne radioactivity in the containment by natural deposition. PPL did not credit fission product reduction due to the initiation of the drywell sprays in the LOCA analysis. Because PPL’s approach is consistent with RG 1.183, the NRC staff finds this approach acceptable.

3.1.1.1.4 Control of pH

PPL credits the manual injection of boron via the stand-by liquid control (SLC) system for suppression pool pH control. The maintenance of a suppression pool pH level above 7.0 is

important to prevent re-evolution of iodine from the suppression pool water. PPL asserts that the initiation of SLC is performed from the CR and that although it is not a new manual action, new procedural guidance is required to address the reliance on SLC for pH control.
PPL's methodology to calculate the post-accident suppression pool water pH is based on NUREG/CR-5950, "Iodine Evolution and pH Control," NUREG-1081, "Post-Accident Gas Generation from Radiolysis of Organic Materials," and NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents," for determining (1) the formation of hydrochloric and nitric acids, (2) the suppression pool pH transient, and (3) long-term iodine re-evolution.

The factors included in the pH calculation are the formation of HI from iodine released into containment, HNO3 from the irradiation of water and air, HCl from the radiolysis of electrical cables in containment, and the injection of sodium pentaborate from the SLC system. Two pH calculations were performed with and without the effects of CsOH on the suppression pool pH. CsOH is a strong base and is formed in the suppression pool during the LOCA. The more conservative pH calculations do not consider the effects of CsOH. PPL calculated the 30-day suppression pool pH of 8.58 subsequent to sodium pentaborate injection and without credit for post LOCA CsOH production in the pool. The NRC staff performed an independent calculation of the pH at the end of 30 days following a LOCA and confirmed PPL's conclusion.

The NRC staff finds that following a LOCA, the re-evolution of iodine will be limited given the buffering action of the sodium pentaborate injected by the SLC system to maintain the suppression pool pH above 7.0. Therefore, the NRC staff finds this approach acceptable.

3.1.1.1.5 Primary containment bypass

The SSES 1 and 2 LOCA analysis assumes that a small amount of the primary containment atmosphere will leak directly to the environment bypassing the secondary containment. The LOCA analysis assumes that the maximum TS combined leakage rate for all miscellaneous secondary containment bypass leakage pathways of nine standard cubic feet per hour (SCFH) exists for the first 24 hours with a 50% reduction at 24 hours post-LOCA. This activity leakage path is modeled as a ground level release from primary containment directly to the environment based on the drywell volume for the first 2 hours and the combined drywell and wetwell volumes for the remainder of the accident. The NRC staff finds this approach acceptable.

3.1.1.2 Assumptions on transport in the secondary containment

The secondary containment structure (also referred to as the RB) completely encloses each of the two primary containment structures such that a dual-containment design is utilized to limit the spread of radioactivity to the environment during a design basis LOCA. Following a LOCA, the secondary containment structure is maintained at a negative pressure ensuring that leakage from primary containment to secondary containment can be collected and filtered prior to release to the environment. The standby gas treatment system (SGTS) performs the function of maintaining a negative pressure within the secondary containment, as well as collecting and filtering the leakage from primary containment. Each of the two redundant SGTS trains consists of a mist eliminator, an electric air heater, a bank of prefilters, two banks of high efficiency particulate air (HEPA) filters, upstream and downstream of the charcoal adsorber, a vertical 8-inch deep charcoal adsorber bed with fire detection temperature sensors, a water spray system for fire protection, and associated dampers, ducts, instruments, and controls.

In the LOCA analysis, PPL credits the SGTS for mitigation of the radiological releases from the RB. The SSES 1 and 2 proposed TS 3.6.4.1.4 bases change will establish a maximum allowable time of less than or equal to 300 seconds for one SGTS subsystem to drawdown the RB, including Zones I, II, and III, to greater than or equal to 0.25 inch of vacuum water gauge. Conservatively, PPL assumed a 10-minute RB drawdown time in the LOCA analysis. PPL assumed that releases into the RB prior to the 10-minute drawdown time leak directly to the environment as a ground level release with no filtration. After the assumed 10-minute drawdown, these releases are filtered by the SGTS and released via the SGTS exhaust vent. The LOCA analysis assumes an SGTS charcoal filter efficiency of 99% for all species of iodine, based on an 8-inch charcoal bed in accordance with Table 2 of RG 1.52, Rev. 2 (ML003740139).

RG 1.183, Appendix A, Section 4.4 states that, "Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%."

The SSES 1 and 2 secondary containment is mixed by a recirculation system to obtain a more uniform concentration of radioactivity following a design basis LOCA. As described in Section 6.5 of the SSES 1 and 2 FSAR, the recirculation system consists of two 100% redundant, vane-axial fans connected to the emergency power supply, associated ductwork, dampers, and controls. The recirculation air is distributed to all areas and rooms through the existing normal ventilation ductwork. Both fans, ductwork used in the recirculation mode, supports, and instruments and controls meet the Seismic Category I requirements. The recirculation system starts automatically on receiving a secondary containment isolation signal. Therefore, the SSES 1 and 2 secondary containment recirculation system meets the guidance to provide adequate means of mixing as stated in RG 1.183. Credit for 50% mixing is a part of the current licensing basis for SSES 1 and 2 and it remains acceptable to the NRC staff for PPL to credit a secondary containment 50% mixing efficiency in the AST LOCA analysis.

3.1.1.3 Assumptions on Engineered Safety Feature System Leakage

To evaluate the radiological consequences of ESF leakage, PPL used the deterministic approach as described in RG 1.183. This approach assumes that except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the suppression pool water. Except for iodine, all of the radioactive materials in the suppression pool are assumed to be in aerosol form and retained in the liquid phase. As a result, PPL assumed that the fission product inventory available for release from ECCS leakage consists of 40% of the core inventory of iodine. This amount is the combination of 5% released to the suppression pool water during the gap release phase and 35% released to the suppression pool water during the early in-vessel release phase. This source term assumption is conservative in that 100% of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the suppression pool concurrently.

ECCS leakage develops when ESF systems circulate suppression pool water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. PPL considered two sources of potential ESF leakage in the release model. The first source evaluated is ESF system leakage directly into secondary containment. Currently, SSES 1 and 2 does not have a TS limit for ESF leakage outside primary containment. TS 5.5.2, "Primary Sources Outside Containment" discusses a program that provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. All significant systems that could contribute to the ESF transport pathway are included in the program which is controlled by station procedure. The program includes preventive maintenance, periodic visual inspection requirements, and integrated leak test requirements for each system at least once per 24 months. By procedure, the current combined maximum allowable limit for these systems is 5 gallons per minute (gpm). PPL indicated in Attachment 8 of its October 13, 2005, application, that they intend to revise this limit to 2.5 gpm prior to the implementation of the AST license amendment. For the LOCA analysis of ESF leakage, PPL used a value of 5 gpm, representing two times the revised limit of 2.5 gpm as specified in RG 1.183, Appendix A, Item 5.2. PPL assumed that ESF leakage starts at T = 0 post-LOCA and continues for the duration of the accident.

For conservatism, PPL evaluated a second source of potential leakage (non-ESF) through the control rod drives (CRDs) and the scram discharge volume (SDV). PPL assumed that the combined contribution from the CRDs and SDV is 15 gpm. This value was not increased by a factor of 2 for this analysis, per RG 1.183, Appendix A, Section 5.2, since this is not an ESF leakage source.

PPL has determined that for all LOCA cases evaluated, the maximum bulk suppression pool water temperature does not exceed 212 °F. RG 1.183, Appendix A, Section 5.5, states that, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid." Therefore, PPL used a flash fraction of 10% to evaluate the contribution due to ESF leakage in the LOCA analysis. In accordance with RG 1.183, PPL assumed that the chemical form of the released iodine is 97% elemental and 3% organic. All ESF leakage is assumed to be released into the secondary containment.

Releases into the secondary containment during drawdown (the first 10 minutes after startup of SGTS) are assumed to leak directly to the environment as a ground level release with no filtration. After the assumed 10-minute drawdown period, ESF releases are assumed to be mixed in the secondary containment with a 50% mixing efficiency, filtered by the SGTS and released via the SGTS vent. PPL has taken no exception or departure from the guidance provided in RG 1.183 for evaluating the radiological consequence resulting from the ESF leakage fission product release pathway. Therefore, the NRC staff finds this approach acceptable.

3.1.1.4 Assumptions on Main Steam Isolation Valve Leakage

The main steam lines in BWR plants, including SSES 1 and 2, contain dual quick-closing MSIVs. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. RG 1.183, Appendix A, Section 6, provides guidance for the evaluation of the radiological consequences from MSIV leakage which should be combined with other fission product pathways to determine the total calculated radiological consequences from a LOCA.

Following the guidance in RG 1.183, PPL assumed that the activity available for release via MSIV leakage is that activity determined to be in the drywell for evaluating containment leakage. PPL did not credit activity reduction by the steam separators or by iodine partitioning in the reactor vessel. The wet-well free air volume was included with the drywell free air volume after 2 hours as previously discussed in the containment leakage section of this SE.
In the SSES 1 and 2 analysis, MSIV leakage is evaluated at the maximum leak rate above which the TS would require the MSIVs to be declared inoperable (less than 100 SCFH from any one valve or less than 300 SCFH total from four valves). The SSES 1 and 2 analysis assumes one MSIV is faulted with a leakage flow rate of 100 SCFH. The remaining leakage flow is evenly split between the remaining MSIVs. The leakage was assumed to continue for the duration of the accident. PPL credited a 50% reduction in the postulated maximum MSIV TS leakage after the first 24 hours.

RG 1.183 allows credit for the reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs as evaluated on a plant-specific basis. The SSES 1 and 2 model for the evaluation of main steam line deposition only credits horizontal runs of piping in determining plateout. Only the portion of the main steam lines from the primary containment isolation valve to the drain piping "tee" is used for plateout. Plateout is not credited in the faulted line. Additionally, since aerosol plateout is a mechanistic settling process, only the bottom half of the inside projected surface area of the lines is credited in the SSES 1 and 2 analysis. To account for uncertainties related to the potential for steam condensation in the piping to wash out and re-evolve some of the settled aerosols, PPL reduced the calculated projected area of the steam lines by a factor of two.

PPL's model for aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of an AST at the Perry Nuclear Power Plant and Clinton Nuclear Station. The aerosol settling model is described in a report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," which was prepared by the NRC Office of Nuclear Regulatory Research. AEB-98-03 (ML011230531) gives a distribution of aerosol settling velocities that are estimated to apply in the main steam line piping. The model used in the Perry and Clinton assessments assumed aerosol settling may occur in the main steam lines upstream of the outboard MSIV, at the median (50^{th} percentile) settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. The NRC staff acknowledges that aerosol settling is expected to occur in the main steam line piping but, because of recent concerns with aerosol sampling and its characteristics as used in AEB-98-03 and the lack of further information, the NRC staff is concerned with how much deposition (i.e., what settling velocity value) is appropriate.

In a meeting with the NRC staff on May 24, 2005, PPL was informed of the NRC staff's concerns regarding the use of the AEB-98-03 median settling velocity of 1.17E-03 m/sec (50^{th} percentile value). In recognition of the uncertainties involved in the determination of an appropriate value for settling velocity, the SSES 1 and 2 MSIV leakage analysis conservatively uses an aerosol settling velocity equal to one quarter of the 10^{th} percentile value from AEB 98-03 or 5.25E-05 m/sec [0.25 x 2.1E-04].

In the analysis of iodine deposition in the MSLs, deposition is enhanced and resuspension is minimized with lower temperatures. PPL evaluated the effect of temperature on the deposition and resuspension of elemental iodine using methods from J. E. Cline, "MSIV Leakage Iodine Transport Analysis," March 26, 1991 (ML003683718). In the AST analysis, PPL conservatively applied the minimum calculated deposition rate and the maximum calculated resuspension rate by assuming a constant temperature of 550 °F for the duration of the accident (0-30 days).

PPL assumed that MSIV leakage was released to the environment as an unprocessed, ground level release without credit for holdup or dilution within the TB.
RG 1.183 allows credit for the reduction in MSIV releases due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). Amendment No. 151 to License NPF-14 and amendment No. 121 to License NPF-22, "Susquehanna Steam Electric Station, Units 1 and 2 (TAC Nos. M91013 and M91014)," dated August 15, 1995 (ML010110048), documents the acceptability of the credited components and piping systems used in the MSIV release path to perform their safety function during and following an SSE.

PPL assumed that the elemental and particulate iodine removal efficiency of the main condenser is 99.6% for the alternate leakage treatment pathway through the main steam drain lines downstream of the outboard MSIV to the isolated main condenser. This efficiency was determined using a methodology in the approved boiling water reactor owners group (BWROG) licensing topical report NEDC-31858P-A, "BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," dated August 1999. This methodology was based on the TID-14844 iodine species fractions of 91% elemental, 4% organic and 5% particulate. The NRC staff believes that the condenser iodine removal efficiency determined using that methodology is bounding for the AST which assumes that the iodine species fractions are 95% aerosol (particulate), 4.85% elemental and 0.15% organic.

The condenser equivalent elemental and particulate (aerosol) decontamination factor (DF) of 250 (effective filter efficiency of 99.6%) was used in the analysis submitted in conjunction with the approval of the MSIV leakage collection system elimination (ML010110048) and is part of the current SSES 1 and 2 licensing basis LOCA analysis. Therefore, the NRC staff finds the assumption of 99.6% efficiency for elemental and particulate iodine removal for the main condenser to be acceptable for the AST LOCA analysis. PPL did not credit any organic iodine removal in the main condenser.

3.1.1.5 CR habitability

3.1.1.5.1 CR ventilation assumptions

Under accident conditions, habitability for the CRHE is provided by the CR Emergency Outside Air Supply System (CREOASS). This system provides habitability zone isolation and a positive pressure for the CRHE. PPL has determined that for the LOCA, the CRHE will automatically isolate and enter the emergency mode in sequence with the SGTS prior to commencement of the release of activity to environment. Per SSES 1 and 2 TS 3.7.3.4 and 5.5.7a, the CREOASS filtered intake flow ranges from 5,229 cfm to 6,391 cfm. PPL evaluated the LOCA for filtered intake flows of 5,229 cfm and 6,391 cfm and determined that the 5,229 cfm flow rate was limiting. The LOCA CRHE analysis assumes that 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 LOCA, 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. PPL did not modify the atmospheric dispersion factors and the associated dose consequence analyses for the LOCA since the results, as originally submitted, are bounding for the new CRHE intake location. Therefore, the NRC staff finds this approach acceptable.

3.1.1.5.2 CR direct shine dose evaluations

The total CR LOCA dose includes direct shine contributions from the effluent plume outside the CR, the direct shine from the buildup of activity on the CR filters, the direct shine from piping containing contaminated fluids as well as the direct shine from radioactive material in buildings adjacent to the control structure including the containment, RB and the TB. PPL performed extensive calculations to bound the total dose to the CR from all significant sources as documented in PPL Calculation No. EC-RADN-1129, Rev. 0, “DBA-LOCA Total Control Room Dose.” This analysis includes contributions to control room dose from the following DBA-LOCA radiation sources:

- Contamination of the CR atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility

- Radiation shine from the external radioactive plume released from the facility

- Contamination of the CR atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the CR envelope

- Radiation shine from radioactive material in buildings adjacent to the control structure; includes containment, RB and TB

- Radiation shine from radioactive material in systems and components inside or external to the CR envelope, e.g., piping, components and radioactive material buildup in heating, ventilation, and air conditioning (HVAC) filters

RG 1.196 defines the control room envelope (CRE) as follows: “The plant area, defined in the facility licensing basis, that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.”

The SSES 1 and 2 CRHE is defined in FSAR Figure 6.4-1A. The areas located inside the SSES 1 and 2 CRHE include the following rooms: Control Room, Technical Support Center, Operational Support Center (OSC), computer, relay, cable spreading and HVAC and battery rooms for both Units 1 and 2. Of these areas within the CRHE, the following are designated as requiring continuous occupancy following a DBA LOCA.

- 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-ENVIR-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 consist of ensuring that personnel maintain a distance of 5 feet from the east wall of the control structure in the STA Office (Room C-401), the Operational Support Center (Room C-402), the electrical equipment room (Room C-413) and the NRC conference room (Room C-414) on elevations 729'-1" and 741'-1" of the CRHE. PPL plans to accomplish this by designating an area 5 feet from the CRHE east wall in these areas as a limited entry zone. PPL has stated that Emergency Plan station procedures shall be revised to address OSC access control prior to AST implementation. PPL asserts and the NRC staff agrees that the access controls described will not impact the performance of critical safety functions or affect the ability of the CR operators to perform the tasks necessary for a safe shutdown. Therefore, the limited access controls within the CRHE, as described herein, are acceptable to the NRC staff.

PPL evaluated the radiological consequences resulting from the postulated LOCA 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 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 5 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 NRC staff finds that the EAB, LPZ, and CR doses estimated by PPL for the LOCA meet the applicable accident dose criteria and are therefore acceptable.

3.1.2 Main Steam Line Break Accident

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. The radiological consequences of an MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The MSLB accident is described in the FSAR Section 15.6.4, "Steam System Piping Break Outside Containment." RG 1.183, Appendix D, identifies acceptable radiological analysis assumptions for an MSLB.

For the DBA, the reactor is assumed to be in hot standby prior to the break. This condition maximizes the calculated liquid mass releases which maximizes the radiological consequences. The assumed displacement of the pipe ends permits a maximum steam line blowdown rate. The only action credited by PPL to reduce the radiological consequence of the MSLB event is the termination of the release upon the automatic closure of the MSIVs. The MSIVs start to close at 0.5 seconds on a high flow signal and are fully closed at 5.5 seconds. The 5.5 second closure time is consistent with the current licensing basis and is supported by TS 3.6.1.3. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount passing through the MSIVs prior to closure. PPL increased the current licensing basis mass releases by 20% to provide additional margin for the EPU conditions. In the LAR, PPL asserts that evaluations of steam line break masses for other extended power uprate plants determined that the increases in mass releases were small compared to the pre-uprate MSLB masses while at power. In response to an RAI dated August 17, 2006 (ML062290377), PPL provided the following additional information concerning the expected mass release for EPU conditions. PPL indicated that the General Electric (GE) Topical Report NEDC-33004P-A, "Licensing Topical Report Constant Pressure Uprate," Revision 4, Class III, July 2003, states that there is no significant increase in the mass released for an MSLB outside containment. PPL stated that this would be expected since, for EPU conditions, there is no 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 2,559 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

CRHE air intake. In addition, the calculation was revised to align the assumptions regarding the atmospheric dispersion factors and the magnitude of activity released. In the original analysis, the worst case atmospheric dispersion factor was selected, among various effluent release scenarios, even though the mass and activity release associated with the worst case value was substantially less than the maximum mass and activity release scenario. The worst case atmospheric dispersion factor was then coupled with the maximum activity release to determine the dose consequence. In the revised analysis, PPL used the release scenario that produced the most conservative dose consequence while maintaining consistency in the amount of mass released for the determination of the atmospheric dispersion factor and the quantity of activity released.

PPL evaluated the radiological consequences resulting from the postulated MSLB accident 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 6 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 NRC staff finds that the EAB, LPZ, and CR doses estimated by PPL for the MSLB meet the applicable accident dose criteria and are therefore acceptable.

3.1.3 Control Rod Drop Accident (CRDA)

The postulated CRDA involves the rapid removal (i.e., drop) of a highest worth control rod resulting in a reactivity excursion. Section 15.4.9 of the SSES 1 and 2 FSAR describes the CRDA as a postulated sequence of events in which a high worth control rod is inserted into the core and subsequently becomes decoupled from its drive mechanism. The drive mechanism is then withdrawn, but the decoupled control rod is assumed to remain in place. A CRDA is initiated when at some later time, the control rod suddenly falls free and drops out of the core, resulting in the insertion of a large positive reactivity and a localized power excursion.

PPL assumed that, as a result of the accident, fuel damage would occur consisting of localized damage to fuel cladding with a limited amount of fuel melt occurring in the damaged rods. In the current licensing basis CRDA evaluation, 1000 rods are assumed to experience cladding damage. PPL expects that the number of rods experiencing clad damage will not substantially change based on favorable results of analyses performed for EPU. In order to establish a conservative bound for assessing the CRDA dose consequence for EPU conditions, PPL assumed that 2000 fuel rods experience cladding failure.

PPL's core performance analyses show the energy deposition that results from this event is below the threshold postulated to melt the fuel pellets. Even though fuel melting is not postulated for the SSES 1 and 2 CRDA, PPL conservatively assumed that 0.77% of the 2000 rods that are assumed to experience cladding damage will also experience fuel melt. PPL stated that the fuel melt assumption is intended to ensure compatibility with the same assumption made in GE's Topical Report NEDO-31400A, which evaluated the elimination of certain main steam line radiation monitor (MSLRMs) safety functions.

3.1.3.1 Source Term

The fuel rod fission product inventory is based on long-term reactor operation at 102% of the EPU thermal power. PPL multiplied the fuel rod fission product inventory by a radial peaking factor of 1.6 to conservatively maximize the fission product release. The source term used for the CRDA analysis is the combination of the release of the gap activity from the fuel rods postulated to be damaged and the 0.77% fuel melt that occurs in the same number of damaged fuel rods. Consistent with the guidance in RG 1.183, Appendix C, Sections 3.1 and 3.2, PPL assumed that the gap activity and the activity from fuel pellet melting mixes instantaneously in the reactor coolant within the reactor pressure vessel with no credit for partitioning or removal by the steam separators.

RG 1.183, Appendix C, Section 1, states that: "For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant."

Consistent with the guidance provided in RG 1.183, Appendix C, PPL assumed that 10% of the core inventory of noble gases and iodine reside in the fuel gap. In addition, PPL assumed that 12% of the core inventory of alkali metals (Cs and Rb) reside in the fuel gap as specified in Table 3 of RG 1.183. Per RG 1.183, PPL assumed that 100% of the noble gases and 50% of the iodines in the melted regions of the fuel is released to the RCS. For additional conservatism, PPL assumed that activity from solids within the fuel matrix would be released to the RCS from the melted fuel regions in proportion to the fractional amounts as specified in Table 1 of RG 1.183.

3.1.3.2 Transport

PPL analyzed two release scenarios for the CRDA. The first case assumes that the accident occurs during full power operation and results in 2000 damaged fuel rods. The termination of this accident scenario is accomplished by automatic safety features with no required operator actions. The main condenser is assumed to isolate on main steam line high radiation along with the mechanical vacuum pump and the steam jet air ejectors. Following the guidance of RG 1.183, Appendix C, Section 3.4, PPL assumed that the activity released from the damaged fuel that reaches the turbine and condenser is released from the TB at ground level at a rate of 1% of the condenser volume per day for the specified release duration of 24 hours. Consistent with RG 1.183, PPL did not credit fission product holdup or dilution in the TB.

PPL evaluated a second case in which the CRDA is postulated to occur at low power operation with the mechanical vacuum pump (MVP) running. As described in calculation EC-RADN-1127 (ML060120487), the MVP draws non-condensable gases from the main condenser and discharges to the TB vent stack. The MVP's period of operation is relatively short, usually during start-up, from critically to a maximum of 5% reactor power. PPL has determined that a minimum of 30 failed rods would be required to generate an MVP trip signal from a CRDA. If a CRDA occurs at low power, with fewer than 30 rods experiencing cladding failure, the steam line dose rates may be too low for the MSLRM to generate an MVP trip signal. Therefore, to be complete in the analysis of the potential radiological consequences of a CRDA, PPL performed a second 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 2,559 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.

3.1.4 Fuel-Handling Accident and Equipment-Handling Accident

As described in Section 15.7.4 of the SSES 1 and 2 FSAR, PPL evaluated two separate cases involving damage to spent fuel. The FHA is assumed to occur as a consequence of the failure of the fuel assembly lifting mechanism resulting in the drop of a channeled fuel assembly, grapple and mast onto other fuel bundles. The analysis considers the total weight of the fuel assembly, channel, grapple and mast (1500 pounds) falling a distance of 32.95 feet (the maximum height that an irradiated fuel assembly can be carried) onto the core or the stored fuel in the spent fuel pool (SFP). PPL assumed that for the FHA with Atrium 10 fuel assemblies, 156 fuel rods would fail releasing their gap activity into the surrounding water. PPL conservatively addressed the issue of lead fuel assemblies (whether in the reactor or in the SFP) by including in the radiological dose results, an evaluation based on the assumption that an additional Atrium 10 assembly, representing a lead use assembly, fails resulting in a total of 254.8 failed rods.

The EHA is assumed to occur as a consequence of a crane failure resulting in the drop of an object onto fuel assemblies in the core or the SFP. The analysis assumes the weight of the dropped object is 1100 pounds with a fall height of 150 feet which is the maximum height that the overhead crane can carry an object. Movement of objects in excess of 1000 pounds are controlled by the Susquehanna Heavy Loads Program. PPL assumed that for the EHA, the number of failed Atrium 10 fuel rods is 366. PPL also evaluated the EHA considering the damage of a lead use assembly resulting in damaging a total of 460.8 rods. PPL calculated the extent of damage for both cases based on the free fall distance and the resulting kinetic energy of the dropped assembly or equipment.

3.1.4.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA or EHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released as a result of the accident. PPL assumed that the affected assemblies are those with the highest inventory of fission products of the 764 assemblies in the core by applying a radial peaking factor of 1.6 and assuming a minimum 24-hour decay time since reactor shutdown.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or SFP depending on their physical and chemical form. RG 1.183, Appendix B, Section 2, allows an overall iodine DF of 200 for a water cover depth of 23 feet. The SSES 1 and 2 evaluation is based on a minimum water cover depth of 21 feet which is in accordance with bases for TS 3.7.7. For a water cover depth of 21 feet, PPL calculated an overall iodine DF of 138 using guidance from RG 1.183 and from a technical paper entitled, "Evaluation of Fission Product Release and Transport for Fuel Handling Accident," G. Burley, 1971 (NRC legacy library accession number 8402080322). Consistent with RG 1.183, PPL credited an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity. In accordance with RG 1.183, PPL did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

PPL analyzed the FHA/EHA based on the fuel rod gap activity release fractions from RG 1.183, Section 3, Table 3. RG 1.183 states that these release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. PPL considered the limitations on peak burnup as stated in RG 1.183 and asserts that for the SSES 1 and 2 extended burnup design, the peak bundle exposure is 54,000 MWD/MTU and the maximum fuel rod pressurization is less than 1200 psig.

3.1.4.2 Transport

As prescribed in RG 1.183, the SSES 1 and 2 FHA/EHA is analyzed based on the assumption that 100% of the fission products released from the reactor cavity or SFP are released to the environment over a 2-hour period. The analysis assumes a 2-hour ground-level release that is filtered by the SGTS. PPL evaluated the FHA/EHA using realistic assumptions and parameters to minimize the activity released and demonstrated that in plant radiation monitors would sense the event and provide the required signals to ensure that all the released activity would be filtered by the SGTS prior to being released to the environment.

Under accident conditions, habitability for the CRHE is provided by the CREOASS. This system provides habitability zone isolation and a positive pressure for the CRHE. PPL has determined that for the FHA/EHA, the CRHE will automatically isolate and enter the emergency mode in sequence with the SGTS prior to commencement of the release of activity to the environment.

Section 3.17 of EC-RADN-1126, "CRHE and Off-Site FHA/EHA Doses - AST," states that, "Per References 22 (EC-RADN-0531) and 23 (EC-RADN-0319), the activity transport from the pool to the environment is via the SGTS filters. Reference 22 provides a conservative analysis, using realistic assumptions and parameters for a fuel handling accident that demonstrates that the Refueling Floor High Exhaust Duct Radiation Monitors Refueling Floor Wall Exhaust Duct Radiation Monitors and the Railroad Access Shaft Exhaust Duct Radiation Monitor will sense the event and provide the required signals to the SGTS. Reference 23 provides an analysis that demonstrates that the isolation damper closure time is less than the air travel time. Therefore, the isolation damper will close prior to the activity reaching the damper."

In response to an RAI, PPL provided an analysis the CRHE and off-site consequences for an FHA that results in an activity release just below the threshold needed to activate the SGTS and CREOASS. The results of the requested analyses indicate that the dose consequences based on the activity release at the setpoint values are bounded by the dose consequences provided in the AST submittal.

3.1.4.3 CR Habitability for the FHA/EHA

Per SSES 1 and 2, TS 3.7.3.4 and 5.5.7a, the CREOASS filtered intake flow ranges from 5,229 cfm to 6,391 cfm. PPL evaluated the FHA/EHA for filtered intake flows of 5,229 cfm and 6,391 cfm and determined that for the FHA/EHA, the 6,391 cfm flow rate was limiting. The FHA/EHA CRHE analysis assumes that 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 FHA/EHA, 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 FHA/EHA since the results, as originally submitted, are bounding for the new CRHE intake location.

PPL evaluated the radiological consequences resulting from the postulated FHA/EHA 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 8 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 NRC staff finds that the EAB, LPZ, and CR doses estimated by PPL for the FHA meet the applicable accident dose criteria and are therefore acceptable.

3.2 Atmospheric Dispersion

PPL calculated new χ/Q values for use in evaluating the radiological consequences of DBAs on the control room, EAB and LPZ using meteorological data collected at the Susquehanna site during the period 1999-2003. PPL used RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," to generate control room χ/Q values. The LOCA, CRDA and FHA/EHA χ/Q values were calculated using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and the MSLB χ/Q values were calculated using the instantaneous puff release methodology, both discussed in RG 1.194. PPL used RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," methodology to calculate the EAB and LPZ χ/Q values. PPL used the WINDOW computer program which is similar to PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Material from Nuclear Power Stations") and made comparative calculations with the PAVAN code to demonstrate the acceptability of the χ/Q values that they used in the EAB and LPZ dose assessments. The resulting set of control room, EAB and LPZ χ/Q values for the LOCA, CRDA, MSLB, and FHA/EHA represent a change from those currently presented in the SSES 1 and 2 FSAR.

3.2.1 Meteorological Data

PPL generated the new LOCA, CRDA, and FHA control room and all offsite χ/Q values for this license amendment request using onsite meteorological data collected during 1999-2003. PPL provided these data by a December 13, 2006, letter in the form of electronic hourly data files for input into the ARCON96 computer code. PPL also provided joint wind speed, wind direction, and atmospheric stability frequency distribution (joint frequency distribution) tables for input into the PAVAN computer code in Attachment 10 of the October 13, 2005, letter.

PPL stated that the 1999-2003 Susquehanna onsite meteorological data were selected based upon the quality of the data, the data recovery rate, the representativeness of long-term conditions and seasonal trends, and that the data meet the requirement of RG 1.23, “Onsite Meteorological Programs.” PPL performed its atmospheric dispersion analyses using wind measurements taken at 10 meters and 60 meters above ground level on the onsite meteorological tower. PPL determined atmospheric stability using the temperature difference measurements between the 60-meter and 10-meter levels on the main meteorological tower.

The NRC staff performed a quality review of the 1999-2003 hourly meteorological database provided by PPL using the methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data.” Further review was performed using computer spreadsheets. The combined data recovery of the wind speed, wind direction, and stability (temperature difference) data was in the upper 90 percentiles at each level during each of the 5 years, thus exceeding the RG 1.23 goal of 90% recovery. With respect to atmospheric stability measurements, the time of occurrence and duration of stable and unstable conditions were consistent with expected meteorological conditions. Stable and neutral conditions were reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year. During 1999-2003, wind speed and wind direction were reasonably similar at each height from year to year. Winds were generally bi-modal, showing the influence of the valley terrain in the Susquehanna site vicinity, and were predominately from the east northeast and southwest directions at the 10-meter level and from the north northeast and northeast and southwest and west southwest at the 60-meter level. A comparison of joint frequency distributions derived by the NRC staff from the ARCON96 hourly data with the joint frequency distributions developed by PPL for input into the PAVAN atmospheric dispersion model showed good agreement.

3.2.2 Control Room Atmospheric Dispersion Factors

PPL initially evaluated 13 release points to the environment for possible modeling of the control room χ/Q values. Following a review of the layout of the SSES 1 and 2 plant structures, exhaust vents and components, and the locations of the postulated release points to the location of the CRHE, PPL determined the limiting locations as follows. The LOCA, primary containment and ESF leakages to the secondary containment were assumed to be released to the environment via the SGTS vent and the primary containment bypass and MSIV leakages via the TB exhaust vent. For the CRDA, the release to the environment was modeled to occur from the TB Unit 2 vent and the FHA release via the SGTS exhaust vent.

PPL used guidance provided in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” to generate new control room atmospheric dispersion factors for the CRHE air intake. PPL calculated the χ/Q values for the LOCA, CRDA and FHA using the ARCON96 computer code (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes”). RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in design basis accident radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the Susquehanna site. PPL executed ARCON96 using the 1999-2003 hourly data from the Susquehanna onsite meteorological tower. Because the heights of all of the release locations (i.e., TB Unit 1 and Unit 2 exhaust vents and SGTS exhaust vent) are less than 2½ times the height of adjacent buildings, they were modeled using the ARCON96 ground-level release option in accordance with RG 1.194 as point sources. PPL provided four sets of χ/Q values, two each for two receptor locations, namely, the CRHE air intake and outside the CRHE (for unprotected CR operator dose determination). These χ/Q values are provided in Table 2 of Section 3.4 of this SE.

The χ/Q values calculated for the CRHE air intake provided in the October 13, 2005, submittal were based upon an assumption that the CRHE air intake is located near the southeast corner at the top of the Unit 2 reactor building. The enclosure to PLA-6124 of the November 14, 2006, letter provided revised χ/Q values for the new planned location of the CRHE air intake to be located on the south wall of the Unit 2 reactor building. The enclosure to PLA-6124 included a comparison of the χ/Q values for both locations. In all cases, it was shown that the CR intake χ/Q values provided in the October 13, 2005, submittal and used in the AST dose assessment described above bound the revised χ/Q values for the CRHE air intake located on the south wall of the Unit 2 reactor building.

The χ/Q values for outside the CRHE provided in the October 13, 2005, submittal were approximated using the ARCON96 methodology assuming postulated releases from the TB Units 1 and 2 and SGTS exhaust vents with a receptor point location on the south wall of the control building. Although the dose assessment assumes that the cloud of effluent surrounds the entire structure housing of the CRHE, the NRC staff determined that PPL's use of the single point receptor location on the south wall of the control room building coupled with assumed releases from the TB Unit 1 and 2 and SGTS exhaust vents for estimation of the χ/Q values does not significantly impact the resultant estimates for this specific dose assessment. The χ/Q values for outside the CRHE were revised as discussed in the November 14, 2006, submittal to correct an input error in the original calculations. Although the revised χ/Q values for the 8 - 24 time periods are slightly higher than in the original calculation, as discussed in Section 3.1.1.5.2 above, the overall dose decreased slightly due to the decreases in the χ/Q values for the other time periods.

PPL used the instantaneous puff release methodology discussed in RG 1.194 to calculate the control room χ/Q values for the MSLB and selected the χ/Q value associated with the highest postulated level of activity released. Two χ/Q values are presented in Table 2 of Section 3.4 of this SE. The χ/Q value in the column titled Revision 0 was the most limiting value, assuming a difference in height between the center of the puff and the control room intake. Since RG 1.194 states that the difference in height should be assumed to be zero for the case at SSES 1 and 2, PPL revised the MSLB to conservatively assume that the height difference between the height of release and the intake height is zero and provided a revised χ/Q value in the enclosure to PLA-6124. This value, listed in Table 2 of Section 3.4 of this SE, under the column titled Revision 1, is not the limiting χ/Q value for all cases considered, but is the χ/Q value which results in the highest dose when combined with the associated activity release.

The NRC staff qualitatively reviewed the inputs to the ARCON96 and MSLB puff calculations and found them generally consistent with site configuration drawings and site practice. The NRC staff also performed a random check of the ARCON96 and puff calculations by generating χ/Q values and obtained results that were consistent with PPL's estimates.

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 WINDOWS 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	7.8E+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−02	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/m³)	
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-ENVIR-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-ENVIR-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)

Table 2 (Continued)
Susquehanna Control Room Atmospheric Dispersion Factors

Receptor/ Source Location / Duration		χ/Q (sec/m^3)	
		EC-ENVIR-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-ENVIR-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 4
SSES Control Room Data and Assumptions

Control structure habitability envelope total volume	518,000 ft^3
CR free air volume	110,000 ft^3
CREOASS pressurization flow rate	5,229 cfm
CR unfiltered inleakage	510 cfm
CR isolation and CREOASS initiation	
LOCA	Automatic
MSLB	Not credited
CRDA	Not credited
FHA/EHA	Automatic
Credited manual actions relative to the CREOASS	None
CREOASS credited filter efficiencies for all iodine species	99%
CRHE operator breathing rate	
0 - 720 hours	3.5E−04 m^3/sec
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4

Table 5 (Page 1 of 2)
SSES Data and Assumptions for the LOCA

Power	4032 MWt
Drywell free volume	239,600 ft^3
Wetwell free volume	148,590 ft^3
Total free volume	388,190 ft^3
Primary containment leak rate	
0 - 24 hours	1%/day
24 - 720 hours	0.5%/day
Primary containment aerosol deposition	10^{th} percentile Powers Model
Containment bypass leak rate	
0 - 24 hours	9 scfh
24 - 720 hours	4.5 scfh
RB free air volume used	4,156,600 ft^3 (Zones I & III)
Zone I	1,488,600 ft^3
Zone II	1,598,600 ft^3
Zone III	2,668,000 ft^3
RB Post-Loca drawdown time	10 minutes
RB mixing efficiency	50%
Iodine chemical form in containment atmosphere	
cesium iodide	95%
elemental iodine	4.85%
organic iodine	0.15%
Iodine chemical form in the RB sump	
elemental	97%
organic	3%
Containment sump pH	≥ 7
SGTS filter efficiency	99% all iodine species
MSIV leak	300 scfm total (4 lines) 100 scfm in faulted line 66.67 scfm each remaining line
Effective condenser volume	98,601 ft^3

Table 5 (Page 2 of 2)
SSES Data and Assumptions for the LOCA

Condenser removal efficiency for drain line pathway	
Aerosols	99.6 %
Elemental iodine	99.6 %
Organic iodine	No removal credit
MSL inlet initial pressure and temperature conditions	
Pressure	50 psia
Temperature	550 °F
ESF	
ESF	5 gpm
CRD	15 gpm
Minimum post-LOCA suppression pool volume	132,000 ft^3
Maximum post-LOCA suppression pool temperature	< 212 °F
ESF flash fraction	10%
CR isolation	Automatic
CRE intake flow	5,229 cfm
CRE unfiltered inleakage	510 cfm
CREOASS filter removal efficiency	99% all species

Table 6
SSES Data and Assumptions for the MSLB Accident

Core thermal power level	4032 MWt
Noble gas design source term	100,000 μCi/sec after 30 min
Design offgas release rate	403,000 μCi/sec after 30 min
Maximum equilibrium iodine	0.2 μCi/gm DE I-131
Pre-accident iodine spike	4.0 μCi/gm DE I-131
Iodine carryover fraction	
Reactor coolant	2%
Steam	8%
MSIV isolation time	5.0 seconds
Total release duration	5.5 seconds
Liquid release	101,808 lbm
Steam release from steam dome	7,980 lbm
Steam release from flashed liquid	7,776 lbm
CR isolation	Not credited
CRE intake flow	6,391 cfm
CRE unfiltered inleakage	510 cfm
CREOASS filter removal efficiency	Not credited

Table 7
SSES Data and Assumptions for the CRD Accident

Core thermal power level	4032 MWt
Radial peaking factor	1.7
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

Table 8
SSES Data and Assumptions for the FHA/EHA

Core thermal power level	4032 MWt
Minimum post shutdown fuel handling time (decay time)	24 hours
Number of fuel assemblies in core	764
Number of equivalent fuel rods per assembly - Atrium 10	87.8
Number of failed rods for fuel handling accident	254.8
Number of failed rods for equipment handling accident	460.8
Core radial peaking factor	1.6
Minimum pool water depth	21 feet
Fuel clad damage	
Gap fractions	
I-131	8%
Remainder of halogens	5%
Kr-85	10%
Remainder of noble gases	5%
Alkali metals	12%
Pool DF	
Noble gases	1
Aerosols	Infinite
Iodines - corrected for 21 ft of water cover	138 (effective DF)
Duration of release	2 hours
Activity transport from pool to environment	SGTS actuation prior to activity release to environment
SGTS filter efficiency	99% all iodine species
CR isolation	Automatic
CRE intake flow	6,391 cfm
CRE unfiltered inleakage	510 cfm
CREOASS filter removal efficiency	99% all species

3.5 NRC Staff Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by PPL to assess the radiological consequences of DBAs with full implementation of an AST at SSES 1 and 2. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that PPL's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that SSES 1 and 2, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the SSES 1 and 2 design basis is superseded by the AST proposed by PPL. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Part 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the SSES 1 and 2 design basis, and modified by the present amendment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 51231). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: John Parillo
Leta Brown
Carolyn Lauron

Date: January 31, 2007