Britt T. McKinney
Sr. Vice President & Chief Nuclear Officer

PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3149 Fax 570.542.1504
btmckinney@pplweb.com

ppl™

SEP 15 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 281 TO LICENSE NPF-14
AND PROPOSED AMENDMENT NO. 251 TO LICENSE NPF-22:
"APPLICATION FOR LICENSE AMENDMENT AND
RELATED TECHNICAL SPECIFICATION CHANGES
TO IMPLEMENT FULL-SCOPE ALTERNATIVE SOURCE
TERM IN ACCORDANCE WITH 10 CFR 50.67 - RESPONSE
TO REQUEST FOR ADDITIONAL INFORMATION"** **Docket Nos. 50-387**
PLA-6114 **and 50-388**

References: 1) *PLA-5963, Mr. B. T. McKinney (PPL) to Document Control Desk (USNRC) Proposed Amendment No. 281 to License NPF-14 and Proposed Amendment No. 251 to License NPF-22: "Application for License Amendment and Related Technical Specification Changes to Implement Full-Scope Alternative Source Term in Accordance with 10 CFR 50.67," dated October 13, 2005.*

2) *Letter from US NRC (R.V. Guzman) to Britt T. McKinney (PPL) "Request for Additional Information (RAI) - Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2) - Application for License Amendment and Related Technical Specification Changes to Implement Full-Scope Alternate Source Term (TAC Nos. MC 8730 and MC 8731)," dated August 17, 2006.*

In accordance with the provisions of 10 CFR 50.90, PPL Susquehanna, LLC (PPL) submitted a request for license amendment and related changes to the Unit 1 and Unit 2 Technical Specifications to implement the full-scope alternate source term in accordance with 10 CFR 50.67.

The purpose of this letter is to provide the PPL responses to the NRC RAI (Reference 2).

Attachment 1 to this letter contains the NRC RAI's and the PPL responses.

A001

If you have any questions regarding this submittal, please contact Mr. Michael H. Crowthers at (610) 774-7766.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 9-15-06

Respectfully,

Britt T. McKinney

Attachments:
Attachment 1 – PPL RAI Responses

cc: NRC Region I
Mr. A. Blamey, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

Attachment 1 to PLA-6114

PPL RAI Responses

Question 1: Section 3.7 of Regulatory Guide (RG) 1.183 states that for boiling-water reactors (BWRs), primary containment 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 technical specification leak rate. Please provide the requisite site-specific information to support the assumption of a 50% reduction in primary containment leakage after 24 hours. Section 3.9 of EC-RADN-1125, Rev. 0 provides some information; however, the bottom of sheet 8 (as provided in the submittal package and partially reproduced below) is blurred and difficult to read.

"Based on the significant reduction of the calculated internal pressure of primary containment at 24 --------------------- Per PPL Drawing C-206-130, Sheet 1, Primary Containment Zones (Reference 29), the Drywell LOCA peak pressure <24 hours equals 41.3 psig and at 24 hours equals approximately 15.3 psig."

PPL Response: The blurred sentence is reproduced below:

"Based on the significant reduction of the calculated internal pressure of primary containment at *24 hours into the LOCA, the 50% reduction in leak rate is taken herein.* Per PPL Drawing ...".

The containment pressures presented in the calculation were obtained from the bounding containment pressure profile used for equipment qualification (EQ) purposes. Containment pressures as a function of time can be found in the SSES FSAR. A review of FSAR Figures 6.2-6-1, 6.2-6-2 and 6.2-6-3 indicates that the maximum containment pressure is 45 psig, and the containment pressure at 24 hours is approximately 9.4 psig. 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. Since the pressure at 24 hours is less than 50% of the peak pressure, the 50% reduction in primary containment leakage is justified.

Question 2: PPL's modeling of the primary containment leakage to the secondary containment pathway appears to treat the drywell and wetwell as a single, well-mixed volume of 388,190 feet cubed (ft ^{3}), from the start of the event. With the AST timing, as described in RG 1.183, Table 4, the initial blow-down of the reactor coolant system would be expected to have occurred prior to the onset of the early in-vessel release phase. Therefore, at the onset of the early in-vessel release phase (T = 30 minutes), the driving force for mixing between the two volumes will be less certain. Because of this uncertainty, the NRC staff has deterministically assumed that complete mixing does not occur until 2 hours of post-loss-of-coolant accident (LOCA), when core re-flood is projected and necessary to end the early in-vessel release phase (as PPL assumed for the main steam isolation valve leakage and the secondary containment bypass pathways).

Please provide further justification describing why PPL's proposed approach is adequately conservative for the primary containment leakage pathway and justify the apparent inconsistency in the application of credit for drywell/wetwell mixing in the LOCA analysis.

PPL Response: The primary containment leakage to the secondary containment pathway was reanalyzed with RADTRAD assuming complete mixing within the primary containment does not occur until two hours post-LOCA. The only pathway result that changed was the primary containment leakage to the secondary containment pathway. The 30-day integrated dose results for this pathway are compared to the values provided in the AST submittal below:

	CRHE	**EAB**	**Time for**	**LPZ**
LOCA	Rem TEDE	Rem TEDE	worst 2 hr EAB dose	Rem TEDE
AST Submittal RADTRAD Values	**1.16**	**3.07**	**@4.4 hr**	**1.63**
RB SGTS 0- 2 hr Drywell mixing only	0.27	1.21	@ 1.5 hr	0.22
RB SGTS 2 -720 hr Drywell+Wetwell mixing	0.89	2.45	@ 5.6 hr	1.41
Total	**1.16**	**3.66**		**1.63**

The CRHE and the LPZ doses are not changed from the values provided in the AST submittal. The worst case 2-hour EAB dose is determined by adding the worst case 2-hour doses for each period. The conclusion that the EAB dose is well within the regulatory limit of 25 Rem TEDE is unaffected by the assumption change.

Question 3: Section 3.1 of RG 1.183 states that the suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. Please provide a description of the site-specific mechanisms available to ensure mixing between the drywell and the wetwell.

PPL Response: The suppression chamber airspace communicates with the drywell via vacuum breakers. There are five pairs of vacuum breakers. Each pair consists of two valves in series. They are attached to the capped downcomers to allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve, which can be remotely operated for testing purposes. The drywell interacts with the suppression pool and wetwell airspace via a total of 87 downcomers. Downcomers are open vertical pipes which connect the containment atmosphere with the suppression pool. Downcomers are submerged in the suppression pool. The purpose of the downcomers is to transport steam in the drywell to the suppression pool during a large break LOCA. The steam is then quenched in the suppression pool. These features are discussed in more detail in FSAR Section 6.2. Additionally, Technical Specification (TS) 3.6.1.6 assures that the vacuum breakers will be available in the event of a large break LOCA.

Question 4: In the main steam line break (MSLB) analysis, the current licensing basis mass releases are increased by 20% to provide additional margin for extended power uprate (EPU) conditions. PPL asserts that evaluations of steam line break masses for other EPU plants determined that the increases in mass releases were small compared to the pre-uprate main steam line break masses while at power. Please provide additional information to show that a 20% increase in MSLB mass release would be bounding for a proposed future increase in rated thermal power of approximately 12%.

PPL Response: From GE Topical Report NEDC-33004P-A "Licensing Topical Report Constant Pressure Power Uprate," Revision 4, Class III, July 2003, there is no increase in the mass released for a Main Steam Line break outside primary containment. This would be expected since for EPU conditions, there is no pressure increase or change to the size of the main steam lines. Based on these parameters remaining the same, the mass released will be equal to the mass released under current conditions, and the 20% increase in mass released is a conservative assumption in the calculation.

Question 5: Table 4.5-1 in PPL's application describes the control rod drop accident (CRDA) assumptions and indicates that PPL assumed 100% of the noble gases and 50% of the iodines in the melted regions of the fuels is released to the reactor coolant system. The amount of solids released from the melted regions is not specified. However, Section 4.5.8 states that as a calculation conservatism, solids released from the melted fuel were included in the analysis. Table 4 of EC-RADN-1127 indicates that activity from solids as a result of the fuel melt is included in the analysis, and references RG 1.183, Table 1. Please clarify the assumptions regarding the inclusion of solids in the fuel melt portion of the CRDA source term. Also, please note that it appears that the fractions for Lanthanides and Ceriums in Table 4 of EC-RADN-1127 may have been inadvertently reversed; however, it is not expected that this would have a significant effect on the results of the calculation.

PPL Response: Solids were included in the fuel melt portion of the CRDA source term. Solids were added as a conservatism to the CRDA evaluation, and their conclusion ensures analytical consistency with General Electric Topical Report, NEDO-31400A "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and SCRAM Function of the Main Steam Line Radiation Monitor" for removal of the trip function for the MSLRM. This is discussed in assumption/input 3.2.2 of EC-RADN-1127.

The equations used to determine the fractional releases appear in the text below Table 4 of EC-RADN-1127. The resulting release fractions for all of the isotopes input to RADTRAD are shown in Table 5 of EC-RADN-1127.

The Lanthanides and Cerium fractions were reversed in the calculation as noted in the RAI. The correct fractions which should have been input to RADTRAD are:

Lanthanides	4.59E-12
Cerium	1.15E-11

The change was made to the RADTRAD input file and the code was rerun. As expected, the results were not changed as shown below:

CRDA 30 Failed Rods with MVP Running			
	AST Submittal Reported Values	Actual RADTRAD Values AST Submittal Output	Actual RADTRAD Values Corrected Cerium/Lanthanum Fractions Output
	Rem TEDE	Rem TEDE	Rem TEDE
EAB	2.3	2.2154	2.2154
LPZ	0.18	0.17753	0.17753
CRHE	1.8	1.7335	1.7335

CRDA 2000 Failed Rods with Condenser Leakage			
	AST Submittal Reported Values	Actual RADTRAD Values AST Submittal Output	Actual RADTRAD Values Corrected Cerium/Lanthanum Fractions Output
	Rem TEDE	Rem TEDE	Rem TEDE
EAB	0.19	0.18669	0.18669
LPZ	0.05	0.04339	0.04339
CRHE	0.49	0.48637	0.48637

Question 6: In Section 4.6.3 of the AST application describing the fuel-handling accident/equipment handling accident (FHA/EHA) analysis, it states that, "For this event, the CRHE [control room habitability envelope] automatically isolates and enters the emergency mode in sequence with the SGTS [standby gas treatment system] prior to commencement of the release of activity to environment." In Section 3.17 of EC-RADN-1126, "CRHE and Off-Site FHA/EHA Doses – AST," it states:

> 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.

From the cited references is an estimate available of the CRHE and off-site consequences for an FHA/EHA that results in an activity release just below the threshold needed to activate the SGTS and the Control Room Emergency Outside Air Supply System as credited in the analysis?

PPL Response: The following provides an estimate of the CRHE and off-site consequences for an FHA/EHA that results in an activity release just below the threshold needed to activate SGTS and CREOASS. The analysis credits manual initiation of SGTS within 30 minutes of the accident but does not credit CREOASS. This is a conservative assumption as the FSAR Section 15.7.4 analysis states that operator actions will begin at 5 minutes. As stated in the FSAR, one action performed by the operator at 5 minutes is to assure that the normal ventilation system has isolated and that SGTS has initiated. The operators are alerted to the accident by Refuel Floor personnel in contact with the control room and also by the HI-HI SPING alarms which are received in the control room. The Emergency Operating procedures require the operators to isolate the normal ventilation system and start SGTS for this condition. The results are presented below:

	CR Rem TEDE	EAB Rem TEDE	LPZ Rem TEDE
	TEDE	TEDE	TEDE
AST Submittal Values (EHA with 460.8 rods failed)	0.13	1.74	0.1
Setpoint Analysis	0.08	0.16	0.01

The dose consequences based on the activity release at the setpoint values are bounded by the EHA dose consequences provided in the AST submittal.

Question 7: In the following sections of the submittal, the terms "control room" and "control room operator," may have been used in a general sense that could lead to confusion as to the areas and the personnel requiring access limitations. RG 1.196 defines the terms Control Room and Control Room Envelope (CRE) and makes a distinction between the two regarding the requirements for occupancy. The NRC staff is reluctant to make any exceptions to General Design Criterion 19 dose limits for the control room (and the technical support center). However, other areas within the CRE that do not support critical safety functions may be evaluated on a case-by-case basis to assess the appropriateness of limited access controls.

Please provide a plan view of the affected areas to ensure that the proposed access controls for designated portions within the CRHE will not impact the control room proper or the technical support center.

PPL Response: The attached figure shows by bubble and crosshatching the layout for the areas of concern. The areas of concern are two office spaces (room designations C-401 and C-402 on the attached drawing) on elevation 729' which are adjacent to the main control room and two areas (room designations C-414 and C-413 on the attached drawing) on elevation 741' which are adjacent to the technical support center (TSC). These areas do not contain any equipment necessary for safe shutdown. During an accident, the control structure would be continuously monitored for habitability, and these areas would be controlled as required. Control of these office spaces within 5 feet of the east wall would not affect the ability of the operator to perform safe shutdown. Additionally, the plant operators would not be expected to spend significant time in an office space during an event. TSC personnel would also not be impacted. The areas on elevation 741' are not active work spaces for PPL personnel during an accident. The conference room on elevation 741' (C-414) will be controlled if conditions warrant, but would not prevent the room from being used or occupied. The other room of concern on elevation 741' (C-413) is an electrical closet and is not typically accessed during normal operation or during TSC activation.