January 16, 2007

Mr. Britt T. Mckinney
Sr. Vice President & Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603-0467

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. MD3021 AND MD3022)

Dear Mr. McKinney:

The U.S. Nuclear Regulatory Commission staff has reviewed the Severe Accident Mitigation Alternatives analysis submitted by PPL Susquehanna, LLC, in support of its application for license renewal for the Susquehanna Steam Electric Station, and has identified areas where additional information is needed to complete its review. Enclosed are the staff's requests for additional information.

We request that you provide your responses to these questions within 90 days of the date of this letter to support the license renewal review schedule. If you have any questions, please contact me at 301-415-1224 or via email at axm7@nrc.gov.

Sincerely,

/RA/

Alicia Mullins, Project Manager
Environmental Branch B
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure:
As stated

cc w/encl: See next page

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Dear Mr. McKinney:

The U.S. Nuclear Regulatory Commission staff has reviewed the Severe Accident Mitigation Alternatives analysis submitted by PPL Susquehanna, LLC, in support of its application for license renewal for the Susquehanna Steam Electric Station, and has identified areas where additional information is needed to complete its review. Enclosed are the staff's requests for additional information.

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OFFICE	LA:DLR	PM:DLR:REBB	PM:DLR:REBB	BC:DLR:REBB
NAME	I. King	J.Davis	A.Mullins	R.Franovich
DATE	01/5/07	01/8/07	01/16/07	01/16/07

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. MD3021 AND MD3022)

DISTRIBUTION:

HARD COPY
A. Mullins
B. Palla

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M. Rubin (RidsNrrDraApla)
R. Franovich (RidsNrrDlrRebb)
E. Benner (RidsNrrDlrReba)
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RidsNrrDlrRlra
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R. Schaaf
F. Monette (fmonette@anl.gov)
R. Guzman
Y. Diaz Sanabria
E. Gettys
A. Blamey, SRI Region 1
N. Sheehan, OPA Region 1
Samuel Lee, EDO Contact, Region I
A. Blamey (ajb3@nrc.gov)
C. Colleli
R. K. Wild
J. Storch

Request for Additional Information
Regarding the Analysis of Severe Accident Mitigation Alternatives
for the Susquehanna Steam Electric Station, Units 1 and 2

1. Provide the following information regarding the Susquehanna Steam Electric Station (SSES) Probabilistic Risk Assessment (PRA) model used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

   a. Provide a summary of the major Level 1 and 2 PRA versions and their core damage frequency (CDFs) from the Individual Plant Examination (IPE) to the present, including the version reviewed by the Boiling Water Reactor Owners Group (BWROG), and the version used for risk-informed submittals such as inservice inspection and allowed outage time extension for offsite power. Also, indicate the major changes to each version from the prior version (including the changes from pre-Extended Power Uprate (EPU) to post-EPU models) and the major reasons for changes in the CDF.

   b. Provide the freeze date for the incorporation of design and/or procedure changes into the PRA.

   c. Provide the CDF contribution due to station blackout (SBO) and anticipated transient without scram (ATWS).

   d. Explain why loss of an AC bus is a very small (approximately 0.2 percent) contributor to the CDF.

   e. The summary of the BWROG peer review overall assessment provided on pages E.2-14 and -15 describes a non-conservatism associated with SBO events. Identify the facts and observations (F&Os) associated with this non-conservatism and discuss their resolution.

   f. Section E.2.3.2 of the Environmental Report (ER) describes a self assessment that considered the open Level B F&Os and concluded that the remaining items and gaps would not have a significant impact on the EPU application. Confirm that the same conclusion can be drawn concerning the impacts of the remaining items and gaps on the SAMA analysis.

   g. Describe the current containment venting capability and procedural directions at SSES (hard pipe, via standby gas treatment system, etc.) and how it is modeled in the PRA.

2. Provide the following information relative to the Level 2 analysis:

   a. Provide a summary description of the current Level 2 model, including: the Level 1/Level 2 interface, the containment event tree (CET), the basis for quantification of CET nodes, the binning process used to assign end states to release categories, and the determination of release fractions for each release category.

ENCLOSURE

b. Describe the steps taken to ensure the technical adequacy of the Level 2 revisions subsequent to the BWROG peer review.

3. Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

   a. The individual plant examination of external events (IPEEE) fire analysis utilized the IPE internal events models to assess system performance. Indicate whether the original IPE models or the revised IPE models were utilized.

   b. Based on a sensitivity study performed by PPL Susquehanna, LLC, the U.S. Nuclear Regulatory Commission concluded in the IPEEE safety evaluation report that the CDF for some fire contributors might be as much as three orders of magnitude higher than the revised values reported in the IPEEE. Discuss this issue and its potential impact on the ER assumption that the fire CDF is about equal to the internal events CDF.

4. Provide the following information concerning the MACCS2 analyses:

   a. Clarify whether separate ORIGEN calculations were performed for pre-EPU and post-EPU conditions and used to determine population doses for the respective cases.

   b. Based on the March 31, 2006, license amendment request, the EPU power level would be approximately 13% above the current licensed power level. As such, the population dose for EPU conditions would be expected to be approximately 13% greater than for pre-EPU conditions. However, from Table E.3-4, the increase in dose for the dominant release categories (e.g., L2-1, L2-2, and L2-5) ranges from 4 to 11%. Explain this result.

5. Provide the following with regard to the SAMA identification and screening process:

   a. Tables E.5-1 and E.5-2 include a number of events that are described as preventative maintenance actions (i.e., 024-N-E-DSL-P, 024-I-A-DSL-P, and 024-II-B-DSL-P). Identify the specific structure, system, and components associated with these maintenance actions.

   b. Section 5.1.5 includes a list of nine enhancements identified in the IPE. The seventh enhancement, revise guidance regarding reactor vessel control, is listed as “not implemented” and only provides the reasoning that the enhancement has been determined not to be required for safe operation of the plant. Provide a further description of the disposition of this enhancement, and any efforts made to identify SAMA candidates exist to address the associated risk contributors.

   c. Section E.5.1.7.1 discusses the contribution to fire CDF from the dominant fire zones. Although two SAMAs from the internal events analysis were identified to address this risk, no SAMAs unique to the fire analysis were identified. For each fire zone, discuss the potential for SAMAs to address the unique cause of the fire risk, such as SAMAs to reduce the fire initiators, to improve fire detection or suppression, or to relocate components or cabling.

6. Provide the following with regard to the Phase 2 cost-benefit evaluations:

   a. For SAMA 3, Proceduralize Reactor Pressure Valve Depressurization When Fire Protection System Injection is the Only Makeup Source, indicate what failure events were included for the failure to provide late low pressure injection via the fire main.

   b. For SAMA 8, Automatic Feedwater Runback for ATWS, the percent reduction in dose risk and offsite economic cost risk (OECR) is much smaller than the reduction in CDF. The reduction in CDF is almost entirely in the low/early release category, which has a very small contribution to dose-risk and OECR. One might expect the reduction in CDF due to ATWS to impact high or medium release categories. Explain this apparent discrepancy.

   c. In the discussion of the costs for SAMA 8, it is implied that the cost estimate does not account for inflation. Clarify whether this cost estimate, or any other cost estimates, accounts for inflation.

   d. For SAMA 12, Containment Venting After Core Damage, the analysis shows very little risk reduction. Since this SAMA would reduce the releases for all drywell overpressure failure sequences, a more significant reduction in risk would be expected. Explain the reasons for the small risk reduction for this SAMA.

7. One of the Mark I plants considered in it's SAMA identification process (Section E.5.1.4) identified the following SAMAs as potentially cost-beneficial:

   a. Develop guidance/procedures for local, manual control of reactor core isolation cooling following loss of DC power.

   b. Procedures to control containment venting to avoid adverse impacts on emergency core cooling system.

   These SAMAs would appear to be applicable to SSES but are not among the Phase 2 SAMAs for SSES. Provide a brief statement regarding the applicability/feasibility of these alternatives for SSES, and a further evaluation (similar to those evaluations provided in the ER) if the alternative could be potentially cost-beneficial at SSES.

Susquehanna Steam Electric Station, Unit Nos. 1 and 2

cc:
Robert A. Saccone
Vice President - Nuclear Operations
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

Aloysius J. Wrape, III
General Manager - Performance
Improvement and Oversight
PPL Susquehanna, LLC
Two North Ninth Street, GENPL4
Allentown, PA 18101-1179

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.
Associate 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
Rachel Carson State Office Building
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

cc:

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

Mr. James Ross
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, D.C. 20006-3708