William H. Spence
President and Chief Nuclear Officer

PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 610.774.3683 Fax 610.774.5019
Whspence@pplweb.com

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

SUSQUEHANNA STEAM ELECTRIC STATION
AMENDMENT REQUEST NO. 305 TO UNIT 1 LICENSE NPF-14
AND AMENDMENT REQUEST NO. 276 TO UNIT 2 LICENSE NPF-22:
ONE-TIME EXTENSION OF TECHNICAL SPECIFICATION 3.8.1
PRA SUPPLEMENTAL INFORMATION **Docket Nos. 50-387**
PLA-6505 **and 50-388**

Reference: 1) PLA-6480, Mr. W. H. Spence (PPL) to Document Control Desk (USNRC), "Susquehanna Steam Electric Station Amendment Request No. 305 to Unit 1 License NPF-14 and Amendment Request No. 276 to Unit 2 License NPF-22: One Time Extension of Technical Specification 3.8.1 Allowable Completion Time for Offsite AC Circuits," dated March 24, 2009.

In accordance with the provisions of 10 CFR 50.90, PPL Susquehanna, LLC (PPL) submitted a request for amendment to the Technical Specifications (TS) for Susquehanna Units 1 and 2 in Reference 1.

During a teleconference on April 20, 2009 between PPL and the Nuclear Regulatory Commission (NRC) staff it was determined that additional probabilistic risk assessment (PRA) information was necessary to support the NRC acceptance review of Reference 1.

The attachment provides PPL's response to each of the specific PRA related questions discussed during the teleconference.

The supplemental PRA information contained herein for each of the responses does not affect the original no significant hazards consideration included in Reference 1.

Any questions regarding the basis or discussion associated with this response should be directed to Mr. D. L. Filchner - Nuclear Regulatory Affairs, at (610) 774-7819.

A001
NRR

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

Executed on: 4/30/09

W. H. Spence

Attachment: Responses to PRA Questions

Copy: NRC Region I
Mr. R. Janati, DEP/BRP
Mr. F. W. Jaxheimer, NRC Sr. Resident Inspector
Mr. B. K. Vaidya, NRC Project Manager

Attachment to PLA-6505

Responses to PRA Questions

NRC PRA QUESTION 1:

RG 1.200 is referenced by the licensee with regard to the technical adequacy of its PRA models. For its internal events analysis, the licensee has properly identified peer review results and disposition of open significant findings for this application, and an ASME standard evaluation and disposition of open significant findings. However, the submittal does not provide, as specifically required by RG 1.200 Section 4.2, any discussion as to which elements of the ASME standard are relevant to this application, what capability category is required for each of those relevant elements, and what capability category is achieved by their baseline PRA model.

PPL RESPONSE:

For the purposes of this application, all of the PRA attributes were considered to be relevant to the ST No. 20 Allowed Outage Time (AOT) submittal. A self-assessment was performed in July 2005 based on the Supporting Requirements of ASME Standard RA-Sa-2003 (and RA-S-2002), as well as associated NEI guidance and NRC staff positions related to the ASME Standard, as noted in Section 4.2 of the Amendment Request. This self-assessment was performed to meet ASME Standard RA-Sa-2003 Capability Category II. The majority of the ASME Supporting Requirements were determined to meet Category II.

The self-assessment also identified the ASME PRA Supporting Requirements that could be enhanced to meet the intent of the Supporting Requirements as stated in the ASME PRA Standard or the NRC's interpretation. Some items identified were equivalent to the then remaining B level findings from the Susquehanna model BWROG Peer Review per the NEI 00-02 Standard. The remaining items were new items that were considered desirable to meet Capability Category II of the ASME Standard. Attachment 5 presents the Self Assessment Gap items remaining open and discusses their impact on the Amendment Request analysis. Table 1 (below) identifies the Supporting Requirements associated with those open items.

TABLE 1: Self-Assessment Open Items – Impact on ST No. 20
Technical Specifications Submittal & ASME SRs

Description of Self Assessment Open Items	Impact on ST No. 20 TS Submittal	ASME Supporting Requirements
Include a discussion of dual unit effects in the event tree /success criteria notebook.	Dual unit concerns are discussed in Section 4.2 of the submittal.	IE-A10, SC-A4
Conduct interviews and walkdowns to verify that the model reflects the as-built, as-operated plant.	The systems modeled use design capacities. If there is a deviation from design, it has been justified. Interviews and walkdowns would help to refine the operating aspects of the systems modeled but would not appreciably impact the risk metric results.	SY-A4, SY-B6, SY-B8
Provide HEPs for flood isolation capability in model or provide rationale for their exclusion.	A Human Error probability (HEP) for flood isolation has conservatively not been included in the model. However, the highest F-V for a flood initiator in the base model is about one-half of one percent. Adding HEPs for flood isolation would reduce the flood F-V and will not significantly impact the risk metric results.	HR-A3, HR-D5, HR-G7
Add common cause mis-calibration for low pressure permissive for RHR and CS injection valves	A common cause mis-calibration for the low pressure permissive for RHR and CS injection valves is not currently modeled. This is not predicted to be impacting for the application for two reasons. First, the pressure switches being out of the calibration range but not failed will still allow the injection valves to open. Second,	HR-A3, HR-D5, HR-G7

Description of Self Assessment Open Items	Impact on ST No. 20 TS Submittal	ASME Supporting Requirements
	both core spray loops have a low pressure permissive bypass switch that can be manually activated from the control room if the core spray valves do not open at the correct pressure.	
Update the component data notebook to incorporate more plant-specific data evaluations, especially for high FV components.	The diesel generators have the highest F-V of any modeled component. The failure rate for diesel generators was based on plant specific data. HPCI is another relatively high F-V system. However, for this application, the risk metric results are relatively insensitive to changes in HPCI failure rates. The HPCI F-V for the base case is about 1% and for ST No. 20 OOS it is about 2%. Thus, changes in the HPCI failure rate would not significantly alter the overall risk results.	DA-A1, DA-A3, DA-C1, DA-C2, DA-C3, DA-C9
The HRA notebook should provide an assessment of the uncertainty in HEPs.	Adding HEP uncertainties will enhance the uncertainty analysis. The risk metrics for this application are "best estimate". Therefore, the uncertainties would not alter the risk metric results.	HR-D6, HR-G9

NRC PRA QUESTION 2:

Unlike the cited precedent licensing action from 2003, a quantitative risk analysis of fire events has been provided to support this application. The submittal does not provide any information of the quality or technical adequacy of the fire analysis used to develop the fire PRA results, other than to identify that it is based on Unit 1, only credits manual suppression, and assumes all equipment in a zone is damaged if not suppressed. There is insufficient information for the staff to make a determination that the risk results for fire are based on a technically adequate, comprehensive, current PRA model. RG 1.200 Section 1.2.4 and Table 3 identify the technical attributes applicable to fire PRA models used to support applications.

PPL RESPONSE:

General Approach

The fire risk was generated by using the internal events model, our cable & raceway database, and the fire frequency and screening criterion from the SSES IPEEE. The cable & raceway program was used to determine what equipment would be failed for an all-encompassing fire in each fire zone. It was assumed that the cables in the fire zone would fail open. Once the fire impacts for each fire zone were determined, the screening criterion was applied.

Screening Criterion

The screening criterion is:

> (HPCI or RCIC) and [2 Divisions of ADS and (2LPCI pumps in different divisions, or 2 CS pumps in different divisions, or 1 LPCI pump and 1 CS pump, may be in same division)]

If the screening criterion was met, the fire zone was considered to be screened out due to sufficient defense in depth. This screening criterion is appropriate since the core damage frequency due to a fire in a zone that met this criterion was calculated to be 5E-10, assuming a transient fire frequency. This is four magnitudes lower that the aggregate core damage fire frequency for all non-screened zones. Therefore, it is concluded that the screening criterion is valid.

Each fire zone that did not screen out had its fire core damage frequency calculated considering the equipment failed due to the fire. This calculation did not credit balance of plant (BOP) equipment since the cable and raceway database was not developed to assess the functionality of this equipment. Not crediting the BOP equipment is conservative since some of it may be functional after a fire.

This Technical Specification change is for ST No. 20. The other source of off site power is from ST No. 10. ST No. 10 is located outdoors in fire zone 0-00. A fire in this fire zone would only fail the two off site power sources and therefore cause a loss of off site power (LOOP). Since a fire in this zone only loses off site power without affecting other equipment, the screening criterion is met.

Quantification of Fire Risk

The fire risk for each zone is the product of the conditional probability of core damage or large early release, the fire frequency, the probability of the fire progressing to a large fire, and the probability of non-suppression.

The fire frequencies for each fire zone were obtained from the IPEEE. The probability of non-suppression is consistent with NEI 00-01, Nuclear Power Plant Fire Protection, for failing manual suppression. The fire analysis only used the probability of failing manual suppression.

The fire risk analysis did consider a fire-induced loss of off site power. If the loss of off site power was caused by a fire, no LOOP recoveries were credited.

Assumptions

An assumption for this analysis is that, if the fire in a zone containing cables affecting power from both ST No. 10 and ST No. 20 caused a LOOP with both off site power sources in service, the risk metrics (CDF and LERF) were doubled. The doubling is to account for the fact that with ST no. 20 out of service only one of the two off site power sources needs to fail in the fire (the source from ST No. 10) to cause a LOOP.

Another assumption is that all fires progress to a large fire i.e., a fire that damages all cables in the fire zone.

Model

The model used for the fire analysis is the model PPL used for the Extended Power Uprate and Licensing Renewal submittals with one exception. Recently, PPL discovered and documented in our corrective action process, a problem with our LERF model. The problem deals with the timing of the release. On an interim basis, until a final revised model is issued, this problem has been conservatively addressed by equating LERF to all large releases (large early, large intermediate and large late releases). This approach is being used until the timing of the large releases can be resolved. The only other changes to the model from the EPU submittal involve enhancements to the BOP fault trees. Since the fire analysis does not credit the BOP equipment these changes are of no consequence.

NRC PRA QUESTION 3:

RG 1.200 Section 4.2 requires identification and justification of any permanent plant changes not yet incorporated into the PRA model to assure the PRA represents the as-built and operated plant to support the application. The submittal does not provide any discussion of this aspect of the PRA model.

PPL RESPONSE:

The installed plant modifications that have not been credited in the risk model are the cooling modifications to the C and D Residual Heat Removal (RHR) pumps. The Unit 1 and Unit 2 C and D RHR pump motor oil coolers are cooled by both loops of Emergency Service Water (ESW). The model used for this submittal only credits one loop of ESW to these pumps. Not crediting the additional cooling to these pumps is conservative because the risk metric values will be higher than they would have been if the additional cooling was credited.