May 14, 2007

Mr. Britt T. McKinney
Sr. Vice President
and Chief Nuclear Officer
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) - SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (SSES 1 AND 2) - EXTENDED POWER UPRATE APPLICATION RE: REACTOR SYSTEMS TECHNICAL REVIEW (TAC NOS. MD3309 AND MD3310)

Dear Mr. McKinney:

In reviewing your letter dated October 11, 2006, concerning the request to increase the maximum steady-state power level at the SSES 1 and 2 from 3489 megawatts thermal (MWt) to 3952 MWt, the Nuclear Regulatory Commission staff has determined that additional information contained in the enclosure to this letter is needed to complete its review. These questions were discussed with your staff during a teleconference on May 7, 2007. As agreed to by your staff, we request you respond by June 15, 2007.

If you have any questions, please contact me at 301-415-1030.

Sincerely,

/RA/

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

Docket Nos. 50-387 and 50-388

Enclosure:
RAI

cc w/encl: See next page

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Accession Number: ML071290652 * RAI provided by memo. No substantive changes made.

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DATE	5/10/07	5/10/07	4/26/07	5/14/07

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REQUEST FOR ADDITIONAL INFORMATION

RELATING TO THE

APPLICATION FOR EXTENDED POWER UPRATE (EPU)

BOILING-WATER REACTOR (BWR) SYSTEMS TECHNICAL REVIEW

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (SSES 1 AND 2)

PPL SUSQUEHANNA, LLC

DOCKET NOS. 50-387 AND 50-388

The Nuclear Regulatory Commission (NRC) staff is reviewing the request from PPL Susquehanna, LLC (PPL, the licensee) to support the application of the EPU for SSES 1 and 2. The NRC staff has determined that additional information requested below will be needed to complete its review.

1. (General): Based on the Power Uprate Safety Analysis Report (PUSAR), it appears that all generic dispositions and specific analyses were performed for one SSES unit and applied to both. This implies that the two units are viewed as functionally congruent. Provide a description of major differences in operation, procedures, system configuration and flow, pressure, and level setpoints between SSES 1 and 2.

2. (Fuel System Design): Many of the methods specified have limited exposure ranges; PPL stated that the equilibrium reference core analyzed for the uprate application remained within these exposure ranges. Confirm that the currently loaded fuel that will remain in the core through the introduction of a full campaign of uprate fuel will also remain within the specified exposure ranges.

3. (Fuel System Design): The NRC staff is unable to determine from Technical Specification (TS) 5.6.5.b, "Core Operating Limits Report," and PUSAR Table 1-1, as to which methods specified perform which function. The NRC staff is also unable to determine whether each specified method is being used in a manner consistent with its NRC approval. Supplement both the Core Operating Limits Report (COLR) references list and Table 1-1 with a specific description of the function of each method and explaining why, in some cases, as many as six codes are required to perform a task or group of tasks.

4. (Nuclear Design): Provide plant- and cycle-specific information to show that the CASMO-4/MICROBURN-B2 code system was applied in a manner such that the predicted results for SSES 1 and 2 constant pressure power uprate analysis were within the range of the measurement uncertainties presented in EMF-2158(P)-A, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."

Enclosure

5. (Nuclear Design): Clarify whether the nuclear data file for CASMO-4/MICROBURN-B2 has been updated to include ENDF/B-VI.

6. (Thermal and Hydraulic Design): Clarify whether the fuel to be used for SSES 1 and 2 constant pressure power uprate operation will remain within the gadolinia and U-235 enrichment limits as specified in Condition 2 of the NRC staff safety evaluation approving EMF-2158(P).

7. (Thermal and Hydraulic Design): The NRC safety evaluation report authorizing the use of the revised SPCB critical power correlation, EMF-2209(P)(A), Rev. 1, indicates that conservatisms in the original correlation were reduced, based on the fuel length assumed by Framatome, Advanced Nuclear Power (FANP). The NRC staff, therefore, authorized a reduction in the Tong factor, based on the fact that prior assumptions about this factor "took on significantly larger values than expected," and authorized a step change in the omega function at the top node of the fuel assembly. The result was an adjustment to the critical power correlation based on the fact that the unrevised correlation calculated an "overly conservative" critical power in the top node of the fuel assembly.

   a. Since the blanket length in the fuel proposed for SSES 1 and 2 uprate is different than the authorized amount in the revision to EMF-2209(P)(A), explain what effect this difference has on the critical power correlation. Provide a technical basis justifying why the recent revision to EMF 2209(P)(A) remains adequately conservative.

   b. During a teleconference on February 6, 2007, with AREVA and PPL, representatives from Areva indicated that the revisions to the SPCB critical power correlation did not affect the critical power as determined for the uprate fuel. Provide a sample comparison of predicted critical power from one revision of EMF-2209(P)(A) to the next, specifically and quantitatively identifying the differences in Tong factor and step changes to the omega function, and demonstrating no change in predicted critical power. This comparison should be performed using a radially limiting fuel rod from each of beginning, middle, and end of cycle.

8. (Thermal and Hydraulic Design): Demonstrate that the statistical process used to determine the safety limit minimum critical power ratio is both statistically rigorous and conservative enough to be applied to the flatter radial power distribution required to achieve an EPU. For the limiting operating state point, characterize the Monte Carlo distribution of safety limit minimum critical power ratio values in terms of the shape of the distribution, its upper and lower tolerance limits, and the number of runs required to develop a 95% confidence level.

9. (Functional Design of Control Rod Drive System): Please discuss how PPL is addressing channel bow at SSES 1 and 2, and what effects channel bow may have on EPU operation.

10. (Residual Heat Removal System): The NRC staff accepted General Electric's (GE) approach in the Constant Pressure Power Uprate (CPPU) Licensing Topical Report (CLTR) regarding how the longer shutdown cooling (SDC) time does not have an effect on plant safety. The CLTR also indicates an expectation that licensees would conduct plant-specific SDC

evaluations at CPPU conditions to demonstrate that plants can meet the required cool down time. PPL has performed such an evaluation. Given the NRC staff's expectation that the plant will meet the required cool down time, provide the following information:

a. Identify conservatisms in the SDC analysis that would lead to the conclusion discussed above.

b. Clarify whether the realistically expected shutdown time would meet the design objective.

c. Discuss whether the design objective will change as a result of SDC cooling analysis.

11. (Standby Liquid Control System): Clarify why the same amount of boron is required to attain the required shutdown worth both before and after implementation of CPPU.

12. (Standby Liquid Control System): Anticipated transient without scram (ATWS) analysis indicates a peak lower plenum pressure of 1220 pounds per-square-inch absolute (psia), when all other pressures given are gauge (PUSAR Page 6-15). Clarify or confirm the following:

a. Was it PPL's intent to express the lower plenum pressure in pounds per square-inch gauge (psig), and is psia a typographical error?

b. What line losses are expected from the Standby Liquid Control System (SLCS) pump to the lower plenum injection point?

13. (ATWS): Provide graphs of the data presented in PUSAR Table 9-4 for the ATWS scenarios (1) main steam isolation valve closure and (2) pressure regulator open failure.

14. (ATWS): Confirm that the ATWS events selected for generic disposition in the CLTR remain bounding for ATWS events with a Framatome core.

15. (Station Blackout (SBO)): Provide Modular Accident Analysis Program (MAAP) benchmarking results to substantiate the conclusion that, "both codes, BWRSAR and MAAP, produced similar results for CPPU."

16. (SBO): Provide documentation that BWRSAR is an acceptable code to use when benchmarking MAPP.

17. (SBO): Discuss the initiating events used when evaluating an SBO Event using MAAP. Provide information about the event sequence that is analyzed using MAAP, and what systems are included in the MAAP SBO model.

18. (SBO): Review the guidance in Regulatory Guide 1.155, "Station Blackout," and confirm that the SBO analysis performed conforms to the guidelines established in Regulatory Position 3.2.

19. (SBO): Confirm that the makeup water inventory assumed in the SBO analysis conforms to the NUMARC 87-00 guidance for SBO.

20. (Fuel Storage): General Design Criteria (GDC) 62 is applicable to the NRC staff's review of the effect on the proposed CPPU on new and spent fuel storage. PUSAR Section 2.3 describes that the uprated fuel will geometrically fit in the current configuration. Verify that the discharged fuel will be equal to the pre-EPU decay power or be bounded by the current anaylsis to prevent criticality as required by GDC 62. If needed, describe and justify any changes.

The following RAIs are from PUSAR Section 9, "Reactivity Safety Performance Evaluations:"

21. On page 9-1, it states, "FANP evaluated the planned change to reduce the percent of rated power at which thermal limit monitoring is required. The evaluation was performed to support the beginning of the thermal limit monitoring at 23% of 3,952 MWt (CPPU rated power) for the ATRIUM-10 fuel." Please provide the method and justification used to obtain 23% of CPPU rated thermal power for beginning the thermal limit monitoring.

22. Please provide the basis for the following statement on Page 9-2: "the thermal power limit of 23% of CPPU rated thermal power for reactor pressure less than 785 psig is justified for the ATRIUM-10 fuel design." Is this conclusion applicable for the scenario with reactor pressure greater than 785 psi?

23. Regarding the threshold power for monitoring operating limits, in the second paragraph of Page 9-2, it states "These conclusions are cycle independent." Please provide the justification for this statement.

24. In Section 9.1.1, the operational limit minimum critical power ratio (OLMCPR) is determined as 1.34 for all CPPU cycle exposures. However, Table 9-2 lists higher OLMCPR values (e.g., 1.43 for Generator load rejection with a recirculation pump trip-out of service (OOS) and 1.38 for Feedwater controller failure max demand with turbine bypass valve (TBV)-OOS). Please explain why 1.43 and 1.38 were not used for OLMCPR. Is there an OLMCPR uncertainty included in determining this value?

25. For the loss of feedwater flow transient, please provide the decay heat model used in the analysis. EPU licensing topical report (ELTR)-1 suggested Decay heat 1979 ANS + 10% be used for this transient evaluation.

26. In Section 9.1.3.2, the reactor scram on low reactor water level (Level 3) is discussed. The analysis showed a Level 3 scram for CPPU for both 99 and 108% rated flow. Is this a requirement or just an expectation for loss of feedwater pump transient? Please explain the significance of this level scram event.

27. In Table 9-1, the parameters used for transient analysis are mostly 100% rated CPPU conditions. As stated in PUSAR Section 1.2.1, "Uprate Analysis Basis", the 2% power uncertainty factor is accounted for either statistically or through the inherent conservatism of the methodology. Please provide individual justification for the transients with 100% rated CPPU power according to the category of "statistically" or "inherent conservatism."

28. For rod withdraw error events, please explain how the OLMCPRs were obtained without the critical power ratio (CPR). In the updated final safety analysis report (UFSAR), Tables 15C.0-1 and 15D.0-1, the CPR is listed for this event. Please provide the CPR and explain any differences for these two analyses (pre-CPPU and CPPU).

29. For rod withdraw error (RWE) events, is the RBM setpoint (111%) a typical value or a conservative one? How does the setpoint affect the results? Why was the RBM not credited in the pre-CPPU analysis but is credited in the CPPU analysis? Please also provide the LHGR increase in the analysis. How do the two transients (RWE and RWE with turbine bypass failure/OOS) justify to be non-limiting compared to other transients in Table 9-2 regarding safety margin increase?

30. In the UFSAR, Table 15C4.9-2 and 15D.4.9.2, the control rod drop accident (CRDA) analysis shows the peak deposited enthalpy for CLTP. The values provided (249.7 calories per gram (cal/gm) for Unit 1 and 269.4 cal/gm for Unit 2) are approaching the acceptance criterion of 280 cal/gm. For the CPPU analysis performed in PUSAR Section 9.2, please provide the peak fuel enthalpies for both units and justify how they meet the acceptance criterion (280 cal/gm).

31. For feedwater controller failure max demand with TBV-OOS, what is the maximum vessel pressure at the bottom of the vessel for this transient? In Figure 9-24, the dome pressure is approaching the TS limit of 1325 psig and American Society of Mechanical Engineers peak vessel limit of 1375 psig.

32. For feedwater controller failure max demand with TBV-OOS, the main steam relief valve (MSRV) flow is greater in high setpoint (Fig. 9-26) while the MSRV position is less (Figure 9-25). This is opposite of the feedwater controller failure max demand without TBV-OOS (Figures 9-18 and 9-19). Please explain.

33. For Tables 9-3 and 9-4, please provide values for CLTP ATWS for comparison. For CPPU calculation, explain how uncertainty of power was considered in the analysis since the power level is 100% rated? Also, explain why the number of safety relief valve-OOS is zero for this analysis?

34. Please explain the steam flow in Figure 9-1 between time (t)=0.7 seconds and t=1.5 seconds since there was no relief flow during that period. Please provide any load reject (or turbine trip) transient plant data for steam flow and reactor pressure available for SSES or similar BWR4 plants.

35. In PUSAR Section 9.3.2, “Station Blackout”, it is stated that the MAAP computer was used for this analysis. According to Table 1-1, “Computer Codes Used for CPPU,” the MAAP code is not approved by the NRC. Please provide an explanation why the use of the MAAP code for CPPU is justified. The BWRSAR code is not listed in Table 1-1, what is the approval status for this code?

36. In the first paragraph of Section 9.3.3 (Page 9-6), it states, “The core design necessary to achieve CPPU operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression.” Please provide a detailed explanation of what this statement means.

   a. On PUSAR Page 9-7 regarding the impact of ATRIUM-10 fuel on ATWS, it states “ fuel design differences are small compared to the 0.3 to 0.5 decay ratio variation associated with the various plant configurations, loading patterns…." Please provide justification for this statement.

b. Provide comparative data (EPU vs. pre-EPU) to substantiate the conclusions drawn with regard to the impact of ATRIUM-10 fuel on ATWS/Stability.

37. In Figure 9-8, relative feed flow appears to be linear. Is there any level control in this calculation?

38. In Figure 9-29, please provide the reactor pressure plot for the Pressure Regulator Downscale Failure transient.

39. It appears that the transients analyzed for CPPU in PUSAR Table 9-2 are not complete compared to Table E-1 in ELTR-1, which provides the minimum set of transients suggested to be evaluated by GE. Please provide the justification for not evaluating the "turbine trip, bypass failure, with scram on high flux" transient (Number 13 in Table E-1).

The following RAIs pertain to the emergency core cooling system loss-of-coolant accident (ECCS-LOCA) analysis and the LOCA Analysis Report, EMF-3242(P):

40. In PUSAR Section 4.2.3, it states, "CPPU has no effect on the core spray distribution in the reactor vessel." For LOCA, CPPU has a flatter power distribution and higher decay power. Shouldn't these points affect the spray flow distribution due to different pressure distribution within core? Please provide justification for your answer.

41. On Page 1-1, the maximum extended load line limit analysis (MELLLA)+ was mentioned as one of the initial conditions used in the analysis, and the limiting break (summarized in Page 2-1) was this MELLLA+ domain point. This operation domain has not been approved. Please justify that the chosen initial condition bounds other initial flow conditions in the MELLLA domain.

42. On Page 1-2, it states, "Even though the limiting break will not change with exposure…." Is this statement referring to limiting break characteristics? If not, please clarify "the limiting break." On Page 2-2, please justify: "Fuel parameters that are dependent on exposure (e.g., stored energy, local peaking) have an insignificant effect on the reactor system response during LOCA." Please clarify the term "reactor system response."

43. In Figures 4-3 and 4-4, it looks like the nodal length is not uniform throughout the channel. State whether the nodal length is reflected on these diagrams? If yes, in Figure 4-4, the peak power is located in the smallest node which could cause an inaccurate void calculation. Please justify the peak power being located in the smallest node.

44. In Table 5.1, please explain why there are still two low-pressure coolant injections (LPCIs) left for SF-LPCI and SF-LOCA since BWR 4 plants have 4 LPCIs.

45. In Table 6.9, the limiting break for the 80 million pounds-mass per hour (Mlbm/hr) case is 1.0 Double Ended Guillotine (DEG) pump suction and for 108 Mlbm/hr is 0.8 DEG pump suction. According to the topical report (EMF-2361P), limiting break occurs at 0.8 DEG. Please provide explanations for the effects of discharge coefficient on the final peak cladding

temperature (PCT). Please also provide the equations for break flow calculation with discharge coefficient (Cd).

46. The limiting break results for two-loop operation (TLO) show maximum local metal-water reaction (MWR) of 0.68% and a maximum planar MWR of 0.28%. Based on engineering judgment stated in Section 6.1, the clad MWR would be less than 1.0%. Please provide any assumptions considered for this engineering judgment. According to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) report, SSES's core wide hydrogen generation must be less than 0.2% of the hypothetical amount because of SSES's hydrogen recombiner capacity. Please provide analysis and assumptions to justify how the MWR acceptance criterion (0.2%) is met in detail. Similar issue applies to single-loop operation (SLO) limiting results.

47. In PUSAR Section 4.3, it states, "An overly conservative assumption used in the pre-CPPU analysis was removed for the CPPU analysis." Is this assumption referring to "No LPCI to BL" in Table 6.10 in the LOCA report? If not, please provide details for this assumption. Why does the CPPU calculation with the same nonconservative assumption still result in a lower PCT (1914 degrees Fahrenheit (°F) vs. 1945 °F)? Please also provide the pre-CPPU limiting PCT results without this assumption, including the initial core flow condition used in the calculation. Also provide the limiting break characteristics in the pre-CPPU LOCA analysis.

48. In the typical LOCA calculation, there are two peaks on the PCT plot. The first peak occurs at the early blowdown phase due to transition to film boiling (dryout). The second peak occurs at the refill/reflood phase due to uncovered core heat up. In SSES's PCT plot, two peaks are shown in SLO LOCA (Figure 8.27) but only one peak is shown at the end of the refill/reflood phase for TLO (Figure 6.27). Please explain this deviation in detail.

49. In the SLO analysis, why is the SLO multiplier only applied in the HUXY code analysis? The heat input should also be applied in RELAX code to generate consistent thermal hydraulic conditions (heat transfer coefficients) for the HUXY code calculation. This inconsistency could result in different limiting break characteristics. Please justify this approach.

50. In the SLO LOCA analysis, the limiting PCT is 1686 °F. The multiplier of MAPLHGR is established so that the limiting PCT for SLO is less than the limiting PCT for TLO. The limiting PCT for TLO is 1803 °F according to Table 8.4. Since there is approximately a 120 °F difference, why can't the multiplier be higher so both limiting PCTs are closer? Are there additional constraints or margins for this multiplier beyond the PCT?

The following RAIs pertain to the MAPLHGR Report, EMF-3243(P):

51. Please explain why the limiting PCT of 1844 ° F is listed in Table 2.1 (and on Page 4-13 of the PUSAR) is different from the one listed in Table 6.10 in the LOCA Analysis Report, EMF-3242(P), which has 1803 ° F.

52. On Page 4-4 of the MAPLHGR report, the recirculation discharge isolation valve was mentioned as being credited in the LOCA calculation relative to the previous calculation. Please provide the explanation and justification for this credit.

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