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JUN 14 2007

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION PROPOSED LICENSE AMENDMENT NO. 285 FOR UNIT 1 OPERATING LICENSE NO. NPF-14 AND PROPOSED LICENSE AMENDMENT NO. 253 FOR UNIT 2 OPERATING LICENSE NO. NPF-22 EXTENDED POWER UPRATE APPLICATION RE: REACTOR SYSTEMS TECHNICAL REVIEW REQUEST FOR ADDITIONAL INFORMATION RESPONSES PLA-6209

Docket Nos. 50-387 and 50-388

References: 1) PLA-6076, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 for Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate," dated October 11, 2006.

2) Letter, R. V. Guzman (NRC) to B. T. McKinney (PPL), "Request for Additional Information (RAI) - Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2) - Extended Power Uprate Application Regarding Turbine Generator Review (TAC Nos. MD3309 and MD3310)," dated May 14, 2007.

3) USNRC Letter, Nerses, Victor (NRC) to Byram, Robert G. (PPL) "Review of Susquehanna Steam Electric Station, Units 1 and 2, Individual Plant Examination Submittal - Internal Events (TAC NOS. M74478 and M74479)," dated August 11, 1998.

4) PLA 5980, R. A. Saccone (PPL), to USNRC, "Proposed Amendment No. 282 to Facility Operating License NPF-14: Proposed Change to Technical Specification 2.1.1.2 MCPR Safety Limit Supplemental Information," dated November 2, 2005.

Pursuant to 10 CFR 50.90, PPL Susquehanna LLC (PPL) requested in Reference 1 approval of amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Operating Licenses (OLs) and Technical Specifications (TSs) to increase the maximum power level authorized from 3489 Megawatts Thermal (MWt) to 3952 MWt, an approximate 13% increase in thermal power. The proposed Constant Pressure Power Uprate (CPPU) represents an increase of approximately 20% above the Original Licensed Thermal Power (OLTP).

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The purpose of this letter is to provide responses to the Request for Additional Information transmitted to PPL in Reference 2.

The Attachments contain the PPL responses.

Attachment 1 contains AREVA NP, Inc. and General Electric Company proprietary information. As such, AREVA NP, Inc. and General Electric Company request that they be withheld from public disclosure in accordance with 10 CFR 2.390 (a) 4 and 9.17 (a) 4. Affidavits supporting this request are contained in Attachment 3. Attachment 2 contains a non-proprietary version of the responses.

There are no new regulatory commitments associated with this submittal.

PPL has reviewed the "No Significant Hazards Consideration" and the "Environmental Consideration" submitted with Reference 1 relative to the Enclosure. We have determined that there are no changes required to either of these documents.

If you have any questions or require additional information, please contact Mr. Michael H. Crowthers at (610) 774-7766.

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

Executed on: 6-14-07

Richard Hagedorn
for B. T. McKinney

Attachment 1: Proprietary Version of the Request for Additional Information Responses
Attachment 2: Non-Proprietary Version of the Request for Additional Information Responses
Attachment 3: AREVA NP, Inc. and General Electric Company Affidavits

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

Attachment 2 to PLA-6209
Non-Proprietary Version of the Request for Additional Information Responses

NRC Question 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 Units 1 and 2.

PPL Response:

The SSES units are analyzed using a single model to represent both units. Based on satisfactory results obtained from SSES transient model development and benchmarking, it was determined that the SSES units were similar enough to not warrant separate modeling of the reactors. Therefore, from a geometry standpoint, the units are treated as being the same.

There are no major differences in the operation and procedures for the SSES units that would impact the safety analysis.

The SSES units have the same number of major components with similar performance characteristics. Each unit has the following major components: three feedwater heater strings, three turbine-driven feedwater pumps, two reactor recirculation pumps, one reactor core isolation cooling pump, sixteen safety relief valves, and two main steam isolation valves per each of the four main steam lines. Each unit also has the same number and type of pumps in the ECCS including one HPCI pump, four core spray pumps, and four residual heat removal pumps. Both units also have Siemens main turbines. Therefore, the system performance of both units is similar.

The normal operating conditions with respect to flow, pressure, and level setpoint are the same for both units. Both units operate using the same pressure regulator and level setpoints. Current steam flow for both units is approximately 14.4 Mlb/hr at their current rated core powers of 3489 MWt. Under CPPU conditions, it is anticipated that both units will continue to have similar steam flows and have the same pressure regulation and level setpoints.

Transient response is governed by system geometry, system performance, and core design. Currently, both SSES cores consist of full cores of ATRIUM-10 fuel with scatter loading. Unit capacity factor which is directly related to the unit's operating history (maintenance outages, unplanned shutdown, scrams, etc.) has one of the largest impacts on core designs. The core design for each unit accounts for its unit specific capacity factor.

Each unit has a core design that is tailored to compensate for its individual operating history. Due to the uniqueness of each unit's core design, the transient analysis for each unit is performed specific to that unit's core design.

In summary, a single model is used to represent the geometry and predict system performance for both SSES units.

NRC Question 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.

PPL Response:

The currently loaded fuel that remains in the core through the introduction of power uprate will be verified to remain within the approved exposure limit during the core design process. Exposure is also monitored during the cycle. Projected transition cycles indicate that exposure limits can be met.

NRC Question 3:

(Fuel System Design): The 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 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.

PPL Response:

The response to this NRC Question will be provided by June 22, 2007 after the NRC audit scheduled for June 5-8 at AREVA NP, Inc. is completed. This date has been discussed with the NRC.

NRC Question 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."

PPL Response:

The response to this NRC Question will be provided by June 22, 2007 after the NRC audit scheduled for June 5-8 at AREVA NP, Inc. is completed. This date has been discussed with the NRC.

NRC Question 5:

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

PPL Response:

The CASMO-4/MICROBURN-B2 methodology being applied to the CPPU analysis continues to use the original nuclear data file which was used in the analysis supporting the topical report (EMF-2158(P)(A)). The nuclear data library has not been updated to ENDF/B-VI.

NRC Question 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 staff safety evaluation approving EMF-2158(P).

PPL Response:

The fuel used for SSES 1 and 2 under CPPU operation will remain within the gadolinia and U-235 enrichment limits specified in EMF-2158(P)(A).

NRC Question 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 February 06, 2007 teleconference 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.

PPL Response 7a:

EMF 2209(P)(A) and the NRC Safety Evaluation Report (SER) did not authorize a specific fuel blanket length. The presence of the blanket rather than the length of the blanket is the important parameter. The NRC SER for EMF 2209(P)(A) provides general guidance regarding "in the region of the uranium blanket at the top six inches of the fuel" and, in fact, continues later in the SER to address "The insertion of natural uranium in the last 6 to 12 inches." The report did not identify absolute blanket lengths. The analysis supporting EMF-2209(P)(A) is impacted only by the presence of a natural blanket, not the particular length of the blanket. The technical basis for EMF-2209(P)(A) is unchanged. The technical basis shows adequate conservatism and that the conservatism is not impacted by the specific length of the blanket.

PPL Response 7b:

The requested comparisons are illustrated in the following table for two different flow rates of an assumed hot assembly. No attempt was made to assure that the powers selected would be at or above a particular operating power limit, rather power was selected that would provide a representative means of comparing what might happen to the bundle at different points in the cycle for different assumed flows. Pressure was set at 1049 psia and f-effective was set at 1.039.

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At the nominal flow condition (.103 Mlbm/h), the beginning of cycle (BOC) condition shows the largest changes that occur with respect to CPR, Tong Factor and Omega for the predicted location of dryout. At the low flow condition (0.014 Mlb/hr), the CPR values are nearly the same when using Revision 1 and Revision 2. The predicted axial location of dryout changes for one of the three exposure conditions for nominal flow and for low flow. Different values of Tong and Omega are observed due to the difference in the predicted axial location of dryout. The Table demonstrates essentially no change in predicted critical power.

NRC Question 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 CPPU. 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.

PPL Response 8:

The response to this NRC Question will be provided by June 22, 2007 after the NRC audit scheduled for June 5-8 at AREVA NP, Inc. is completed. This date has been discussed with the NRC.

NRC Question 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.

PPL Response 9:

PPL has implemented a channel management action plan to monitor and assess the impact of channel bowing on control rod performance (see Reference 4). Actions are taken based on the results of the control rod performance tests. Channel bowing can result in an unacceptable operability condition that may ultimately require the replacement of fuel channels in the affected control cells to regain acceptable control rod performance. The susceptible fuel channels are planned to be replaced with new 100 mil Zr-4 fuel channels that will have better resistance to channel bow before CPPU is implemented.

Some 100 mil Zr-2 channels will be left in SSES Unit 1 peripheral control cells. However, these channels are not expected to present any channel bow issues since they are expected to remain below cell friction thresholds based on assembly exposures and control histories.

NRC Question 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 Susquehanna has performed such an evaluation. Given the 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.

PPL Response 10:

10 (a)

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]] The Technical Specification normal operating limit is 85°F. Typically, the temperature is less than the TS limit.

10 (b)

The heat removal rate from the heat exchangers is dependent upon the RHR service water inlet temperature. Lower temperatures will reduce the time to achieve the design objective temperature. Parametric studies were not specifically conducted for this task to determine at what RHR service water temperature the design objective would be achieved. [[

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10 (c)

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NRC Question 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.

PPL Response 11:

The same amount of boron is not required to attain the required shutdown worth both before and after implementation of CPPU. As described in the NRC Safety Evaluation Report for Technical Specification Amendments 240 and 217, the amount of boron currently required is based on operation at the CPPU power level and is not based on the pre-CPPU power level. Thus, the current amount of boron required has already been changed to implement CPPU.

The Standby Liquid Control System (SLCS) analysis performed for the CPPU core design indicated that the requirements of the analysis could be met with a boron concentration of 660 ppm. The SLCS analysis is performed on a cycle specific basis to confirm that the 660 ppm requirement is still adequate for maintaining shutdown. Based on the analyses performed, it is concluded that the boron concentration required to maintain shutdown before and after implementation of CPPU does not have to be changed.

NRC Question 12:

(Standby Liquid Control System): Anticipated Transient Without Scram (ATWS) analysis indicates a peak lower plenum pressure of 1220 pounds per square inch (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 1220 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?

PPL Response 12:

12 (a)

The lower plenum pressure of 1220 psia is correct. To provide consistency with the General Electric verified analysis, it was not converted to psig in Section 6.5 of Attachment 4 (Reference 1).

12 (b)

It is expected that the line losses with one SLCS pump in operation will be approximately 51.3 psi with an elevation head difference of -19.1 psi (the SLCS pump sits higher than the lower plenum region of the reactor).

NRC Question 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.

PPL Response 13:

The tables below provide the sequence of events for limiting the main steam isolation valve closure (MSIVC) and pressure regulator failure – open (PRFO) ATWS events.

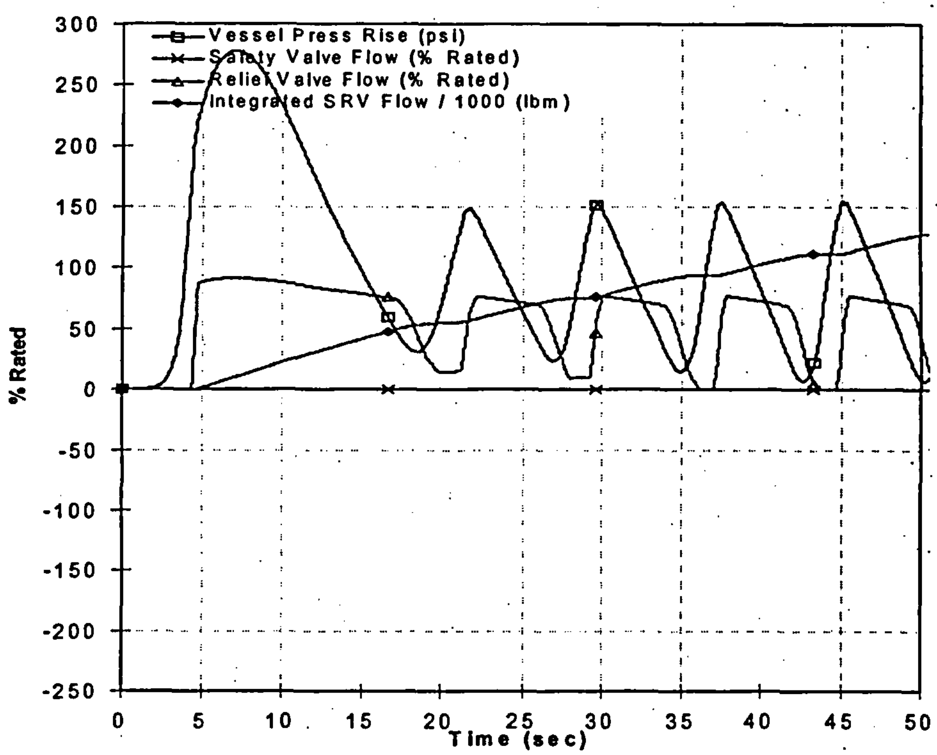
In addition, the following graphs provide the trends for: 1) the change in vessel dome pressure, relative to the initial value; 2) peak cladding temperature; 3) peak suppression pool temperature; and 4) peak containment pressure for the limiting MSIVC and PRFO events.

Sequence of Events For Limiting MSIVC ATWS

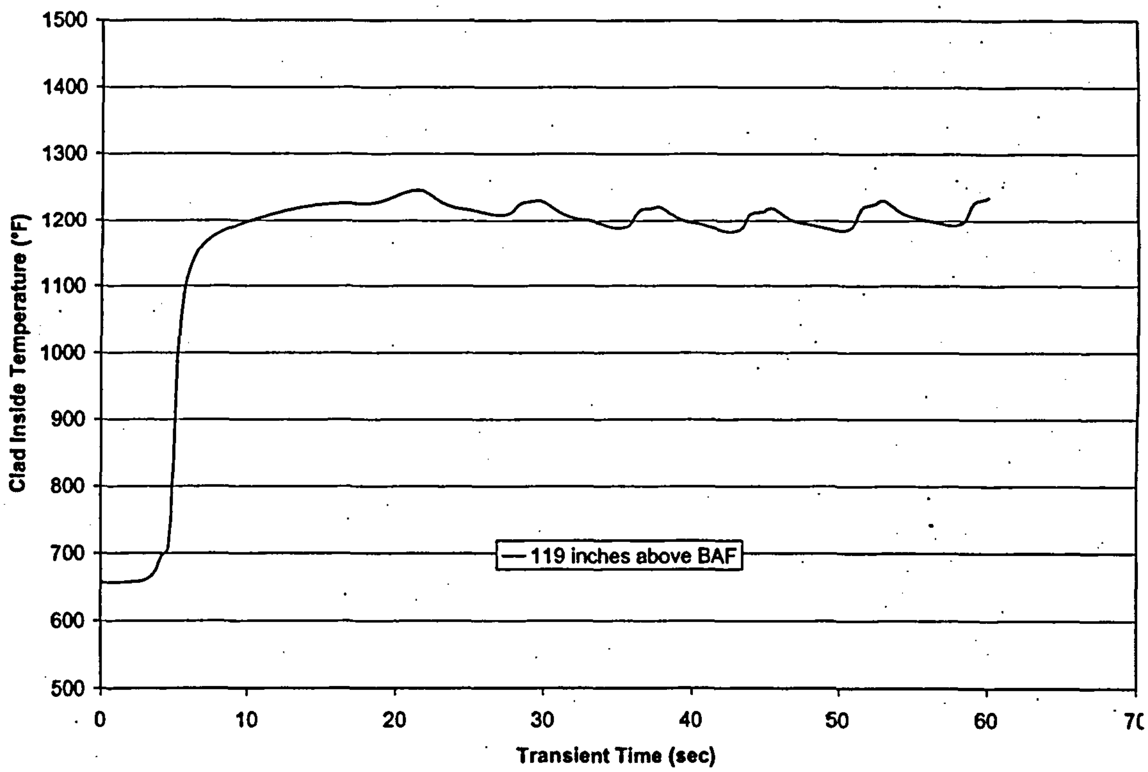
Response	Event Time (sec)
MSIV Isolation Initiates	0.0
MSIVs Fully Closed	4.0
Peak Neutron Flux	4.02
High Pressure ATWS Setpoint	4.17
Opening of the First Relief Valve	4.34
Recirculation Pumps Tripped	4.70
Peak Heat Flux Occurs	4.81
Peak Vessel Pressure	6.84
Feedwater Reduction Initiated (feedwater stopped completely)	104
SLCS Pumps Start	124
RHR cooling initiated (first train/second train)	1100 / 1600
Peak Suppression Pool Temperature	1508
Hot Shutdown Achieved (Neutron flux remains <0.1%)	1618

Sequence of Events For Limiting PRFO ATWS

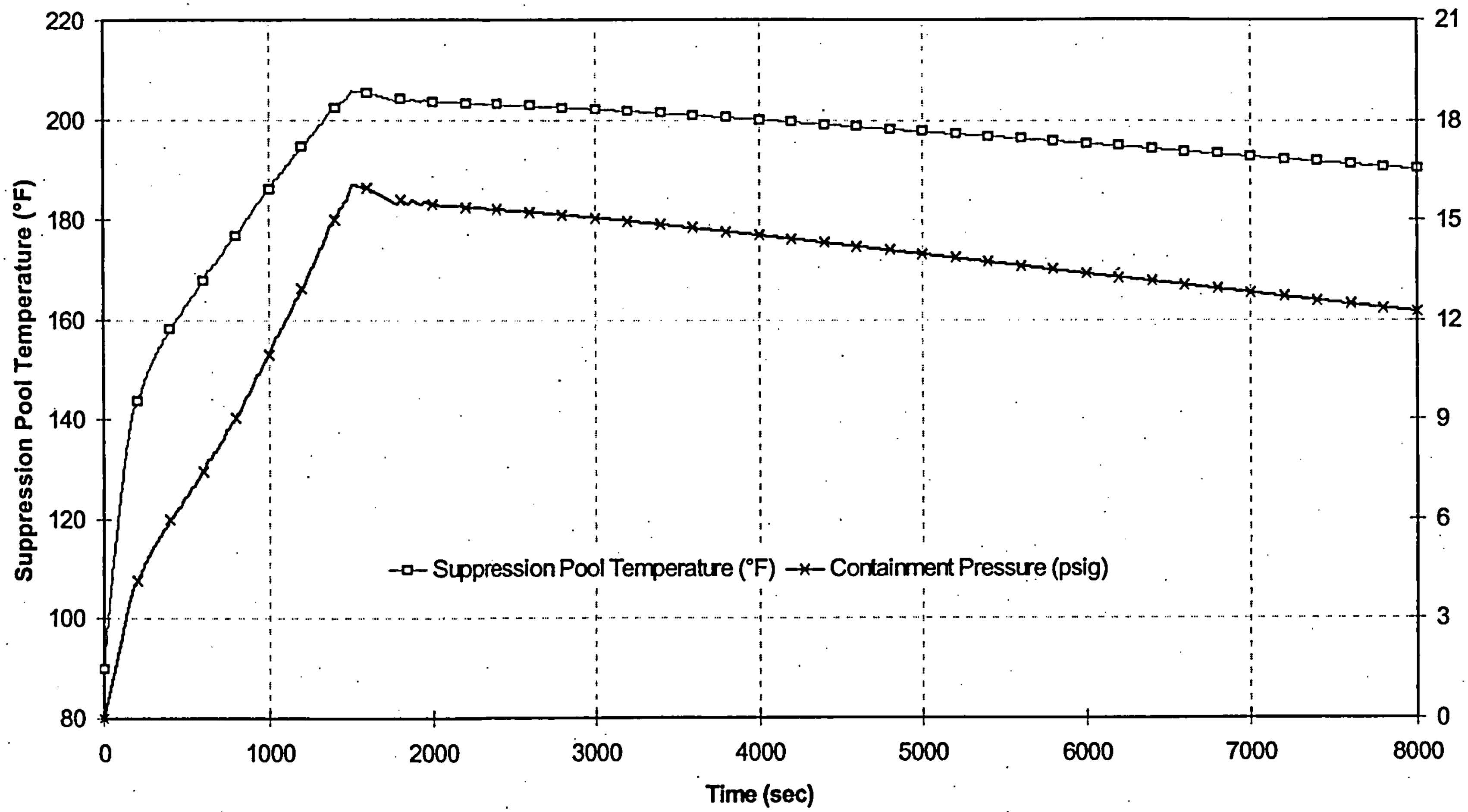
Response	PRFO
Turbine Control and Bypass Valves Start Open	0.11
MSIV Closure Initiated by Low Pressure	12.6
Peak Neutron Flux	16.60
MSIVs Fully Closed	16.6
High Pressure ATWS Setpoint	18.70
Opening of the First Relief Valve	18.92
Peak Heat Flux Occurs	19.23
Recirculation Pumps Tripped	19.23
Peak Vessel Pressure	21.32
Feedwater Reduction Initiated (feedwater stopped completely)	118
SLCS Pumps Start	139
RHR cooling initiated (first train/second train)	1100 / 1600
Peak Suppression Pool Temperature	1959
Hot Shutdown Achieved (Neutron flux remains <0.1%)	1656



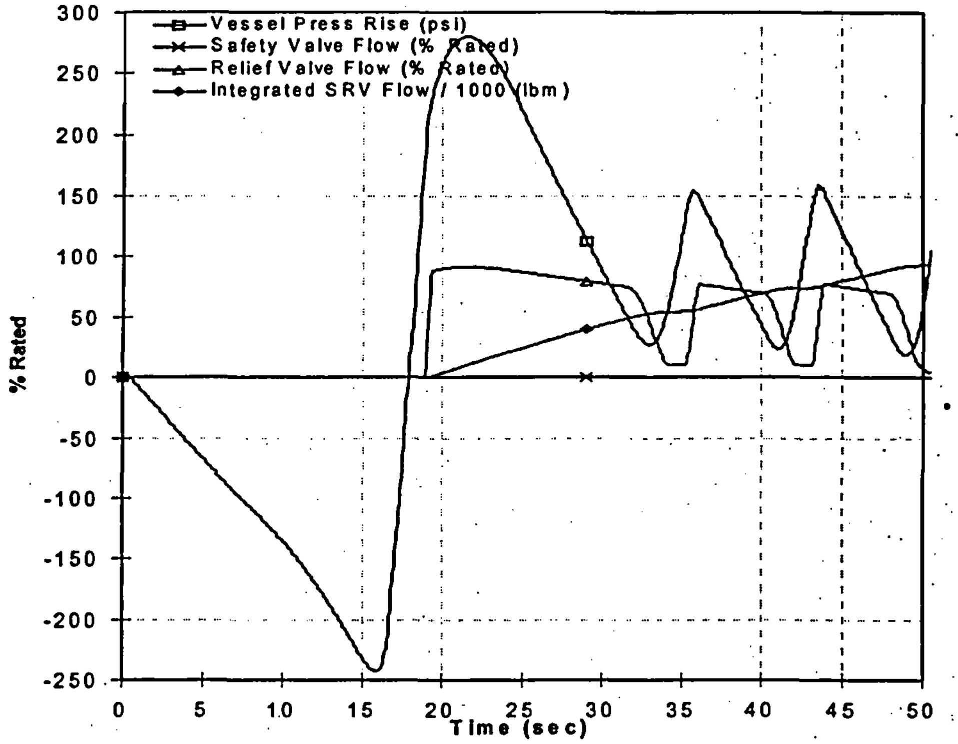
Peak Vessel Dome Pressure For Limiting MSIVC ATWS



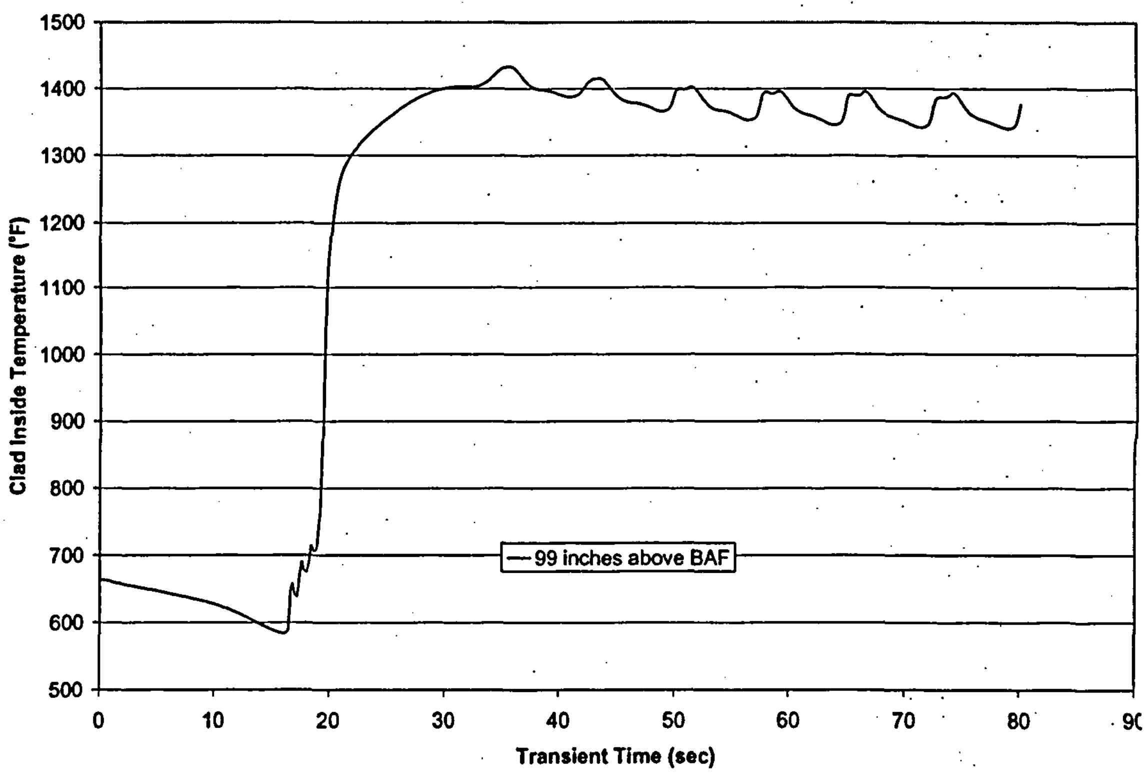
Peak Cladding Temperature For Limiting MSIVC ATWS



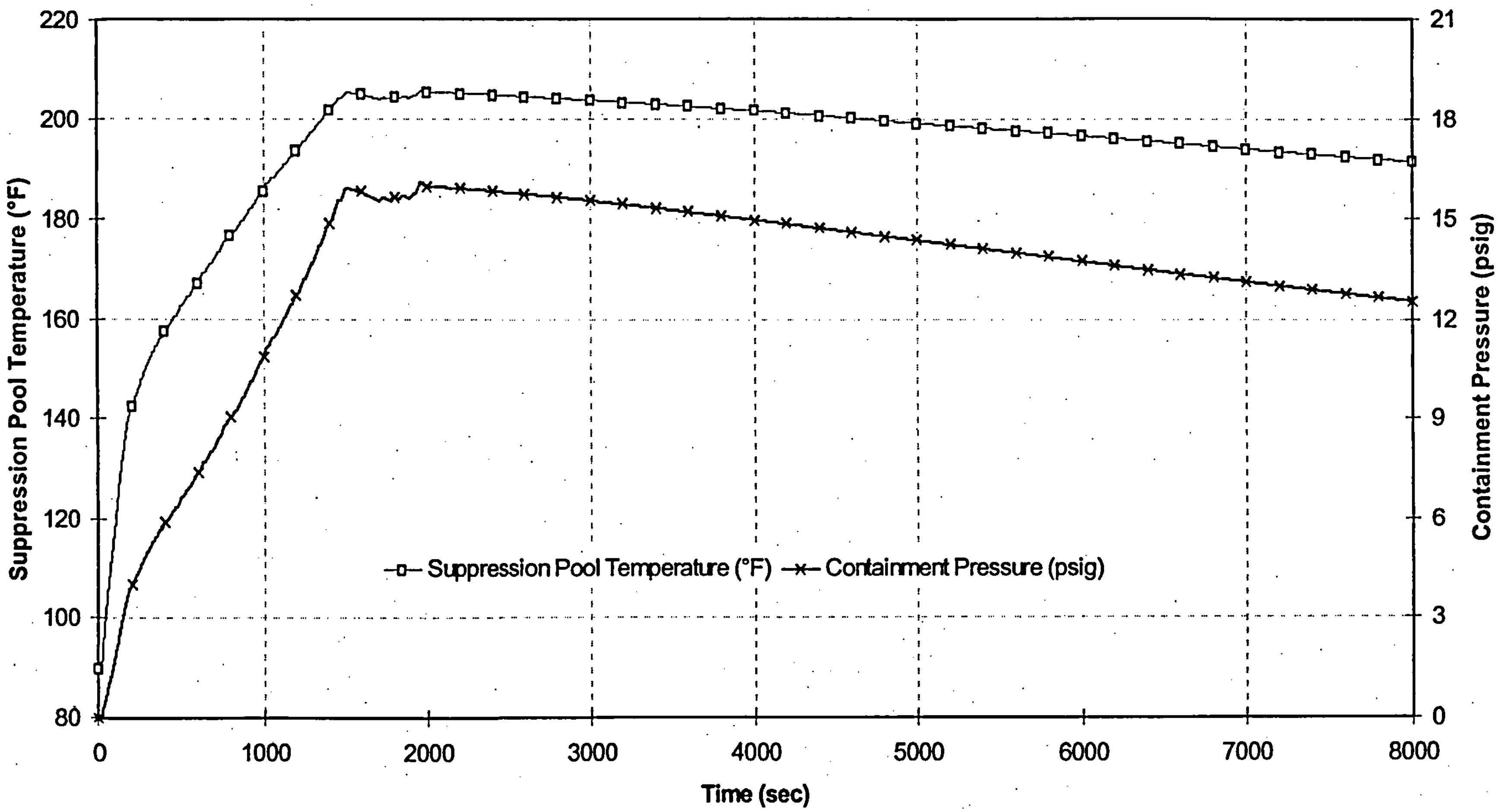
Peak Containment Parameters For Limiting MSIVC ATWS



Peak Vessel Dome Pressure For Limiting PRFO ATWS



Peak Vessel Dome Pressure For Limiting PRFO ATWS



Peak Containment Parameters For Limiting PRFO ATWS

NRC Question 14:

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

PPL Response 14:

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See the response to Question #13 above for additional information regarding the MSIVC and PRFO events.

NRC Question 15:

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

PPL Response 15:

The current SBO evaluation uses the BWRSAR computer code to analyze the reactor and containment response to the event. With the reactor water level being maintained within normal operating limits, this evaluation calculates the peak drywell pressure, peak suppression pool temperature, and the amount of makeup water required for the reactor during a four-hour event. The results from both codes for CPPU operation are:

	BWRSAR	MAAP
Peak Drywell Pressure (psig)	10.8	11.3
Peak Suppression Pool Temperature (°F)	195.2	156.6
Makeup Water (gal)	< 135,000	132,000

The BWRSAR results differ from the MAAP results in the behavior of the containment parameters. Multiple cases were run for both codes with varying input parameters (i.e., pump seal leakage, no pump seal leakage, etc.). Each of the individual MAAP cases show higher drywell pressure at 4 hours than the corresponding BWRSAR case. However, each MAAP case shows that the suppression pool heats up more slowly than the corresponding BWRSAR case. This relative behavior of the two models in the prediction of containment parameters has been noted in several Probabilistic Risk Assessment (PRA) calculations. The peak suppression pool temperature is much lower for the MAAP case because the MAAP code can model passive heat sinks in containment better than the BWRSAR code.

NRC Question 16:

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

PPL Response 16:

The BWRSAR code was developed by Oak Ridge National Laboratory and funded by the NRC. The current SSES Individual Plant Examination (IPE), which used the BWRSAR code, was approved by NRC (Reference 3). This approved model was used to perform the current SSES SBO analysis. MAAP is an industry-accepted code used for thermal-hydraulic analysis for many IPE submittals to the NRC, especially with recent Extended Power Uprate submittals. The MAAP code has been used by other utilities for their SBO analysis (ML062690086).

NRC Question 17:

(SBO): Discuss the initiating events used when evaluating a 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.

PPL Response 17:

The initiating events for the current SBO evaluation are not revised for the MAAP code analysis. The analysis assumes as the initiating event, a complete loss of alternating current electric power to the essential and nonessential switchgear buses.

The event sequence for the current SBO evaluation is not revised for the MAAP code analysis. The sequence is: reactor scram from 100% power, reactor vessel level is maintained between Level 4 and Level 7 by RCIC with suction from the Condensate Storage Tank (CST), recirculation pump seal leakage of 100 gpm, and reactor pressure maintained through the duration of the four-hour event using SRVs. The systems used in the current limiting SBO evaluation are the same for the MAAP SBO analysis: RCIC, SRVs, Containment Instrument Gas, CST, and 250 and 125 VDC systems.

NRC Question 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.

PPL Response 18:

The guidelines of Regulatory Guide 1.155, Regulatory Position 3.2 were reviewed and the SSES analysis performed for CPPU meets these guidelines. In addition to the statements made in PUSAR Section 9.3.2, no new operator actions are needed for the event. CPPU operation does not affect the response time for operators to perform any manual actions, such as equipment load shedding.

NRC Question 19:

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

PPL Response 19:

The MAAP code accurately calculates the amount of makeup water to the vessel. The current SSES SBO evaluation uses the BWRSAR code to calculate the required makeup for the event. Therefore, the general guidance for calculating the amount of makeup water from NUMARC 87-00 was not needed for the SBO evaluation.

NRC Question 20:

(Fuel Storage): General Design Criteria (GDC) 66 is applicable to the staff's review of the affect 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 discharge fuel will be equal to the pre-EPU decay power or be bounded by the current analysis to prevent criticality as required by GDC 62. If needed, described and justify any changes.

PPL Response 20:

The criticality safety analyses for both new and spent fuel storage performed for SSES are independent of core power. These criticality safety analyses specify maximum lattice enrichment and minimum gadolinia loadings that must be met to remain within the bounds of the analyses. The fuel bundle designs determined for the CPPU core design meet the requirements of the criticality safety analyses. Bundle designs are verified on a cycle specific basis to ensure that they meet the requirements of the new and spent fuel storage criticality safety analyses.

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

NRC Question 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.

PPL Response 21:

The evaluation considers critical power data at low flows for a large number of different fuel designs and considers assembly flow variation as a function of power and pressure drop. LHGR and APLHGR limits are compared to values of LHGR and APLHGR that would be expected at the low flow conditions. The evaluation process confirms use of

23% of CPPU rated thermal power for the beginning of the thermal limit monitoring is appropriate. This justification is based on showing that the radial peaking factor that might be associated with a lower bound on assembly critical power (when the downcomer and bypass water level exceeds the active fuel length) is both a large value and a value similar to values that are reported for other BWRs.

NRC Question 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?

PPL Response 22:

The proposed thermal power limit of 23% of CPPU power is a Safety Limit defined in the SSES Unit 1 and 2 Technical Specifications (TS) 2.1.1.1. It applies to reactor conditions with the reactor steam dome pressure < 785 psig or core flow < 10 million lbm/hr. Per Technical Specifications, with reactor steam dome pressure ≥785 psig and core flow ≥10 million lbm/hr, the MCPR Safety Limit (MCPRSL) specified in TS 2.1.1.2 is required to be met. Under the latter conditions, the critical power correlation is directly useable to ensure that the MCPRSL is not violated. Therefore, the conclusion is applicable when the pressure is ≥785 psig and core flow is < 10 million lbm/hr. If dome pressure is ≥785 psig and core flow is ≥10 million lbm/hr, the MCPRSL requirement of TS 2.1.1.2 is required to be met and the thermal power limit requirement of TS 2.1.1.1 is not applicable.

NRC Question 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.

PPL Response 23:

The basis for this statement was derived from the evaluation that was performed. The evaluation for the thermal power limit was based on critical power data. Critical power data is cycle independent. The comparison of MCPR between CPPU conditions and current conditions shows a slight reduction in MCPR for an assembly with high power peaking (from 4.1 to 3.9). Either value is considered to be a value with significant margin. Both values were obtained based on the use of similar radial and axial conditions and were not related to any particular cycle, and consequently would be cycle independent. With respect to the LHGR and APLHGR evaluation, conservatively high peaking factors were used. These factors were selected to be cycle independent so that conditions of a presumed high-powered assembly could be considered. At reactor startup conditions, a significant inventory of control rods will limit the core maximum peaking factors to values much lower than those assumed above. Differences in fuel and core design will not challenge the conservatively high core peaking factors used in the cycle independent evaluation discussed on page 9-2.

NRC Question 24:

In Section 9.1.1, operating 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 recirculation pump trip-out of service (OOS) and 1.38 for Feedwater controller failure maximum demand with turbine bypass valve (TBV) -OOS). Please explain why 1.43 and 1.38 did not use the OLMCPR. Is there a OLMCPR uncertainty included in determining this value?

PPL Response 24:

The OLMCPR of 1.34 is for rated CPPU power for all cycle exposures and normal operation with no equipment Out-Of-Service (OOS). Table 9-2 lists higher OLMCPRs for various equipment OOS options. The OLMCPR of 1.43 would be applicable to protect the safety limit minimum critical power ratio (SLMCPR) during a limiting generator load rejection if the recirculation pump trip (RPT) was OOS. Similarly, an OLMCPR of 1.38 would be applicable to protect the SLMCPR during a limiting feedwater controller failure to maximum demand if the turbine bypass valve (TBV) was OOS. The SSES Technical Specifications 3.3.4.1 and 3.7.6 require application of these equipment OOS MCPR limits.

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NRC Question 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.

PPL Response 25:

The decay heat model used in the loss of feedwater analysis is ANSI/ANS-5.1-1979 with corrections based on GE SIL 636 Rev. 1. The uncertainty used was two times the standard deviation which was calculated as defined by the standard. [

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NRC Question 26:

In Section 9.1.3.2, the reactor scram on low reactor water level (Level 3) is discussed. The analysis showed 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.

PPL Response 26:

Reaching the Level 3 scram setpoint for the loss of a feedwater pump transient is an expectation. Analyses performed by General Electric predict that little or no margin will exist to the Level 3 scram setpoint for this transient. Therefore, the expectation is that the plant may scram if this event occurs.

This event is not significant from a safety standpoint since the event causes a reduction in power through a runback or a scram, which increases thermal margins. The expectation for this event is significant from an operational perspective. Previous expectations for this event were that a scram would not occur for the loss of a feedwater pump event. For CPPU, it is expected that this event may now result in a scram. Identifying that this event may result in a scram for CPPU conditions provides the correct operational perspective.

NRC Question 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."

PPL Response 27:

The transient analyses that were initiated at 100% rated CPPU are listed in the following table. The individual treatment of the 2% power measurement uncertainty (statistically or inherent conservatism) is also identified in this table and is defined below.

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NRC Question 28:

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

PPL Response 28:

There is no difference in the methodology used to analyze the Rod Withdrawal Error (RWE) event for pre-CPPU and CPPU conditions. The ΔCPR values listed in the FSAR for the RWE event are calculated by subtracting the MCPR Safety Limit (MCPRSL) from the OLMCPR. The ΔCPR value for the CPPU RWE case listed in Table 9-2 is calculated as 1.32 – 1.07 or 0.25. Thus, the only difference is in the way that the results are reported for the two analyses (pre-CPPU and CPPU).

NRC Question 29:

For rod withdraw error 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 these two transients justify to be non-limiting compared to other transients in Table 9-2 regarding safety margin increase?

PPL Response 29:

The core loading and fuel bundle design process takes into account the impact of the RWE event on OLMCPR. These values are analytically determined as a function of RBM setpoint. The RBM setpoints are then chosen to optimize operating margin and MCPR margin.

The setpoint of 111% is slightly conservative in this example since the RWE OLMCPR of 1.32 is less than the limiting OLMCPR of 1.34. Based on RWE analysis results, the RWE setpoint could be raised to 114% to make the RWE OLMCPR the same as the most limiting events. The LHGR increase from an unblocked RWE was calculated to be 31%. This value is below the 35% increase allowed for the PAPT limit provided by AREVA. Thus, the increase in LHGR during an RWE is within the transient LHGR limit.

Ultimately, a balance must be achieved when choosing the final setpoint. There is a "band" of acceptable setpoints that protect the MCPR Safety Limit and the LHGR limits. An unnecessarily low setpoint can produce unnecessary operator burden due to excessive rod blocks. An unnecessarily high setpoint reduces available thermal margin and results in an economic penalty caused by reduced core operating efficiency.

The RBM is now credited in the RWE analysis for Unit 2, with the NRC approval of ARTS/MELLLA for SSES (Amendments 242 and 220 Unit 1 and 2, respectively). The RBM will also be credited in the SSES Unit 1 RWE analysis when ARTS/MELLLA is implemented for SSES Unit 1in the spring of 2008. The RBM was not previously credited in the SSES RWE analyses due to hardware issues that have since been corrected.

Based on the above discussion the RBM setpoints could be reduced to make the RWE event non-limiting or raised until the RWE OLMCPR is as limiting as the most limiting events. The RWE event with bypass out of service may be the limiting event when compared to other events that are analyzed with bypass out of service and therefore set the OLMCPR for bypass out of service conditions.

NRC Question 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 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 ca/g).

PPL Response 30:

AREVA performed a CRDA analysis for the representative equilibrium cycle design at CPPU conditions using NRC approved methodology. The results show a peak fuel rod deposited enthalpy of 174 cal/g for this event. This satisfies the acceptance criteria of 280 cal/g. For comparison, AREVA calculated peak fuel rod deposited enthalpies of 190 cal/g for Unit 1 Cycle 15 and 223 cal/g for Unit 2 Cycle 14. The CRDA event is analyzed on a cycle-specific basis to demonstrate compliance to the acceptance criteria. The UFSAR Table 15C.4.9-2 and 15D.4.9.2 values cited were for prior cycles calculated with PPL methodology which is no longer used.

NRC Question 31:

For feedwater controller failure maximum 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 ASME peak vessel limit of 1375 psig.

PPL Response 31:

The vessel pressure (lower plenum) for the limiting feedwater controller failure at maximum demand with TBV-OOS is 1290 psig which is less than the 1375 psig ASME limit. The dome pressure for the same case is 1266 psig which is less than the 1325 psig TS Safety limit. Note that the pressures in Figure 9-24, of the PUSAR report, are in units of psia. Note that this event is conservatively analyzed using the higher Safety Relief Valve (SRV) safety settings versus the normal lower SRV relief mode settings that could be used for Anticipated Operating Occurrence events such as the feedwater controller failure maximum demand with TBV-OOS.

NRC Question 32:

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

PPL Response 32:

Each MSRV has the same flow capacity and the flow through each valve increases as the steam line pressure increases and as the valve opens. However, the flows which are plotted are the flow through all of the valves with the same setpoint which are defined as a bank of valves. Since there are more valves in the high setpoint bank than in the medium setpoint bank (8 versus 4), the flow for the high setpoint bank may be higher than the flow for the medium setpoint bank depending on the amount that each valve bank is open.

Figures 9-18 and 9-19 show that while the 8 high setpoint valves are less than 20% open, the flow through these valves is almost as high as the flow through the 4 medium setpoint valves which are about 40% open. Therefore, each bank flow is expected to be about the same (8 * 20% versus 4 * 40%). Figures 9-25 and 9-26 show that the 8 high setpoint valves reach about 50% open, and this provides more flow than the 4 medium setpoint values which reach about 80% open. Therefore, the high setpoint bank flow is expected to be higher (8 * 50% versus 4 * 80%). In summary, the MRSV flows make sense when the number of valves in each bank is considered.

NRC Question 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 SRV OOS is zero for this analysis.

PPL Response 33:

The tables below provide the inputs and results for the CLTP ATWS analysis, along with the CPPU information provided in Table 9-3 (SSES Key Inputs for ATWS Analysis) and Table 9-4 (SSES Results of ATWS Analysis) of the Safety Analysis for the SSES CPPU licensing amendment request (Attachment 4 of Reference 1).

SSES Key Inputs for ATWS Analyses

Input Variable	CLTP	CPPU
Reactor power (MWt)	3489	3952
Reactor dome pressure (psia)	1049	1050
SRV capacity (Mlbm/hr @ 1175 psig with 3% accumulation)	14.143	14.143
High pressure ATWS-RPT (psig)	1170	1170
Number of SRVs Out-of-service (OOS)	0	0

SSES Results of ATWS Analyses

Acceptance Criteria	CLTP	CPPU	Acceptance Criteria
Peak vessel bottom pressure (psig)	1288	1336	≤1500
Peak cladding temperature (°F)	1420	1434	≤2200
Peak suppression pool temperature (°F)	207	206	≤220
Peak containment pressure (psig)	16.5	16.1	≤53

From the tables above, it is seen that an increase in the analyzed power results in an increase in the peak vessel pressure of about 48 psig, and an increase in peak cladding temperature of 14°F. These differences occur in the short-term, and are consequences of the higher initial power. However, results similar to CLTP are obtained for the CPPU long term peak suppression pool temperature and containment pressure due to the use of enriched boron (approved for use in the NRC Safety Evaluation Report for Technical Specification Amendments 240 and 217).

With respect to analytical uncertainties, the SSES CPPU ATWS analysis is performed assuming a nominal initial power 3952 MWt (100% CPPU). Appendix L, Section L.3.1 of NEDC-32424P-A (ELTR-1) states: "Reactor operating conditions will be equal to the uprated power conditions". Also, Appendix L, Section L.3.4 states that "nominal operating and equipment parameters are utilized for the evaluation of this special situation". This assumption has been utilized in previous SSES ATWS evaluations, including the analysis submitted with the SSES ARTS/MELLLA license amendment, which was approved by the staff in SSES TS Amendments 242 and 220. Thus, the use of nominal values is consistent with the current SSES licensing basis and analysis of design record.

The use of nominal values in ATWS analyses is generally accepted due the special circumstances and the multiple failures postulated to result in the analyzed event. As such, acceptable performance is assumed for equipment which is credited. Thus, for the calculation of peak vessel pressure, all main steam relief valves are assumed to function properly in the pneumatically assisted relief mode. Nonetheless, as discussed in Section 10.5.4 of the CPPU safety analysis (Attachment 4, Reference 1), an acceptable peak RPV pressure is still maintained during ATWS events with 4 SRVs out of service.

NRC Question 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.

PPL Response 34:

The steam flow plotted in Figure 9-1 is the steam flow at the vessel exit (entrance of the steam line). The steam flow trend observed between 0.7 and 1.5 seconds is caused by the pressure wave oscillating back and forth between the closed control valve (end of steam line) and the reactor vessel (beginning of the steam line). After the SRVs open at approximately 1.95 seconds into the transient, the steam flow continues to oscillate but at a higher value due to the SRV flow.

The turbine bypass system is designed to mitigate the severity of a generator load rejection and has worked as designed when a generator load rejection occurred for SSES. Similarly, the analysis only credits opening the SRVs in safety mode (higher pressure set points when compared with the normal relief mode of operation). Analyzing this event with no credit for turbine bypass and with no credit for opening the SRVs in relief mode causes the peak power and maximum pressure to be significantly higher than plant data. Therefore, a comparison between plant data and the analysis results would not prove useful.

NRC Question 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 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?

PPL Response 35:

10CFR50.63, "Loss of All Alternating Current Power," and RG1.115, "Station Blackout," do not require the use of an NRC approved code for the SBO evaluation. The use of the MAAP code for the CPPU SBO evaluation has been demonstrated to be acceptable for the SBO evaluation as described in PPL Responses 15 and 16. The BWRSAR code is the current code used for the SBO evaluation. The BWRSAR code was not the basis for determining the results for CPPU operation; therefore, it is not listed in Table 1-1.

NRC Question 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 (EPU vs. pre-EPU) to substantiate the conclusions drawn with regard to the impact of ATRIUM-10 fuel on ATWS/Stability.

PPL Response 36:

The industry assessment of ATWS relative to BWR core thermal-hydraulic stability is presented in Reference 36.1 (listed below), and the mitigation of BWR core thermal-hydraulic instabilities in ATWS presented in Reference 36.2, and the Brookhaven assessment is presented in Reference 36.3. Our assessment of the ATWS/Instability event is additionally supported by analyses with reduced order models based on first principles.

We conclude that the major parameter affecting the consequences of an ATWS/Instability transient is the amplitude of the global mode limit cycle, which is in turn dependent on the "linear" hydraulic decay ratio and neutron-coupled global decay ratio. For example, Reference 36.4 relates the maximum amplitude of the highly nonlinear limit cycle oscillation to the hydraulic and global decay ratios at the oscillation inception, namely,

$$\Phi_{min/max} = \frac{1-2k\dfrac{\ln DR}{\ln DR_{hyd}}}{1-2\dfrac{\ln DR}{\ln DR_{hyd}} \pm \sqrt{\left(1-2\dfrac{\ln DR}{\ln DR_{hyd}}\right)^2 - 1}} \qquad (36.1)$$

Where:

Φ	relative global mode flux (or power) amplitude
DR	global mode decay ratio at the time of oscillation inception (>1)
DR_{hyd}	core hydraulic decay ratio (<1)
k	power-reactivity coefficient

The statement cited from PUSAR Section 9.3.3 (Page 9-6) reflects the fact that core design changes can potentially change the linear stability of the core; i.e., its neutron-coupled as well as hydraulic decay ratios, due to potential changes in void-reactivity and power distribution, even when the same fuel type is used. It is common practice to evaluate the core stability for each cycle at several exposure points. This is true for standard as well as transition cycles to CPPU conditions. In the case of a CPPU cycle, the power profile is flatter than in a pre-CPPU cycle, which compensates for the other destabilizing effects. [

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The statement regarding the event progression refers to the fact that when the power oscillation amplitude grows significantly, the flow oscillation amplitude grows as well. The oscillation amplitude is limited by the strong damping effect when the top of the core is dry (steam quality close to unity) depriving the density wave of density variation as its driving source (Reference 36.5). [

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a. The statement cited from PUSAR (Page 9-7) reflects the fact that fuel designs are constrained by design requirements (fuel cycle economics, hydraulic compatibility, etc.) that result in minor variations in parameters significant to core stability. The dominant factors in determining the consequences of ATWS events with core instabilities are the reactor system and core boundary conditions (e.g., feedwater temperature) and core characteristics which are largely independent of specific fuel designs (e.g., axial and radial power distributions). [

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b. The comparative data is provided as natural circulation power levels corresponding to given decay ratios for pre-CPPU (PPL Unit 2 Cycles 13 and 14) and CPPU cores, all with ATRIUM-10 fuel. Lower power levels indicate a less stable condition for a given decay ratio. [

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References:

36.1. EDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," GE Nuclear Energy, June 1995.

36.2. NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," GE Nuclear Energy, December 1992.

36.3. W. Wolf et al., "BWR Stability Analysis with the BNL Engineering Plant Analyzer," NUREG/CR-5816, October 1992.

36.4. Y. M. Farawila and D.W. Pruitt, "A Study of Nonlinear Oscillations and Limit Cycles in Boiling Water Reactors – I: The Global Mode," Nuclear Science and Engineering, 154 302-315 (2006).

36.5. Y. M. Farawila and D. W. Pruitt, "A Study of Nonlinear Oscillations and Limit Cycles in Boiling Water Reactors – II: The Regional Mode," Nuclear Science and Engineering, 154 316-327 (2006).

NRC Question 37:

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

PPL Response 37:

Yes, the level control is present in the calculation. However, due to the time response characteristics of the level controller and the feedwater system, the change in the feedwater flow is slow relative to the duration of this transient.

This event (generator load rejection with turbine bypass failure and EOC-RPT-OOS) is a potentially limiting MCPR event and the primary purpose of the analysis is to calculate ΔCPR. The ΔCPR for this transient is calculated to occur early, at 1.98 seconds into the transient. Due to the time required for changes in the feedwater flow to reach the core, the feedwater control system has negligible effect on the ΔCPR.

NRC Question 38:

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

PPL Response 38:

The requested pressure plots are included below:

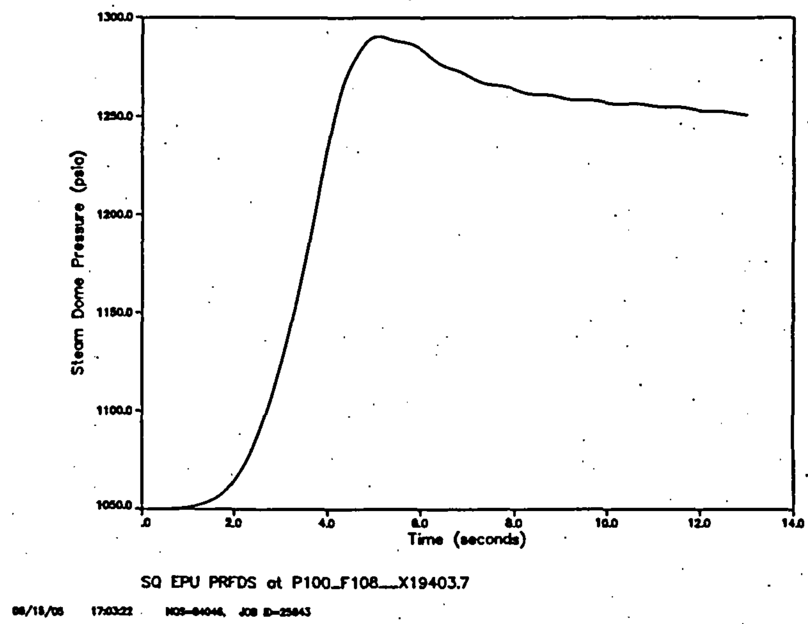

Figure 1 Reactor Vessel Steam Dome Pressure

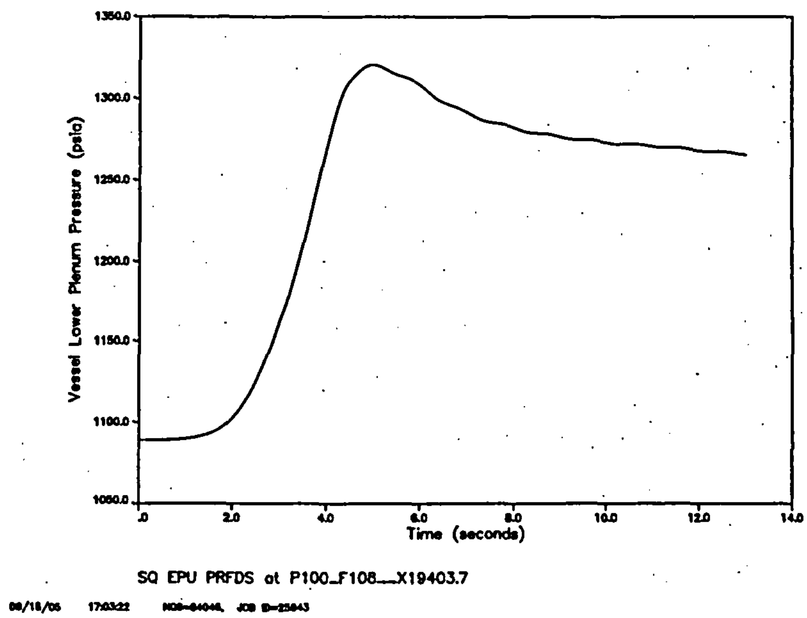

Figure 2 Reactor Vessel Lower Plenum Pressure

NRC Question 39:

It appears that the transients analyzed for CPPU in PUSAR Table 9-2 is 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).

PPL Response 39:

The "turbine trip, bypass failure, with scram on high flux" was analyzed as part of the ASME overpressurization analysis. This is mentioned in Section 3.1 (Page 3-2) of the PUSAR report. The results of this event were not included in the PUSAR report because this was not the limiting event for overpressurization. The closure of MSIVs, with scram on high flux was the limiting event and the results were presented in Figures 3-1 through 3-7 of the PUSAR.

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):

NRC Question 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.

PPL Response 40:

To explain the basis for this statement, the long term and short term affects need to be addressed separately. Core Spray distribution is not directly credited in the short-term cooling LOCA analyses.

The short and long term impacts are addressed below.

Short-term Impact: The effects of CPPU on actual core spray distribution in the reactor vessel are not directly credited in the LOCA pre or post CPPU analyses. This is consistent with the ECCS evaluation models specified in Appendix K to 10 CFR Part 50. Therefore, the convective heat transfer coefficients used during the short-term spray cooling period are the conservative values specified in Appendix K.

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NRC Question 41:

On Page 1-1, 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.

PPL Response 41:

LOCA analyses were performed at maximum CPPU power and two core flows, 80 Mlbm/hr and 108 Mlbm/hr, which represent the boundaries of the MELLLA+ domain (the lowest and the highest attainable core flows at full CPPU power). The results from analyses of these bounding core flows support operation at intermediate core flows. At full CPPU power, the boundaries of the MELLLA domain are 99 Mlbm/hr and 108 Mlbm/hr. Therefore, the initial core flows which were analyzed support operation within the MELLLA+ domain and they support operation within the currently licensed MELLLA domain.

NRC Question 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."

PPL Response 42:

Yes, the statement on Page 1-2 refers to limiting break characteristics (location, size, single failure, etc.). The term reactor system response means the thermal hydraulic response of the reactor vessel, recirculation system and ECCS to a LOCA event.

The key events of a LOCA scenario - a blowdown in which reactor pressure decreases rapidly toward atmospheric pressure as the coolant inventory is lost through the break, initiation of ECCS based principally on indications of pressure and water level, and recovery of water inventory - are controlled by the break characteristics. Exposure dependent fuel parameters, such as stored energy and local peaking, differ throughout the core but ultimately provide the total 102% of rated power assumed in the analysis. It is this aggregate power that drives the system response for LOCA events.

NRC Question 43:

In Figures 4-3 and 4-4, it looks like the nodal length is not uniform throughout the channel. State if 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.

PPL Response 43:

Yes, the nodal length is reflected in Figures 4.3 and 4.4. The shortest node lengths are consistent with those used in calculations to support the AREVA (then Siemens) Licensing Topical Report, EMF-2361(P)(A) EXEM BWR-2000 ECCS Evaluation Model. Figure 4.2 of that report shows a nodal diagram for a top peaked hot channel which illustrates the use of [ ] nodes at and adjacent to the peak power location. This allows for finer detail in the calculations at and near the peak power location while maintaining a reasonable nodal length (L) to diameter (D) ratio, L / D > 1, providing for a well behaved numeric solution. Figures 4.3 and 4.4 of EMF-3261(P) are similar, illustrating the use of [ ] nodes at and adjacent to the peak power location. Section 4.2.1 of EMF-2361(P)(A) describes nodalization sensitivity studies.

NRC Question 44:

In Table 5.1, please explain why there are still two LPCIs left for SF-LPCI and SF-LOCA since BWR 4 plants have 4 LPCIs.

PPL Response 44:

For SSES, there are two independent loops of LPCI (i.e. injection lines), each with 2 pumps.

The SF-LPCI condition represents the failure of a LPCI injection valve. In this case, the failure of a LPCI injection valve still leaves one loop of LPCI available with 2 pumps. Thus, 2 LPCI (2 pumps in one loop) are left for the SF-LPCI condition.

The SF-LOCA condition represents a false LOCA condition on the non-LOCA unit. Under this condition, each unit may only use two LPCI pumps. In this situation, 1 LPCI pump is left available in each of the 2 LPCI loops. So, 2 LPCI (1 pump in each loop) are available in the SF-LOCA condition.

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

PPL Response 45:

The discharge coefficient is used to calculate critical mass flow. Critical flow occurs as the velocity in a junction approaches the sonic velocity. When this occurs, the flow rate is dependent only on the upstream pressure and independent of the downstream pressure. The equation used to calculate critical mass flow is:

$$W = Cd\,Aj\,Gx\,(P,h)$$

where

Cd = a user-supplied critical flow multiplier (typically ≤ 1.0)
Aj = junction area, ft^2
P = donor volume pressure, psi
h = donor volume enthalpy, Btu/lbm, and
$Gx\,(P,h)$ = critical flow from table lookup, lbm/ft^2-sec.

The critical flow tables used in AREVA's approved evaluation model are either Moody or Henry-Fauske. The Appendix K criteria require the Moody discharge model for the two-phase region. In the subcooled region, the criteria do not specify the discharge model to be used. In AREVA's approved evaluation model, the Henry-Fauske model is used for the subcooled region.

The critical mass flow at the break location is a linear function of the discharge coefficient. As required by Appendix K, a range of values are used in calculations to assure that the maximum clad temperature is found. The complex combination of models and model requirements in an evaluation model do not always present a monotonic trend in analysis results. Table 6.9 of EMF-3242(P) summarizes the maximum clad temperatures from calculations summarized in Tables 6.3 through 6.8 of EMF-3242(P), performed as part of a broad spectrum of calculations to find the limiting combination of parameters reasonably expected to have a significant impact on maximum clad temperature for SSES.

Section 4.1 of the topical report (EMF-2361(P)(A)) presents the application of the EXEM BWR-2000 methodology to break spectrum analyses of a BWR/3. The results in Section 4.1 are not from analyses of SSES. [

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<u>NRC Question 46</u>:

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 CMWR would be less than 1.0%. Please provide any assumptions considered for the engineering judgment. According to the Maximum Planar Linear Heat Generation Rate (MAPLHGR) report, Susquehanna's core wide hydrogen generation must be less than 0.2% of the hypothetical amount because of Susquehanna's hydrogen re-combiner capacity. Please provide analysis and assumptions to justify how the MWR acceptance criterion (0.2%) is met in detail. Similar issue applies to SLO limiting results.

PPL Response 46:

Both the maximum local and the maximum planar MWR are calculated to be less than 1%. Therefore, in the worst case scenario, all the rods in the core would have a local MWR equal with the maximum local MWR which is less than 1%. The resulting core wide MWR (CMWR), which is an averaging of these rods, would also result in a value less than 1%.

In reality, the rest of the rods in the core will have a local MWR less than the maximum reported value and therefore the CMWR will be much less than the maximum local MWR reported.

Following is a summary of the process AREVA used to calculate CMWR. [

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Since the MWR for TLO LOCA is higher than the MWR for SLO LOCA, the CMWR calculated for TLO is conservatively applicable for SLO.

To clarify the MAPLHGR report statement about the re-combiner capacity, it should be noted that the limit of 0.2% is used to ensure that the assumptions in the combustible gas analysis remain valid. Per Regulatory Guide 1.7, either 1% of the clad or 5 times the CMWR results from the LOCA analysis (which ever is larger) must be assumed in the calculation for combustible gas. The current combustible gas analysis uses 1% as the limiting value. If the LOCA analysis CMWR is calculated to be greater than 0.2%, then the combustible gas analysis must be re-evaluated since the required factor of 5 applied to the LOCA analysis results would be more limiting than the current assumption of 1%.

NRC Question 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 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.

PPL Response 47:

Yes, the pre-CPPU analyses assumed "No LPCI to BL" while the CPPU analysis credited the injection of LPCI in the BL (broken loop).

The comparative analysis of the two cases (pre-CPPU with a PCT of 1945°F and CPPU with a PCT of 1914°F) showed that the main reasons for the higher PCT are a later reflood time (approximately 5 seconds later) and slightly more conservative initial conditions for the pre-CPPU analysis [

] A later reflood time results in a higher PCT because the heatup time is longer.

[

]

NRC Question 48:

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

PPL Response 48:

The AREVA BWR EXEM-2000 methodology does not always produce two peaks in the PCT plot. Dryout results in a decrease in the heat transfer and an increase in the clad temperature. Furthermore, after dryout is predicted to occur re-wetting is not allowed even if the thermal hydraulic conditions were such that it would occur. In the limiting TLO case, the increase in clad temperature at about 20 seconds corresponds to the decrease in heat transfer (Figure 6.25). This decrease in heat transfer was the result of dryout and stagnation of the core flow. Dryout conditions are experienced earlier during SLO operation because the core flow decreases more rapidly when there is no inertia in the intact loop.

[

] For SLO, the hot node experiences dryout early in the event and may show a two peaked PCT plot (as shown in Figure 8.27). For TLO, the dryout will not occur as soon and the PCT plot may only have one peak (as shown in Figure 6.27).

NRC Question 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.

PPL Response 49:

[

]

However, even with this conservatism, acceptable PCTs are obtained through application of a SLO multiplier to the TLO MAPLHGR limit.

For LOCA, the most important difference between TLO and SLO is the early CHF that occurs due to the more rapid decrease in core and assembly flow when there is no forward inertia in the intact (inactive) recirculation loop. [

]

The LOCA modeling requirements in Appendix K of 10 CFR 50 state that a fuel heat conductor cannot return to a pre-CHF heat transfer mode after CHF is predicted. In the limiting Susquehanna SLO LOCA, CHF was predicted to occur 2.1 seconds after the break. Post-CHF heat transfer correlations are used during the rest of the blowdown period as required by Appendix K. During the period after core spray reaches rated flow, the heat transfer coefficients required by Appendix K are used. Therefore, the heat transfer coefficients following the early CHF are not strongly dependent on the core thermal-hydraulic conditions calculated in the SLO LOCA analysis.

[

]

NRC Question 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?

PPL Response 50:

There are no additional constraints or margins for this multiplier beyond PCT. Several things were considered when determining this multiplier. Operation under single-loop conditions is expected to occur on an infrequent basis. There is no technical reason why a higher multiplier could not be justified. However, it is not expected that a higher multiplier for SLO would provide more operational flexibility since power is significantly reduced during SLO.

The current multiplier has produced acceptable PCT biases between two- and single-loop conditions for several LOCA analyses and therefore the need to iterate to a new higher multiplier has not been deemed to be necessary.

The following RAIs are specifically from the MAPLHGR Report, EMF-3243(P):

NRC Question 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.

PPL Response 51:

The difference between the two PCTs noted in the question is primarily due to different local peakings between the two analyses. [

]

NRC Question 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.

PPL Response 52:

The recirculation isolation discharge valve has been credited in previous SSES LOCA analyses. An example is the LOCA analysis which was reviewed and approved by NRC for the original SSES stretch power uprate. The following discussion provides justification for the continued crediting of this valve in the LOCA analyses.

During large break LOCA scenarios, the reactor recirculation system (RRS) pump discharge and discharge bypass valves close to divert LPCI flow through the vessel jet pumps. The valves automatically close when vessel pressure is reduced to the low vessel pressure permissive, with either a concurrent high drywell pressure, or low vessel level signal.

The safety function for these valves to close during large break LOCAs is part of the plant's original design and licensing basis. These valves are included in the Generic Letter 89-10 Program, and as such, their motors/actuators are sized for the anticipated LOCA operating conditions. Both the "A" and "B" loop valves are powered by separate, independent, single failure proof power supplies from the plant's 480 VAC Class 1E electrical distribution system. The valves and their actuator assemblies are seismically and environmentally qualified for the postulated accident conditions.

The recirculation pump discharge and discharge bypass valves, which are credited to close during large break LOCA scenarios, meet the intent of the applicable regulatory criteria, guidelines, and industry standards.

Attachment 3 to PLA-6209
AREVA NP, Inc. and General Electric Company Affidavits

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the PPL letter PLA-6209, *Susquehanna Steam Electric Station Proposed License Amendment No. 285 for Unit 1 Operating License No. NPF-12 and Proposed License Amendment No. 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Re: Reactor Systems Technical Review Request for Additional Information Responses*, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Jerald S Holm

SUBSCRIBED before me this 30th

day of May, 2007.

Susan K McCoy

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/2008

SUSAN K. McCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
1-10-08
STATE OF WASHINGTON

General Electric Company

AFFIDAVIT

I, Bradley J. Erbes, state as follows:

(1) I am Manager Services Engineering, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of the GE-SSES-AEP-322, Larry King (GE) to Mike Gorski (PPL), *GE Responses to BWR Systems RAIs 10, 12, 14, 33 and 40; Mechanical and Civil RAIs 1, 3, 7, 12, 20 and 21; Containment and Venting RAIs 1, 3, 5, 9 and 14*, GE Proprietary Information, dated May 21, 2007. The Enclosure 1 (*GE Responses to BWR Systems RAIs 10, 12, 14, 33 and 40; Mechanical and Civil RAIs 1, 3, 7, 12, 20 and 21; Containment and Venting RAIs 1, 3, 5, 9 and 14*) proprietary information is delineated by a dotted underline inside double square brackets. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from evaluations, utlizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic modes and computer codes were achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 18TH day of May 2007

Bradley J. Erbes

Bradley J. Erbes
General Electric Company