Exelon Generation.

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

10 CFR 50.90
10 CFR 2.390

August 22, 2013

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Extended Power Uprate License Amendment Request – Supplement 9
Response to Request for Additional Information

Reference:
1. Exelon letter to the NRC, "License Amendment Request - Extended Power Uprate," dated September 28, 2012 (ADAMS Accession No. ML122860201)
2. NRC letter to Exelon, "Request for Additional Information Regarding License Amendment Request for Extended Power Uprate (TAC Nos. ME9631 and ME9632)," dated July 23, 2013 (ADAMS Accession No. ML13203A100)

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (EGC) requested amendments to Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3, respectively (Reference 1). Specifically, the proposed changes would revise the Renewed Operating Licenses to implement an increase in rated thermal power from 3514 megawatts thermal (MWt) to 3951 MWt. During their technical review of the application, the NRC Staff identified the need for additional information. Reference 2 provided the Request for Additional Information (RAI).

This letter addresses requests from the staff of Reactor Systems (SRXB) and Accident Dose (AADB) Branches of the U. S. Nuclear Regulatory Commission to provide information in support of the request for amendment for the extended power uprate. Responses to these questions are provided in the attachments to this letter:

Attachment 1 - Reactor Systems Branch question responses, proprietary version

Attachment 2 – Reactor Systems Branch question responses, non-proprietary,

Attachment 3 – Accident Dose Branch question responses

Attachment 1 contains Proprietary Information.
When separated from Attachment 1, this document is decontrolled.

A001
NRR

GE Hitachi Nuclear Energy America (GEH) considers portions of the information provided in the responses in Attachment 1 to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390. The proprietary information in Attachment 1 is identified; a non-proprietary version of this information is provided in Attachment 2. In accordance with 10 CFR 2.390, EGC requests Attachment 1 be withheld from public disclosure. An affidavit supporting this request for withholding is included as Attachment 4.

EGC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the U. S. Nuclear Regulatory Commission in Reference 1. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. Further, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the Commonwealth of Pennsylvania and the State of Maryland of this application by transmitting a copy of this letter along with the non-proprietary attachments to the designated State Officials.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Mr. David Neff at (610) 765-5631.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 22nd day of August 2013.

Respectfully,

Kevin F. Borton
Manager, Licensing – Power Uprate
Exelon Generation Company, LLC

Attachments:
1. Response to Request for Additional Information – SRXB - Proprietary
2. Response to Request for Additional Information – SRXB
3. Response to Request for Additional Information – AADB
4. Affidavit in Support of Request to Withhold Information

cc:	USNRC Region I, Regional Administrator	w/attachments
	USNRC Senior Resident Inspector, PBAPS	w/attachments
	USNRC Project Manager, PBAPS	w/attachments
	R. R. Janati, Commonwealth of Pennsylvania	w/o proprietary attachment
	S. T. Gray, State of Maryland	w/o proprietary attachment

Attachment 2

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

Response to Request for Additional Information – SRXB

Response to Request for Additional Information

Reactor Systems Branch

By letter dated September 28, 2012, Exelon Generation Company, LLC (Exelon) submitted a license amendment request for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendment would authorize an increase in the maximum power level from 3514 megawatts thermal (MWt) to 3951 MWt. The requested change, referred to as an extended power uprate (EPU), represents an increase of approximately 12.4 percent above the current licensed thermal power level.

The response to SRXB RAI-1 was provided in Supplement 2, dated May, 7, 2013 (ADAMS Accession No. ML13129A143). The NRC staff has reviewed the information supporting the proposed amendment and by letter dated July 23, 2013 (ADAMS Accession No. ML13203A100) has requested additional information. The response to that request is provided below.

Note – an Acronym List is provided at the end of this Attachment.

SRXB RAI-2

Section 2.8.2.1 of the Power Uprate Safety Analysis Report (PUSAR[2]) indicates that the average bundle power would increase from the current licensed thermal power (CLTP) value of 4.60 Megawatts (MW)/bundle to a value of 5.17 MW/bundle under EPU conditions. This 12.4% change in average bundle power level corresponds to the same percent increase of total core power from CLTP to EPU conditions. For constant pressure power uprate (CPPU) for boiling-water reactors (BWRs), it is assumed that the additional core power is obtained by flattening the core power profile (i.e., raising the average bundle power, but keeping the peak bundle power the same). However, past BWR EPU operations have demonstrated that peak bundle power can increase by a limited amount. Please provide the current peak bundle power level and the expected value of peak bundle power for EPU operation at PBAPS.

RESPONSE

The peak bundle power for CLTP (3514 MWt) operation is 7.05 MW (based on Unit 2 Cycle 19), 7.34 MW (based on Unit 2 Cycle 20), 7.11 MW (based on Unit 3 Cycle 19) and 7.24 MW (based on Unit 3 Cycle 20). The peak bundle power for the representative equilibrium GNF2 core design in the PUSAR (Reference 2-1) is 7.65 MW. Small variation is expected depending upon the cycle specific nuclear design and associated operating limits.

---

[2] A proprietary (i.e., non-publicly available) version of the PUSAR is contained in Attachment 6 to the application dated September 28, 2012. A non-proprietary (i.e., publicly available) version of the PUSAR is contained in Attachment 4 to the application dated September 28, 2012.

References

2-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-3

As stated in the PUSAR, PBAPS EPU analyses assumed a representative "equilibrium" core comprised of GNF2 fuel only. Describe the PBAPS core design for the first EPU cycle. If the actual EPU core will be comprised of other GE fuel designs, for example GE14, in addition to GNF2, then justify why using a GNF2 equilibrium core for EPU calculations provides bounding results for EPU transient and accident analyses, in particular the safety limits (minimum critical power ratio (MCPR), linear heat generation rate, maximum average planar linear heat rate, peak cladding temperature (PCT), etc.).

RESPONSE

The PBAPS core at the time of EPU implementation for each PBAPS unit is expected to consist only of GNF2 fuel. The response to SNPB-RAI-1, part d, (Reference 3-1) concerning the PBAPS core design for the first cycle, and the response to SNPB-RAI-7 (Reference 3-1), concerning details of the PBAPS equilibrium core which supported the analysis for EPU, provides the justification for the use of this equilibrium core for EPU transient and accident analyses.

References

3-1 Exelon letter to USNRC, "Extended Power Uprate License Amendment Request – Supplement 6 – Response to Request for Additional Information – SNPB", dated July 30, 2013.

SRXB RAI-4

Pellet clad interaction (PCI) and stress-corrosion cracking (SCC) phenomena can cause clad perforation resulting in leaking fuel bundles and resultant increased reactor coolant activity. Therefore, the NRC staff requests the licensee to provide the following additional information regarding PCI/SCC for PBAPS at EPU conditions:

a) Describe any differences in operating procedures associated with PCI/SCC at EPU conditions versus pre-EPU operations.

b) From the standpoint of PCI/SCC, discuss which of the Anticipated Operational Occurrences (AOOs), if not mitigated, would most affect operational limitations associated with PCI/SCC.

c) For the AOOs in part (b), discuss the differences between the type of required operator actions, if any, and the time to take mitigating actions between pre-EPU and EPU operations.

d) If the EPU core will include fuel designs with non-barrier cladding which have less built-in PCI resistance, then demonstrate by plant-specific analyses that the peak clad stresses at EPU conditions will be comparable to those calculated for the current operating conditions.

e) Describe operator training on PCI/SCC operating guidelines.

RESPONSE

a) There are no differences in operating procedures associated with PCI/SCC at EPU conditions versus pre-EPU conditions. The fuel vendor, Global Nuclear Fuel (GNF), provides operating recommendations associated with PCI/SCC. Those operating recommendations are the same for both pre-EPU and post-EPU conditions. There is extensive successful operating experience with the GNF operating recommendations at both EPU and non-EPU conditions.

b) From the standpoint of PCI/SCC the AOO that would most affect operating limitations associated with PCI/SCC, if not mitigated, is the Loss of Feedwater Heater AOO. This is a relatively slow transient, however the power increases associated with this event might exceed the ramp rate increases recommended by the 'soft duty guidelines'. This could reduce the margin to fuel failures associated with PCI/SCC. However, with GNF barrier fuel, no fuel failures associated with PCI/SCC are expected associated with this AOO; and no fuel failures in GNF barrier fuel have occurred to date in any plants from a loss of feedwater heating AOO. Furthermore, operating procedures specify a reduction in core recirculation flow early in the event, which reduces the increases in nodal powers, so that failures, even with non-barrier fuel, are not likely or expected.

Operating procedures also specify that any control rods (after recirculation flow reduction) be continuously and fully inserted to 00, to avoid the axial power peaking that can occur at the tip of partially inserted control rods.

c) There are no differences in the type or timing of the operator actions in response to the Loss of Feedwater (LOFW) AOO between pre-EPU and EPU operation.

d) The PBAPS EPU core will only consist of fuel with barrier cladding.

e) Operator training on PCI is integrated into the Core Thermal Limits lesson and SCC is covered in the Operations chemistry lesson in the initial licensed and non-licensed operator training programs. Continuing training includes topics selected to reinforce fundamental knowledge; PCI is currently included in the biennial thermal limits review.

SRXB RAI-5

Characterize the expected amount of bypass voiding under CPPU conditions. Provide the expected bypass void level at points C, D, and E of Figure 1-1 of the PUSAR, using a methodology equivalent to that used by ISCOR for both hot and average channel.

RESPONSE

ISCOR was used to characterize the expected amount of bypass voiding under CPPU conditions. ISCOR conservatively calculates the hot channel bypass voiding using its direct moderator-heating model and provides no credit for cross flow while applying conservative hot channel bypass heating. Points C, D, and E of PUSAR (Reference 5-2) Figure 1-1 are at the following Power/Flow Statepoints:

Table SRXB RAI-5 – PBAPS EPU Bypass Voiding (ISCOR)

Point	Power (EPU) (% Rated)	Flow (% Rated)	Core Average LPRM D-Level Bypass Voids (%)	Hot Channel LPRM D-Level Bypass Voids (%)	Core Average TAF Bypass Voids (%)	Hot Channel TAF Bypass Voids (%)
E	100	100	[[			
D	100	99				
C	54.9	38				]]

For Points E and D, the Core Average and Hot Channel LPRM D-Level Bypass Voiding associated with the representative equilibrium GNF2 core is shown to be less than 5%

when operating at steady state conditions within the MELLLA boundary. For Point C, the Core Average and Hot Channel Bypass Voiding values at the LPRM D-level are shown above. The Core Average and Hot Channel Bypass Voiding values are also shown at the Top of Active Fuel (TAF) location and confirmed to be less than [[ ]] and [[ ]], respectively. These values are not specific criteria, but ranges for bypass voiding for the MELLLA operating domain as shown in Sections 5.4, 6.1.1.1, and 6.2 of the Reference 5-1.

References

5-1 GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173-P-A, Revision 4, November 2012.

5-2 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-6

Reliability of the local power range monitor (LPRM) instrumentation and accurate prediction of in-bundle pin powers typically requires operation with bypass voids lower than 5% at nominal conditions (e.g., point E of Figure 1-1 of the PUSAR). If the expected bypass void conditions at CPPU are greater than 5%, evaluate the impact on: (1) reliability of LPRM instrumentation, (2) accuracy of LPRM instrumentation, and (3) in-bundle pin powers.

RESPONSE

Per results noted in Section 2.8.2.4.1 of Reference 6-1 (and in response to SRXB RAI-5), bypass void conditions at EPU are not expected to be greater than 5% at Point D and Point E of the power/flow map.

References:

6-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-7

The presence of bypass voids affects the LPRM calibration. Evaluate the expected calibration error on Oscillation Power Range Monitor (OPRM) and Average Power Range Monitor cells induced by the expected level of bypass voids. Document the impact of this error on the detect-and-suppress Option III scram setpoint.

RESPONSE

The effect of LPRM calibration errors on the OPRM system scram amplitude due to bypass voiding would be less than 5% (see Section 6.2 of Reference 7-1). This translates to an approximate 0.01 difference in OPRM amplitude setpoint.

In accordance with the Stability Setpoints Adjustment Limitation in Section 6.2 of the Safety Evaluation in Reference 7-1, a 5% penalty was applied to the calculated OPRM amplitude setpoint, which translates to an approximate 0.01 decrease. The OPRM amplitude setpoints presented in Section 2.8.3.1.2 and Table 2.8-2 of Reference 7-2 include the 5% setpoint penalty due to LPRM calibration errors.

The Average Power Range Monitor (APRM) system is not used for detection and suppression of thermal-hydraulic oscillations; therefore, there is no effect of APRM calibration errors on the Detect and Suppress Option III scram setpoint.

References:

7-1 GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

7-2 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-8

PUSAR Table 2.8-2 only shows the Option III Setpoints Demonstration. Please provide an example setpoint calculation for the EPU cycle including an uncertainty term reflecting the possible LPRM miscalibration under bypass void conditions.

RESPONSE

Table 2.8-2 of Reference 8-1 shows the Option III setpoint for the EPU equilibrium core. The following is an example of the calculation method used to determine the OPRM setpoint for the two-pump trip event, which is the more limiting of the two bounding stability events for the EPU equilibrium core.

For an OPRM setpoint of 1.12, the hot channel oscillation magnitude was calculated as Δh = [[ ]] for PBAPS based on the statistical methodology described in Reference 8-2. The DIVOM slope was calculated as [[ ]] for the PBAPS EPU equilibrium core. In order to protect SLMCPR, the initial MCPR is determined using the following relationship (Section 4.5.1 of Reference 8-2),

IMCPR ≥ SLMCPR / (1 - ΔCPR/ICPR)

ΔCPR/ICPR in the above equation is the product of Δh and DIVOM slope. For the EPU equilibrium core, SLMCPR is 1.09 including the steady-state uncertainties resulting from LPRM calibration. Therefore, MCPR at the start of the oscillations is calculated as follows,

IMCPR ≥ [[ ]]

It was determined from a PANAC11 analysis that this IMCPR after the pump trip will be attained if the MCPR at rated conditions prior to the pump trip, OLMCPR(2PT), is [[ ]].

This OLMCPR(2PT) is to be compared to the transient-based OLMCPR. Per Limitation and Condition 9.19 of Reference 8-3, 0.01 is added to OLMCPR(2PT),

OLMCPR(2PT) = [[ ]]

The OLMCPR(2PT) value of [[ ]] was calculated for an OPRM setpoint of 1.12. In order to account for impact of the setpoint uncertainties resulting from bypass voiding discussed in the SRXB RAI-7 response, 0.01 is subtracted from the setpoint and the result is conservatively reported as applying to the reduced setpoint (as reported in Table 2.8-2 of Reference 8-1). Hence, the minimum OLMCPR that can be supported based on a two recirculation pump trip event is [[ ]] for an OPRM setpoint of 1.11.

Thus, this OLMCPR value accounts for the LPRM calibration uncertainty due to bypass voiding. Exelon will apply the above methodology to the EPU implementation cycle core design to determine the cycle-specific OPRM setpoint. This setpoint will be reported in the PBAPS Supplemental Reload Licensing Report.

References:

8-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.
8-2 GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, Class I (Non-proprietary), August 1996.
8-3 GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

SRXB RAI-9

The Delta Critical Power Ratio (CPR) over Initial CPR Versus Oscillation Magnitude (DIVOM) slope is not included in PUSAR Table 2.8-2 under CPPU conditions. Please document which DIVOM slope will be used for future CPPU cycles and which methodology will be used to: (1) calculate it, or (2) evaluate the adequacy of an older slope.

RESPONSE

The DIVOM slope calculated for the EPU equilibrium core is [[ ]]. The DIVOM slope for each Peach Bottom Unit 2 and 3 operation cycles is calculated as part of the cycle-specific reload licensing analysis and the DIVOM slope will be evaluated on a cycle-specific basis per References 9-1 and 9-2. It is limited to no less than the generic DIVOM slope of 0.45 as prescribed in References 9-2 and 9-3.

References:

9-1 GE Hitachi Nuclear Energy, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465 Supplement 1, September 2011.
9-2 Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel (GESTAR II), "NEDE-24011-P-A-19 and the US Supplement NEDE-24011-P-A-19-US, May 2012.
9-3 GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, Class I (Non-proprietary), August 1996.

SRXB RAI-10

Assuming a conservative OPRM setpoint of 1.15, provide the hot-spot fuel temperature as a function of time before the scram. Evaluate this fuel temperature oscillation against PCI limits. Assume that the steady-state fuel conditions before the oscillations are those of point A1 of PUSAR Figure 2.8-21 (the highest power point in the backup stability protection (BSP) scram region).

RESPONSE

The current licensing criteria applicable to SRXB RAI-10 are [[

]]. Additionally, the current licensing criterion that cladding fatigue life usage be less than or equal to 1.0 applies to SRXB RAI-10. These criteria are addressed in this response. This response also addresses the issue of the potential for increased pellet-cladding interaction (PCI) raised in SRXB RAI-10. Because design or licensing criteria for PCI currently do not exist, the issue is addressed qualitatively in terms of impact on reliability.

A thermal-mechanical based power-exposure limits envelope is specified [[

]] The LHGR limits are specified to assure compliance with several primary fuel rod thermal-mechanical licensing criteria; these criteria address fuel centerline temperature, [[
]], and fuel rod internal pressure. [[

]]

A major GNF fuel rod design objective is to specify the LHGR limits curves to achieve balanced margins and a balanced design with high reliability over the rod lifetime.

During the core design process, a specified margin is typically maintained between the LHGR limits and the anticipated operation for each bundle. Operation under power uprate conditions will result in more rods in some bundles operating near the specified margin for a larger fraction of the bundle lifetime, thus increasing the potential for fuel failure. The potential for increased failure under power uprate conditions is assessed in terms of available GNF operational experience and experimental information below. The impact of power uprate on thermal-mechanical licensing analyses for the GNF2 fuel design is also discussed below.

[[

]] The results from this Severe Power Ramp testing, as compared to the LHGR limits curves for the fuel designs noted above, are also provided in Figure 10-1. It is observed from Figure 10-1 that significant margin exists to the apparent failure threshold represented by the available ramp test results. In addition to barrier fuel's resistance to ramping, ramp rates at power uprate conditions versus non-power uprate conditions are not appreciably different. Thus it is judged that the possible increased cladding mechanical duty associated with operation under power uprate conditions will have negligible impact on the reliability of GNF fuel. It is further noted that the margin to failure is reasonably well-balanced over the entire exposure range, consistent with the design objective noted above.

In addition to possible increased fuel duty, other potential effects of power uprate are small changes in core conditions such as increased coolant pressure (and temperature) and changes in flow conditions. [[

]]

For instability oscillations indicated in SRXB RAI-10, the incremental fatigue usage due to the oscillations is negligible in an absolute sense and relative to the margin to the limit (1.0) calculated for the cyclic loading assumed in the fuel rod thermal-mechanical licensing analyses. This criterion is based upon preventing wide spread cladding fatigue failures during normal operation. The fuel rod time constant is higher than the period of the power oscillations. As a result, the power oscillations result in insignificant fuel temperature oscillations relative to the PCI margin shown in Figure 10-1. These results indicate that the instability oscillations will have negligible impact on fuel reliability.

In summary, on the basis of the generic licensing analyses and the specific analyses to address operation under power uprate conditions summarized above, it is concluded that the [[ ]] fuel design is fully compliant with existing licensing requirements for operation under power uprate conditions. Based upon available operational experience and experimental data, it is also concluded that operation under power uprate conditions will not significantly affect GNF2 fuel reliability.

Figure 10-1 LHGR Limits and Severe Ramp Test Failure Data

[[

]]

References

10-1. H. Sakurai, et. al., 'Irradiation Characteristics of High Burnup BWR Fuels', paper presented at the ANS Light Water Reactor Fuel Performance Conference held at Park City, Utah, April 10-13, 2000.

SRXB RAI-11

Describe any effects or impacts of EPU on the long-term stability implementation.

RESPONSE

The EPU expands the operating domain in a region of the power to flow map where the plant is not susceptible to thermal-hydraulic instability events. The effects of PBAPS EPU implementation for the Option III stability solution are described in Section 2.8.3 of Reference 11-1. Furthermore, the OPRM setpoints and Backup Stability Protection regions are generated, and the OPRM Trip-enabled region boundaries are confirmed each reload. Therefore EPU does not affect the applicability of the Option III solution.

References:

11-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-12

For the BSP calculations, describe how the stability curves for the scram region and the controlled entry region shown in PUSAR Figure 2.8-21 are calculated for EPU conditions. Specifically, provide the associated feedwater temperature assumptions that allow the use of the same decay ratio criteria shown in Table 2.8-3 for the Scram and Controlled Entry boundary.

RESPONSE

The BSP Scram and Controlled Entry region for Option III methodology are calculated in the fuel cycle reload stability analysis (References 12-1 to 12-3). The same methodology is applied for EPU. To calculate the BSP Scram and Controlled Entry Region boundaries, ODYSY decay ratio calculations are performed on the highest licensed flow control line and on the natural circulation line. Rated feedwater temperature and rated xenon concentrations are assumed for calculating the BSP Scram Region boundary points, and the points where a 0.8 core wide decay ratio is calculated are connected using well defined Shape Function (i.e., Generic or Modified Shape Function) to define the Scram region boundary. The BSP Controlled Entry Region is calculated in a similar manner, also using a core wide decay ratio of 0.8 to define the region boundary; the difference being that the decay ratio calculation of the point on the highest flow control line assumes equilibrium feedwater temperature at off-rated operating conditions and xenon concentration (rather than rated), and the point on the natural circulation line assumes equilibrium feedwater temperature and xenon free conditions. This is why the two different curves can have almost identical calculated core wide decay ratios.

References:

12-1 "Backup Stability Protection (BSP) for Inoperable Option III Solution", OG 02-0119-260, July 2002.

12-2 "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," NEDE-33213P-A, April 2009.

12-3 Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A-19 and the US Supplement NEDE-24011-P-A-19-US, May 2012.

SRXB RAI-13

Provide plant-specific information relevant to an anticipated transient without scram (ATWS) event under EPU conditions. Specifically, provide the location of the boron injection, a description of the standby liquid control system actuation logic and its operability requirements, boron enrichment level, turbine bypass capacity, and location of the steam extraction points for the feedwater heaters.

RESPONSE

The equipment performance parameters used in the PBAPS EPU ATWS analysis are provided below:

a. The RPV lower plenum is the location for boron injection from the SLC system (SLCS). This is not a change from the current plant configuration.

b. There is no automatic actuation logic for SLCS. Operators manually initiate SLCS via key lock switches in the main control room consistent with PBAPS emergency operating procedures. ATWS analysis assumes SLCS is manually initiated at the later of either: 1) the time of high pressure ATWS RPT plus 120 seconds operator action time, or 2) the time at which the suppression pool temperature reaches the Boron Injection Initiation Temperature (BIIT). SLCS operability requirements are stated in PBAPS Technical Specification 3.1.7. Note that revised SLCS Technical Specifications for EPU are contained in the EPU LAR Attachments 2 (Unit 2) and 3 (Unit 3).

c. B10 enriched to at least 92%.

d. Turbine bypass capacity at EPU rated thermal power is 2.82×10^6 lbm/hr, which is unchanged from the turbine bypass capacity at current licensed thermal power. The turbine bypass is not credited in the PBAPS rated power ATWS analysis.

e. Steam extraction points for FW heaters are downstream of the MSIVs, such that FW heating is lost following isolation. The steam extraction points are listed below:

FWH	Extraction Steam Point Location
5th Stage FWH	HP Exhaust (Cross-around Steam)*
4th Stage FWH	AS2 stage Low Pressure Turbine*
3rd Stage FWH	AS3 stage Low Pressure Turbine*
2nd Stage FWH	AS6 stage Low Pressure Turbine*
1st Stage FWH	AS8 stage Low Pressure Turbine*
Drain Cooler	N/A – No Extraction Point

* see PBAPS Piping & Instrumentation Drawing M-304

SRXB RAI-14

Provide a short summary of the Solution III hardware currently installed in PBAPS. Provide justification that the hot channel oscillation magnitude portion of the Option III calculation is not affected by EPU because the OPRM hardware does not change.

RESPONSE

The stability Option III hardware for PBAPS is fully integrated into the NUMAC™ Power Range Neutron Monitoring (PRNM) System. The licensing basis for the PRNM retrofit at PBAPS is contained in References 14-1 and 14-2. The Option III Oscillating Power Range Monitor (OPRM) Channel is integral with each channel of the PRNM. [[

]]

References:

14-1 GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A, Volume 1 and 2, October 1995.

14-2 GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A Supplement 1, November 1997.

14-3 GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004-A, Revision 4, Class I (Non- Proprietary), July 2003.

SRXB RAI-15

Provide a summary of the ATWS emergency operating procedure (EOP) actions. What version of emergency operating guidelines is currently implemented in PBAPS? Provide a short description of the process used to ensure that the emergency procedure guideline variables (e.g., hot shutdown boron weight, heat capacity temperature limit, etc.) are adequate under CPPU conditions.

RESPONSE

The ATWS related EOP operator actions are summarized as follows:

- Manually SCRAM the reactor by placing the mode switch in shutdown.
- Manually initiate alternate rod insertion (ARI) to insert the control rods
- Manually trip the reactor recirculation pumps.
- Manually initiate boron injection with the SLCS if sustained power oscillations exceed 25% peak to peak.
- Manually initiate boron injection with the SLCS before torus temperature reaches 110°F.
- Perform actions to manually insert the control rods.
- Perform manual actions to minimize coolant injection to the Reactor Pressure Vessel (RPV) in order to lower RPV water level to between below -60 inches and above top of active fuel until all control rods are inserted or sufficient boron injection has occurred.
- Inhibit Automatic Depressurization System.
- Bypass Main Steam Isolation Valve isolation.

Revision 2 of the BWROG Emergency Procedure and Severe Accident Guidelines (EPG/SAGs) is currently implemented at PBAPS.

Calculation revisions to address these EPG variable changes are being made in accordance with the EGC configuration control process. This process also ensures impacted EOPs are updated to reflect the changes from the calculations as applicable.

SRXB RAI-16

Provide a short description of how the Stability Mitigation Actions (e.g., immediate water level reduction and early boron injection) are implemented in PBAPS. Does operation at CPPU conditions require modification of any operator instructions?

RESPONSE

PBAPS implemented the Option III stability solution to address Stability Mitigation Actions. This includes use of the power range neutron monitoring (PRNM) system to provide a signal to shut down the reactor when a thermal-hydraulic instability (THI) condition is detected. Oscillations in the neutron flux are used as an indicator of THI. The Oscillation Power Range Monitor (OPRM) Upscale Function provides compliance with GDC 12 by providing a hardware system that detects and acts to suppress THI conditions. If a transient occurs (e.g., trip of a reactor recirculation pump at 100% RTP), the PRNM system will automatically trip the reactor when the OPRM trip setpoint is exceeded.

If THI conditions are observed, procedures direct the operators to take the following actions:

- Manually insert control rods until a THI condition no longer exists and monitor indications for THI.
- SCRAM the reactor if APRM flux oscillations exceed an amplitude of 15% RTP.
- If a THI condition exists and a reactor SCRAM is unsuccessful (i.e., an ATWS event), then the operators will respond as follows:
  - Manually initiate Alternate Rod Insertion (ARI) to insert the control rods.
  - Manually trip the operating reactor recirculation pump(s).
  - Manually initiate boron injection with the SLCS if sustained power oscillations exceed 25% peak to peak.
  - Manually initiate boron injection with the SLCS before torus temperature reaches 110°F.
  - Perform actions to manually insert the control rods.
  - Perform manual actions to minimize coolant injection to the Reactor Pressure Vessel (RPV) in order to lower RPV water level to between below -60 inches and above top of active fuel until all control rods are inserted or sufficient boron injection has occurred.
  - Inhibit Automatic Depressurization System.
  - Bypass Main Steam Isolation Valve isolation.

There are no changes to operator instructions for the stability mitigation actions discussed above. However, due to the increase in boron-10 enrichment, EOP operator instructions will be revised to reflect a reduction in the percentage of SLC tank volume required to be injected by the SLCS to achieve hot shutdown boron weight for EPU conditions.

SRXB RAI-17

PBAPS currently operates under the Option III solution. Please provide clarification for the following areas:

a) Describe the process that was followed by PBAPS to implement Option III Long-Term Stability Solution and to verify that Option III is still applicable under CPPU operation.

b) Describe the expected effects of CPPU operation on Option III.

c) Describe any alternative method to provide detection and suppression of any mode of instability other than through the current OPRM scram.

d) Provide a summary of the PBAPS Technical Specifications affected by the Option III implementation and future CPPU operation.

e) List the approved methodologies used to calculate the OPRM setpoint by the current operation and future PBAPS CPPU operation.

RESPONSE

a) The process followed by PBAPS to implement the Option III Long Term Stability Solution is described in the PRNM LARs (References 17-1 and 17-2) and the related NRC Safety Evaluation Report (Reference 17-8.) Validity of the Option III solution at EPU conditions has been shown generically in Reference 17-3. The Option III solution has plant and cycle-specific features, such as the OPRM Trip-Enabled region, OPRM trip setpoints, and Backup Stability Protection regions. Section 2.8.3 of Reference 17-4 established the basis for the plant-specific feature, namely the OPRM Trip-Enabled region at EPU conditions. A demonstration analysis for the EPU conditions is also presented in Section 2.8.3 of Reference 17-4. The cycle-specific features are included with the reload analysis.

b) The EPU expands the operating domain in a region of the power to flow map where the plant is not susceptible to thermal-hydraulic instability events. The effects of PBAPS EPU implementation for the Option III stability solution are described in Section 2.8.3 of Reference 17-4. Furthermore, the OPRM Setpoints and Backup Stability Protection regions are generated, and the OPRM Trip-Enabled region boundaries are confirmed each reload. Therefore EPU does not affect the applicability of the Option III solution.

c) The Backup Stability Protection at PBAPS is discussed in the response to RAI-3 for the PRNM LAR (Reference 17-5.) There is no change to the Backup Stability Protection implementation with EPU.

d) The changes to the Technical Specifications (TS) due to implementation and activation of the Option III long-term stability solution for CLTP conditions at PBAPS are described in the related NRC SER (Reference 17-8). EPU affects TS LCO 3.3.1.1 Condition J, SR 3.3.1.1.19 and Table 3.3.1.1-1 Function 2.f. The changes to update the PRNMS TS for EPU are described in Section 3.1.8 of the EPU LAR (Reference 17-4, Attachment 1) and in EPU LAR Supplement No. 5 (Reference 17-9).

e) There are no differences in the methodology used to determine the OPRM setpoints for either CLTP or EPU conditions; that methodology is as specified in References 17-6 and 17-7. However, it should be noted that the setpoint penalties discussed in the SRXB RAI-8 response are applied at EPU conditions. The OPRM setpoints are determined each cycle as a part of the reload analysis.

References:

17-1 PECO Energy Company letter to the NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 License Change Request ECR 98-01802," dated March 1, 1999.

17-2 Exelon letter to the NRC, "License Amendment Request, Activation of the Trip Outputs of the Oscillation Power Range Monitor Portion of the Power Range Neutron Monitoring System," dated February 27, 2004 (NRC Accession Number ML0407008073.)

17-3 GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004-A, Revision 4, Class I (Non- Proprietary), July 2003.

17-4 Exelon letter to the NRC, "License Amendment Request – Extended Power Uprate," dated September 28, 2012, Attachments 4 and 6 (NRC Accession No. ML122860201.)

17-5 Exelon letter to the NRC, "Responses to Request for Additional Information, License Amendment Request, Activation of the Trip Outputs of the Oscillation Power Range Monitor Portion of the Power Range Neutron Monitoring System," dated September 13, 2004 (NRC Accession No. ML042580401.)

17-6 GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, Class I (Non-proprietary), August 1996.

17-7 GE Hitachi Nuclear Energy, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465 Supplement 1, September 2011.

17-8 NRC letter to Exelon, "Activation of Oscillation Power Range Monitor Trip (TAC Nos. MC2219 and MC2220)," dated March 21, 2005 (NRC Accession No. ML050270020.)

17-9 Exelon letter to the NRC, "Supplemental Information Supporting Request for License Amendment Request - Extended Power Uprate – Supplement No. 5," dated June 27, 2013 (NRC Accession No. ML042580401.)

SRXB RAI-18

Provide a table of hot channel and core-wide decay ratios at the most limiting state point for the last cycles and the proposed CPPU condition. The purpose is to evaluate the impact of CPPU on relative stability of the plant, and the applicability of Option III to PBAPS under these new conditions.

RESPONSE

The core decay ratio and hot channel decay ratio were calculated at the intersection of the Natural Circulation Line and High Flow Control Line for the same EPU equilibrium core that was used in the demonstration analyses in Section 2.8.3 of Reference 18-1. The decay ratios were also calculated for the current Peach Bottom 2 Cycle 20 reload core design, at the same absolute power / flow values. The results are summarized in the table below. As can be seen from the table, the difference between the decay ratios calculated for EPU and CLTP conditions is small. These results are representative of both Peach Bottom Units 2 and 3.

Also note that the Backup Stability Protection regions are generated for each reload. Therefore, while a change in decay ratio may affect the size of the scram and controlled entry regions, it does not affect the applicability of Option III.

Rated Power (MWt)	Power		Core Flow		Core Decay Ratio	Hot Channel Decay Ratio
	(MWt)	%	(Mlb/hr)	%		
3951	1953.1	49.4	32.08	31.3	1.03	0.35
3514	1953.9	55.6	32.08	31.3	1.08	0.36

References:

18-1 Exelon letter to the NRC, "License Amendment Request – Extended Power Uprate," dated September 28, 2012 (NRC Accession No. ML122860201), Attachments 4 and 6.

SRXB RAI-19

Describe the effects or impacts, if any, of EPU on suppression pool cooling during isolation ATWS events and/or EOPs.

RESPONSE

As noted in the NRC Safety Evaluation for the GEH Constant Pressure Power Uprate Licensing Topical Report (Reference 19-1), "[[

]]" These actions are consistent with the BWROG EPG/SAGs.

The peak suppression pool temperature response to an ATWS event at PBAPS is lower at EPU conditions as compared to CLTP conditions due to elimination of Containment Accident Pressure (CAP) credit (Reference 19-2, Table 2.8-8).

The EOPs will require revision to incorporate the changes associated with modifications for CAP credit elimination (i.e., the enriched boron-10 modification and the Condensate Storage Tank modification), as described in PUSAR Section 2.11, Human Factors (Reference 19-3, Attachment 4).

References

19-1 GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004-A, Revision 4, Class I (Non- Proprietary), July 2003.
19-2 Exelon letter to the NRC, "License Amendment Request – Extended Power Uprate," dated September 28, 2012, Attachments 4 and 6 (NRC Accession No. ML122860201.)
19-3 Exelon letter to the NRC, "Supplemental Information Supporting Request for License Amendment Request - Extended Power Uprate – Supplement No. 5," dated June 27, 2013 (NRC Accession No. ML042580401.)

SRXB RAI-20

Please provide a short description of the simulator neutronic core model. Also, provide the schedule to show when the PBAPS simulator will be upgraded for EPU conditions.

RESPONSE

The PBAPS simulator neutronic core model is a Studsvik Simulate 3 Real-Time (S3R) model, based upon the Studsvik-Scandpower CASMO-SIMULATE engineering code. S3R is the real-time version of SIMULATE-3K, a best-estimate transient analysis code.

The upgrades to the PBAPS simulator for EPU conditions are currently scheduled to be completed in May 2014 in order to support operator training prior to the EPU implementation outage (Fall 2014.)

SRXB RAI-21

PUSAR Section 2.8.5.7.3 states that the highest calculated PCT for ATWS events is 1342 °F, during the Pressure Regulator Failure Open event. The submittal states that:

> Local cladding oxidation is not explicitly analyzed because, with PCT less than 1600 °F, cladding oxidation has been demonstrated to be insignificant compared to the acceptance criteria of 17% of cladding thickness. Therefore, the local cladding oxidation for the PBAPS ATWS events is qualitatively evaluated to show compliance with the acceptance criteria of 10 CFR 50.46.

Please provide a reference to show where cladding oxidation has been demonstrated to be insignificant when the PCT is less than 1600 °F during ATWS events.

RESPONSE

The discussion that follows provides references and discussion for why cladding oxidation has been demonstrated to be insignificant when PCT is less than 1600°F.

Section 3.4.3 of the Safety Evaluation for Reference 21-1 originally restricted upper bound PCT to 1600°F because: (a) the range of test data submitted as part of the code qualification extended only to 1600°F, and (b) the Monte Carlo Simulation presented in the SAFER Licensing Topical Report (LTR) was performed over a temperature range where effects such as metal-water reaction are negligible. Reference 21-2 was issued to remove the 1600°F limitation for the licensing basis PCT. The following is an excerpt describing the metal-water reaction as a function of temperature:

> *"The metal-water reaction does not become a factor until the cladding temperatures reach 1700°F and does not become significant until the cladding temperatures exceed 1800°F. When the upper bound PCT approaches 1800°F (where metal-water reaction is just beginning to become significant), the licensing basis PCT will be approaching 2200°F where it would be restricted by the 50.46 limit."*

The following figure from Reference 21-3 illustrates the Baker-Just zircaloy-water reaction equation used in the SAFER method which demonstrates that cladding oxidation is not significant below 1800°F.

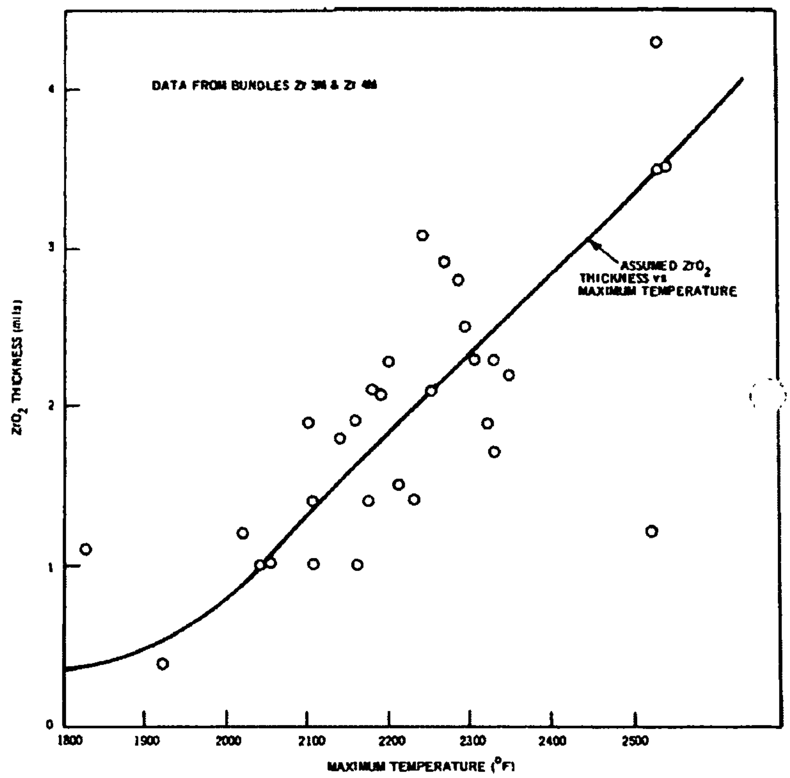

Figure B-1 ZrO_2 Thickness as a Function of Maximum Temperature

Because the criteria to assure coolable core geometry (2200°F PCT and 17% local cladding oxidation thickness limit) for a loss of coolant accident are also applicable to an ATWS, the above references and discussion that demonstrate cladding oxidation is insignificant when PCT is less than 1600°F are also applicable to the ATWS analysis.

References

21-1 GE Nuclear Energy, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-Of-Coolant Accident Volume III," NEDE-23785-1-PA, Revision 1, (Proprietary), October 1984.

21-2 GE Nuclear Energy, "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, March 2002.

21-3 BWR FLECHT Final Report, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," GEAP-13197, June 1971.

SRXB RAI-22

With respect to overpressure protection (i.e., Section 2.8.4.2 of the PUSAR), if an analysis was performed for the Turbine Trip with Bypass Failure and Scram on High Flux (TTNBPF) event, as it is required in Table E-1 of ELTR-1, please provide a plot comparing the pressure transients for the Main Steam Isolation Valve Closure with Scram on High Flux and the TTNBPF events. If a TTNBPF analysis was not performed for EPU, then justify why not.

RESPONSE

An analysis was performed for the TTNBPF event, as required by Table E-1 of ELTR-1. The comparison plot of the MSIVF and TTNBPF events is provided in Figure RAI-22-1. The MSIVF event is clearly more limiting for both dome and reactor vessel bottom pressure.

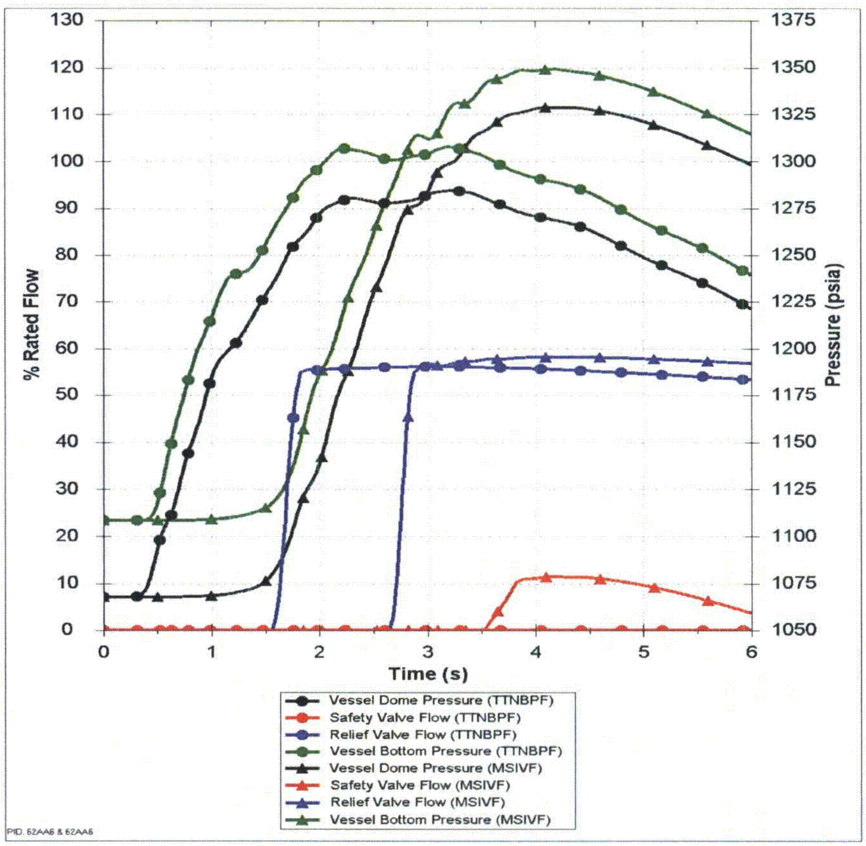

Figure RAI-22-1: Results Comparison for MSIVF and TTNBPF Events

SRXB RAI-23

Do the decay heat removal requirements change between current and EPU power levels due to any changes in decay heat load or suppression pool temperature? If so, what are the new requirements and how does the reactor core isolation cooling (RCIC) system meet the new criteria without updating the system performance?

RESPONSE

As stated in Section 2.8.4.3.1 of Reference 23-1, the only design requirement of the RCIC system is to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of flow from the FW system. The system design injection rate must be sufficient for compliance with the system limiting criteria to maintain the reactor water level above Top of Active Fuel (TAF) at EPU conditions. EPU does increase the amount of decay heat for the RCIC system to remove, however the requirements remain unchanged. The RCIC system design capabilities (flow, head, etc.) are sufficient to accomplish this design requirement as demonstrated by analysis. The results of the analysis presented in Section 2.8.5.2.3 of Reference 23-1 demonstrate that the RCIC system meets this design requirement at EPU conditions with no changes to the RCIC hardware or flow capability that currently exists at PBAPS current licensed thermal power level (CLTP).

The analysis of events in which RCIC operation may be credited, Appendix R Method A (described in Reference 23-1 Section 2.5.1.4), Station Blackout (described in Reference 23-1 Section 2.3.5), and ATWS (described in Reference 23-1 Section 2.8.5.7), did not assume any increased flow capability of the RCIC system from CLTP. CLTP performance characteristics of the RCIC system are adequate to mitigate these events for EPU.

References

23-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-24

Table 4.7.1 in the PBAPS Updated Final Safety Analysis Report (UFSAR) shows that the RCIC system pump has a design temperature range of 40 °F to 140 °F. Are there any instances under EPU conditions where the pump would be operating outside of this temperature range? If so, what are the conditions and how are they addressed for this EPU?

RESPONSE

The only EPU analyses in which RCIC operation is credited are: Appendix R Method A (described in Reference 24-1 Section 2.5.1.4), Station Blackout (described in Reference 24-1 Section 2.3.5), ATWS (described in Reference 24-1 Section 2.8.5.7) and Loss of Feedwater Flow Event (described in Reference 24-1 Section 2.8.5.2.3.1). For the loss of feedwater flow event, there is no elevated suppression pool temperature. For the Appendix R, Station Blackout and ATWS analyses, the RCIC pump suction source credited is exclusively from the condensate storage tank, which has a temperature range of 40 °F to 140 °F. Therefore, there are no safety analyses for EPU where RCIC would operate outside the design temperature range of 40 °F to 140 °F.

References

24-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-25

What is the effect on net positive suction head for the reactor recirculation system for EPU? The PUSAR stated this result is based on past uprate analyses. Explain the past analyses and their relevance.

RESPONSE

Plant-specific evaluation of the Reactor Recirculation (RR) Pump NPSH was performed for the PBAPS EPU. The RR pump NPSH available at EPU conditions increases to 563.03 feet from the CLTP value of 514.71 feet. This increase in NPSH available is due to the higher feedwater flow contribution as a function of total core flow, resulting in colder recirculation pump flow. The net increase in subcooling increases the RR pump NPSH available at EPU conditions. Because the maximum core flow does not change at EPU conditions and the core flow resistance at EPU conditions is only slightly increased, the NPSH required for the RR pumps is essentially unchanged from CLTP. Therefore, the RR pump NPSH margin for PBAPS (available NPSH minus required NPSH) increases at EPU conditions.

The statement, "Based on past uprate analyses, the NPSH required at full power does not significantly increase or reduce the NPSH margin because the required increase in recirculation flow is small," contained in Section 2.8.4.6.1 of Reference 25-1 essentially refers back to the [[

]] concerning RR pump NPSH. The plant-specific evaluation performed for PBAPS reconfirms that the Reference 25-2 [[
]]

References

25-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

25-2 GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004-A, Revision 4, Class I (Non- Proprietary), July 2003.

SRXB RAI-26

Section 2.8.5.6.2.5 of the PUSAR states that the licensing basis PCT is 1925 °F based on the most limiting Appendix K case, including a variable plant uncertainty term. Please provide further explanation regarding the "plant variable uncertainty term."

RESPONSE

The ECCS-LOCA analysis is performed with the SAFER model, which is considered as a representative or non-mechanistic model. Nominal plant operating parameters and conditions are input as the basis of the analysis to calculate the peak cladding temperature. Conservatisms are then explicitly added to the peak cladding temperature based on nominal conditions so the result assures compliance with the regulatory requirements. This is summarized in an equation in Reference 26-1 Section 3.1 with the peak cladding temperature based on nominal conditions ($PCT_{Nominal}$) calculated by the SAFER model and conservatisms explicitly added (ADDER) to determine the licensing basis peak cladding temperature ($PCT_{Licensing\ Basis}$). The equation from Reference 26-1 Section 3.1 is also shown below.

$$PCT_{Licensing\ Basis} = PCT_{Nominal} + ADDER$$

The ADDER in the above equation to account for conservatisms is calculated as follows:

$$ADDER^2 = [PCT_{Appendix\ K} - PCT_{Nominal}]^2 + \Sigma(\delta PCT_i)^2$$

Where:

$PCT_{Appendix\ K}$ = Peak cladding temperature from calculation using Appendix K specified models and inputs.

$PCT_{Nominal}$ = Peak cladding temperature from the nominal conditions case.

$\Sigma(\delta PCT_i)^2$ = Plant variable uncertainty term.

The nominal case ($PCT_{Nominal}$) is based on the Appendix K case with the most limiting PCT for all analyzed operating conditions, break locations, break sizes and single failure. The ADDER comprises two terms:

- The first term, $[PCT_{Appendix\ K} - PCT_{Nominal}]^2$, in the ADDER incorporates model specifications required by Appendix K that are not already included in the nominal calculation. The Appendix K model specification for calculation of Licensing Basis PCT using the SAFER model is listed in Table 26-1.
- The second term, $\Sigma(\delta PCTi)^2$, in the ADDER is the plant variable uncertainty term. The intent of the plant variable uncertainty term is to include uncertainties in plant variables not specifically required in the Appendix K model specifications listed in Table 26-1. Reference 26-1 documents a survey spanning many sources of variable uncertainty from which a set of prominent items were justified for inclusion in the standard methodology. The uncertainties in plant variables are fuel product line dependent and listed in Table 26-2.

The plant variable uncertainty term is a sum of squares regarding the change in calculated PCT when a single plant variable listed in Table 26-2 is perturbed to an upper bound value while the other plant variables listed in Table 26-2 are at best estimate values.

Table 26-1 - Appendix K Model Specification for Licensing PCT Using SAFER

1971 ANS + 20% Decay Heat
Moody Slip Flow Model with discharge coefficients of [[ ]]
Baker-Just Metal Water Reaction Rate
Transition boiling allowed during blow down only until cladding superheat exceeds [[ ]]
102% bundle power and at least 102% core power
Technical Specification MCPR limit
PLHGR consistent with Technical Specification MAPLHGR for selected bundle type
Worst Single Failure
Fuel Exposure which maximizes PCT or stored energy

Table 26-2 - Plant Variables Perturbed

[[
]]

Reference

26-1 GE Nuclear Energy, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss of Coolant Accident Volume III," NEDE-23785-1-PA, Class III, Revision 1, October 1984.

SRXB RAI-27

Page 2-396 of the PUSAR states that, independent of the EPU, the licensee will be replacing the recirculation system pump motor power supplies from motor/generator set power supplies to adjustable speed drives (ASDs). Section 2.8.5.2.1 of the PUSAR discusses Loss of External Load and Turbine Trip events with specific evaluations for the Generator Load Rejection with Steam Bypass Failure (LRNBP) event and the Trip with Steam Bypass Failure event. Results of the transient analysis, shown in PUSAR Table 2.8-12, indicates that LRNBP is the limiting event with a delta CPR of 0.27. Section 2.8.5.2.1 indicates that, [[

]] Please specify the resulting delta CPR
[[ ]]

RESPONSE

The delta CPR with the ASDs installed and the EOC-RPT out of service is 0.30. This result cannot be directly compared to the LRNBP delta CPR of 0.27 in Table 2.8-12 of Reference 27-1 to determine the effect of the ASD because the Table 2.8-12 result considers the EOC-RPT in service. When the ASD with EOC-RPT out of service delta CPR (0.30) result is compared to the M/G set with EOC-RPT out of service delta CPR (0.30), the effect of the ASD is negligible. This is due to the ATWS-RPT occurring approximately one second into the transient, thus limiting any benefit due to the ASD, similar to the effect of the EOC-RPT when in service.

PBAPS is not planning to install the ASD modification until 2015 for Unit 3 and 2016 for Unit 2. EPU does not rely on this modification, nor is approval of this modification requested.

References

27-1 Letter from K. F. Borton (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request – Extended Power Uprate," dated September 28, 2012. (ML122860201), Attachments 4 and 6.

SRXB RAI-28

Recent operating experience has shown that, at a similar BWR/4, the events that follow a loss of stator cooling (LOSC) could cause a situation that is limiting with respect to the MCPR. Please explain whether the LOSC has a potential to be CPR-limiting at PBAPS. If the LOSC is non-CPR limiting, explain what design features exist to provide protection from a LOSC. If the LOSC is a CPR-limiting event, please explain what affect the EPU could have on the severity of the event, and how the EPU safety analyses address the event.

RESPONSE

The LOSC event was evaluated for PBAPS and was determined to not be potentially limiting with respect to the minimum critical power ratio (MCPR). At the high power conditions (including EPU conditions), the plant is operating closer to the high pressure reactor scram (mitigating scram) and the LOSC is not limiting at these conditions. At off-rated conditions, the plant is operating further away from the high pressure scram. Evaluations at these off-rated conditions demonstrate that the PBAPS off-rated critical power ratio limits bound the LOSC event.

The design feature that caused the MCPR to be limiting for the LOSC for the similar BWR/4 was a recirculation pump trip at the initiation of the LOSC. The LOSC sequence of events for PBAPS does not have an automatic recirculation system pump trip or recirculation runback.

ACRONYM LIST

ACRONYM	DEFINITION
ΔCPR	Delta critical power ratio
2PT	Two pump trip
ADS	Automatic Depressurization System
AOO	Anticipated operating occurrence
APLHGR	Average planar linear hear generation rate
APRM	Average power range monitor
ARI	Alternate rod insertion
ASD	Adjustable speed drive
ATWS	Anticipated transient without scram
BIIT	Boron injection initiation temperature
BSP	Backup stability protection
BWROG	Boiling Water Reactor Owners Group
CAP	Containment accident pressure
CASMO	Computer code name
CLTR	Constant Pressure Power Uprate topical report – Reference 2
CPPU	Constant pressure power uprate
CPR	Critical Power Ratio
DIVOM	Delta CPR over Initial CPR Versus Oscillation Magnitude
ECCS	Emergency core cooling system
EGC	Exelon Generation Company
ELTR1	Extended power uprate topical report – Reference 3
ELTR2	Extended power uprate topical report – Reference 6
EOC-RPT	End of cycle recirculation pump trip
EOP	Emergency operating procedure
EPU	Extended power uprate
FW	Feedwater
FWH	Feedwater heater

ACRONYM	DEFINITION
GEH	General Electric - Hitachi
GESTAR-II	Core design topical report – Reference 5
GE*xx*	A fuel type (e.g., GE9, GE14)
GNF	Global Nuclear Fuel
GNF2	A fuel type
GWD/MTU	Unit of exposure; gigawatt day per metric ton uranium
HP	High pressure
ICPR	Incremental critical power ratio
IMCPR	Initial minimum critical power ratio
ISCOR	A computer code
LAR	License amendment request
LHGR	Linear heat generation rate
LOCA	Loss of coolant accident
LOFW	Loss of feedwater
LOSC	Loss of stator cooling
LPRM	Local power range monitor
LRNBP	Generator load rejection with steam bypass failure event
LTR	Licensing Topical Report
MAPLHGR	Maximum average planar linear heat generation rate
MCPR	Minimum critical power ratio
MELLLA	Maximum extended load line limits analysis; current operating domain
M/G	Motor – generator
Mlb/hr	Thousand pounds per hour
MSIV	Main steam isolation valve
MSIVF	Main steam isolation valve closure with SCRAM on high flux
MW	Megawatt
MWt	Megawatt thermal
N/A	Not applicable

ACRONYM	DEFINITION
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NUMAC	Trademark brand of the Power Range Neutron Monitoring System
ODYSY	A computer code
OLMCPR	Operating limit minimum critical power ratio
OPRM	Operating power range monitor
PANAC	A computer code
PBAPS	Peach Bottom Atomic Power Station
PCI	Pellet-clad interaction
PCT	Pellet clad temperature
PLHGR	Planar linear heat generation rate
PRNM	Power range neutron monitoring
PUSAR	Power uprate safety analysis report – Reference 1
RAI	Request for additional information (from NRC)
RCIC	Reactor core isolation cooling system
RPV	Reactor pressure vessel
RR	Reactor recirculation
S3R	A computer code
SAFDL	Specified acceptable fuel design limit
SAFER	A computer code
SAG	Severe accident guidelines
SCC	Stress corrosion cracking
SLCS	Standby liquid control system
SLMCPR	Safety limit minimum critical power ratio
SNPB	Performance and Code Review Branch of the NRC
SRXB	Reactor Systems Review Branch of the NRC
TAF	Top of active fuel
THI	Thermal-hydraulic instability

ACRONYM	DEFINITION
TS	**Technical Specification**
TTNBPF	**Turbine trip with bypass failure and SCRAM on high flux event**
UFSAR	**Updated Final Safety Analysis Report**

Attachment 3

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

Response to Request for Additional Information – AADB

Response to Request for Additional Information

Accident Dose Branch

By letter dated September 28, 2012, Exelon Generation Company, LLC (Exelon) submitted a license amendment request for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendment would authorize an increase in the maximum power level from 3514 megawatts thermal (MWt) to 3951 MWt. The requested change, referred to as an extended power uprate (EPU), represents an increase of approximately 12.4 percent above the current licensed thermal power level.

Exelon provided a response to an initial request for additional information from the AADB in its EPU LAR Supplement 3, dated May 24, 2013 (ADAMS Accession No. ML13149A145.) The NRC staff has reviewed the information supporting the proposed amendment and by letter dated July 23, 2013 (ADAMS Accession No. ML13203A100) has requested information to clarify the submittal. The response to that request is provided below.

AADB-RAI-2

In Section 2.1.5 of Attachment 1 to Exelon's letter dated May 24, 2013, the licensee stated that:

> The EPU Main Steam Line Break (MSLB) Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors are updated using a site-specific X/Q calculation. This differs from the CLB [current licensing basis] MSLB evaluation which used X/Q values calculated using guidance from RG [Regulatory Guide] 1.5.

Please provide a description of the calculation used for the updated MSLB X/Q values. Include a discussion of how it differs from the CLB MSLB evaluation, a justification for its use, and all inputs and assumptions used to make the calculation.

RESPONSE

All atmospheric dispersion factors (χ/Q) utilized by the PBAPS EPU dose calculations were previously supplied to the NRC Staff during the PBAPS AST submittal (Reference 2-1). The supplied information included code inputs, code outputs, and calc notes describing their generation and use.

The CLB AST LOCA, CRDA and FHA EAB and LPZ χ/Q values were generated by a site-specific PAVAN calculation. For each location, all three values are the same, and for simplicity, they are called "LOCA" in this response.

The CLB AST MSLB EAB and LPZ atmospheric dispersion factors were based upon RG 1.5.

Atmospheric dispersion factors are a function of release point, receptor point, and other site-specific geography, layout, and meteorological data. Because there is no dependence upon licensed core thermal power, the AST atmospheric dispersion factors generated at CLB are applicable to EPU.

For the EPU MSLB dose calculation, additional conservatism was added to the dose results by applying the higher PAVAN-calculated AST LOCA, 0-2 hr ground release, values rather than the RG 1.5 AST MSLB values. Consequently:

$$(\chi/Q)_{EAB,MSLB,EPU} = (\chi/Q)_{EAB,LOCA,AST} = 9.11 \times 10^{-4}\ s/m^3 \text{ and}$$

$$(\chi/Q)_{LPZ,MSLB,EPU} = (\chi/Q)_{LPZ,LOCA,AST} = 1.38 \times 10^{-4}\ s/m^3$$

Larger χ/Qs are more conservative because the χ/Q is a multiplier within the dose calculation. Therefore, larger χ/Qs generate higher dose results, and it is acceptable to use the larger CLB LOCA χ/Q values for the EPU MSLB accident.

References

2-1. Exelon letter to U. S. Nuclear Regulatory Commission, "License Amendment Request - Application of Alternative Source Term," dated July 13, 2007.

Attachment 4

Peach Bottom Atomic Power Station Units 2 and 3

NRC Docket Nos. 50-277 and 50-278

AFFIDAVIT

Note

Attachment 1 contains proprietary information as defined by 10 CFR 2.390. GEH, as the owner of the proprietary information, has executed the enclosed affidavit, which identifies that the proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information has been faithfully reproduced in the attachment such that the affidavit remains applicable.

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am the Vice President Fuel Licensing of GE-Hitachi Nuclear Energy Americas LLC (GEH), 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 GEH letter, GEH-PBAPS-EPU-427, "GEH Response to NRC SRXB RAIs 2-14, 17-19, 21-28," dated August 16, 2013. The GEH proprietary information in Enclosure 1, which is entitled "GEH Response to NRC SRXB RAIs 2-14, 17-19, 21-28," is identified by a dark red dotted underline inside double square brackets. [[This sentence is an example.{3}]]. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* (FOIA), 5 U.S.C. Sec. 552(b)(4), and the Trade Secrets Act, 18 U.S.C. 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 qualifies 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, 975 F.2.d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2.d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that 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 GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH or other companies.

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

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, 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 GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to 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 for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH 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 or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains results of analyses performed using the GEH EPU methodology including proprietary technical methods and processes. Development of these methodologies and the supporting analysis techniques and information, and their application to the design, modification, and processes were achieved at a significant cost to GEH.

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

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH'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 GEH. 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. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH 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 GEH 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 GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining 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 16th day of August, 2013.

James F. Harrison

James F. Harrison
Vice President Fuel Licensing
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd
Wilmington, NC 28401
james.harrison@ge.com