February 16, 2007

Mr. William Levis
Senior Vice President & Chief Nuclear Officer
PSEG Nuclear LLC-X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST FOR EXTENDED POWER UPRATE (TAC NO. MD3002)

Dear Mr. Levis:

By letter dated September 18, 2006 (Agencywide Documents and Management System (ADAMS) Accession No. ML062680451), as supplemented on October 10, 2006 (Accession No. ML062920092), and October 20, 2006 (Accession No. ML063110164) PSEG Nuclear, LLC submitted an amendment request for an extended power uprate at the Hope Creek Nuclear Generating Station. The proposed amendment would increase the authorized maximum power level by approximately 15 percent, from 3339 megawatts thermal (MWt) to 3840 MWt.

The Nuclear Regulatory Commission (NRC) staff has been reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The questions were sent by e-mail to you on January 26, 2007 (Accession No. ML070300277), to ensure that the questions were understandable, the regulatory basis was clear and to determine if the information was previously docketed. In subsequent discussions with your staff some questions were deleted, revised or removed for further clarification. Paul Duke of your staff agreed to respond within 30 days from the date of this letter for all questions with the exception of questions 3.18 and 3.29 where you requested a 60 day response.

Please note that if you do not respond to this letter within the prescribed response times or provide an acceptable alternate date in writing, we may reject your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Section 2.108. If you have any questions, I can be reached at (301) 415-1388.

Sincerely,

/RA/

James J. Shea, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

cc w/encl: See next page

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Hope Creek Generating Station

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING TECHNICAL SPECIFICATION CHANGES FOR

EXTENDED POWER UPRATE

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

By letter dated September 18, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062680451), as supplemented on October 10, 2006 (Accession No. ML062920092), and October 20, 2006 (Accession No. ML063110164) PSEG Nuclear, LLC (PSEG or the licensee) submitted an amendment request for an extended power uprate (EPU) at the Hope Creek Nuclear Generating Station (Hope Creek). The proposed amendment would increase the authorized maximum power level by approximately 15 percent, from 3339 megawatts thermal (MWt) to 3840 MWt.

The Nuclear Regulatory Commission (NRC) staff has been reviewing the submittal and has determined that additional information is needed to complete its review.

1) Vessels & Internals Integrity Branch (CVIB)

1.1 Section 3.2.1, “Fracture Toughness,” of Attachment 4 to your submittal dated September 18, 2006, indicates that Hope Creek is participating in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), and will comply with the withdrawal schedule specified in the program, which is documented in BWRVIP-86, “Updated BWR Integrated Surveillance Program (ISP) Implementation Plan.” The ISP specifies that Hope Creek shall remove the next surveillance capsule when its fluence is approximately equal to the reactor pressure vessel 1/4 thickness (1/4T) end of license (EOL) fluence. The next Hope Creek surveillance capsule is scheduled to reach this fluence and be withdrawn at 22 effective full power years (EFPY). The licensee states that the withdrawal schedule is not changed by the EPU. However, under EPU conditions, the licensee estimates that the next Hope Creek surveillance capsule will reach the unit's 1/4T EOL fluence at approximately 23 EFPY. Since BWRVIP-86, Tables 4-4 and 4-5 specify a withdrawal schedule in the year 2014 based on 22 EFPY, provide the following information.

a) Under EPU conditions, what is the projected fluence for this surveillance capsule at 23 EFPY?

b) Explain how 23 EFPY corresponds to the BWRVIP-86 surveillance capsule withdrawal date of 2014. If 23 EFPY does not correspond to the surveillance capsule withdrawal date of 2014, then determine whether any changes to BWRVIP-86 and BWRVIP-116, "Updated BWR Integrated Surveillance Program (ISP) Implementation Plan. License Renewal" are required.

1.2 As addressed in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix H, Section III (C)(1)(d), and in the NRC staff-approved BWRVIP ISP (BWRVIP-86), maintaining adequate contingencies to support potential changes to the program is an important part of any ISP. It should be noted that the BWRVIP considers the "standby" Hope Creek surveillance capsule to be a license renewal candidate for the ISP as documented in BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal." Therefore, the staff requests the licensee to confirm whether the third capsule (i.e., the one designated as a "standby" capsule in the Hope Creek Updated Final Safety Analysis Report (UFSAR), Section 5.3.1.6.1) will continue to reside in the reactor vessel and be tested in accordance with BWRVIP-116 to support the ISP, and satisfy the contingency requirement of 10 CFR Part 50, Appendix H, Section III (C)(1)(d).

1.3. Section 3.2, "Reactor Vessel," of Attachment 4 to your submittal dated September 18, 2006, states that the upper shelf energy evaluation of the reactor pressure vessel is provided in Table 3-2, "Hope Creek Upper Shelf Energy - 40 Year Life (32 EFPY)." In Table 3-2, heat 10024/1 for the LPCI nozzle forging specifies a copper content of 0.14 percent. However, in the Hope Creek UFSAR, Appendix 5A, Table 5A-6, "Heat Treatment and Chemical, Mechanical Properties of Nozzle Material," specifies a copper content of 0.15, while the NRC Reactor Vessel Integrity Database (RVID) specifies a copper content of 0.35 percent. In order for the NRC staff to verify the upper shelf energy evaluation, confirm the copper content of heat 10024/1 for the LPCI nozzle forging, and the basis for the copper content determination. Include a revised upper shelf energy evaluation, if necessary.

1.4 Section 10.7, "Plant Life," of Attachment 4 to your submittal dated September 18, 2006, identifies irradiation-assisted stress corrosion cracking (IASCC) as a degradation mechanism influenced by increases in neutron fluence. The licensee also stated that it has a procedurally controlled program that is consistent with the BWRVIP issued documents for the augmented nondestructive examination of selected reactor vessel internal components (core spray piping, core spray spargers, core shroud and core shroud supports, jet pumps and associated components, top guide, lower plenum, vessel ID (inner diameter) attachment welds, and feedwater sparger) in order to ensure their continued structural integrity. In addition, only two components, the top guide and the shroud, are predicted to exceed the BWRVIP-26 threshold fluence level of 5×10^{20} n/cm^2 (E > 1 MeV). This section indicates that the current inspection strategy for reactor internal components is expected to be adequate to manage any potential effects of EPU. Based on this, provide the following:

a) Confirm that the core plate, in-core flux monitoring guide tubes, and control rod guide tubes were considered in the determination of which components exceed the BWRVIP-26 threshold fluence level and become susceptible to cracking due to IASCC.

b) For each vessel internals component that exceeds the IASCC threshold, clarify the current inspection program to be utilized in managing IASCC of the component. Identify the scope, sample size, inspection method, frequency of examination and acceptance criteria for the inspection programs.

c) What BWRVIP documents are credited for these inspections? List any deviations or exceptions to these documents.

1.5 Section 10.7 of the Attachment 4 to the licensee's letter dated September 18, 2006, indicates Hope Creek injects hydrogen in the primary system for intergranular stress corrosion cracking (IGSCC) mitigation in the recirculation piping. Reactor water chemistry conditions are maintained consistent with Electric Power Research Institute (EPRI) and established industry guidelines. However, the NRC staff notes that other reactor vessel internals may be susceptible to stress corrosion cracking (SCC) and IGSCC. Therefore, what are the other water chemistry conditions, or other mitigation strategies (i.e., BWRVIP guidelines) that are used to mitigate the potential for SCC and IGSCC for the internal components?

2) <u>Electrical Engineering Branch A (EEEA)</u>

2.1 Attachment 9 of the license's amendment request states that PJM studies are documented in the Artificial Operating Guide (PSEG Engineering Evaluation A-5-500-EEE-1686). Demonstrate that this PJM study, completed in May 2003 and followed up with a facility study in February 2005, bounds the current grid conditions.

2.2 Describe how the Hope Creek EPU power levels have been coordinated with the Mid-Atlantic Area Reliability Council to assure that tripping of Hope Creek at the EPU level will not cause inadequate post-trip voltages at the safety-related buses.

2.3 Describe the impact of the Hope Creek modifications (identified in Attachment 5 of the license amendment request), both completed and planned, on the grid impact analyses. Have these modifications been coordinated with the Mid-Atlantic Area Reliability Council?

2.4 Describe the 'case-specific analysis' that was performed in determining that the radiation life of Target Rock solenoid valves inside containment could be extended up to the design life of Hope Creek.

2.5 Describe the 'case-specific analysis' that was performed in determining that the radiation life of Barksdale Pressure Switches outside containment could be extended up to the design life of Hope Creek.

3) <u>BWR Systems Branch (SBWB)</u>

<u>Fuel Storage</u>

3.1 As shown in NRC Review Standard RS-001, BWR Template Safety Evaluation as revised for Hope Creek, Section 2.8.6, "Fuel Storage," General Design Criteria (GDC)-62 is applicable to the NRC's review of the effect of the proposed EPU on fuel storage. This GDC requires the licensee to address prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. The NRC staff did not find any discussion on criticality of new and/or spent fuel storage in the licensee's submittals (PUSAR). Therefore, please provide this information considering the following:

a) These calculations should consider higher enrichment level, lattice exposure and the change in elements/isotopes including plutonium as a result of EPU operation.

b) In light of the fact that the void fraction history at uprated power will be appreciably higher than at current power level, clarify if the assumption made in your calculation regarding void history is consistent with the EPU conditions.

3.2 Question removed for Revision.

3.3 In the Hope Creek PUSAR Table 6-3 (Spent Fuel Pool Parameters) indicates a very small margin in temperature limits for batch offload. It is stated that the 135 °F limit was selected to assure operator comfort and to provide ample margin against inventory loss due to evaporation or boiling. What measures are planned to remain below this limit?

Core

3.4 Please provide the following information applicable to Cycle 15 (first EPU operation):

a) The average bundle power before and after the EPU.

b) The peak bundle power before and after the EPU.

c) Which fuel type (GE14 or SVEA-96+) will be limiting with respect to the thermal limits, safety limit for minimum critical power ratio (SLMCPR), operating limit maximum critical power ratio (OLMCPR), linear heat generation rate (LHGR), maximum average planar linear heat-generation rate (MAPLHGR), and peak cladding temperature (PCT) for Cycle 15?

LOCA

3.5 Question removed for Revision.

3.6 Question Deleted.

3.7 Please discuss the impact of increased discharge piping water leg height and temperature on SRV operation for EPU (flow and response time characteristics) as a result of the higher expected suppression pool temperature and containment pressure during a postulated LOCA and anticipated transient without scram (ATWS).

3.8 Question removed for Revision after the GE Interim Methods (NEDC-33173P) draft SE is completed.

3.9 Question Deleted.

Functional Design of Control Rod Drive System

3.10 In a letter dated July 14, 2005 (ADAMS Accession No. ML052000328), GE recommended a surveillance program for monitoring Channel-Control Blade Interference for BWR/2-5 (C/D-Lattice) plants. HC was among the plants recommended for the surveillance program. HC-PUSAR does not address Channel-Control Blade Interference at constant-pressure power uprate (CPPU) conditions. Please discuss the affects of Channel-Control Blade Interference at CPPU conditions and whether the recommended GE surveillance program has been implemented. GE has indicated that channel bow evaluation will be eventually implemented in all reload core designs. Has the channel bow evaluation been included in the EPU reload core design?

3.11 In Section 4.6.1.2.4.1, Hydraulic Requirements, the Hope Creek UFSAR states that a drive pressure of 260 psi (minimum) above reactor vessel pressure is required for a flow rate of approximately 4 gpm to insert and 2 gpm to withdraw a control rod. The generic evaluation in ELTR- 2 states "Normal CRD header pressure is maintained approximately 250 psig above the lower head pressure." Please address whether the drive pressure difference has an affect on the insert/withdraw drive flows.

Standby Liquid Control System

3.12 In the Hope Creek PUSAR, Section 9.3.1, Anticipated Transients Without Scram, it states that Hope Creek meets the ATWS mitigation requirements defined in 10 CFR 50.62 for boron injection equivalent to 86 gpm. Please provide the calculations that demonstrate the 86 gpm equivalency requirement of the rule satisfies the following relationship:

$(Q/86) \text{ x } (M251/M) \text{ X } C/13) \text{ X } (E/19.8) > 1$

where:

Q = expected standby liquid control system (SLCS) flow rate (gpm)

M = mass of water in the reactor vessel and recirculation system at hot rated condition in lbs

C = sodium pentaborate solution concentration (weight percent)

E = Boron-10 isotope enrichment (19.8% of natural boron)

M251 = mass of water in a BWR/4 251 inches diam. reactor vessel (lbs) = 628,300 lbs

3.13 The licensee performed a plant-specific ATWS analysis and found that the peak calculated vessel pressure during SLCS operation is 1179 psig. Please provide the results including the maximum pump discharge pressure and operating pressure margin for the SLCS pump discharge relief valves.

3.14 Question Deleted.

3.15 In the Hope Creek PUSAR Sections 10.5.3 and 10.5.4.1 discuss the automatic initiation of the Standby Liquid Control System design feature at Hope Creek. Section 9.3.5.1 of the Hope Creek UFSAR describes the automatic actuation of a timer upon receipt of a signal from the Redundant Reactivity Control System (RRCS). Please confirm that the current timer setting is still valid for EPU.

Residual Heat Removal (RHR) System

3.16 Following abnormal events, the standby pressure control (SPC) function controls the long-term suppression pool temperature such that the maximum operating temperature limit is not exceeded. The proposed EPU would increase the reactor decay heat, which increases the heat input to the suppression pool during a LOCA, and results in a higher peak suppression pool temperature. Please address the SPC function to control the long-term suppression pool temperature and provide the results that demonstrate that the maximum operating temperature limit is not exceeded.

3.17 What is the available RHR Heat Exchanger margin with the increased decay heat load due to EPU?

Reactor Core Isolation Cooling (RCIC) System

3.18 RCIC turbine steam exhaust trip is currently set at 25 psig. EPU analysis used a revised analytical model which predicted a maximum pressure of 38 psig. Please address the affect of the predicted maximum pressure will have on the system components. Has this revised analytical model been used in any other EPU analysis?

Transients and Accidents

3.19 As discussed in a November meeting, please provide a reference roadmap to the licensing topical reports (ELTR1, ELTR2, CLTR), PUSAR section or UFSAR for the analyzed transients according to the order in Review Standard for EPU (RS 001) Section 2.8.5.

3.20 In the Hope Creek PUSAR Table 1-3, the core inlet enthalpy is lower in CPPU than in current limiting thermal power (CLTP) condition. Please explain why core inlet enthalpy is lower when feedwater temperature is higher in constant pressure power uprate operation. In the table on page 3-15 of PUSAR, loop flow (17.35 Mlbm/hr) at CPPU is higher than it is in CLTP (16.76 Mlbm/hr) at 100% core flow case (same for CLTP and CPPU). Please explain how jet pumps induce less suction flow with increased jet flow.

3.21 The Hope Creek PUSAR does not reference CLTP as the basis for areas involving reactor systems and fuel issues. The transients analyzed in Table 9-2 and Figures 9-1 to 9-4 are not complete compared to Table E-1 in ELTR-1, which is the minimum set of transients suggested by GE. Please provide the rationale for the missing ones, i.e.,

No 4 Pressure Regulator Downscale Failure
No 6 Inadvertent HPCI (high-pressure coolant injection) start
No 13 Turbine trip, bypass failure, with scram on High Flux.

3.22 In the Hope Creek PUSAR, In the second paragraph on page 9-2, the 25% of rated thermal power value for Technical Specification (TS) safety limit, limiting condition for operation thresholds and Surveillance Requirement thresholds is modified to 24% based on the uprated bundle power. The 25% is based on generic analysis with highest average bundle power for BWR6, which is 4.8 Mwt/bundle. The staff is concerned that this modified value (24%) will be less conservative for Hope Creek (BWR4) core since this value is obtained from the ratio of "highest" bundle power value of BWR6 (4.8 MWT/bundle) and "average" BWR4 uprated bundle power. Shouldn't the divisor be the "highest" uprated bundle power instead of "average" uprated bundle power? Please address this concern. Please also list the affected safety limits and thresholds for the staff to evaluate.

3.23 Based on Table 9-2 In the Hope Creek PUSAR, what will be the final value of OLMCPR in this EPU core, Option A values or B? Please explain why option A and B for OLMCPR are the same for Loss of Feedwater Heating and Rod Withdrawal Error accidents. Please explain why Option A OLMCPRs in Table 9-2 anticipated operational occurrences are not simply SLMCPR (1.10) plus ΔCPR. What is the ΔCPR for main steam isolation valve (MSIV) closure with flux scram?

3.24 Please provide CPPU loss of feedwater flow and loss of one feedwater pump transient key parameter plots like Figures 9-1 to 9-4. For loss of feedwater transient, Hope Creek EPU mixed core (non GE-14 equilibrium) has different decay heat value than the one (GE-14 equilibrium) used in the analysis. How much does it affect the level recovery time? In terms of uncertainties, how does decay heat model used in your loss of feedwater flow analysis compare with the decay heat (decay heat >= 1979 ANS + 10%) suggested in ELTR1? Please also provide an estimate of lowest inside core shroud level in this transient. How does the calculated lowest level compare to Level 1? In loss of one feedwater pump transient, was Level 3 scram avoided?

3.25 Question Deleted.

3.26 Is GE 14 equilibrium core used in all section 9.2 design basis accidents analysis? Please clarify this assumption used in 9.1.1. Please clarify what is meant by "CPPU core inventory" cited throughout section 9.2.

3.27 In Table 9-9, with higher power density in constant pressure power uprate operation, why is the PCT (1446) is lower than the value in CLTP (1589)?

3.28 Please document the sequence of events for the analyzed ATWS transients and identify the most limiting event in section 9.3.1.

3.29 In section 9.3.2 second paragraph, you stated "Decay heat was conservatively evaluated assuming end-of-cycle and GE-14 fuel." Why is the SVEA 96 fuel not more limiting? In station blackout (SBO) transient, please provide more details about "coping capabilities" and how they are justified in constant pressure power uprate operation. Please also provide the sequence of events and operation of the safety features (i.e., RCIC) for the entire transient.

3.30 Please explain in more details why “the core design necessary to achieve CPPU operation may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillation at natural circulation condition” described in first paragraph of section 9.3.3?

3.31 Question Deleted.

3.32 In feedwater heater out of service and final feedwater temperature reduction transients, feedwater temperature is lower and thus the power is supposed to be higher during the transient. Even with less moderator reactivity being inserted (due to smaller difference between initial and final feedwater temperatures), isn’t the higher power at same low flow rate supposed to have higher ATWS oscillation?

3.33 Figure 9-1 (Turbine Trip with Bypass Failure), it showed steam flow (70%) out of vessel (peak around 1.2 sec) before t = 2 sec when no relief flows were shown according to the upper right plot. The staff is concerned that the large steam flow after reactor dome being pressurized is unrealistic because steam line is pressurized soon after stop valve closes and pressure difference between reactor dome and steam line should be smaller than the model’s estimate. Please justify this calculation and provide real plant turbine trip steam flow and dome pressure data for Hope Creek or a similar plant.

3.34 In Figure 9-4, the core inlet flow plot does not seem to show reactor pump trip out of service. Please explain the difference between this transient and the one in Figure 9-3.

3.35 Please provide the maximum vessel dome pressure for the transients shown in Figure 9-1 to 9-4. The plots show pressure rise in % rated. What is the rated pressure rise for Hope Creek?

Stability

Bypass Voiding

3.36 Characterize the expected amount of bypass voiding under CPPU conditions. Provide the expected bypass void level at points J, D, and E of Figure 1.1 of NEDC-33076P, Rev. 2, using a methodology equivalent to that used by the Interagency Steering Committee on Radiation Standards (ISCORS) for both hot and average channel.

Effect of bypass voids on instrumentation during normal operation

3.37 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 Fig 1.1 of NEDC-33076P, Rev. 2). 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.

Effect of bypass voids on instrumentation to detect and suppress (D&S) unstable oscillations

3.38 The presence of bypass voids affects the LPRM calibration. Evaluate the expected calibration error on operating power range monitor (OPRM) and average power range monitor (APRM) cells induced by the expected level of bypass voids. Document the impact of this error on the D&S Option III scram setpoint.

D&S Setpoint calculations

3.39 NEDO-33190 and NEDO 33188 show an example setpoint calculation for Cycles 13 and 14 respectively. These setpoints do not appear to include an uncertainty term reflecting the possible LPRM miss-calibration under bypass void conditions. How is this uncertainty accounted for?

DIVOM applicability to future cycles

3.40 NEDC-33185P states that the Cycle 13 DIVOM correlation may be used for future cycles based on an evaluation. Revision 1 NEDC-33185P shows a significantly larger (non-conservative) DIVOM slope for Cycle 14 under CPPU conditions, and it does not state that the Cycle 14 DIVOM slope may be used for future cycles. 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.

DIVOM slope for Cycle 14 under CPPU conditions

3.41 The recommended DIVOM slope in NEDC-33185P Revision 1 for Cycle 14 under CPPU conditions uses a radial peaking uncertainty that is half of that assumed for Cycle 13 in NEDC-33185P under OLTP conditions. Justify the use of this smaller uncertainty.

Interface between GE Methods and existing Solution III hardware

3.42 Provide a short summary of the Solution III hardware currently installed in Hope Creek. Are there any issues related to the interface between the existing hardware and GE methods?

ATWS EPGs

3.43 What version of emergency operating guidelines is currently implemented in Hope Creek? Provide a short description of the process used to ensure that the emergency procedure guideline (EPG) variables (e.g., hot shutdown boron weight (HSBW), heat capacity temperature limit (HCTL) are adequate under CPPU conditions.

ATWS/Stability

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

Plant-Specific OPRM System

3.45 Hope Creek currently operates under Option III solution. Please provide a clarification for the following areas:

a) Describe the process that was followed by Hope Creek to implement Option III L/T 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.

e) Provide a summary of the Hope Creek TSs affected by the Option III implementation and future CPPU operation.

f) List the approved methodologies used to calculate the OPRM setpoint by the current operation and future Hope Creek CPPU operation.

Hot Channel and Core-Wide Decay Ratio

3.46 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 Hope Creek under these new conditions.

4) Operator Lic & Human Performance Branch (IOLB)

4.1 Describe any changes to abnormal operating procedures (AOPs) and emergency operating procedures (EOPs) as a result of the proposed EPU.

a) Include any changes to setpoints and alarms that will be incorporated into operating procedures and training materials as a result of the proposed EPU.

b) Describe any procedural changes that credit manual actions due to the EPU that are not currently credited in the Final Safety Analysis Report (FSAR).

c) If any changes are identified, describe how these changes may affect operator action response times credited in the safety analyses in the UFSAR.

4.2 Identify and describe the effect on manual actions sensitive to the proposed EPU that are credited in the safety analyses in the UFSAR.

a) If any manual actions are affected, has the licensee performed an evaluation of the environment of the manual actions for the applicable accident scenarios?

b) If any manual actions are affected, describe how the EPU affects the amount of available time for the operators to perform their tasks.

c) How will the EPU affect the actual time for the operators to perform their tasks during accident scenarios?

d) The staff requests the current amount of time available and the post-EPU times for the operators to perform their actions provided in the EOPs and AOPs. In the response, include a discussion of how the new times will be validated to ensure that operators have enough time to perform the actions affected by the EPU.

4.3 Identify and describe any changes to human interfaces for control room controls, displays, and alarms that will affect the operator's ability to interpret, read or visually identify the information required from the instrumentation.

a) Describe any controls, displays, alarms that may be upgraded from analog to digital as a result of the proposed EPU and how operators will be tested to determine proficiency.

4.4 In the submittal, the licensee stated that the Safety Parameter Display System (SPDS) will be reviewed to determine effects from the EPU. Please describe any changes to the SPDS that have been identified thus far including monitored points, alert and trip set points, and various changes in EOP curves and limits.

4.5 Provide the implementation schedule for making the changes to the operator training program and the plant referenced control room simulator.