50-387

September 23, 1993

MEMORANDUM FOR: C. L. Miller, Project Directorate (14E-21)
Project Directorate I-2
Division of Reactor Projects I, II

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety & Analysis

SUBJECT: SUSQUEHANNA-1 & 2 PROPOSED LICENSE AMENDMENT
POWER UPRATE REVIEW (TAC NOs. M83426 & M83427)

Enclosed is the Reactor Systems Branch input to the Safety Evaluation Report being prepared by your Project Directorate for the subject power uprate license amendment. It is my understanding that you will use this and other technical branch inputs for developing the overall staff safety evaluation for this license amendment.

/s/

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
As stated

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ENCLOSURE

INTRODUCTION

Pennsylvania Power & Light Company (PP&L), the licensee for Susquehanna Steam Electric Station (SSES), Units 1 & 2, submitted a request by letter on June 15, 1992 to uprate the licensed power level from 3293 MWt to 3441 MWt. This represents approximately a 4.5% increase in thermal power with a 5% increase in rated steam flow. The planned approach to achieve the higher power level consists of (1) an increase in the core thermal power to create an increased steam flow, (2) a corresponding increase in feedwater flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along extension of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in General Electric report NEDC 31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991. The operating pressure will be increased approximately 30 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

2.4 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Extended Load Line Limit Analysis (ELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100% rated power.

2.5 Stability

Ongoing activities by the BWR Owners' Group and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions (as required by General Design Criteria 12 of 10 CFR 50 Appendix A). GE has documented information and cautions concerning this possibility in Service Information Letter (SIL)

380 and related communications. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, Technical Specifications and associated implementing procedures, as requested by the NRC Bulletin, have been incorporated by the licensee which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions have been established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

The restrictions recommended by NRC Bulletin 88-07 and Supplement 1 to that Bulletin will continue to be followed by the licensee for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWR Owners' Group and the NRC. This is acceptable to the staff.

2.6 <u>Reactivity Control</u>

2.6.1 <u>Control Rod Drives and CRD Hydraulic System</u>

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion will be slower due to the high pressure. As the scram continues, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. The licensee has indicated that CRD performance during power uprate will meet current Technical Specification requirements. The licensee will continue to monitor by various surveillance requirements the scram time performance as required in the plant Technical Specifications to ensure that the original licensing basis for the scram system is preserved.

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The minimum drive water pressure for power uprate conditions is therefore 1325 psig. Recent operating data show a range of CRD pump discharge pressures from 1435 to 1455 psig. The licensee's calculations indicate that the CRD system insert and withdraw operations will be satisfactory with these discharge pressures.

The CRD system will therefore continue to perform all its safety-related functions at uprated power with ICF, and will function adequately during insert and withdraw modes.

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) and the high pressure reactor scram provide this protection. The changes in the nuclear system pressure relief for power uprate are increases in the SRV setpoints as described below, and a decrease in the number of valve groups from five to three.

The operating steam dome pressure is defined to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require a change in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuation.

3.2 Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is an MSIV closure with a failure of the valve position scram. The MSIV closure will be analyzed by the licensee using the NRC-approved methods, with the following exceptions: (1) the MSIV closure event be analyzed at 102% of the uprated core power and 108 million lbm/hr core flow, and (2) the maximum initial reactor pressure will be assumed to be the Technical Specification maximum value.

The number of SRVs which will be assumed to be out of service is based on the maximum allowed by Technical Specification. Uprated conditions will produce a higher peak RPV pressure, and with reduced valve grouping, the reload analysis must show that it remains below the 1375 psig ASME code limit. The licensee's analysis plan is acceptable.

3.4 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with allowance for increased core flow. The cycle-specific core reload analyses will consider the full core flow range, up to 108 million lbm/hr. The evaluation by the licensee of the reactor recirculation system performance at uprated power with ICF determined that the core flow can be maintained.

The cavitation protection interlock will remain the same in absolute thermal power, since it is based on the feedwater flow rate. These interlocks are based on subcooling in the external recirculation loop and thus are a function of absolute thermal power. With power uprate, slightly more subcooling occurs in the external recirculation loop due to the higher RPV dome pressure. It would therefore be possible to lower the cavitation interlock setpoint slightly, but this change would be small and is not necessary.

An evaluation by the licensee of recirculation pump net positive suction head (NPSH) found that at full power, power uprate alone does not increase NPSH required (NPSHr), and that the secondary effect of the 30 psi increase in RPV pressure increases NPSH available (NPSHa), so that power uprate alone increases the NPSH margin.

Increased core flow both increases NPSHr and reduces NPSHa, and thereby reduces the NPSH margin. Despite this reduction, NPSHa will remain at least three times the NPSHr with uprated power, with power uprate and increased core flow, or with increased core flow alone.

The recirculation drive flow stops were reviewed by the licensee for application to uprated power and ICF conditions. Due to the increased core flow (8%) the pump motor-generator set scoop tube electrical and mechanical stops will be adjusted upward from 102.5% and 105% of 100 million lbm/hr, respectively, to 109.5% and 110.5% of 100 million lbm/hr.

An estimate by the licensee of the required pump head and pump flow indicates that the power demand of the recirculation motors increases up to 2.5% with power uprate, and up to 30% with both increased core flow and power uprate. These increases are within the capability of the recirculation system. The licensee has committed in the document titled, "Power Uprate Startup Test Specification," M-1515 Rev. A, to perform tests on the recirculation flow control system, a) to demonstrate the flow control capability of the plant over the entire pump speed range, including individual local manual and combined Master-Manual operation; and b) to determine that all electrical compensators and controllers are set for desired system performance and stability. Tests will also be performed to enable a complete calibration of the installed recirculation system flow instrumentation and will include specific signals to the plant process computer.

3.7 Main Steam Isolation Valves (MSIVs)

The main steam isolation valves (MSIVs) have been evaluated by the licensee, and are consistent with the bases and conclusions of the generic evaluation.

Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. Performance will be monitored by surveillance requirements in the Technical Specification to ensure original licensing basis for MSIV's are preserved.

3.8 Reactor Core Isolation Cooling System (RCIC)

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. The RCIC system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee has indicated by letter dated January 25, 1993 that the recommendations of GE SIL No. 377 have been implemented on the RCIC system on each SSES unit. Note that instead of adding a startup bypass line, the licensee chose to modify the control circuit of the RCIC steam admission valve. This modification is intended to achieve the turbine speed control/system reliability desired by SIL 377, and is consistent with the requirements in the staff SER of the generic topical report. The purpose of the modification is to mitigate the concern that a slightly higher steam pressure and flow rate at the RCIC turbine inlet will challenge the system trip functions such as turbine overspeed, high steam flow isolation, low pump suction pressure and high turbine exhaust pressure. The licensee also plans to perform startup testing on RCIC during the initial startup after being licensed at uprated power. Further details of the startup testing plan will be provided with the proposed license amendment. The staff requires that licensee provides assurance that the RCIC system will be capable of injecting their design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the licensee must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads.

3.9 Residual Heat Removal System (RHR)

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes are discussed in the following paragraphs.

3.9.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125°F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to be insignificant.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," requires demonstration of cold shutdown capability (200°F reactor fluid temperature) within 36 hours. FSAR Section 15.2.9 indicates that cold shutdown can be reached in a much shorter time even considering the availability of only one RHR heat exchanger. For power uprate, licensee analysis of the alternate path for shutdown cooling based on the criteria of Regulatory Guide 1.139 shows that the reactor can be cooled to 200°F in 28 hours, which meets the 36-hour criterion.

3.9.2 Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the FSAR is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the peak suppression pool temperature analysis by the licensee confirms that the pool temperature will stay below its design limit at uprated conditions.

3.9.3 Containment Spray Cooling Mode

The containment spray cooling mode provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. Power uprate increases the containment spray temperature by only a few degrees. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these parameters reach peak values prior to actuation of the containment spray.

4.0 ENGINEERED SAFETY FEATURES

4.2 Emergency Core Cooling Systems (ECCS)

The effect of power uprate and the increase in RPV dome pressure on each ECCS system is addressed below.

As discussed in the FSAR compliance to the NPSH requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit. Assuming a LOCA occurs during operation at the uprated power, the suppression pool temperature will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

4.2.1 High Pressure Core Injection System (HPCI)

The HPCI system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. In response to a staff request the licensee has indicated by letter dated Jan. 25, 1993 that the modifications on the HPCI system on each unit in response to GE SIL 480 have been installed, and is consistent with the requirements in the staff SER of the generic topical report. The purpose of this modification is similar to that of the RCIC system as discussed in Section 3.8. The licensee also plans to perform startup testing on HPCI during the initial startup after being licensed at uprated power. Further details of the startup testing plan will be provided with the proposed license amendment. The staff requires that licensee provides assurance that the HPCI system will be capable of injecting their design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, licensee must also provide assurance that the reliability of the HPCI system will not be decreased by the higher loads placed on the system or because of any modifications made to this system to compensate for these increased loads.

4.2.2 Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system do not experience different operating conditions due to power uprate, there is no impact due to power uprate.

4.2.3 Core Spray System (CS)

The hardware for the low pressure core spray are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since these systems do not experience different operating conditions due to power uprate, there is no impact due to power uprate. Also, the impact of power uprate on the long term response to a LOCA will continue to be bounded by the short term response.

4.2.4 <u>Automatic Depressurization Systems (ADS)</u>

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCI failure. This function allows low pressure coolant injection (LPCI) and core spray (CS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. Plant design requires a minimum flow capacity for the SRVs, and that ADS initiate after a time delay on either low water level plus high drywell pressure, or on low water level alone. The ability to perform either of these functions is not affected by power uprate. This assessment is based on the analysis of system response under various LOCA conditions presented in the GE report NEDC-32071P, "SAFER/GESTR-LOCA Report", which has been provided with Reference 1.

4.3 <u>ECCS Performance Evaluation</u>

The emergency core cooling systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCAs) caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Appendix K. The Siemens Nuclear Power 9x9 fuel, used in SSES Units 1 and 2, was analyzed by the licensee with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

The licensee used the staff approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for SSES Units 1 and 2 was performed by the licensee with SNP 9x9-2 fuel in accordance with NRC requirements and demonstrates conformance with the

ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K PCT as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The SSES specific analysis was performed with a conservatively high Peak Linear Heat Generation Rate (PLHGR) and a conservatively low Minimum Critical Power Ratio (MCPR). In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is below 1050°F. The statistical Upper Bound PCT is below 1320°F. The Licensing Basis PCT for SSES is 1510°, which is well below the acceptance criteria of 10 CFR 50.46 PCT limit of 2200°F. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in Table 4-4 of the PP&L Licensing Topical Report. Therefore, SSES Units 1 and 2 meet the NRC S/G-LOCA licensing analysis requirements.

The licensee also reevaluated the ECCS performance for single loop operation (SLO) using the S/G - LOCA methodology. The DBA size break is also limiting for SLO. Using the same assumptions in the S/G - LOCA calculation with no MAPLHGR reduction, yields a calculated nominal and Appendix K PCT of 1160° and 1661°F, respectively. Since the PCT was below the 10 CFR 50.46 limit of 2200°F, the licensee claimed that no MAPLHGR reduction is required for SLO. The staff asked the licensee to reconcile the fact that the S/G - LOCA analysis PCT results for SLO were higher than those presented for two loop operation, and no statistical analysis of the Upper Bound PCT had been provided for this case. The licensee reviewed this staff question, and has proposed in a letter dated April 2, 1993 to impose a LHGR reduction (multiplier) of 0.70 during SLO. Based on this reduction, the calculated SLO Licensing Basis PCT and Upper Bound PCT are lower than their respective values for two loop operation. The proposed Technical Specification markup reflecting the LHGR reduction (multiplier) has been transmitted to the NRC in Reference 2.

A S/G-LOCA analysis for the ELLLA region was performed by the licensee at a core flow of 87 Mlb/hr and uprated power for SSES with SNP 9x9-2 fuel. A DBA recirculation suction line break coincident with a false LOCA signal from the opposite unit was assumed. The results of the analysis show that early dryout of the high power node would not occur and the MAPLHGR multipliers as a function of flow are not required. Consistent with the Appendix K Licensing Basis calculations performed by the licensee, the high power mode is assumed to experience early dryout for the Appendix K ELLLA analysis. The nominal and Appendix K results both show a small increase in the PCT when compared to the base 100 Mlb/hr core flow cases, however, the PCT is still well below the 10 CFR 50.46 limit. The nominal and Appendix K values for the base case are 916 F and 1499 F, respectively, and for the ELLLA case they are 937 F and 1514 F, respectively. The increase in PCT for the ELLLA case is due to (1) the lower heat transfer rate during flow coastdown from the lower initial core flow; and (2) more subcooling in the downcomer which results in increased break flow and earlier core uncovery. No statistical Upper Bound PCT was provided for the ELLLA case. In response to a staff question to give an explanation for not providing the Upper Bound PCT for the ELLLA case, the licensee presented additional clarifying information in a letter dated August 5, 1993. The licensee indicated that the Upper Bound PCT documented in the submittal (NEDC-32071P) is not based on ELLLA. If it were, the event would begin at a slightly lower core flow, but would otherwise be essentially the same. As shown in NEDC-32071P Table 5-5, the nominal PCT is only 21 F higher when ELLLA is taken into account. The statistical uncertainties between the two cases do not change. Therefore, based on the results reported in the submittal, the ELLLA case will not impact the 1600 F limit on the Upper Bound PCT, nor the 2200 F limit on the Licensing Basis PCT, and the Licensing Basis PCT will continue to be greater than the Upper Bound PCT. This explanation is acceptable to the staff.

The licensee also evaluated the applicability of the S/G-LOCA methodology to SSES, Units 1 and 2 which operates with Siemens Nuclear Power (SNP) 9x9-2 fuel. The dimensions and characteristics of the SNP fuel are similar to GE fuels. The reactor and core response during a LOCA are not strongly dependent on fuel design. This is because for most BWRs, including BWR/4 (SSES is a

BWR/4), the core heatup, and corresponding PCT, occurs late in the event, well after the stored energy in the fuel is released. Hence, the PCT is more dependent on the decay heat power level and the heat transfer coefficient in the core. The maximum cladding temperature (or PCT) occurs during a period that is governed predominantly by steam cooling and eventually by core reflooding, both of which are well understood in fuel bundle geometries. The fuel specific input geometry and characteristics for the SNP fuel were input directly into S/G-LOCA following the same procedures used for GE fuel. The results of the break spectrum analysis show that the large break PCT was second peak limited, i.e., late in the event following core uncovery, and that the PCT was similar to the second peak PCT for the generic BWR/4 with GE fuel.

Since, the geometry and characteristics of the SNP fuel used in SSES are similar to GE fuels, and that the S/G - LOCA results for SSES are similar to those of the generic BWR/4 S/G-LOCA analysis and also similar to those for a typical GE BWR/4 plant, the S/G - LOCA methodology is applicable to SSES with SNP fuel.

9.0 REACTOR SAFETY PERFORMANCE FEATURES

9.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant cause by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for the Susquehanna units were identified. The relatively small changes in rated power and maximum allowed core flow are not expected to effect the selection of limiting events. The events which will be explicitly evaluated for cycle specific reload analyses are:

1. Loss of Feedwater Heating
2. Feedwater Controller Failure (FWCF)

3. Generator Load Rejection without Bypass (GLRWOB)
4. Turbine Trip without Bypass (TTWOB)
5. Rod Withdrawal Error
6. Recirculation Flow Controller Failure, Increase (RFCF)
7. Fuel Loading Error

The limiting events which establish the minimum critical power ratio (MCPR) operating limits are currently GLRWOB, FWCF, and RFCF. These events are expected to remain limiting. The licensing analyses will be performed by the licensee up to a maximum power level of 102% of the uprated power level, or 3510 MWt, to account for power uncertainty.

Parametric studies were conducted as part of developing licensee's licensing methods. These studies lead to the following expectations. The GLRWOB delta CPR (critical power ratio) is determined based on a parametric analysis up to the maximum power level, and the FWCF is analyzed as a function of power. Thus, the increase in core power only changes the maximum power level considered. It is expected that the increased flow rate for the GLRWOB and the FWCF will produce slightly higher delta CPRs. This expectation will be confirmed as part of the reload licensing analyses. The RFCF is analyzed as a function of core flow. The effect of increased core flow on the RFCF event will be evaluated as part of the reload licensing analyses. In response to a staff question, the licensee in a letter dated August 5, 1993 has indicated that it has decided not to take credit for the flow biased simulated thermal power trip in the RFCF analysis for power uprate.

The safety limit minimum critical power ratio (SLMCPR) is calculated by the licensee as part of the reload licensing analyses using the NRC-approved Siemens Nuclear Power (SNP) methodology. No change will be made to this methodology due to power uprate or increased core flow. The analysis plan proposed by the licensee is acceptable. The staff will verify the acceptability of the results when the reload document is submitted.

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transients Without Scram (ATWS)

Although General Electric has performed generic bounding ATWS analyses, they cannot be used because SSES (1) uses non-GE fuel, and (2) has taken exceptions to the Rev. 4 Emergency Procedure Guidelines (EPGs) for responding to ATWS, which are used in the GE generic analyses.

A plant-specific analysis is currently being performed by the licensee for ATWS under uprated conditions. The results of this analysis will be provided by the licensee with the actual license amendments proposed in support of power uprate implementation. The results will also be included in the ongoing project to upgrade SSES Emergency Operating Procedures. Staff will provide evaluation when the licensee makes the submittal.

9.3.2 Station Blackout

Per the NUMARC 87-00 methodology, SSES is classified as a 4-hour-duration station blackout (SBO) plant based on an offsite power design characteristics group of "P1," an emergency AC power configuration group of "D", and a target emergency diesel generator reliability of 0.975. Power uprate conditions will not affect this 4-hour-duration classification.

The limiting parameters for SBO events lasting longer than four hours are water inventory for decay heat removal, class 1E battery capacity, compressed air capacity, and the effects of loss of ventilation. Power uprate will result in more decay heat which will require a slightly larger water inventory. However, the current SBO analysis provides for adequate water inventory to meet the additional requirements of power uprate.

Class 1E battery capacity and the compressed air system are unaffected by power uprate, and power uprate will not increase demand on these systems for SBO scenarios. The capacity of these systems will therefore remain adequate.

Power uprate will have a slight effect on loss of ventilation since slightly more heat will be transferred to the containment. This will result in slightly higher compartment temperatures. The Compartment Transient Temperature Analysis Program (COTTAP) computer code developed by the licensee was run for the station blackout scenarios using revised heat inputs from major equipment affected by power uprate. It simulates the control room and reactor building thermal response under loss-of-HVAC conditions. The licensee indicates that the results of this calculation show that the compartment temperatures only rise 2 or 3 °F due to power uprate, and that the temperatures during a SBO event will not exceed the 180 °F limit identified in Appendix F of NUMARC 87-00 Rev. 1.

The equipment with revised heat inputs used for the power uprate SBO evaluations includes motors, electrical cabinets, piping, and miscellaneous mechanical equipment such as heat exchangers. The rest of the equipment whose heat load changes with power uprate, but which was not included in these calculations, adds very little to the heat loads already considered, and will not contribute significantly to the increase in compartment temperatures.

REFERENCE

1. PLA-3788, H. W. Keiser to C. L. Miller, "Submittal of Licensing Topical Report on Power Uprate with Increased Core Flow", dated June 15, 1992.

2. PLA-3948, R. G. Byram to C. L. Miller, "Revisions to PP&L Power Uprate Submittal", dated April 2, 1993.