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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING "LICENSING TOPICAL REPORT NE-092-001,

REVISION 0, POWER UPRATE WITH INCREASED CORE FLOW"

PENNSYLVANIA POWER & LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter of June 15, 1992, as revised and supplemented by letters of July 24, September 17, and December 18, 1992, and January 8, January 25, April 2, August 5, August 12, and September 29, 1993, the Pennsylvania Power & Light Company (PP&L or the licensee) requested approval of "Licensing Topical Report NE-092-001, Revision 0, Power Uprate With Increased Core Flow," for the Susquehanna Steam Electric Station, Units 1 and 2. The topical report describes the licensee's intention to change the licensed thermal power level of the reactor from the current limit of 3293 megawatts thermal (MWt) to an increased limit of 3441 MWt. This request is made in accordance with the generic boiling-water reactor (BWR) power uprate program established by General Electric Nuclear Energy (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter of September 30, 1991. This request is similar to a request made on September 24, 1991, by the Detroit Edison Company for the Fermi-2 facility.

2.0 DISCUSSION

By letter of June 10, 1991, GE submitted Revision 1 to "Licensing Topical Report (LTR) NEDC-31897P, Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (Reference 1). In this LTR, GE proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent. The LTR contained a proposed outline for individual license amendment submittals, as well as discussions of the scope and depth of reviews that would need to be performed and the methodologies that would be used in these reviews. By letter of September 30, 1991, the NRC issued a staff position concerning the LTR (Reference 2), which approved the proposed program, provided that individual power uprate amendment requests meet certain requirements contained in the document.

The generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level was generally based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the

design power level is often referred to as "stretch power." Since the design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling system (ECCS), increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment, nor does it significantly impact the reliability of this equipment.

The licensee's topical report proposes to increase the current licensed power level of 3293 MWt to a new limit of 3441 MWt which represents an approximate 4.5-percent increase in thermal power with a corresponding 5-percent increase in rated steam and feedwater flows. The planned approach to achieving the higher power level consists of (1) an increase in the core thermal power level to increase steam production in the reactor; (2) an increase in feedwater flow corresponding to the increase in steam flow; (3) an 8-percent increase in maximum allowable core flow; and (4) operation of the reactor along extensions of current rod position/flow rate control lines. This approach is consistent with the generic guidelines for BWR power uprate presented in Reference 1 and approved by the staff. The increased core power will be achieved by utilizing a slightly flatter radial power distribution while maintaining the most limiting fuel bundles within their operating constraints. The operating pressure of the reactor will be increased approximately 30 psi to ensure satisfactory turbine pressure control and pressure drop characteristics with the increased steam flow.

3.0 EVALUATION

In its review of the Susquehanna power uprate topical report, the NRC staff used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions regarding the topics being evaluated. Additionally, the staff evaluated the Susquehanna submittal for compliance with the generic BWR power uprate program as defined in Reference 1. Detailed discussions of individual review topics follow.

3.1 Reactor Core and Fuel Performance

The effect of power uprate was evaluated for potential impact on various areas related to reactor thermohydraulic and neutronic performance. These included changes to the power/flow operating map, core stability, reactivity control, fuel design, control rod drives, and scram performance. Additionally, the staff considered the impact of power uprate on reactor transients, anticipated transients without scram (ATWS), emergency core cooling system (ECCS) performance, and peak cladding temperature (PCT) for design-basis-accident (DBA) break spectra.

3.1.1 Fuel Design and Operation

The licensee has stated that no new fuel designs would be needed to achieve power uprate. This statement is consistent with information submitted by GE in LTR NEDC-31984P (Reference 3). Fuel operating limits, such as the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) for future fuel reloads will continue to

be met after power uprate. The methods used for calculating MAPLHGR and OLMCPR limits will not be changed by power uprate, although the actual thermal limits may vary between cycles. Cycle-specific thermal limits will be included in the plant Core Operating Limits Report (COLR).

3.1.2 Power/Flow Operating Map

The power/flow operating map described in the topical report includes operating domain changes for both uprated power and increased core flow operations. Specifically, the licensee proposed to permit plant operations within an operating domain consisting of an increased core flow (ICF) range and a revised 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-percent rated power. The staff has concluded that the proposed extension of the power/flow operating map will not degrade plant operations.

3.1.3 Stability

The BWR Owners Group (BWROG) and the NRC continue to address methods to minimize the occurrence and potential effects of core power oscillations which have occasionally been observed for certain boiling-water reactor (BWR) operating conditions. Until this issue is resolved, the licensee has adopted the generic interim operating constraints proposed by GE. Existing plant procedures have been incorporated in accordance with NRC Bulletin 88-07 and Supplement 1 to that bulletin which restrict plant operation in the high-power/low-flow region of the power/flow operating map. Since plant operation after power uprate will primarily extend the power/flow map to a higher power level (with corresponding higher flow), the current restricted operation regions of the power/flow map will remain essentially unchanged, and operator actions upon entry into these regions will likewise remain the same. This is consistent with information presented in the generic evaluations provided by GE in Reference 3.

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 BWROG and the NRC. The staff considers this approach acceptable.

3.1.4 Control Rod Drives and Scram Performance

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, rods will insert more slowly because of 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 stated that CRD performance during power uprate will meet current technical specifications 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 staff concludes that the CRD system will continue to perform all its safety-related functions at uprated power with increased core flow, and will function adequately during insert and withdraw modes.

The licensee assured the adequacy of the control rod drive mechanisms (CRDMs) in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1971 Edition, through Winter 1972 Addenda (Reference 9). The limiting components of the CRDM were identified to be the indicator tubes. The maximum stress and fatigue usage factors are below the allowable limits, and provide safety factors of about 1.5 and 6.6, respectively, for the design-basis conditions.

The increase in the reactor dome pressure, operating temperature, and steam flow rate as a result of the power uprate are bounded by the conditions assumed in the GE generic guidelines for the power uprate. The increase in core flow rate has no adverse effects on the control rod drive mechanism (CRDM). The CRDM was originally evaluated for a normal maximum reactor dome pressure of 1045 psig, which is higher than the power uprate dome pressure of 1035 psig.

On the basis of its review, the staff concludes that the CRDM will continue to meet its design-basis and performance requirements at uprated power conditions.

3.2 Reactor Coolant System and Connected Systems

In reviewing the mechanical engineering portions of the Susquehanna power uprate topical report, the staff focused on the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components.

GE based its generic guidelines for BWR power uprate effects on a 5-percent higher steam flow, an operating temperature increase of 5 °F, and an operating

pressure increase of 40 psig or less. For Susquehanna, the maximum reactor vessel dome pressure increases from 1005 psig to 1035 psig (30 psi), the dome temperature increases from 547 °F to 550.5 °F (3.5 °F) and the steam flow rate increases from 13.483 million pounds-mass per hour (lbm/hr) to 14.139 million lbm/hr (approximately 5%). The maximum core flow rate increases from 100 million lbm/hr to 108 million lbm/hr (8%) for the Susquehanna power uprate conditions, while GE generic guidelines assumed no change in core flow.

3.2.1 Nuclear Steam 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 offer 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 in order 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 an increase 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 actuations.

3.2.2 Reactor Overpressure Protection

The results of the overpressure protection analysis are cycle specific and will be incorporated in the Core Operating Limits Report. 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 will be analyzed at 102 percent 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 specifications maximum value.

The number of SRVs which will be assumed to be out of service is based on the maximum allowed by the technical specifications. Uprated conditions will produce a higher peak pressure in the reactor pressure vessel (RPV), and with reduced valve grouping, the cycle-specific analysis must show that it remains below 1375 psig, which is the ASME Code limit.

3.2.3 Reactor Vessel and Internal Components

The licensee evaluated the reactor vessel and internal components considering load combinations that include reactor internal pressure difference (RIPD), loss-of-coolant accident (LOCA), safety-relief valve (SRV), seismic, annulus pressurization (AP), jet reaction (JR), and fuel lift loads.

The licensee evaluated LOCA loads such as pool swell, condensation oscillation (CO), and chugging for the Susquehanna power uprate with increased core flow, and found that the original LOCA analyses did not change because the LOCA dynamic loads for Susquehanna were defined on the basis of the Kraftwerk Union (KWU) test conditions which bound the power uprate blowdown conditions with respect to vent mass and energy flow rate, and suppression pool water temperature. The design-basis SRV containment dynamic loads that affect the reactor vessel and piping systems are defined in accordance with the original KWU-specified SRV load boundary specification. The licensee stated that there is adequate conservatism in the design-basis loads to accommodate the slight increase in reactor pressure due to the power uprate. The licensee's review of the original analyses for the AP and jet loads indicated that the assumptions, analysis methodologies, and input parameters are conservative for the power uprate. On this basis, the staff concurs with the licensee's evaluation that the LOCA, SRV, AP, and jet design-basis loads remain bounding for power uprate with increased core flow.

In analyzing the potential for lifting of fuel as a result of the power uprate with increased core flow, the licensee considered load combinations that include SRV, LOCA, AP, JR, pipe restraint, seismic loads, and scram loads for the power uprate conditions. On the basis of its review, the staff concludes that the potential increase in fuel lift due to the power uprate is negligible.

The licensee evaluated stresses and fatigue usage factors for reactor vessel components in accordance with the requirements of the 1968 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB, 1968 Edition through Summer 1970 Addenda (Reference 8), to ensure compliance with the original code of record for Susquehanna, Units 1 and 2. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. A limiting fatigue usage factor of 0.92 was calculated for the reactor vessel head flange region for 40 years of operation based upon the uprated power level. There were no new assumptions used in the analysis for the power uprate conditions from those utilized by the licensee in previous evaluations. On the basis of the staff's review, the maximum stresses and fatigue usage factor, as provided by the licensee, are within the code's allowable limits and are, therefore, acceptable.

The licensee assessed the effects of increased core flow on flow-induced vibration by reviewing the startup test data for the valid prototype plant in comparison with the Susquehanna power uprate condition. The licensee stated that 113 percent of rated core flow, versus 108 percent of core flow for the power uprate, was tested and that the measured data do not show any indication

of potential fluid-elastic instability. Therefore, the staff concurs with the licensee's assessment that the reactor internal response to flow-induced vibration will remain within acceptable limits.

3.2.4 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 (ICF). The cycle-specific core reload analyses in the Core Operating Limits Report will consider the full core flow range, up to 108 million lbm/hr. In evaluating the performance of the reactor recirculation system at uprated power with ICF, the licensee 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 as a result of 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 (i.e., without an increase in core flow) 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 licensee reviewed the recirculation drive flow stops for application to uprated power and ICF conditions. Because of the increase in core flow (up to 108 percent of rated), the recirculation pump motor-generator set scoop tube electrical and mechanical stops will be adjusted upward from 102.5 percent and 105 percent of 100 million lbm/hr, respectively, to 109.5 percent and 110.5 percent 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 percent with power uprate, and up to 30 percent with both increased core flow and power uprate. These increases are within the capability of the recirculation system. The licensee has committed to provide a startup test plan with the proposed license amendment application. The staff expects that the recirculation flow control system will be tested as part of this startup test plan and will review the test program in conjunction with review of the amendment application.

3.2.5 Reactor Coolant Piping

The licensee evaluated the effects of the power uprate conditions, including higher flow rate, temperature and pressure for thermal expansion, dynamic loads, and fluid transient loads on the Class 1 reactor coolant pressure boundary (RCPB) piping systems, including such in-line components as equipment nozzles, valves and flange connections, and pipe supports. The licensee performed the evaluation in accordance with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB-3600, 1971 Edition through Winter 1972 Addenda (Reference 9).

The licensee stated that stresses and fatigue usage factors were calculated for the power uprate, based on Equations 9 through 14 of the ASME Code (Reference 9), for the design, normal, upset, emergency, and faulted conditions. The revised stresses caused by power uprate were compared with the code allowables for acceptability. The licensee concluded that the code requirements are satisfied for the evaluated piping systems and that power uprate will not have an adverse effect on the Class 1 piping system design.

The licensee evaluated pipe supports, equipment nozzles, and in-line components by comparing the increased piping interface loads on the system components due to the power uprate thermal expansion, with the margin in the original design-basis calculation. The licensee concluded that sufficient margin exists between the original design stresses and the code limits to accommodate the stress increase caused by the power uprate. The licensee also evaluated the effect of power uprate conditions on thermal and vibration displacement limits for struts, springs, and pipe snubbers, and found them acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified.

On the basis of its review of the licensee's submittal, the staff concludes that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop in the power uprate conditions.

3.2.6 Main Steam Isolation Valves

The performance of the main steam isolation valves (MSIVs) with regard to reactor coolant pressure boundary requirements, such as closure time and leakage, could potentially be impacted by the increased reactor operating pressure. However, the pressure increase is relatively small (less than 3%) and MSIV performance will be monitored by surveillance requirements in the plant technical specifications to ensure that the original licensing basis for the MSIVs is preserved.

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 specifications to ensure that the original licensing basis for MSIVs is preserved.

3.2.7 Balance-of-Plant Piping

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design-basis conditions with those for the proposed uprated conditions and by performing stress analyses in accordance with requirements of the code and the code addenda of record under the power uprate conditions. The BOP piping systems were determined from the uprated reactor and BOP heat balances. These systems include lines that are affected by power uprate, but not evaluated in Section 3.5.1 of the letter (Reference 13), such as main steam bypass lines, reactor feed pump turbine lines, and SRV discharge lines.

On the basis of a review of the existing design-basis calculation, the licensee determined that a majority of the BOP systems were originally designed to maximum temperatures and pressures that bounded the increased operating temperature and pressure due to the power uprate, and, therefore, are acceptable.

For the other portions of systems whose design temperature and pressure did not envelope the conditions of uprated power, the licensee performed stress analyses based on the power uprate conditions, and concluded that the actual calculated pipe stresses and support loads remained within the code-allowable limits.

The licensee evaluated the original pipe break analyses in accordance with the Standard Review Plan Section 3.6 guidance based on the revised fatigue analysis and concluded that the existing postulated break configurations and locations in these systems were not affected. No new postulated pipe break locations were identified in any system evaluated.

On the basis of its review of information submitted by the licensee, the staff concurs with the licensee's evaluations and concludes that the BOP systems will operate at the proposed power uprate conditions without adverse effects on the piping system and pipe supports.

3.2.8 Reactor Core Isolation Cooling System

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 licensee evaluated the RCIC system, and it is consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee indicated in a letter of January 25, 1993, that the recommendations of GE Service Information Letter (SIL) No. 377 have been implemented on the RCIC system at each Susquehanna unit. The staff noted 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's safety evaluation report (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 the RCIC system during the initial startup after being licensed at uprated power. Further details of the startup testing plan will be submitted with the proposed license amendment. The staff requires that the licensee provide assurance that the RCIC system will be capable of injecting the 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. This may be done during startup testing following implementation of the power uprate.

3.2.9 Residual Heat Removal System

The residual heat removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to perform primary system decay heat removal following reactor shutdown for both normal and postaccident 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 (except for LPCI which is discussed in 3.3.2.2) are discussed in the paragraphs that follow:

(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. The Final Safety Analysis Report (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.

(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) Containment Spray Cooling Mode

In the containment spray cooling mode, water is pumped from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during postaccident 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 reach peak values before the actuation of the containment spray.

3.2.10 Reactor Water Cleanup System

The operating pressure and temperature of the reactor water cleanup (RWCU) system will increase slightly as a result of power uprate. The licensee has evaluated the impact of these increases and has concluded that the power uprate will not impair the integrity of the RWCU system. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, current technical specifications limits for reactor water chemistry will not be changed as a result of a power uprate. Therefore, the power uprate will not significantly impact the operation or coolant boundary integrity of the RWCU system.

3.3 Engineered Safety Features

The staff reviewed the impact of the power uprate on containment system performance, the standby gas treatment system, (due to increased iodine loading), post-LOCA combustible gas control, the main steam isolation valve leakage control system, the control room atmosphere control system, and the emergency cooling water system. This review was performed to ensure that the ability of these systems to perform their safety function when responding to or mitigating the effects of design-basis accidents was not impaired by the approval of power uprate. Additionally, the effects of power uprate on high-energy line breaks, fire protection, and station blackout were considered.

3.3.1 Containment System

Primary containment temperature and pressure response following a postulated LOCA is important when determining the potential for offsite release of radioactive material, in determining ECCS pump NPSH requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment. Short-term and long-term containment analyses results are reported in the FSAR following a large break inside the drywell. The short-term analysis is directed primarily at determining the peak drywell pressure responses during the initial blowdown of the reactor vessel inventory to the containment following a DBA LOCA. The long-term analysis is directed primarily at determining the peak pool temperature response. The licensee indicated that the analyses were performed in accordance with Regulatory Guide 1.49 and Reference 1.

The effect of power uprate on the events which yield the limiting containment pressure and temperature response is evaluated in the following sections:

(1) Long-Term Suppression Pool Temperature Response

Bulk Pool Temperature

The licensee stated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA at 102 percent of the uprated power by using the SHEX computer code. The SHEX code utilizes more refined models than are used by the M3CPT/HXSIZ code in the original analysis to determine the suppression pool temperature. The SHEX code is capable of modeling containment response to more accident scenarios than the HXSIZ code. Many of the models used in the SHEX code are the same as, or very similar to, those used in the M3CPT code to calculate the short-term containment temperature and pressure response following a LOCA.

In a July 11, 1992, safety evaluation regarding GE Licensing Topical Report NEDC-31984P (Reference 3), the staff stated that although the SHEX code was not yet formally approved on a generic basis, use of the SHEX code in place of the M3CPT/HXSIZ code would be acceptable on a plant-specific basis, if adequate information is given to justify its use. In a letter of July 13, 1993 (Reference 12), the staff confirmed and clarified its position regarding the use of SHEX and the ANSI/ANS 5.1-1979 decay heat source term in containment response analyses for BWRs.

The licensee has submitted the results of three analyses performed using the SHEX code.

One analysis was performed to show how the current analysis methods compare with the analysis methods used in the FSAR. This case was evaluated with input parameters that match those used in the FSAR analysis as closely as possible. This analysis predicted a peak suppression pool temperature of 209 °F, while the original FSAR analysis had predicted 208.2 °F. Since SHEX and M3CPT/HXSIZ predicted essentially identical peak suppression pool temperatures, the use of SHEX for the analysis of long-term suppression pool response at power uprate is acceptable for Susquehanna.

The second analysis was performed to demonstrate the effects of power uprate with no other changes. This analysis predicted a long-term peak pool temperature of 211 °F.

The third analysis was performed at uprated power and included updated plant parameters. In this analysis, the licensee has updated the long-term cooling parameters for RHR service water temperature, RHR heat exchanger K-factor (Susquehanna design factor in lieu of generic BWR-4 factor), and ECCS pump heat (does not include HPCI system heat, because HPCI does not operate long term for a DBA LOCA). This analysis predicted a peak long-term suppression pool temperature of 203 °F, which remains below the suppression pool design temperature of 220 °F.

The licensee has also analyzed the highest bulk pool temperature response from an alternate shutdown cooling event according to Regulatory Guide 1.139 for power uprate assuming only one RHR heat-exchanger. This analysis predicted a peak bulk pool temperature of 208 °F, which remains below the design limit of 220 °F.

On the basis of its review (as discussed above), the staff concludes that the use of the SHEX code for calculating containment long-term peak bulk suppression pool temperature response is acceptable, and that the long-term peak bulk suppression pool temperature will remain acceptable after power uprate.

Local Pool Temperature Response With SRV Discharge

The licensee stated that the maximum local pool temperature with SRV discharge was previously calculated at 104.3 percent of current rated power to demonstrate compliance with NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." This analysis predicted a peak local pool temperature of 212.2 °F. The above event was reanalyzed at 102 percent of uprated power as reported in the Susquehanna Design Assessment Report. This analysis predicted a peak local pool temperature of 214 °F, which remains below the NUREG-0783 local pool temperature design limit of 216 °F. Both analyses were performed for 90 °F initial pool temperature.

On the basis of these results, the staff concludes that the local pool temperature will remain acceptable after the power is uprated.

Containment Gas Temperature Response

The licensee stated that the design temperatures for the containment drywell and wetwell will not be affected by the power uprate. The containment drywell design temperature of 340 °F was based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA and predicted a maximum temperature of 318 °F. The licensee stated that since the vessel dome pressure of 1055 psia (1040 psig) assumed for the FSAR containment analysis bounds the power uprate vessel dome pressure of 1053 psia (1038 psig), the initial break flow rate for this event will not change and therefore, power uprate will have no impact on the containment drywell design temperature.

The licensee also stated that the wetwell gas space peak temperature response is calculated assuming thermal equilibrium between the pool and wetwell gas space. Since the power uprate analysis has not changed the wetwell temperature response, it will have no effect on the wetwell space design temperature.

On the basis of this review, the staff concludes that the containment gas temperature response will remain acceptable after power uprate.

(2) Short-Term Containment Pressure Response

The licensee stated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation line to demonstrate that operation with power uprate will not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occur. This analysis was performed at 102 percent of the uprated power level using the GE M3CPT computer code. The calculated maximum containment pressure at uprated conditions is 44.6 psig. This code was also used in the original FSAR case which predicted a maximum containment pressure of 40.5 psig. The Susquehanna containment was designed for a maximum pressure of 53 psig.

The licensee also performed three analyses using the updated methods described above (see "Bulk Pool Temperature"). One was performed at current power level and predicted a maximum pressure of 43.4 psig. The updated methods show an increase of 2.9 psi in peak drywell pressure at current power due to the different assumptions (shorter MSIV closure time and use of the Moody slip flow model with different subcooled flow assumptions) used in performing the evaluations. The second analysis was performed at uprated power which predicted a maximum pressure of 44.6 psig. The third analysis was also performed at uprated power level in which the long-term cooling parameters were also updated. This analysis predicted no change in the maximum pressure of 44.6 psig because the additional parameter changes only affect the long-term results, and do not affect the peak containment pressure which is a short-term response.

On the basis of its review, the staff concludes that the pressure response following a postulated LOCA will remain acceptable after power uprate.

(3) Containment Dynamic Loads

LOCA Containment Dynamic Loads

NEDC-31897 (Reference 1) requires that the power uprate applicant determine if the containment pressure, temperature, and vent flow conditions calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate, and thus do not require further analysis.

The licensee stated that the results of the short-term LOCA containment pressure and temperature analysis were used to evaluate the LOCA dynamic loads such as pool swell, vent thrust, condensation oscillation, and chugging. The change in the short-term containment response with power uprate is small, and the loads remain bounded by the test conditions used to define the original loads except the pool swell loads. The licensee reported that a detailed evaluation of the wetwell components within the pool swell zone has been

completed. The power uprate loads and stresses were compared to the component's allowable values to determine component qualification. This comparison showed that all wetwell components within the pool swell zone are qualified for the power uprate pool swell loads. On the basis of its review, the staff finds that the LOCA dynamic loads and their effect on the components qualification at power uprate are acceptable. The staff is currently reviewing the effect of LOCA dynamic loads on spent fuel pool cooling components in response to the November 27, 1992 10 CFR Part 21 report described in section 3.5.1. NRC conclusions on the acceptability of LOCA dynamic loads on SFP cooling systems will be documented in separate correspondence.

SRV Containment Dynamic Loads

Safety-relief valve (SRV) containment dynamic loads include loads on the SRV discharge lines, quenchers and quencher supports, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration, and suppression pool configuration. Of these parameters, only the SRV setpoint is affected by power uprate. NEDC-31897 states that if the SRV setpoints are increased, the power uprate applicant will show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

The licensee stated that the original SRV load specification was based on a maximum reactor pressure of 1276 psig, and that the original SRV load specification has adequate conservatism to accommodate the slight increase in reactor pressure. The results of the reanalysis indicate that the loads remain below their design-allowable values and are not affected by power uprate.

Subcompartment Pressurization

A postulated pipe break in the annulus region between the reactor vessel and biological shield wall produces asymmetric pressure loads on the vessel, attached piping, and biological shield wall. The licensee stated that the original pressure loads were calculated using conservative mass and energy release rates and a computer code that predicted conservative pressure responses within the annulus and that a review of the original pressure load analysis has verified that adequate margin exists to accommodate the slight increase in reactor pressure due to power uprate. The staff agrees with the licensee's position that subcompartment pressurization effects will remain acceptable for uprated power.

(4) Containment Isolation

The licensee stated that the containment isolation capability is not affected by power uprate. The peak drywell pressure resulting from power uprate remains bounded by the original design conditions. Therefore, containment isolation valves and actuators will meet closure and leakage requirements at uprated containment pressure, temperature, and flows. On the basis of its

review, the staff agrees with the licensee that the operation of the plant at the uprated power level will not impact the containment isolation system.

(5) Post-LOCA Combustible Gas Control System

The licensee stated that the Susquehanna units have nitrogen-inerted containments even though worst-case hydrogen concentrations for the original power level did not require inerting. The worst-case concentration at the original power level is 3.5 volume percent. The design-basis hydrogen will increase by about 4.5 percent with power uprate. The staff has verified that the hydrogen recombiners have sufficient capacity to accommodate this increased load.

On the basis of its review, the staff concludes that the power uprate will not impact the post-LOCA combustible gas control system.

3.3.2 Emergency Core Cooling Systems

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

As discussed in the FSAR, compliance with the NPSH requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig (no containment overpressure) 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.

(1) High-Pressure Coolant Injection (HPCI) System

The licensee evaluated the HPCI system and determined that operation of this system at uprated conditions will be consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee has reported, in a letter of January 25, 1993, that it had installed the modifications on the HPCI system on each unit in response to GE SIL 480. These modifications were performed during the Unit 1 fourth and Unit 2 third refueling outages. As discussed in Section 4.2 of the GE letter (Reference 3), the modifications will avoid the possibility of turbine overspeed trips at the higher reactor pressure associated with power uprate. 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 submitted with the proposed license amendment. The staff requires that the licensee provide assurance that the HPCI system will be capable of injecting the design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the 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. This may be accomplished during startup testing following implementation of power uprate.

(2) RHR System (Low-Pressure Coolant Injection)

The hardware for the low-pressure portions of the RHR is 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 does not experience different operating conditions upon power uprate, there is no impact from power uprate.

(3) Core Spray System

The hardware for the low-pressure core spray (CS) system is 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 upon power uprate, there is no impact from 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) Automatic Depressurization System

The automatic depressurization system (ADS) uses safety/relief valves to reduce reactor pressure following a small-break LOCA and failure of the HPCI system to maintain reactor water level. 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 power 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 was submitted in the licensee's June 15, 1992, submittal (Reference 13).

3.3.3 Emergency Core Cooling System Performance

The emergency core cooling systems (ECCSs) 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 Part 50 (Appendix K). The licensee analyzed the Siemens Nuclear Power (SNP) 9x9-2 fuel, used in Susquehanna Units 1 and 2, using 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 Susquehanna 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 10 CFR Part 50 (Appendix K). A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K peak cladding temperature (PCT) as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The Susquehanna-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 Susquehanna is 1510 °F, which is well below the 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 the PP&L Licensing Topical Report. Therefore, the ECCS/LOCA analysis contained in the topical report for Susquehanna, Units 1 and 2, meets 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 °F 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 of April 2, 1993, to impose an LHGR reduction (multiplier) of 0.70 during SLO. On the basis of 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 19 and will be incorporated in the licensee's proposed amendment application.

An S/G-LOCA analysis for the ELLLA region was performed by the licensee at a core flow of 87 million lbm/hr and uprated power for Susquehanna 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 node is assumed to experience early dryout for the Appendix K Extended

Load Line Limit Analysis (ELLLA). The nominal and Appendix K results both show a small increase in the PCT when compared to the base 100 million lbm/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 of August 5, 1993 (Reference 20). The licensee stated that the upper bound PCT documented in 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. The licensee reported that 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, on the basis of 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 Susquehanna, 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 those of 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/4s (Susquehanna 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 at Susquehanna are similar to GE fuels, and since the S/G-LOCA results for Susquehanna 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 Susquehanna with SNP fuel.

3.3.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to ensure controlled and filtered release of particulates and halogens from primary and secondary

containment to the environment during abnormal and accident situations in order to maintain offsite thyroid doses within the 10 CFR Part 100 limits. The SGTS consists of two 100-percent-capacity, parallel, redundant flow trains. Each flow train consists of a mist eliminator, an electric air heater, a bank of prefilters, a high-efficiency particulate air (HEPA) (pre)filter, an upstream and downstream charcoal adsorber, a HEPA (after)filter, a vertical 8-inch-deep charcoal adsorber bed with fire-detection temperature sensors, a water spray system for fire protection, and one 100-percent-capacity exhaust fan. Each train is sized to change one secondary containment (SC) air volume per day while maintaining the SC at a slight negative pressure of 0.25-inch water gauge with respect to the outside atmosphere. Maintaining this negative pressure serves to prevent unfiltered release of radioactive material from the SC to the environment. The staff agrees with the licensee that the proposed slight increase in power (4.5%) by itself will not impair the capability of the SGTS to meet the design objective as stated above, since it does not change the ventilation design aspect of the SGTS.

The licensee stated that the proposed power uprate will increase the loss-of-coolant accident source term by 4.5 percent which will increase the loading on the SGTS filter trains by 4.5 percent. The staff recognizes that iodine loading in the filters will increase marginally (4.5%) due to the proposed power uprate. The SGTS design utilizes filters that meet the intent of Regulatory Guide (RG) 1.52 guidelines with respect to the design, testing, and maintenance criteria of enginered safety features (ESFs) grade filters. The staff notes that one of the criteria deals with the filter loading capability. Since the two SGTS filter trains have more than 500-percent excess capacity at the original power level, the licensee has determined that the slight increase (4.5%) in iodine loading will remain well below the original design capacity of the filters. The licensee stated and the staff agrees that even with a slight increase in the previously calculated limiting offsite thyroid dose due to the uprated power, filter design capacity will sustain the thyroid dose well below the 10 CFR Part 100 limit.

On the basis of these findings, the staff concludes that the uprated power level operation will not have any impact on the ability of the SGTS to meet its design objectives.

3.3.5 Other Engineered Safety Features Systems

(1) Main Steam Isolation Valve Leakage Control System

The licensee's containment analysis calculated that the peak post-LOCA pressures at uprated power conditions do not increase beyond the original design basis. On the basis of its review of those calculations, the staff agrees with the licensee's assertion that the operation of the MSIV leakage control system will not be affected by power uprate.

(2) Post-LOCA Combustible Gas Control System

In its submittal, the licensee confirmed the ability of the combustible gas control system (CGCS) to maintain oxygen and hydrogen concentrations within acceptable levels following a LOCA. This conclusion is consistent with that reached by GE in Reference 3. The licensee stated that although the amount of oxygen liberated by radiolytic decomposition of water is expected to increase slightly because of the power uprate, the expected concentrations are well within the capacity of the CGCS. The licensee also stated that hydrogen recombiners may need to be started sooner following a postulated LOCA after uprate; however, current procedures which direct control room operators to initiate the recombiners are based on combustible gas concentrations, not on a fixed time following a LOCA.

Additionally, the revised hydrogen generation calculations submitted by the licensee indicate that less hydrogen will be liberated due to corewide metal-water reactions than previously predicted. This slight decrease is primarily due to significantly lower predicted fuel cladding temperatures during a postulated LOCA. The decrease in expected PCT is a result of the use of more realistic calculational methods in the ECCS/LOCA analysis (see Section 3.3.3). On the basis of its review of the licensee's submittals, the staff concludes that the existing post-LOCA combustible gas control systems will continue to perform their design function after power uprate.

(3) Main Control Room Atmosphere Control System

The control room atmosphere control system (CRACS) is one of the control room habitability systems. The CRACS includes an emergency filtration system which in turn contains an emergency makeup air filter train and an emergency recirculation filter train. The emergency makeup air filter train filters the radioiodine and radioactive material in particulate form present in the outside makeup air intake during an emergency situation such as a design-basis accident (DBA). The emergency recirculation filter train filters a mixture of the control room recirculated air and already once-filtered outside makeup air.

The emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure (1/8-inch water gauge) relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room during an accident situation. The system accomplishes this design objective by bringing in controlled and filtered outside air and filtering the recirculated air to keep the control room operator doses within the General Design Criteria (GDC) 19 limits during an accident. Since power uprate does not change the design aspect of the control room emergency filtration system, the staff concludes that the proposed uprate in power (4.5%) by itself will not cause a significant increase in unfiltered inleakage of contaminated outside air into the control room during an accident.

The staff recognizes that iodine loading in the makeup air filters and recirculation air filters will increase marginally (4.5%) under uprate conditions and has concluded that the control room emergency filtration system filters are designed, tested, and maintained in accordance with RG 1.52 guidelines. On this basis, the staff concludes that the filters will continue to be valid for the CRACS at uprated power operation.

On the basis of these findings, the staff concludes that the uprated power level will have little or no impact on CRACS meeting its design objectives.

3.4 Instrumentation and Control

The staff's evaluation of setpoint changes associated with power uprate was limited to those setpoint changes for instrumentation identified in the licensee's submittals. The staff has completed its review of GE Topical Report NEDC-31336P, "General Electric Instrument Setpoint Methodology," October 1986 (Reference 6) and has approved the application of these methods to plant-specific data within the limits stated in the topical report.

A review of the licensee's submittals indicates that GE performed plant-specific calculations for PP&L using methods recommended by the Instrument Society of America (ISA) as outlined in GE Topical Report NEDC-31336P.

The licensee is considering the following setpoint changes:

(1) Flow-biased simulated thermal power for two-Loop operation

Change trip from (0.58W + 59%) to (0.555W + 56.5%).

Change allowable value from (0.58W + 62%) to (0.555W + 59.4%).

(2) Flow-biased simulated thermal power for one-loop operation

Change trip from (0.58W + 54%) to (0.555W + 51.7%).

Change allowable value from (0.58W + 57%) to (0.555W + 54.6%).

(3) Reactor vessel steam dome pressure high

Change trip from 1037 psig to 1087 psig.

Change allowable value from 1057 psig to 1093 psig.

(4) Main steam high flow

Change trip from 107 psid to 113 psid.

Change allowable value from 110 psid to 121 psid.

(5) Rod block for two-loop operation

Change trip from (0.58W + 50%) to (0.555W + 47.9%).

Change allowable value from (0.58W + 53%) to (0.555W + 50.8%).

(6) Rod block for one-loop operation

Change trip from (0.58W + 45%) to (0.555W + 43%).

Change allowable value from (0.58W + 48%) to (0.555W + 46%).

(7) Turbine stop valve and turbine control valve fast closure scram bypass

The turbine first-stage pressure setpoint will be changed to reflect the expected pressure at the new 30-percent power point.

(8) Average power range monitor (APRM) rod block, APRM simulated thermal power high-power clamps, and APRM neutron flux scram

These setpoints will not be physically changed. However, the change in the definition of rated thermal power (from 3293 MWt to 3441 MWt) will result in an increase of approximately 148 MWt to each of these points.

To verify the results of licensee-sponsored calculations and to better understand the quantitative effects of the assumed instrument errors, the staff audited the calculations for the reactor vessel steam dome high-pressure trip, the main steam high-flow trip, and the APRM trips (both fixed and flow-biased). The review demonstrated that the instrumentation errors assumed in the analyses were conservative with respect to the manufacturers' ratings and that the methods of analysis generally conform to those described in Reference 6. Exceptions to the methods described in Reference 6 are based on plant-specific data and instrumentation calibration procedures. The staff also acknowledges that these changes represent more current knowledge than was available when the topical report was issued in 1986.

The proposed setpoint changes are designed to maintain the existing margins between the proposed operating conditions and the new trip points. The same margins to the new safety limits are also maintained. These new setpoints do not significantly increase the likelihood of a false trip or a failure to trip upon demand. Therefore, the staff finds the setpoint changes, as described in the licensee's submittals, to be acceptable for power uprate. The licensee has stated that some of the setpoints described in the topical report may be changed; any such changes will be evaluated with the licensee's technical specification amendment submittal.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling

The fuel pool cooling and cleanup system (FPCCS) consists of fuel pool cooling pumps, heat exchangers, skimmer surge tanks, filter demineralizers, associated piping, valves, and instrumentation. The system is designed to cool the fuel storage pool water by transferring the decay heat of the irradiated fuel through heat exchangers to the service water system.

The licensee stated that the fuel pool storage capacity will not be changed for power uprate, and that the fuel pool cooling and cleanup system, its filter demineralizer system, the service water system, and the fuel pool cooling assist mode of RHR will not require modification. The licensee stated that cycle-specific calculations will ensure that cooling loads on the normal pool cooling system and fuel pool cooling assist mode of RHR will remain within their design capacities. The condensate system and the emergency service water (ESW) system will provide the necessary makeup flow to the fuel pool to maintain level if required, and normal makeup requirements are not significant. The licensee stated that power uprate will not adversely affect fuel pool water chemistry, and that the fuel pool cooling and cleanup system will be adequate for all required functions after power uprate.

On November 27, 1992, two former contract engineers for PP&L filed a 10 CFR Part 21 report contending that significant deficiencies exist in the design of the spent fuel pool cooling and cleanup system for Susquehanna, Units 1 and 2. The individuals asserted that the design of the FPCCS fails to meet numerous regulatory requirements, and that, following a design-basis LOCA, assuming the RG 1.3 source term for 100-percent core damage, or following a LOCA with an extended loss-of-offsite-power event, spent fuel pool cooling could not be restored and would lead to boiling of the water in the spent fuel pool. The NRC staff is reviewing the issues raised by the contract engineers, including the potential decay heat loads associated with the 4.5-percent increase in thermal power described in the topical report. Any issues raised by the contract engineers will be resolved separately from the staff's assessment of the power uprate amendment application.

3.5.2 Water Systems

The licensee evaluated the impact of power uprate on the various plant water systems, including the safety-related and non-safety-related service water systems, closed-loop cooling systems, circulating water system, and the plant ultimate heat sink. The licensee's evaluations considered increased heat loads, temperatures, pressures, and flow rates. The staff's review of these evaluations is discussed below.

(1) <u>Safety-Related Service Water Systems</u>

These systems include the emergency service water (ESW) system and the residual heat removal service water (RHRSW) system. All heat removed by these systems is rejected to the atmosphere via the ultimate heat sink (UHS). The staff's evaluation of the effects of uprated power level operation on each of these systems appears below.

<u>Emergency Service Water System</u>

The emergency service water (ESW) system removes heat from HVAC coolers, diesel generators, emergency core cooling (ECCS) and engineered safety feature (ESF) components, and other equipment required to operate under normal and accident conditions, including loss of offsite power (LOOP) and loss-of-

coolant accident (LOCA) conditions. The licensee revised the rated heat-removal capacities and flow requirements based on the effect of the increased design temperature on the system heat exchangers due to power uprate.

On the basis of its review, the staff finds that the ESW system heat exchangers can satisfy the uprated power cooling requirements at the new design temperature resulting from uprated power operation. The ESW system piping and components meet all their safety and design objectives at the uprated design temperature. Therefore, the staff concludes that the uprated power level operation has no impact on the ESW system operation.

Residual Heat Removal Service Water System

The residual heat removal service water (RHRSW) system provides a safety-related cooling water source for the RHR system under normal or postaccident conditions. The system pumps water from the ultimate heat sink (UHS) spray pond through the RHR heat exchangers and returns it to the pond via a spray network. The system may also be used to flood the reactor core or the primary containment following an accident.

The licensee stated that power uprate increases the heat loads on the RHRSW system proportional to the increase in reactor power level. The effect of higher UHS design temperature on the RHRSW system heat removal capacities has been considered in the RHR system and containment safety analysis and reviews and has been found to be acceptable.

On the basis of its review, the staff finds that the RHRSW system piping and components meet their safety and design objectives. In addition, the post-accident reactor core and containment flooding functions of the RHRSW are not affected by power uprate. Therefore, the staff concludes that the uprated power level has no significant impact on the RHRSW system design.

Ultimate Heat Sink

The ultimate heat sink (UHS) provides a safety-related cooling water source for the emergency service water (ESW) system and the residual heat removal service water (RHRSW) system during testing, normal shutdown, and accident conditions. The UHS consists of an 8-acre, 25-million-gallon concrete-lined spray pond. Following a design-basis accident (DBA), the UHS provides enough cooling water at or below the ESW and RHRSW design temperature for a minimum of 30 days without makeup.

The licensee stated that as a result of operation at the uprated power level, the post-LOCA UHS water temperature for minimum heat transfer (MHT) meteorology and system alignment will increase. The licensee performed an updated power uprate spray pond analysis for maximum water loss (MWL) conditions which revealed that the UHS contains sufficient water inventory to sustain a DBA for 30 days without makeup. The licensee determined that the revised UHS MHT and MWL analyses shows that the current technical specifications for normal operation maximum pond temperature and minimum water

level remain adequate to ensure that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions.

On the basis of its review, the staff agrees with the licensee's conclusion that the UHS design is adequate for the uprated power operation and no modification to the UHS system is required.

(2) Non-Safety-Related Service Water System

The service water (SW) system is designed to continuously supply cooling water to various heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation, and has no safety-related function.

The licensee stated that the service water system will support power uprate with no equipment or setpoint changes. The design SW heat load bounds the power uprate conditions. Therefore, the cooling tower is able to dissipate the service water heat load at uprated conditions without affecting the existing design service water temperature.

Since the SW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the SW system design and performance.

Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system cools various auxiliary plant components in the reactor and radwaste buildings during normal and loss-of-offsite power (LOOP) conditions, and has no safety-related function. The licensee stated that the increase in heat load due to uprated power operation is insignificant to the RBCCW system design.

Since this system does not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the RBCCW system.

Turbine Building Closed Cooling Water System

The turbine building closed cooling water (TBCCW) system supplies cooling water to auxiliary plant equipment in the turbine building. The licensee stated that the increase in heat load from the equipment due to the uprated power level operation is insignificant and that the TBCCW system design cooling capacity will not be exceeded.

Since the TBCCW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level to the TBCCW system design and performance.

Gaseous Radwaste Recombiner Closed Cooling Water System

The licensee stated that the 4.5-percent power uprate will increase the heat loads from the offgas recombiner condenser, the steam jet condenser, and condensate cooler of the offgas system by the same percentage. The licensee performed an evaluation of the offgas system and determined that it will remain within its original design capacities. Therefore, the licensee concluded that the increase in offgas system heat load on the gaseous radwaste recombiner closed cooling water (GRRCCW) system is also within the original GRRCCW system capacity.

On the basis of its review, the staff concludes that the effect of uprated power operation on the GRRCCW system is negligible and that there is adequate operating margin for this system to perform at uprated power operation.

River Water Makeup

The river water makeup system consists of four river water pumps and their screens, the intake structure and pump house, piping, valves, and controls. It supplies raw water to compensate for cooling tower and spray pond blowdown and evaporation, and for makeup to the plant water treatment and storage systems.

The licensee determined that during the peak demand periods, power uprate will increase the maximum system design above the original system design capacity with three of the four river water pumps running. However, the licensee performed a preliminary evaluation which concluded that the fourth makeup pump can be operated to maintain sufficient margin without adversely affecting intake structure HVAC performance, electrical distribution, system piping, or traveling screen operation. The licensee intends to further evaluate and test the need for four-pump operation after power uprate.

Since the river water makeup system does not perform any safety function, the staff has not reviewed the impact of the uprated power level on the river water makeup system design and performance.

(3) Chilled Water Systems

Reactor Building Chilled Water (RBCW) System

The RBCW system supplies chilled water to various reactor building and drywell heating, ventilation, and air conditioning (HVAC) systems and equipment loads during normal plant operation. The RBCW system does not perform any safety function.

The licensee stated that the actual peak loads during hot weather conditions can exceed the original calculated design load on the RBCW system. In order to meet the peak heat load demands (even for current power rating), the licensee has developed and implemented system operating strategies, such as tandem chiller operation to address this situation. The licensee stated that

these system operating strategies will be implemented as necessary for power uprate conditions.

Since the RBCW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level to the RBCW system design and performance.

Control Structure Chilled Water System

This system serves the control structure HVAC system and is addressed in Section 3.5.4.

Radwaste Building Chilled Water System

This system serves the radwaste building HVAC system and is addressed in Section 3.5.4.

Turbine Building Chilled Water System

This system serves the turbine building HVAC system and is addressed in Section 3.5.4.

3.5.3 Standby Liquid Control System (SLCS)

The ability of the SLCS to achieve and maintain safe shutdown is not directly affected by core thermal power; rather, it is a function of amount of excess reactivity present in the core; and as such, is dependent upon fuel loading techniques and uranium enrichment. The SLCS system is designed to inject at a maximum pressure equal to that of the lowest safety/relief valve setpoint. The SLCS pumps are positive displacement pumps, and the small (29 psig) increase in the lowest safety/relief valve setting as a result of uprate will not impair the performance of the pumps. The staff concludes that the ability of the SLCS system to inject to the reactor will not be impaired by the power uprate.

The SLCS shutdown requirements for future operating cycles will be evaluated by the licensee on a cycle-specific basis.

3.5.4 Heating, Ventilation and Air-Conditioning (HVAC) Systems

The licensee evaluated the impact of higher process fluid temperature in piping for all HVAC systems including the drywell cooling system, reactor building HVAC system, control structure HVAC system, radwaste building HVAC system, turbine building ventilation system, engineered safeguards service water (ESSW) pump house heating and ventilation system, and the diesel generator building ventilation system.

The licensee indicated that adequate margin exists in the drywell cooling system and the reactor building HVAC system capacities to meet the additional heat loads imposed by power uprate and increased core flow. The licensee

performed an evaluation which confirmed that power uprate has no significant effects on the control structure HVAC, the radwaste building HVAC system, and the turbine building ventilation system. The licensee confirmed that the system functions and performance of the ESSW pump house heating and ventilation system and the diesel generator building ventilation and their interfacing and supporting systems will not be affected as a result of power uprate. The licensee stated that the uprated heat loads would have no impact on maintaining the design environmental temperature parameters for these systems.

On the basis of its review, the staff agrees with the licensee that uprated power level operation will have no impact to the plant HVAC systems.

3.5.5 Fire Protection

The licensee stated that the operation of the plant at the uprated power level does not affect the fire-suppression or fire-detection systems. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe-shutdown systems and equipment used to achieve and maintain cold-shutdown conditions do not change, and are adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire-protection systems and analyses are not affected by power uprate.

On the basis of its review, the staff finds that the fire-suppression and fire-detection systems and their associated analyses are not affected by power uprate.

3.6 Power Conversion Systems

The steam and power conversion systems and associated components (e.g., the turbine/generator, condenser and steam jet air ejectors, turbine steam bypass, feedwater and condensate systems) were originally designed to utilize the energy available from the nuclear steam supply system and to accept the system and equipment flows resulting from continuous operation at 105 percent of the currently licensed rated power. Therefore, these systems will not be affected by power uprate.

On the basis of its review, the staff agrees that operation at uprated power should not have a significant impact on the steam and power conversion systems and associated components.

3.7 Radwaste Systems and Radiation Sources

The licensee evaluated the radiological impact of the proposed uprate to show that the applicable regulatory acceptance criteria continue to be satisfied for the uprated power conditions. In conducting this evaluation, the licensee considered the effect of the higher power levels on source terms, onsite and offsite doses, and control room habitability during both normal and accident conditions.

3.7.1 Liquid Waste Management

The licensee stated that influent to the liquid radwaste processing system would increase approximately 4.7 percent due to uprate. On the basis of plant experience obtained in 1991, the licensee has determined that the liquid radwaste system has sufficient capacity to handle the increased influent.

The licensee also noted that a 10-percent increase in activated corrosion products would be expected because of the power uprate, but that the total volume of processed waste would not be expected to increase appreciably. The licensee concluded, from reviewing plant operating effluent reports and after considering the expected slight increase in effluents as a result of power uprate, that the requirements related to 10 CFR Part 20 and 10 CFR Part 50 (Appendix I) will continue to be satisfied. Having reviewed available plant data and experience with previous power uprates, the staff concludes that the pwoer uprate will have no significant adverse effect on liquid effluents.

3.7.2 Gaseous Waste Management

The licensee noted that gaseous wastes generated during both normal and abnormal operation are collected, controlled, processed, stored, and disposed of by means of the gaseous waste processing treatment systems. These systems include the standby gas treatment system, the offgas recombiner system, and the ambient temperature charcoal treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. On the basis of its review of available plant data and previous experience with other power uprates, the staff concludes that no significant adverse effect on airborne effluents will occur as a result of power uprate.

3.7.3 Radiation Sources in the Core and Coolant

Radioactive materials in the reactor core are produced in direct proportion to the fission rate. Thus, the expected increase in the levels of radioactive materials (for both fission and neutron activation products) produced are expected to increase by a maximum of 4.5 percent. The licensee noted that experience to date with operation of Susquehanna Units 1 and 2 indicates that concentrations of fission and activation products in the reactor coolant will not increase significantly above those currently experienced. Current experience with operation of the Susquehanna units indicates that both units operate well below the 0.1 Curie/sec design basis and that current offsite radiological release rates are well below the original design basis. On the basis of its review of available plant data and experience with previous power uprates, the staff concludes that no significant adverse effect on radiation sources in either the core or reactor coolant will occur due to power uprate.

3.7.4 Radiation Levels

The licensee considered the effects of power uprate on radiation levels in the Susquehanna facility during normal operation as well as during postaccident conditions. The licensee concluded that radiation levels from both normal operation and accident conditions could increase slightly. However, any such increase would be small and would be bounded by conservatisms in the original plant design and analysis. Further, the licensee noted that the calculated offsite radiological consequences are well below the regulatory limits given in 10 CFR Part 20 and 10 CFR Part 50 (Appendix I). On the basis of its review of plant data and previous experience with other power uprates, the staff finds that no significant adverse effect on radiation levels (either onsite or offsite) will result from the proposed power uprate.

3.8 Reactor Safety Performance Evaluations

3.8.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances in the plant, caused by a malfunction, a single failure of equipment, or a 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 affect the selection of limiting events. The following events will be explicitly evaluated for cycle-specific reload analyses:

(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 percent of the uprated power level, or 3510 MWt, to account for power uncertainty.

Parametric studies were conducted as part of developing the licensee's licensing methods. These studies lead to the following expectations. The GLRWOB delta CPR (critical power ratio) is determined on the basis of 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. The increased flow rate for the GLRWOB and the FWCF is expected to 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 question from the staff, the licensee, in a letter of 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 as a result of power uprate or increased core flow. The analysis plan proposed by the licensee is acceptable.

3.8.2 Design-Basis Accidents

The licensee reanalyzed a number of events to determine the whole-body and thyroid doses at the exclusion area boundary and in the low population zone. In evaluating the effects of power uprate on accident consequences, the licensee reanalyzed the loss-of-coolant accident, the main steamline break accident, the fuel handling accident, and the control rod drop accident. The analysis was performed based upon operation at 105 percent of uprated power, using current NRC-approved methodologies. The staff has reviewed the information submitted by the licensee, and concludes that the analyzed consequences of postulated accidents will remain well within the staff acceptance criteria and are, therefore, acceptable.

3.8.3 Anticipated Transients Without Scram (ATWS)

Although General Electric has performed generic bounding ATWS analyses, these analyses cannot be used for Susquehanna because the licensee: (1) uses non-GE fuel and (2) has taken exceptions to Revision 4 of the Emergency Procedure Guidelines (EPGs) for responding to ATWS, which are assumed in the GE generic analyses.

The licensee is currently performing a plant-specific analysis of the plant response to an ATWS under uprated conditions. The licensee will submit the results of this analysis with the proposed license amendment application in support of power uprate implementation. The results will also be included in the ongoing project to upgrade Susquehanna Emergency Operating Procedures. The staff will evaluate the plant-specific analysis when the licensee submits it.

3.8.4 Station Blackout

Per the NUMARC 87-00 methodology, Susquehanna is classified as a 4-hour-duration station blackout (SBO) plant based on an offsite power design characteristic 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 4 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 stated that the results of this calculation show that the compartment temperatures only rise 2 or 3 °F as a result of power uprate, and that the temperatures during an SBO event will not exceed the 180 °F limit identified in Appendix F of NUMARC 87-00, Revision 1.

The equipment with revised heat inputs used for the power uprate SBO evaluations includes motors, electrical cabinets, piping, and such miscellaneous mechanical equipment as heat exchangers. The rest of the equipment whose heat load changes with power uprate, but that 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.

3.9 Additional Aspects of Power Uprate

3.9.1 High-Energy Line Break

The slight increase in the operating pressure and temperature caused by the uprated power condition results in a small increase in the mass and energy release rates following a high-energy line break (HELB). This results in a small increase in the subcompartment pressure and temperature profiles and a negligible change in the humidity profile. The licensee performed a reanalysis of high-energy line breaks for all systems currently evaluated in the FSAR. The licensee has reevaluated the HELB for the main steam system, high-pressure coolant injection system, reactor core isolation cooling system, reactor water cleanup system, and the residual heat removal system. As a result of this reevaluation, the licensee has concluded that the affected compartments that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following an HELB at uprated power conditions. The staff has reviewed the results of the licensee's reanalysis and finds them acceptable.

The licensee is currently evaluating the calculations supporting the disposition of potential targets of pipe whip and jet impingement from the postulated HELBs to determine the effects of power uprate. The licensee expects the evaluation to yield results that confirm the adequacy of the existing design under power uprate conditions.

Since the licensee has not completed calculations supporting the disposition of potential targets of pipe whip and jet impingement from postulated HELBs to confirm the adequacy of the existing design under power uprate conditions, the staff has not reviewed the impact of the uprated power level operation on HELBs. The licensee stated that the results of its evaluation will be included with the proposed license amendment.

3.9.2 Moderate-Energy Line Break and Internal Flooding

The licensee determined that the existing moderate-energy piping experiences no appreciable pressure or temperature increases due to power uprate. The high-pressure, moderate-energy HPCI and RCIC pump discharge piping does experience a small increase in core injection mode pressure, but it is within the existing design pressure and their status as moderate-energy lines is not affected.

On the basis of its review, the staff concludes that the moderate-energy line break (MELB) water spray and flooding evaluation of the plant is not affected by the uprated conditions and is acceptable for uprated power operation.

3.9.3 Equipment Qualification

(1) Environmental Qualification of Electrical Equipment

The licensee evaluated safety-related electrical equipment to ensure qualification for the normal and accident conditions expected in the areas in which the equipment is located. For equipment located inside the containment, the licensee indicated that current accident and normal design conditions for temperature, pressure, and humidity are unchanged for power uprate. Accident and normal radiation levels increase in proportion to the increase in power. For equipment outside the containment, normal operational temperature, pressure, and humidity conditions are unchanged. However, accident temperatures increase less than 5 °F and pressures increase less than 1 psi. Normal operational and accident radiation levels increase in relationship to the increase in power.

On the basis of the evaluation, the licensee determined that no safety-related equipment was identified as unqualified for power uprate environmental conditions. The qualified life of certain equipment may be reduced, but a revised aging analysis will assure replacement before the equipment exceeds qualified life.

On the basis of its review, the staff finds the licensee's approach to qualification of safety-related electrical equipment for power uprate conditions acceptable.

(2) Environmental Qualification of Mechanical Equipment With Non-Metallic Components

The licensee stated that operation at the uprated power level is expected to increase the normal process temperatures by less than 6 °F. As in the case of electrical equipment, normal operational and accident radiation levels also increase slightly due to uprate.

The licensee stated that its reevaluation is not expected to identify any components which are unqualified for the uprated environmental conditions. The qualified life of certain equipment may be reduced, but a revised aging analysis will assure replacement before the equipment exceeds its qualified life.

On the basis of its review, the staff finds the licensee's approach to qualifying mechanical equipment with non-metallic components for power uprate conditions acceptable.

(3) Mechanical Component Design Qualification

Having reviewed the licensee's submittals, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

(a) Seismic loads are unchanged by power uprate.

(b) The original LOCA load conditions bound the power uprate conditions as stated in Section 3.2.3.

(c) The slight increase (about 1 to 2 percent) in AP, JR and SRV loads as delineated in Section 3.2.3 has a negligible effect on equipment dynamic response.

(d) No new pipe break locations resulted from the uprated conditions.

3.10 Evaluation of Impact on Responses to Generic Communications

In Reference 3, GE assessed the impact of power uprate on licensee responses to generic NRC and industry communications. GE reviewed both NRC and industry communications to determine whether parameter changes associated with power uprate could potentially affect previous licensee commitments or responses. Of the large number of documents reviewed (more than 3000 items), GE found that only a small number were potentially affected by power uprate. The list of affected topics was then divided into those that could be bounded generically by GE, and those that would require plant-specific reevaluation.

The NRC staff audited the GE assessment in December 1991, and approved the assessment in Reference 4.

In addition to assessing those items requiring a plant-specific reevaluation, the licensee also reviewed the potential effects of uprate on pending licensing actions and internal commitments, such as nonconformance reports and engineering deficiency reports. The licensee found no additional commitments that require modification to accommodate power uprate.

4.0 CONCLUSION

The staff has completed its review of PP&L's "NE-092-001, Revision 0, Licensing Topical Report for Power Uprate with Increased Core Flow," (Reference 13) and subsequent submittals, and has concluded that operation of the Susquehanna Steam Electric Station, Units 1 and 2, in the manner described in the topical report will continue to comply with all applicable regulations and is, therefore, acceptable. As discussed in this safety evaluation, there are four open items that PP&L will address when it submits the proposed license amendment application. These four items are: (1) the startup test plan (3.2.4), (2) the ATWS analysis (3.8.3), (3) the pipe whip and jet impingement evaluation (3.9.1), and (4) upgrading the Emergency Operating Procedures (3.8.3).

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