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See Proposed Change To Tech Specs.

SUBJECT: Forwards Proposed Amend 117 to License NPF-22 re power uprate w/increased flow.

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NOV 24 1993

Director of Nuclear Reactor Regulation
Attention: Mr. C. L. Miller, Project Director
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Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 117 TO LICENSE NO. NPF-22 :
POWER UPRATE WITH INCREASED FLOW
PLA-4055 FILES A17-2/R41-2 Docket No. 50-388

Dear Mr. Miller:

References:

1. *NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow", Pennsylvania Power & Light, December 1992.*

2. *NEDC-31894P, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," GE Nuclear Energy, June 1991.*

3. *NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," GE Nuclear Energy, July 1991.*

4. *NEDC-32064P-1, Revision 1, "Power Uprate With Increased Core Flow Safety Analysis for Susquehanna 1 & 2", GE Nuclear Energy, July 1993.*

5. *NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.*

6. *GE-NE-523-107-0893, "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," GE Nuclear Energy, September 1993.*

Supporting Correspondence:

1. *PP&L Letter PLA-3788, "Submittal of Licensing Topical Report for Power Uprate with Increase Core Flow," dated June 15, 1992.*

2. *PP&L letter PLA-3816, "Response to 7/10/92 Questions on Power Uprate," dated July 24, 1992.*

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3. *PP&L letter PLA-3890, "Submittal of Revision 1 to Power Uprate Licensing Topical Report," dated December 18, 1992.*

4. *PP&L letter PLA-3900, "Correction to Revision 1 of Power Uprate Licensing Topical Report," dated January 8, 1993.*

5. *PP&L letter PLA-3908, "Confirmation of RCIC/HPCI Mods-Power Uprate," dated January 25, 1993.*

6. *PP&L letter PLA-3948, "Revisions to PP&L Power Uprate Submittal," dated April 2, 1993.*

7. *PP&L letter PLA-4010, "Response to Request for Information by NRR/SRXB on Power Uprate LTR," dated August 5, 1993.*

8. *PP&L letter PLA-4014, "Response to NRC Request for Additional Information Regarding the Power Uprate Containment Response Evaluation," dated August 12, 1993.*

9. *PP&L letter PLA-4028, "Response to Request for Additional Information, Licensing Topical Report for Power Uprate," dated September 29, 1993.*

The purpose of this letter is to propose changes to the Susquehanna SES Unit 2 Technical Specifications to uprate the current licensed power level from 3293 MWt to a new limit of 3441 MWt.

INTRODUCTION

The proposed license amendment consists of many changes that will permit uprated power operation for Susquehanna Unit 2. The unit is a General Electric (GE) BWR/4. Unit 2 is currently licensed for operation at 3293 MW_t(100%). The original design power rating for the Susquehanna unit was 3439 MW_t. This amendment redefines Rated Thermal Power at 3441 MW_t which represents an approximate increase of 4.5% over the previously licensed power level. This power supports a steam flow to the main turbine of 14.14 million pounds per hour that will be considered the uprated "rated steam flow." The uprated steam flow represents a 5% increase over the rated steam flow associated with the current licensed power. Analyses and methodology to support this change are contained in PP&L Licensing Topical Report NE-092-001, Reference 1, which is currently under review by the NRC. This report provided an overall assessment of the impact of the proposed uprate on the overall design and operation of Susquehanna SES, and formed the basis for this proposed license amendment.

A number of GE BWR plants are considering uprated power operation and GE has developed generic guidelines NEDC-31894P-1, Reference 2, (commonly called "LTR1") which provide a systematic methodology for the assessment of BWR power uprate capability. This submittal follows the guidelines presented in that document. Reference 1 is the plant specific report that

summarizes all of the significant safety and operational evaluations performed to support this proposed amendment. Reference 1 utilizes the generic evaluations provided in Reference 2 and NEDC-31984P, Reference 3, (commonly called LTR2) which have been approved by the NRC.

The proposed amendment also includes changes to allow operation in the increased core flow region (8% above the present rated core flow of 100 million lbm/hr). This insures that operational flexibility that presently exists is maintained after the power uprate. The portion of the proposed amendment to widen the core flow operating range is consistent with the power/flow range previously licensed for the BWR/4 product line.

Power Uprate raises (expands) the top portion of the operating map (power vs. reactor flow) consistent with the NRC approved generic guidelines for BWR power uprates in Reference 2. The 100% rod line will remain virtually unchanged. Although the definition of the flow biased APRM rod block and scram will not change, they will be increased by 4.5% by virtue of the 4.5% increase in core thermal power. This allows additional margin above the present (and post power uprate) 100% rod line. The safety analyses contained in Reference 1 have analyzed and the reload analysis to support the Unit 2 Cycle 7 Core Operating Limits Report (COLR) will analyze operation at both uprated power with higher rod lines and increased core flow conditions.

The setpoints for the flow biased Rod Block Monitor (RBM) rod block have changed. On an absolute megawatt vs. reactor flow basis, the flow biased rod block remains approximately the same in order to maintain the pre-power uprate flow biased rod block. This flow biased rod block is used in the Rod Withdrawal Error (RWE) analyses. The power uprate RWE analyses require that the flow biased rod block remain approximately the same (in Mw_t) in order to produce acceptable MCPR operating limits. This produces the same flow window at the uprated power as for the current cycles.

Performance improvement features previously licensed for Susquehanna Unit 2 include Single-Loop Operation (SLO) and Extended Load Line Limit Analysis (ELLLA). These features along with Increased Core Flow will be maintained with power uprate. The safety analyses contained in Reference 1 were performed and the reload safety analyses are being performed in an integrated manner, to establish these features as standard operating conditions for power uprate.

The proposed power uprate license amendment included with this submittal is applicable to Susquehanna Unit 2 and is non-cycle specific. The Technical Specification changes reference reports which identify significant changes to the LOCA analysis (SAFER/GESTR-LOCA) and minor changes to other licensing analysis methodology which will be utilized for Unit 2, Cycle 7 and future power uprate cycles. The cycle specific thermal limits which will result from the reload licensing analysis will be included in the cycle specific COLR.

SAFETY ANALYSIS

Proposed Changes

Reference 1 identified many potential Technical Specification changes required to support operation at uprated power. Each of the final Operating License and Technical Specification changes being proposed is evaluated below:

1. Rated Thermal Power is increased to 3441 megawatts thermal on page 3 of the Operating License and is redefined accordingly in Technical Specification Definition 1.33, "Rated Thermal Power".

   Evaluation

   This increase in rated thermal power level at Susquehanna Unit 2 follows the generic guidelines of Reference 2. Reference 2 provides generic licensing criteria, methodology, and a defined scope of analytical and equipment review to be performed to demonstrate the ability to operate safely at the uprated power level. Technical Specification parameters that are expressed as a percentage of reactor power or steam flow were not changed since the uprated definition of those parameters was used in the bounding analyses and evaluations required by Reference 2 unless otherwise specified in this submittal. Reference 1 provides the description of the power uprate licensing analysis methodology changes and the results of the evaluations performed to support the proposed uprated power operation consistent with the methodology presented in Reference 2. That report provides a description of the power uprate licensing analysis methodology which will be used to determine cycle specific thermal limits for Unit 2, Cycle 7 and future cycles and concludes that an uprated power level of 3441 megawatts thermal can be achieved without a significant effect on equipment or safety analyses.

2. The reference to "rated core flow" in Technical Specification 2.1.1 and 2.1.2 has been replaced with a reference to actual core flow. The references to "rated core flow" have been deleted to avoid confusion since the allowable core flow is being increased by 8%. 10 Mlbm/hr is being used in these specifications to be consistent with other similar Technical Specification changes described below in Item 8.

   Evaluation

   The basis for Technical Specification 2.1.1 is that boiling transition will not occur in bundles if core power is less than 25% of Rated Thermal Power regardless of pressure or core flow. Specification of less than 10% rated core flow is not crucial to the basis and, thus, use of 10 Mlbm/Hr is acceptable.

   For Technical Specification 2.1.2, the XN-3 critical power correlation is valid for pressure $\geq$ 580 psig and bundle flow $\geq$ 0.25 Mlbm/Hr-ft^2. As stated in the basis for Technical Specification 2.1.1, if vessel downcomer water level is above TAF, and core power > 25%, bundle flows for potentially limiting bundles will be > 0.25 Mlbm/Hr-ft^2 due to natural circulation. In addition, Technical Specification 3.4.1.1.1 requires at least 1 recirculation loop in operation to run in Condition 2, which

would produce a core flow in excess of 30 Mlbm/Hr. Therefore, core flows below about 30 Mlbm/Hr-ft^2 are prohibited when the reactor is at power. Thus, the change from "10%" to "10 million lbm/hr" is acceptable.

3. Technical Specifications 3.2.2, 3.4.1.1.2.a.2, 3.4.1.1.2.a.3, 3.4.1.1.2.a.5.b and Tables 2.2.1-1 (Item 2.a, 2.b, and 2.c) and 3.3.6-2 (Item 2.a.1, 2.c, and 2.d) do not change because of power uprate. However since the setpoints in these technical specifications are referenced to Rated Thermal Power, which changes, they therefore represent a change from current limits. The change in Rated Thermal Power has the effect of raising the top portion of the operating map (power vs. reactor flow) by 4.5%. This change is being made to maintain operational flexibility.

   Evaluation

   The safety analyses contained in Reference 1 analyze operation at both uprated power with 4.5% higher rod lines and increased core flow. In addition, GE has analyzed and received approval for BWR/4 product line operation in the Maximum Extended Operating Domain (MEOD). Operation at the 4.5% higher rod lines is bounded by the MEOD analysis. Additional justification for this small increase in Power/Flow operating range is contained in Section C.2.3 of Reference 2.

   The cycle specific reload analyses will evaluate operation at the increased power vs. flow conditions (100% uprated power vs. 87% core flow to 100% uprate power vs. 108% core flow). These analyses will ensure that the limits established in the Core Operating Limits Report are applicable to rated power operation from 87% to 108% core flow.

4. The reactor steam dome pressure-high scram (Technical Specification Table 2.2.1-1, Item 3) trip setpoint and allowable values are being changed to $\leq$ 1087 psig and $\leq$ 1093 psig respectively.

   Evaluation

   This scram function is designed to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor vessel maximum operating pressure and the specified analytical limit is used in the transient analyses. The analytical limit of 1105 psig is used in the uprated transient analyses. The results of the overpressure protection analysis indicate that the peak pressure will remain below the 1375 psig ASME limit which meets plant licensing requirements. In accordance with the methodology described in Reference 1, the transient analyses will be performed using the analytic limit and the results will be incorporated into the Core Operating Limits Report.

5. Bases Section 2.1.1 is changed to indicate that at 25% of the uprated thermal power, a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of approximately 3.0 in lieu of a bundle radial peaking factor of greater than 3.0. This change is being made to more accurately characterize the bundle radial peaking factor after the uprate.

Evaluation

This basis change reflects the slight decrease in the allowed bundle radial peaking factor due to the uprated thermal power. The conclusion that the expected peaking factor will be significantly lower than the calculated peaking factor remains unchanged for the uprate.

6. Bases Section 2.2.1 is expanded to include the basis for the automatic bypass of the RPS functions on turbine stop valve closure and control valve fast closure below 30% of rated power. Bases Sections 2.2.1 and 3/4.3.4 are both expanded to equate the turbine first stage pressure analytical limit of 147.7 psig with 22% of rated turbine load. This was done to provide the operator with a redundant, independent confirmation of turbine first stage pressure.

Evaluation

These proposed changes to the Technical Specifications Bases sections provide additional information only and do not change existing bases for the Reactor Protection System or Recirculation Pump Trip instrumentation setpoints.

7. Technical Specification 4.1.5.c is being revised to require the Standby Liquid Control (SLC) pumps to develop a discharge pressure of $\geq$ 1224 psig.

Evaluation

The SLC pump test discharge pressure acceptance criteria are based on the lowest relief valve setpoint. The lowest setpoint is being increased by 30 psi (to 1106) due to power uprate. Operating with increased core flow will result in additional friction losses through the core and a slightly larger core differential pressure (approximately 4 psi).

8. The references to "rated core flow" in Technical Specifications 3.2.2, 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3, and Figure 3.4.1.1.1-1 have been replaced with references to actual core flows. The references to "rated core flow" have been deleted to avoid confusion since the allowable core flow is being increased by 8%.

Evaluation

Technical Specification 3.2.2 contains the definition of "W" for the flow biased APRM scram equation. The word "rated" is being deleted from the definition of "W" since rated core flow is being increased. The definition of "W" is not altered. The change is being made for editorial purposes to avoid confusion.

Technical Specifications 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3, and Figure 3.4.1.1.1-1 specify performance requirements and limits for the Reactor Recirculation System. These specifications are referenced to the current rated core flow. The references to "rated core flow" are being replaced with actual equivalent core flows. The specifications are equivalent and unchanged. This change is being made for editorial purposes to avoid confusion since rated core flow is being increased.

9. The turbine first stage pressure scram bypass at 30% power in Technical Specification Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) is revised to indicate that the uprated equivalent allowable value of first stage turbine pressure is 136 psig. This value ensures that the analytical limit of 147.7 psig, which represents 30% rated thermal power, is not exceeded.

   As currently written Note (j), Note (b) and Table 3.3.1-1, ACTION 6 are unclear and could be misinterpreted. They apply only when RPS scram functions and End-of-Cycle Recirculation Pump Trip on turbine main stop valves (MSV) closure or control valve (CV) fast closure are not automatically bypassed. ACTION 6 provides no guidance in the event the bypass fails to lift when thermal power is above 30%. In the worst case, the action statement could be interpreted literally to allow full power operation with the RPS function still bypassed. Such operation would violate the licensing basis analysis for the MCPR operating limit (for the Generator Load Rejection Without Bypass transient), which takes credit for operation of the anticipatory scram on CV fast closure at greater than 30% of rated thermal power.

   Evaluation

   The revisions to Table 3.3.1-1, ACTION 6, Table 3.3.1-1, Note (j), and Table 3.3.4-1 Note (b) clarify the current action statements; they do not change the intent of the action statement.

   FSAR Chapter 15 transient analyses and reload licensing analyses take credit for operation of the anticipatory scram function on turbine MSV closure and CV fast closure for power levels greater than 30% of rated thermal power. The proposed revision to Table 3.3.1-1, ACTION 6 provides better assurance of the availability of the anticipatory scram function, since the current specifications could be interpreted literally to allow full power operation with the RPS function bypassed.

   The proposed revision to Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) does not change the operation of the RPS and EOC-RPT bypasses on turbine MSV closure and CV fast closure below 30% power. The turbine first stage pressure switches will still be calibrated in the same manner, and, by procedure, the reactor operator will not exceed 30% power if the trip bypass annunciator does not clear.

   The setpoints for the RPS and EOC-RPT bypass functions were selected to allow sufficient operating margin to avoid scrams during low power turbine generator trips. As discussed in Reference 2, Section F.4.2(c) and in Section 5.1.2.8 of the GE Power Uprate Safety Analysis, Reference 3, this small absolute pressure setpoint increase maintains the safety basis for the setpoint.

10. The main steam line flow high differential pressure setpoint and allowable value, as shown in Technical Specification Table 3.3.2-2, Item 3.d, are revised to read trip and allowable values of 113 psid and 121 psid respectively. Footnote "**" was added to Table 3.3.2-2 to indicate that these values will be confirmed during the power uprate start-up testing. If revisions to the setpoint and allowable value are required, they will be forwarded to the Commission for approval within 90 days of completion of the test program.

    Evaluation

    The main steam line flow high differential pressure setpoint changes reflect the redefinition of rated main steam line flow that occurs with power uprate. The allowable value is maintained at the same percentage of rated steam flow as the differential pressure changes due to the increased uprate steam flow. The analytical limit of 140% of uprated steam flow is maintained for the uprated analyses. The relationship between the allowable value and the analytical limit was retained to ensure that a trip avoidance margin is maintained for the normal plant testing of MSIV's and turbine stop valves. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which meets the original intent of the design.

11. The RWCU Flow - High isolation (Technical Specification Table 3.3.2-2, Item 4.f1) trip setpoint and allowable value are being changed to $\leq$ 462 gpm and $\leq$ 472 gpm respectively.

    The RWCU System flow is being increased by 10% to maintain reactor coolant water chemistry at a level equal to pre uprate levels. The isolation setpoint change mentioned above is required to adequately maintain operating margin between normal process values and the isolation setpoints.

    Evaluation

    The basis for the RWCU Flow - High isolation is to ensure a RWCU System isolation in case of a pipe break. The high flow setpoint is set high enough to avoid spurious trips from normal operating transients but low enough to ensure an isolation during a pipe break. The new Technical Specification limits will result in a negligible reduction in the margin between the RWCU isolation setpoint and the 4350 gpm flow postulated during a RWCU line break and will avoid spurious isolations.

12. The HPCI and RCIC Steam Line Flow - High Technical Specifications (Table 3.3.2-2, Items 5.a and 6.a) are being changed to account for changes in steam conditions and flows that result from operation at the uprated conditions. The setpoint and allowable value for HPCI Steam Line Flow - High isolation are $\leq$ 387 inches H_2O and $\leq$ 399 inches H_2O respectively. The setpoint and allowable value for the RCIC Steam Line Δ Pressure - High isolation are $\leq$ 138 inches H_2O and $\leq$ 143 inches H_2O respectively.

Evaluation

The bases for these setpoints are contained in the General Electric Design Specification Data Sheets for the HPCI and RCIC systems. The Design Specification Data Sheets specify that the setpoint and allowable value be set so that the isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. GE has historically seen start-up transients as high as 272% of normal steam flow. Setting the isolation above this value prevents spurious isolations and insures availability of the system and its safety function. Setting the isolation at less than or equal to 300% of normal steam flow insures that the isolation will occur if a steam line should rupture.

The original setpoints were calculated using information obtained during the Susquehanna start-up program. The revised setpoints and allowable values were calculated using the same start-up data and adjusted for uprate conditions in accordance with additional guidance provided in GE Service Information Letter (SIL) No. 475, Revision 2, NEDC-31336, "General Electric Setpoint Methodology," and GE Letter SPU-9378, "HPCI and RCIC Steam Line Break Detection Setpoints".

13. Footnote "**" to Technical Specification Table 4.3.2.1-1 is being modified to delete the reference to reactor pressure.

Evaluation

The original purpose of Footnote "**" to Technical Specification Table 4.3.2.1-1 was to describe the functioning of the permissive circuitry that allowed the MSIV low condenser pressure isolation to be bypassed. The original circuitry required the Mode Switch not be in Run and the Turbine Stop Valves closed, and reactor pressure to be above setpoint. In the start-up phase of the Susquehanna Units, GE deleted the reactor pressure setpoint input to the bypass circuitry. Therefore, this change is being made to make the footnote conform to the installed configuration. The revised footnote is the same as found in the BWR/4 Standard Technical Specifications (NUREG 1433).

14. The Rod Block Monitor (RBM) flow biased rod blocks are being changed as follows:

    a. Technical Specification Table 3.3.6-2, Item 1.a is revised to read trip setpoint and allowable values of $\leq 0.63\ W + 41\%$ and $\leq 0.63\ W + 43\%$ respectively.

    b. Technical Specification 3.4.1.1.2.a.5.a is being revised to read trip setpoint and allowable values of $\leq 0.63\ W + 35\%$ and $\leq 0.63\ W + 37\%$ respectively.

Evaluation

These Technical Specification changes made because of power uprate do not represent a change from current limits. That is, an actual change is made to the Technical Specifications to reflect the rescaling made necessary by the re-definition of rated thermal power on those Technical Specifications.

The RBM flow biased rod blocks are used in the Rod Withdrawal Error (RWE) analysis. In order to maintain Critical Power Ratio (CPR) margins similar to previous Susquehanna cycles, the flow biased rod blocks were not appreciably changed in terms of megawatts thermal. Therefore, the rescaling of the RBM flow biased rod block to reflect the re-definition of Rated Thermal Power maintains the same level of protection as previously provided.

15. The APRM rod block upscale value listed in Table 3.3.6-2, Items 2.a is changed to add a high flow clamp setpoint at 108% with a high flow clamped allowable value at 111%.

Evaluation

The addition of the high flow clamp to the flow biased APRM rod block function maintains the normal margins between the rod block and the scram power levels in the increased core flow regions. When the reactor core flow is greater than 100 million lbm/hr, the APRM clamp provides an alarm to help the operator avoid scrams while operating in the increased core flow (ICF) region.

16. The Reactor Coolant System recirculation flow upscale rod block setpoint and allowable value, contained in Technical Specification Table 3.3.6-2, Item 6.a, are increased to 114/125 divisions of full scale and 117/125 divisions of full scale respectively from 108/125 divisions and 111/125 divisions.

Evaluation

The Reactor Coolant System recirculation flow upscale rod block setpoint and allowable value are being increased to allow operation in the ICF region. The 114/125 divisions setpoint and 117/125 divisions allowable value, specified by General Electric, are based on BWR operating history.

The purpose of the Reactor Coolant System recirculation flow upscale rod block is to prevent rod movement when an abnormally high increase in reactor recirculation flow exists. An increase in reactor recirculation flow causes an increase in neutron flux that results in an increase in reactor power. However, this increase in neutron flux is monitored by the Neutron Monitoring System that can provide a rod block. No design basis accident or transient analysis takes credit for rod block signals initiated by the Reactor Coolant Recirculation System.

17. The reactor recirculation pump MG set scoop tube electrical and mechanical overspeed stop setpoints in Technical Specification 4.4.1.1.1.2 and 4.4.1.1.2.5 are being increased to a core flow of 109.5 million lbm/hr and 110.5 million lbm/hr, respectively.

Evaluation

The reactor recirculation pump MG set scoop tube stops are being increased to allow operation at core flows in the ICF region of up to 108 million lbm/hr.

The electrical stop is maintained above the maximum operating core flow and below the mechanical stop. The 109.5 million lbm/hr electrical stop setpoint, specified by General Electric, is based on BWR operating history. The electrical stop is a system design feature and is not used in any safety analyses.

The 110.5 million lbm/hr mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. The 110.5 million lbm/hr mechanical stop setpoint, specified by General Electric, is also based on BWR operating history. The cycle specific analyses, performed for power uprate, used the 110.5 million lbm/hr mechanical stop setpoint.

18. Technical Specification Figure 3.4.1.1.1-1 is redrawn to reflect the new definition of Rated Thermal Power to retain the same stability operating restrictions in terms of megawatts thermal as were previously described by this graph. Because of the incorporation of ICF, the 100% and 80% rod line remain approximately the same. The word "STABILITY" is being added to the figure title to provide clarification with regard to the figure's purpose.

Evaluation

The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided by these Technical Specifications. This position is consistent with Reference 2, Section 5.3.3 and with Reference 3, Section 3.2.

19. Technical Specification 3.4.1.1.2.5 is renumbered to 3.4.1.1.2.6. A new Technical Specification 3.4.1.1.2.5 is being added to specify a 0.70 LHGR multiplier be added to Specification 3.2.4 when in single recirculation loop operation.

Evaluation

Operation with one recirculation loop out of service is allowed, but it is not considered a normal mode of operation. Single loop operation (SLO) is a special operational condition when only one of the two recirculation loops is operable. In this operating condition, the reactor power will be limited to less than 80% of rated by the maximum achievable core flow, which is typically less than 60% of rated core flow. A postulated LOCA occurring in the active recirculation loop during SLO would cause a more rapid coastdown of the recirculation flow than would occur in two loop operation, where one active loop would remain intact. This rapid coastdown causes an earlier boiling transition and deeper penetration of boiling transition into the bundle, which tends to increase the calculated PCT. However, the PCT effects of early boiling transition are substantially offset by the mitigating effect of the lower power level achievable at the start of such an event. The SAFER/GESTR-LOCA analysis results for Susquehanna for SLO and two loop operation are well below 2200° F and are documented in Reference 4.

The ECCS performance for Susquehanna under SLO was evaluated using SAFER/GESTR-LOCA. Calculations for the DBA were performed using both nominal and Appendix K inputs. The SLO SAFER/GESTR-LOCA analysis for the DBA assumes that there is essentially no period of recirculation pump coastdown. Thus, dryout is assumed to occur simultaneously at all axial locations of the hot bundle shortly after initiation of the event. Dryout is assumed to occur in one second for the nominal case and 0.1 second for the Appendix K case. These assumptions are very conservative and provide bounding results for the DBA under SLO.

The two-loop Appendix K break spectrum documented in Reference 4 is representative of SLO because the two-loop spectrum was analyzed assuming a one second dryout time for all axial locations of the hot bundle. As shown by the two-loop break spectrum,the DBA is the limiting case for SLO. In addition, SLO will affect the DBA results more than the smaller breaks. With breaks smaller than the DBA, there is a longer period of nucleate and/or film boiling prior to fuel uncovery to remove the fuel stored energy.

An LHGR reduction (multiplier) of 0.70 will be imposed when the plant is in SLO. As shown in Table 5-6 of Reference 4, the SLO results are less limiting (i.e., lower PCT's) than the results for the two loop DBA LOCA.

Thus, the licensing PCT is based appropriately on two loop operation rather than SLO.

20. Footnote "****" to Technical Specification 4.4.1.1.2.3 is being changed to reference the power uprate startup test program.

Evaluation

Footnote "****" provided a mechanism for changing the power limits specified in Technical Specification 4.4.1.1.2.3 if the results of the initial startup test program determined that it was necessary. The footnote is being modified to allow operation at uprated power with the present power limits. Should the power uprate startup test program determine a need to change the power limits they will be submitted to the Commission within 90 days as required by the revised footnote. This is consistent with the original BWR startup test program philosophy.

21. The Safety/Relief valve Technical Specification 3.4.2 is changed by reducing the number of setpoint groups from 5 to 3. Two valves will be set at 1175 psig $\pm$ 1%, 6 will be set at 1195 psig $\pm$ 1%, and 8 will be set at 1205 psig $\pm$ 1%. Also, the required number of Operable safety valves has been increased from 10 to 12.

Evaluation

The Safety /Relief valves (SRV's) are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. By rearranging the valve groupings, the lowest SRV setpoint is increased by 29 psig. The highest existing SRV setpoint (1205 psig $\pm$ 1%) will not change after power uprate. This rearrangement of setpoints ensures that adequate differences are maintained so that the increase in dome pressure during normal operation does not result in an

increase in the number of SRV actuations. Analytical values of 103% of the nominal setpoints were used in the transient overpressure analyses along with the assumption that the 4 lowest SRV's will be out of service. The Susquehanna Unit 2 cycle 7 specific overpressure analysis has confirmed that the peak RPV pressure remains below the 1375 psig ASME limit.

The adequacy of BWR safety relief valves to operate at uprated temperatures and pressures has been evaluated generically in Section 4.6 of Reference 3. The reactor operating pressure and temperature increase of less than 40 psi and 5 °F respectively used in that evaluation, bound the uprated operating conditions for Susquehanna.

The effect of revised SRV pressure setpoints on fluid transient forces and piping response was evaluated in detail for piping inside the drywell and wetwell. The affected SRV Discharge Lines (SRVDL's) were evaluated for these revised transient loads in addition to the normal operating design loads. Piping stresses were calculated and compared against the Code of Record allowables using original design basis load combinations. The pipe supports on the SRVDL's were evaluated for these increased loads by checking against the margins available in the existing design or by detailed calculations to the original Code of Record requirements. The piping in the wetwell was also evaluated for fatigue as was done in the original plant design. The qualification of the in-line components such as vacuum breakers, diaphragm slab flued head, quenchers, quencher support base plate and 3-way restraint, etc. was performed to the same methodology as was used in the original design.

The impact of power uprate on the Susquehanna units containment dynamic loads due to a SRV discharge has been evaluated as discussed above. As discussed in Section 4.1.2 of Reference 1, the vent thrust loads with power uprate have been evaluated to be less than the loads used in the containment analysis. The effects of power uprate on SRV air-clearing, discharge line, pool pressure boundary and submerged structure drag loads are also discussed in Section 4.1.2 of Reference 1, which concludes that the small increase in the effective pressure setpoint is well within the margin in the SRV loads defined in the Mark II Containment Program. Therefore, the safety/relief valve changes in conjunction with power uprate do not impact the Susquehanna SRV load definitions used in the containment analysis.

22. Technical Specification 3.4.3.2.d is revised to indicate that the 1 gpm leakage rate applies at the uprated maximum allowable pressure of 1035 psig.

Evaluation

The steam dome pressure for leakage is being increased by 35 psig to 1035 psig (reactor design pressure). This pressure is chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Increasing the leakage rate pressure to 1035 psig is consistent with the expected uprated operating pressure.

23. The reactor steam dome pressure limits in Technical Specification 3.4.6.2 and Technical Specification 4.4.6.2 are changed to 1050 psig.

Evaluation

Operating pressure for uprated power is increased by a minimum amount necessary to assure that satisfactory reactor pressure control is maintained. The operating pressure is chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1032 psig is expected and is being assumed in the transient analyses. The 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram for trip avoidance. This limit is the initial pressure value used in the overpressure protection safety analysis for power uprate, for which all licensing criteria have been met.

24. Technical Specification 4.5.1.b.3 is being revised to permit a test line pressure for the flow surveillance of ≥ 1140 psig at nominal reactor operating conditions.

Evaluation

Currently, the HPCI pump test acceptance criteria discharge pressure is ≥ 1266 psig. This is based, in part, on the lowest SRV setpoint of 1146 psig plus a 1% tolerance and line flow losses. For this test, the HPCI turbine is supplied with steam at the nominal operating reactor pressure of 920 +140/-20 psig. Therefore, the test requires the HPCI pump/turbine to produce an output that exceeds that which would be commensurate with the input conditions. Stated differently, HPCI would be required to develop a pump discharge pressure associated with a steam dome pressure of 1187 psig (1175 ± 1% psig), while being supplied with a steam dome pressure as low as 900 psig.

The purpose of this specification is to demonstrate that the system is capable of producing the required flow at the required pressure. The concern with this approach is that while it demonstrates the required capability by achieving the actual Technical Specification value, it requires the pump/turbine to "over perform." It also reduces the margin available to compensate for normal wear and tear that occurs and is monitored under the ASME Section XI Pump and Valve Test Program. Power uprate will be further increasing the demand because of the increase in reactor steam dome pressure.

The intent of Surveillance 4.5.1.b.3 is to demonstrate that the HPCI System will produce its design flow rate at an expected reactor pressure during a LOCA. Confirmation of the capability to achieve the required flow and pressure can be satisfactorily demonstrated without requiring the pump/turbine to "over perform". This can be done by producing the nominal operating design pressure from the pump with steam supplied to the turbine at nominal reactor operating pressure. From these conditions extrapolation via pump affinity laws will show the pump discharge pressure

that would be developed at emergency reactor operating conditions (i.e. lowest SRV setpoint). This value could then be compared to the calculated required value required for assuring adequate core cooling in both SSES specific and generic evaluations. The HPCI System has been evaluated and shown to be capable of achieving the required pressure and flow conditions for power uprate.

Applying the method of pump affinity laws, the new Technical Specification pump discharge pressure would become ≥ 1140 psig. This value is determined based on the maximum allowable test steam dome pressure of 920 +140 = 1060 psig, plus head losses. Through the use of pump affinity laws it has been shown by calculation that achieving a value of 1140 psig at nominal reactor operating conditions will produce the required flow and pressure during emergency conditions.

Therefore, the Technical Specification HPCI pump discharge pressure at power uprate conditions is changed to ≥ 1140 psig.

25. Bases Table B 3/4.4.6-1 is being changed to provide revised Heat Slab identification number, copper content, nickel content, Starting RT_{NDT}(°F), ΔRT_{NDT}(°F) and Max. RT_{NDT}(°F) for the limiting beltline plate. These changes reflect the calculated increase in Neutron Fluence and the results of in-vessel surveillance coupon sample analysis contained in GE-NE-523-107-0893, Reference 6. An editorial reference to R.G. 1.99, Revision 2 is being added to NOTE "*". This is being done to clarify the bases for the note.

Bases Figure B 3/4.4.6-1 is being revised to reflect the calculated Fast Neutron Fluence of 7.8 x 10^{+17} n/cm^2 based on the uprated power level at the lower shell plate per Reference 6. The units for Neutron Flux are being changed from "n/cm^2 X 10^{-17}" to "n/cm^2 X 10^{+17}". This corrects a typographical error.

Evaluation

Power uprate results in higher neutron flux that increases fluence over the plant's life. The estimated fluence is increased conservatively over the FSAR peak end of life value. The effect of this high fluence was evaluated against the requirement of 10CFR50 Appendix G. Those evaluations indicate that:

a. Upper shelf energy remains greater than 50 ft.-lb for the design life of the vessel.
b. The beltline material reference temperature nil-ductility transition (RT_{NDT}) remains well within the 200°F regulatory requirement.
c. The 32 EFPY shift is slightly increased which requires a change in the adjusted reference temperature that is RT_{NDT} plus the shift.

The actual effect of neutron fluence is determined based on the results of in-vessel surveillance coupon sample evaluations. Since the vessel is still in compliance with regulatory requirements, power uprate will not have a significant effect on reactor vessel fracture toughness, nor does it change the Unit 2 original P/T curve operational parameters because the non-beltline limits are still limiting.

26. Bases Sections 3/4.5.1 and 3/4.5.2 are changed to indicate the maximum pressure required value of 1187 psig for HPCI injection to the reactor pressure vessel.

Evaluation

The revision to this basis is required since the lowest SRV setpoint value (with a 1% tolerance) is being raised as a result of power uprate. This value is used to establish the nominal operating discharge pressure identified in the revised technical specification for which this basis applies.

27. Bases Section 3/4.6.2 is changed to indicate that a reactor coolant system blowdown is initiated from a reactor pressure of 1053 psia in lieu of 1055 psig.

Evaluation

The containment accident analysis performed for both the original accident analysis and the power uprate accident analysis, assume that the reactor pressure is the value specified in the valves wide open reactor heat balance. The original Susquehanna reactor heat balances specify a reactor pressure at the valves wide open condition of 1055 psig. The uprated reactor heat balance (Figure 1-2 of Reference 1) at the valves wide open condition specifies a reactor pressure of 1053 psia. This change is being made to keep Bases Section 3/4.6.2 current with the power uprate analyses design basis.

28. Design Features Section 5.4.2 is changed to show that the nominal T_{ave} has changed from 528°F to 532°F.

Evaluation

The change to the nominal T_{ave} in Design Features Section 5.4.2 is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure.

29. Design Features Table 5.7.1-1 is being modified to specify that the upper limit for a heatup cycle is 551°F in lieu of 546°F.

Evaluation

The change in the heatup upper limit is being made to reflect the increased reactor temperature that results from the 30 psi increase in reactor design pressure.

30. Administrative Control Section 6.9.3.2 describes and lists topical reports that are used to determine core operating limits. Topical reports 15 through 19 are LOCA methodology reports and are being deleted. These reports describe Siemens LOCA methodology. As stated in Reference 1, The GE SAFER/GESTR LOCA methodology is being used for uprated cycles. In addition, other minor methodology changes were made for power uprate transient analysis. GE topical report NEDC-32071P, PP&L topical report NE-092-001 (when approved), and the NRC Safety Evaluation Report on the PP&L power uprate licensing topical (when issued) are proposed to be added as Topical Reports No. 15, 16, and 17, respectively.

Evaluation

The Siemens topical reports are no longer used to determine core operating limits. The appropriate NRC approved Topical Reports used in power uprate cycles to determine core operating limits are being added to Design Feature 6.9.3.2 to keep this Section current with the power uprate analyses design bases.

NO SIGNIFICANT HAZARDS CONSIDERATIONS:

The following three questions are addressed for each of the proposed Technical Specification Changes:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

3. Does the proposed change involve a significant reduction in a margin of safety?

Section 1.0, Definitions, Definition 1.33, Rated Thermal Power

This change redefines Rated Thermal Power as 3441 megawatts thermal.

1. **No.** Neither the probability (frequency of occurrence) nor consequences of any accident previously evaluated is significantly affected by the increased power level because the design and regulatory criteria established for plant equipment remain imposed for the uprated power level. The PP&L assessment to increase the rated thermal power level at Susquehanna SES Unit 2, followed the guidelines of NEDC-31879P ("Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," G.E. Nuclear Energy, June 1991). NEDC-31879P provides generic licensing criteria, methodology, and a defined scope of analytical and equipment review to be performed to demonstrate the ability to operate safely at the uprated power level which have been approved by the NRC. NE-092-001 ("Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992) provides the description of the power uprate licensing analysis methodology and the results of the evaluations performed to support the proposed uprated power operation consistent with the methodology presented in NEDC-31879P. NE-092-001 provides a

description of the power uprate licensing analysis methodology which will be used to determine cycle specific thermal limits for Unit 2, Cycle 7 and future cycles and concludes that an uprated power level of 3441 megawatts thermal can be achieved without significant effect on equipment or safety analyses.

2. **No.** The methodology and results described above do not indicate that a possibility for a new or different kind of accident from any previously evaluated has been created by uprated operation.

3. **No.** Based on the response to Question 1 above, the methodology and results do not indicate a significant reduction in a margin of safety.

<u>Section 2.1, Safety Limits</u>

The reference to "rated core flow" in Technical Specification 2.1.1 and 2.1.2 has been replaced with a reference to actual core flow. The references to "rated core flow" have been deleted to avoid confusion since allowable core flow is being increased by 8%. 10 Mlbm/hr is being used in these specifications to be consistent with other similar Technical Specification changes (Technical Specifications 3.2.2, 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3 and Figure 3.4.1.1.1-1).

1. **No.** The probability and consequences of accidents previously evaluated are not affected by this change. The basis for Technical Specification 2.1.1 is that boiling transition will not occur in bundles if core power is less than 25% of rated thermal power, regardless of pressure or core flow. Consequently, the specification of less than 10% rated core flow is not crucial to the basis and, thus, the use of 10 Mlbm/hr. is acceptable and has no effect on the probability or consequences of a previously evaluated accident.

   For Technical Specification 2.1.2, the XN-3 critical power correlation is valid for pressure $\geq$ 580 psig and bundle flow $\geq$ 0.25 Mlbm/hr-ft^2. As stated in the basis for Technical Specification 2.1.1, if vessel downcomer water level is above TAF, and core power > 25%, bundle flows for potentially limiting bundles will be > 0.25 Mlbm/hr-ft^2 due to natural circulation. In addition, Technical Specification 3.4.1.1.1 requires at least one (1) recirculation loop in operation to run in Condition 2, which would produce a core flow in excess of 30 Mlbm/hr. Therefore, core flows below about 30 Mlbm/hr-ft^2 are prohibited when the reactor is at power. Thus, the change from "10%" to "10 million lbm/hr" is acceptable and has no effect on the probability or consequences of a previously evaluated accident.

2. **No.** The basis for Technical Specification 2.1.1 is that boiling transition will not occur in bundles if core power is less than 25% of rated thermal power, regardless of pressure or core flow. The proposed change is not crucial to this basis. The XN-3 critical power correlation is valid for pressures $\geq$580 psig and bundle flow $\geq$0.25 Mlbm/hr-ft^2. The specification is based upon vessel downcomer water level being above TAF and core power >25% which yields a bundle flow for potentially limiting bundles >0.25 Mlbm/hr-ft^2 due to natural circulation. Based on Technical Specification 3.4.1.1.1, core flows below about 30 Mlbm/hr-ft^2 are prohibited when

the reactor is at power. Therefore, the change to a limit of 10 Mlbm/hr is acceptable and does not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. **No.** As explained above, the margin of safety has not been reduced.

Table 2.2.1-1 (Items 2.a, 2.b, and 2.c) and Specifications 3.2.2, 3.4.1.1.2.a.2, 3.4.1.1.2.a.3, 3.4.1.1.2.a.5.b and 3.3.6-2 (Item 2.a.1, 2.c, and 2.d), APRM Flow Biased Setpoints and Allowable Values

Although the equation for determining these setpoints does not change as a result of the power uprate, because the setpoints in these technical specifications are referenced to rated thermal power, the current limits do change in that the top portion of the operating map (power vs. reactor flow) is raised by 4.5%.

1. **No.** The safety analyses contained in NE-092-001 evaluated operation at both uprated power with 4.5% higher rod lines and increased core flow. In addition, General Electric Co. has analyzed and received generic approval for their BWR/4 product line operation in the Maximum Extended Operating Domain (MEOD). Operation at the 4.5% higher rod lines is bounded by the MEOD analysis. Additional justification for this small increase in the power flow operating range is contained in Section C.2.3 of NEDC-31984P.

   Cycle specific reload analyses will evaluate operation at the increased power vs. flow conditions (100% uprated power vs. 87% core flow to 100% uprate power vs. 108% core flow). These analyses will ensure that the limits established in the Core Operating Limits Report are applicable to rated power operation from 87% to 108% core flow.

   Based on the above analyses, increasing the current limits do not represent a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The analyses described above in response to Question 1 do not indicate that a possibility for a new or different kind of accident from any previously evaluated has been created by the proposed change.

3. **No.** Based on the response to Question 1 above, the proposed change does not result in a reduction in the margin of safety.

Table 2.2.1-1, Item 3, Reactor Steam Dome Pressure - High Scram

The reactor steam dome pressure-high scram trip setpoint and allowable values are being changed to ≤1087 psig and ≤1093 psig respectively.

1. **No.** This scram function is designed to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor

vessel maximum operating pressure and the specified analytical limit is used in the transient analyses. The analytical limit of 1105 psig is used in the uprated transient analyses. The results of the overpressure protection analysis indicate that the peak pressure will remain below the 1375 psig ASME limit which meets plant licensing requirements. In accordance with the methodology described in NE-092-001, transient analyses will be performed using the analytic limit and the results will be incorporated into the Core Operating Limits Report. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The purpose of this scram function is to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor vessel maximum operating pressure and the specified analytical limit is used in the transient analysis. 1105 psig is being used as the analytical limit in the uprated transient analysis. The results of the overpressure protection analysis indicate peak pressure will remain below the ASME limit of 1375 psig which satisfies plant licensing requirements. Based upon that result, it is concluded that the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. **No.** The results of the overpressure protection analysis indicate peak pressure will remain below the 1375 psig licensing limit, therefore, it is concluded that the proposed change does not result in a significant reduction in a margin of safety.

<u>Specification 4.1.5.c, Standby Liquid Control System</u>

This specification has been revised to require SLC pumps to develop a discharge pressure of ≥1224 psig.

1. **No.** The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. The SLC pump test discharge pressure acceptance criteria are based on the lowest relief valve setpoint. The lowest setpoint is being increased by 30 psi (to 1106) due to power uprate. Operating with increased core flow will result in additional friction losses through the core and a slightly larger core differential pressure (approximately 4 psi). Therefore, increasing the SLC pump test discharge pressure acceptance criteria ensures the capability of SLC injection. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. Therefore, the proposed change does not result in a new or different kind of accident from any previously evaluated.

3. **No.** The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. As stated in the response to question 1 above, the SLC pump discharge pressure acceptance criteria are

based upon the lowest relief valve setpoint. The lowest setpoint is being increased by 30 psi. As the SLC pumps are positive displacement pumps, the uprate will not adversely affect the performance of the pumps to achieve proper injection. Based on above, the proposed change does not result in a significant reduction in a margin of safety.

Specifications 3.2.2, 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3 and Figure 3.4.1.1.1-1, Rated Core Flow References

Technical Specification 3.2.2 contains the definition of "W" for the flow biased APRM scram equation. The word "rated" is being deleted from the definition of "W" since rated core flow is being increased. The definition of "W" is not altered. The change is being made for editorial purposes.

Technical Specifications 4.4.1.1.1.1.2, 4.1.1.1.2.5, 3.4.1.3, and Figure 3.4.1.1.1-1 specify performance requirements and limits for the Reactor Recirculation System. These specifications are referenced to the current rated core flow. The references to "rated core flow" are being replaced with actual equivalent core flows. The specifications are equivalent and unchanged. This change is being made for editorial purposes to avoid confusion since rated core flow is being increased. These changes are also consistent with the changes made in Section 2.1.

1. **No.** The proposed changes are editorial and do not effect the probability or consequences of an accident previously evaluated.

2. **No.** The proposed changes are editorial and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **No.** The proposed changes are editorial and do not involve a significant reduction in a margin of safety.

Specification Table 3.3.1-1, Note (j) and Action 6, Reactor Protection System Instrumentation, and Table 3.3.4.2-1, Note b, End-of-Cycle Recirculation Pump Trip System Instrumentation

The turbine first stage pressure scram bypass at 30% power in Technical Specification Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) is revised to indicate that the uprated equivalent allowable value of first stage turbine pressure is 136 psig. This value ensures that the analytical limit of 147.7 psig, which represented 30% rated thermal power, is not exceeded.

As currently written Note (j), Note (b) and Table 3.3.1-1, ACTION 6 are unclear and could be misinterpreted. They apply only when RPS scram functions and End-of-Cycle Recirculation Pump Trip on turbine main stop valves closure or control valve fast closure are not automatically bypassed. ACTION 6 provides no guidance in the event the bypass fails to lift when thermal power is above 30%. In the worst case, the action statement could be interpreted literally to allow full power operation with the RPS function still bypassed. Such operation would violate the licensing basis analysis for the MCPR operating limit (for the Generator Load Rejection Without Bypass transient), which takes credit for operation of the anticipatory scram on control valve fast closure at greater than 30% of rated thermal power.

1. **No.** The revisions to Table 3.3.1-1, ACTION 6, Table 3.3.1-1, Note (j), and Table 3.3.4-1 Note (b) clarify the current requirements; they do not change their intent.

   FSAR Chapter 15 transient analyses and reload licensing analyses take credit for operation of the anticipatory scram function on turbine stop valve closure and control valve fast closure for power levels greater than 30% of rated thermal power. The proposed revision to Table 3.3.1-1, ACTION 6 provides better assurance of the availability of the anticipatory scram function, since the current specifications could be interpreted literally to allow full power operation with the RPS function bypassed.

   The proposed revision to Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) does not change the operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power. The turbine first stage pressure switches will still be calibrated in the same manner, and, by procedure, the reactor operator will not exceed 30% power if the trip bypass annunciator does not clear.

   The setpoints for the RPS and EOC-RPT bypass functions were selected to allow sufficient operating margin to avoid scrams during low power turbine generator trips. As discussed in NEDC-31894P, Section F4.2(c) and in Section 5.1.2.8 of NEDC 31948P, this small absolute setpoint increase maintains the safety basis for the setpoint.

   Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The changes proposed are clarifications and do not change specification intent. The proposed change to Table 3.3.1-1, Action 6 provides better assurance of the availability of the anticipatory scram function as the specification could currently be interpreted to allow full power operation with the RPS function bypassed. The proposed changes to Table 3.3.1-1, Note (j) and Table 3.3.4-1, Note (b) do not change the operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power. Therefore, the possibility for a new or different kind of accident is not created.

3. **No.** The proposed changes are clarification and do not change intent. Operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power is not changed. Therefore, there is no reduction in the margin of safety.

Specification Table 3.3.2-2, Item 3.d, Main Steam Line Flow Differential Pressure Setpoint

The main steam line flow high differential pressure setpoint and allowable value are revised to read trip setpoint and allowable values of 113 psid and 121 psid respectively. Footnote "**" was added to Table 3.3.2-2 to indicate that these values will be confirmed during the power uprate start-up testing. If revisions to the setpoint and allowable value are required, they will be forwarded to the Commission for approval within 90 days of completion of the test program.

1. **No.** The main steam line flow high differential pressure setpoint changes reflect the redefinition of rated main steam line flow that occurs with power uprate. The allowable value is maintained at the same percentage of rated steam flow as the differential pressure changes due to the increased uprate steam flow. The analytical limit of 140% of uprated steam flow is maintained for the uprated analyses. The relationship between the allowable value and the analytical limit was retained to ensure that a trip avoidance margin is maintained for the normal plant testing of MSIV's and turbine stop valves. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which satisfies the original intent of the design. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for the main steam line break accident which satisfies the original intent of the design and, therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **No.** The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which satisfies the original intent of the design and, therefore, does not involve a significant reduction in a margin of safety.

Specification Table 3.3.2-2, Item 4.f1, Isolation Actuation Instrumentation Setpoints

The RWCU system flow-high isolation trip setpoint and allowable value are being changed. System flow is being increased by 10% to maintain reactor coolant water chemistry at a level equal to pre uprate levels. The isolation setpoint change is being made to adequately maintain operating margin between normal process values and the isolation setpoints.

1. **No.** The basis for the RWCU flow-high isolation is to ensure a RWCU System isolation in case of a pipe break. The high flow setpoint is set high enough to avoid spurious trips from normal operating transients but low enough to ensure an isolation during a pipe break. The proposed Technical Specification limits will result in a negligible reduction in the margin between the RWCU isolation setpoint and the 4350 gpm flow postulated during a RWCU line break and will avoid spurious isolations. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** As stated above, the proposed change will result in only a negligible reduction in the margin between the RWCU isolation setpoint while avoiding spurious isolation. Therefore, this change maintains the original design intent and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **No.** See 1. above.

Specification Table 3.3.2-2, Items 5.a and 6.1, Isolation Actuation Instrumentation Setpoints

The HPCI and RCIC Steam Line Flow-High Technical Specifications are being changed to account for changes in steam conditions and flows that result from operation at the uprated conditions. The setpoint and allowable value for HPCI Steam Line Flow-High isolation are ≤ 387 inches H_2O and ≤ 399 inches H_2O respectively. The setpoint and allowable value for the RCIC Steam Line Δ Pressure-High isolation are ≤ 138 inches H_2O and ≤ 143 inches H_2O respectively.

1. **No.** The bases for these setpoints are contained in the General Electric Design Specification Data Sheets for the HPCI and RCIC systems. The Design Specification Data Sheets specify that the setpoint and allowable value be set so that the isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. General Electric has historically seen start-up transients as high as 272% of normal steam flow. Setting the isolation above this value prevents spurious isolations and ensures availability of the system and its safety function. Setting the isolation at less than or equal to 300% of normal flow insures that the isolation will occur if a steam line should rupture.

   The original setpoints were calculated using information obtained during the Susquehanna start-up program. The revised setpoints and allowable values were calculated using the same start-up data and adjusted for uprate conditions in accordance with additional guidance provided in General Electric Information Letter (SIL) No. 475, Revision 2, NEDC-31336, "General Electric Setpoint Methodology," and GE Letter SPU-9378, "HPCI and RCIC Steam Line Break Detection Setpoints".

   Based on the above approach, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The setpoint and allowable value are set so that isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. Setting the isolation at less than or equal to 300% of normal flow ensures that the isolation will occur if a steam line rupture should occur. Therefore, no new events are postulated as a result of this change.

3. **No.** The proposed change does not involve a significant reduction in a margin of safety as the setpoint and allowable value are set to isolate at greater than 272% normal steam flow and less than 300% steam flow which are the setpoints contained in the General Electric Design Specification Data Sheets for the HPCI and RCIC systems.

Specification Table 4.3.2.1-1, footnote "**"

The footnote is being changed to delete reference to reactor pressure.

1. **No.** The original purpose of Footnote "**" to Technical Specification Table 4.3.2.1-1 was to describe the functioning of the permissive circuitry that allowed the MSIV low condenser pressure isolation to be bypassed. The original circuitry required the Mode Switch not be in Run, the Turbine Stop Valves closed, and reactor pressure to be above setpoint. In the

start-up phase of the Susquehanna Units, General Electric deleted the reactor pressure setpoint input to the bypass circuitry. Therefore, this change is being made to make the footnote conform to the installed configuration. The revised footnote is the same as found in the BWR/4 Standard Technical Specifications (NUREG 1433). This change is editorial in nature and, therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** Based on the response to Question 1 above, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **No.** Based on the response to Question 1 above, the proposed change does not involve a significant reduction in a margin of safety.

Specification Table 3.3.6-2, Item 1.a and Specification 3.4.1.1.2.a.5.a, Rod Block Monitor Flow Biased Rod Blocks

The Rod Block Monitor (RBM) flow biased rod blocks are being changed as follows:

a. Technical Specification Table 3.3.6-2, Item 1.a is revised to read trip setpoint and allowable values of $\leq 0.63\ W + 41\%$ and $\leq 0.63\ W + 43\%$ respectively.

b. Technical Specification 3.4.1.1.2.a.5.a is being revised to read trip setpoint and allowable values of $\leq 0.63\ W + 35\%$ and $\leq 0.63\ W + 37\%$ respectively.

1. **No.** These Technical Specification changes do not represent a change from current limits. The change reflects the rescaling made necessary by the re-definition of rated thermal power.

   The RBM flow biased rod blocks are used in the Rod Withdrawal Error (RWE) analysis. In order to maintain Critical Power Ratio (CPR) margins similar to previous Susquehanna cycles, the flow biased rod blocks were changed in terms of megawatts thermal but the change was not appreciable. The rescaling of the RBM flow biased rod block to reflect the re-definition of Rated Thermal Power maintains the same level of protection as previously provided. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** These changes do not represent a change from current limits but are rather a rescaling made necessary by the re-definition of rated thermal power.

3. **No.** These changes do not represent a change from current limits but are rather a rescaling made necessary by the re-definition of rated thermal power. The rescaling of the RBM flow biased rod block maintains the same level of protection as previously provided.

Specification Table 3.3.6-2, Item 2.a, Control Rod Block Instrumentation Setpoints

The APRM rod block upscale value has been changed to add a high flow clamp setpoint at 108% with a high flow clamped allowable value at 111%.

1. **No.** The addition of the high flow clamp to the flow biased APRM rod block function maintains the normal margins between the rod block and the scram power levels in the increased core flow regions. When the reactor core flow is greater than 100 million lbm/hr, the APRM clamp provides an alarm to help the operator avoid scrams while operating in the ICF region. This action does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** The changes maintain the normal margins between the rod block and the scram power levels in ICF regions. The clamp provides an alarm to avoid scrams in the ICF region.

3. **No.** The changes maintain the normal margins between the rod block and the scram power levels.

Specification Table 3.3.6-2, Item 6.a, Reactor Coolant System Recirculation Flow Upscale Rod Block Setpoint and Allowable Value Change

The reactor coolant system recirculation flow upscale rod block setpoint and allowable value are being increased to 114/125 divisions of full scale and 117/125 divisions of full scale respectively.

1. **No.** The Reactor Coolant System recirculation flow upscale rod block setpoint and allowable value are being increased to allow operation in the ICF region. The 114/125 divisions setpoint and 117/125 divisions allowable value, specified by General Electric, are based on BWR operating history.

   The purpose of the Reactor Coolant System recirculation flow upscale rod block is to prevent rod movement when an abnormally high increase in reactor recirculation flow exists. An increase in reactor recirculation flow causes an increase in neutron flux that results in an increase in reactor power. However, this increase in neutron flux is monitored by the Neutron Monitoring System that can provide a rod block. No design basis accident or transient analysis takes credit for rod block signals initiated by the Reactor Coolant Recirculation System. Therefore, this change does not increase the probability or consequences of an accident previously evaluated.

2. **No.** Rod block signal initiation by the Reactor Coolant Recirculation System is not taken credit for in the mitigation of a design basis accident or in any transient analysis.

3. **No.** Rod block signal initiation by the Reactor Coolant Recirculation System is not taken credit for in any transient analysis or in the mitigation of a design basis accident.

Specification 4.4.1.1.1.2 and 4.4.1.1.2.5 Reactor Coolant System

The reactor recirculation pump motor generator set scoop tube electrical and mechanical overspeed stop setpoints are being increased to a core flow of 109.5 million lbm/hr. and 110.5 million lbm/hr., respectively.

1. **No.** The reactor recirculation pump motor generator set scoop tube stops are being increased to allow operation at core flows in the ICF region of up to 108 million lbm/hr.

   The electrical stop is maintained above the maximum operating core flow and below the mechanical stop. The 109.5 million lbm/hr. electrical stop setpoint, specified by General Electric, is based on BWR operating history. The electrical stop is a system design feature and is not used in any safety analyses.

   The 110.5 million lbm/hr. mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. The 110.5 million lbm/hr. mechanical stop setpoint, specified by General Electric, is also based on BWR operating history. The cycle specific analyses, performed for power uprate, used the 110.5 million lbm/hr. mechanical stop setpoint.

   Based on the above, this change does not involve a significant increase of the probability or consequences of an accident previously evaluated.

2. **No.** Increasing the reactor recirculation motor generator set scoop tube electrical and mechanical overspeed stop setpoints is being done to allow operation at core flows in the ICF region up to 108 Mlbm/hr. The electrical stop setpoint is a design feature and is not used in any safety analysis. The mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. Changing of this setpoint was considered in appropriate transient analyses, and will not create the possibility of a new or different kind of accident from any previously evaluated.

3. **No.** See 1. above. This change does not significantly reduce the margin of safety.

Specification Figure 3.4.1.1.1-1, Thermal Power Restrictions

This figure has been redrawn to reflect the new definition of Rated Thermal Power to retain the same stability operating restrictions in terms of megawatts thermal as were previously described by this graph.

1. **No.** The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided by these Technical Specifications. This position is consistent with NEDC-31894P, Section 5.3.3 and with NEDC-31984P, Section 3.2.

2. **No.** The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are changed to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided and does not create the possibility for a new or different kind of accident.

3. **No.** The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, there is no detrimental effect on the level of protection provided, and therefore no significant decrease in any margin of safety.

Specifications 3.4.1.1.2.5, 3.4.1.1.2.6, Reactor Coolant System, Recirculation Loops - Single Loop Operation

Specification 3.4.1.1.2.5 is being renumbered to 3.4.1.1.2.6. A new specification 3.4.1.1.2.5 is being added to specify that a 0.70 LHGR multiplier has been added to Specification 3.2.4 when in single recirculation loop operation.

1. **No.** Operation with one recirculation loop out of service is allowed, but it is not considered a normal mode of operation. Single loop operation (SLO) is a special operational condition when only one of the two recirculation loops is operable. In this operating condition, the reactor power will be limited to less than 80% of rated by the maximum achievable core flow, which is typically less than 60% of rated core flow. A postulated LOCA occurring in the active recirculation loop during SLO would cause a more rapid coastdown of the recirculation flow than would occur in two loop operation, where one active loop would remain intact. This rapid coastdown causes an earlier boiling transition and deeper penetration of boiling transition into the bundle, which tends to increase the calculated PCT. However, the PCT effects of early boiling transition are substantially offset by the mitigating effect of the lower power level achievable at the start of such an event. The SAFER/GESTR-LOCA analysis results for Susquehanna for SLO and two loop operation are well below 2200° F and are documented in NEDC-32064P-1, Revision 1, "Power Uprate with Increased Core Flow Safety Analysis for Susquehanna 1 and 2", GE Nuclear Energy, July 1993.

   The ECCS performance for Susquehanna under SLO was evaluated using SAFER/GESTR-LOCA. Calculations for the DBA were performed using both nominal and Appendix K inputs. The SLO SAFER/GESTR-LOCA analysis for the DBA assumes that there is essentially no period of recirculation pump coastdown. Thus, dryout is assumed to occur simultaneously at all axial locations of the hot bundle shortly after initiation of the event. Dryout is assumed to occur in one second for the nominal case and 0.1 second for the Appendix K case. These assumptions are very conservative and provide bounding results for the DBA under SLO.

The two-loop Appendix K break spectrum documented in NEDC-32064P-1 is representative of SLO because the two-loop spectrum was analyzed assuming a one second dryout time for all axial locations of the hot bundle. As shown by the two-loop break spectrum, the DBA is the limiting case for SLO. With breaks smaller than the DBA, there is a longer period of nucleate and/or film boiling prior to fuel uncovery to remove the fuel stored energy.

An LHGR reduction (multiplier) of 0.70 will be imposed when the plant is in SLO. As shown in Table 5-6 of NEDC-32064P-1, the SLO results are less limiting (i.e., lower PCT's) than the results for the two loop DBA LOCA.

Thus, the licensing PCT is based appropriately on two loop operation rather than SLO.

2. **No.** The licensing PCT is based upon two loop operation rather than SLO, thus the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **No.** Based on the response to Question 1 above, the proposed change does not involve a significant reduction in a margin of safety.

<u>Specification 4.4.1.1.2.3, Reactor Coolant System</u>

Footnote **** to this Specification is being changed to reference the power uprate startup test program.

1. **No.** This footnote provided a mechanism for changing the power limits specified if the results of the initial startup test program determined that it was necessary. The footnote is being modified to allow operation at uprated power with the present power limits. Should the power uprate startup test program determine a need to change the power limits they will be submitted to the Commission within 90 days as required by the revised footnote. This is consistent with the original BWR startup test program philosophy and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **No.** See 1. above; this change is administrative in nature and does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **No.** See 1. above; this change is administrative in nature and does not involve a significant reduction in a margin of safety.

<u>Specification 3.4.2, Reactor Coolant system, Safety Relief Valves</u>

The safety relief valve specification is being changed to reduce the number of setpoint groups from 5 to 3. Two valves will be set at 1175 psig ± 1%, 6 will be set at 1195 psig ± 1%. Also, the number of Operable safety valves is being increased from 10 to 12.

1. No. This change does not increase the probability of occurrence of an accident previously evaluated as, with one exception, the accidents described in FSAR Sections 5.2.2, 7.2.3, 15.1, 15.2 and 15.3 do not document any cases where the SRV's are designated as the cause or initiator of an accident. The exception is inadvertent safety relief valve opening which results in a decrease in reactor coolant inventory and/or reactor coolant temperature. The revised setpoints and proposed groupings will not increase the probability of occurrence of this type of accident.

   The change does not increase the probability of occurrence of a malfunction of equipment important to safety as previously evaluated in the FSAR. The margin between peak allowable pressure and the maximum safety setpoint is unchanged. The reactor vessel and components were evaluated for the setpoint change to assure continued compliance with the structural requirements of the ASME Code. Analysis was performed on the effects of the setpoint change for the design conditions, the normal and upset conditions and the emergency and faulted conditions. The increasing RPV dome pressure does not affect the design condition and, therefore, stresses remain unchanged.

   The proposed change will also not adversely affect HPCI and RCIC system performance.

   There is no indication that changed setpoints contribute to an increase in probability of SRV malfunction. Reduction in the simmer margin will be compensated for by more stringent leak test requirements during valve refurbishment.

2. No. This change does not involve any hardware changes or changes in system function. Relief and safety setpoints are only slightly increased and the maximum safety setpoint remains unchanged, thus the margin between peak allowable pressure and the setpoint remains unchanged.

3. No. The technical specifications were reviewed for margins of safety applicable to the components and systems affected by the change. Analysis has been performed that demonstrates that reactor pressure will be limited to within ASME Section III allowable values for the worst case upset transient. The margin of safety is inherent in the ASME Section III allowable pressure values.

<u>Specification 3.4.3.2.d, Reactor Coolant System, Operational Leakage</u>

This specification is being revised to indicate that the 1 gpm leakage rate limit currently applicable applies at the uprated maximum allowable pressure of 1035 psig.

1. No. The steam dome pressure for leakage is being increased by 35 psig to 1035 psig (reactor design pressure). This pressure is chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Increasing the leakage rate pressure to 1035 psig is consistent with the expected uprated operating pressure. Increasing the reactor steam dome

pressure has been analyzed and found to be within allowable limits. Maintaining the leakage rate limit at 1 gpm does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. This change does not involve any hardware changes or change in safety function. The reactor steam dome pressure has been analyzed and found to be within allowable limits.

3. No. Maintaining leakage the rate limit at 1 gpm is conservative and does not involve a reduction in the margin of safety.

Specifications 3.4.6.2 and 4.4.6.2, Reactor Coolant System, Reactor Steam Dome

The reactor steam dome pressure limits have been changed to 1050 psig.

1. No. Operating pressure for uprated power is increased by a minimum amount necessary to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1032 psig is expected and is being assumed in the transient analyses. The 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram for trip avoidance. This limit is the initial pressure value used in the overpressure protection safety analysis for power uprate, for which all licensing criteria have been met. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. Based on the response to Question 1. above, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. As described in 1. above, the 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram. This limit is the initial pressure value used in the over pressure protection safety analysis for power uprate, for which all licensing criteria have been met. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Specification 4.5.1.b.3, Emergency Core Cooling Systems

This specification has been revised to permit a test line pressure for the flow surveillance of ≥ 1140 psig at nominal reactor operating conditions.

1. No. Currently, the HPCI pump test acceptance criteria discharge pressure is ≥ 1266 psig. This is based, in part, on the lowest SRV setpoint of 1146 psig plus a 1% tolerance and line flow losses. For this test, the HPCI turbine is supplied with steam at the nominal operating reactor

pressure of 920 +140/-20 psig. Therefore, the test requires the HPCI pump/turbine to produce an output that exceeds that which would be commensurate with the input conditions. Stated differently, HPCI would be required to develop a pump discharge pressure associated with a steam dome pressure of 1187 psig (1175 ± 1% psig), while being supplied with a steam dome pressure as low as 900 psig.

The purpose of this specification is to demonstrate that the system is capable of producing the required flow at the required pressure. The concern with this approach is that while it demonstrates the required capability by achieving the actual Technical Specification value, it requires the pump turbine to "over perform". It also reduces the margin available to compensate for normal wear and tear the occurs and is monitored under the ASME Section XI Pump and Valve Test Program. Power uprate will be further increasing the demand because of the increase in reactor steam dome pressure.

The intent of Surveillance 4.5.1b.3 is to demonstrate that the HPCI System will produce its design flow rate at an expected reactor pressure during a LOCA. Confirmation of the capability to achieve the required flow and pressure can be satisfactorily demonstrated without requiring the pump/turbine to "over perform". This can be done by producing the nominal operating design pressure from the pump with steam supplied to the turbine at nominal reactor operating pressure. From these conditions extrapolation via pump affinity laws will show the pump discharge pressure that would be developed at emergency reactor operation conditions (i.e. lowest SRV setpoint). This value could then be compared to the calculated value required for assuring adequate core cooling in both SSES specific and generic evaluations. The HPCI System has been evaluated and shown to be capable of achieving the required pressure and flow conditions for power uprate.

Applying the method of pump affinity laws, the new Technical Specification pump discharge pressure would become ≥ 1140 psig. This value is determined based on the maximum allowable test steam dome pressure of 920 + 140 = 1060 psig, plus head losses. Through the use of pump affinity laws it has been shown by calculation that achieving a value of 1140 psig at nominal reactor operating conditions will produce the required flow and pressure during emergency conditions.

Therefore, the Technical Specification HPCI pump discharge pressure at power uprate conditions is changed to ≥ 1140 psig.

2. **No.** The methodology and the supporting change described above in the response to Question 1 above do not alter the function nor the operation of the HPCI system. Therefore, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **No.** The methodology and the supporting change described above in response to Question 1 do not involve a significant reduction in a margin of safety.

Specification 5.4.2, Design Features, Reactor Coolant System, Volume

This specification is being changed to show that the nominal T_{ave} is being changed from 528°F to 532°F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure.

1. No. The effects of power uprate have been evaluated to ensure that the increase in system temperatures causes minor increases in thermal loadings on pipe supports, equipment nozzles, and in-line components. The results of analyses show that at uprated conditions all ASME components will satisfy design specification requirements and code limits when evaluated to the rules of Subsection NB-3600 of the ASME Boiler and Pressure Vessel Code Section III. The effects of thermal expansion as a result of power uprate were found to be insignificant. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. Increases in system temperatures as a result of power uprate have been evaluated to show that increase in thermal loadings on pipe supports, equipment nozzles and in-line components are minor. Analysis shows that at all uprated conditions all ASME components will satisfy design specification requirements and code limits when evaluated to the rules of subsection NB-3600 of Section IV to the Boiler and Pressure Vessel Code. The effects of power uprate with respect to thermal expansion were found to be insignificant and, therefore, not found to create the possibility of a new or different kind of accident.

3. No. As stated above, the effects of thermal expansion as a result of power uprate were found to be insignificant. Consequently, the nominal increase in T_{ave} does not involve a significant reduction in a margin of safety.

Specification Table 5.7.1-1, Component Cyclic or Transient Limits

This specification is being changed to raise the upper limit for a heat cycle from 546°F to 551°F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure.

1. No. The purpose of this specification is to limit the number of heatup and cooldown cycles. The effects of power uprate have been evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. The analyses were performed for the design, normal, upset, emergency and faulted conditions. The increase in the temperature limitation is not significant with respect to the affect it has upon the RPV and associated components.

2. No. The effects of uprating power have been evaluated for the design, normal, upset, emergency and faulted conditions to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. The increase in the temperature limitation has been found not to be significant and, therefore,

does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **No.** This specification is intended to limit the number of heatup/cooldown cycles. The increase in the temperature limitation has not been found to be significant with respect to its effects upon the RPV and its associated components and, therefore, does not significantly reduce the margin of safety.

Specification 6.9.3.2, Core Operating Limits Report

Administrative Control Section 6.9.3.2 describes and lists topical reports that are used to determine core operating limits. Topical reports 15 through 19 are LOCA methodology reports and are being deleted. These reports describe Siemens LOCA methodology. As stated in Reference 1, the GE SAFER/GESTR LOCA methodology is being used for this uprated cycle. In addition, other minor methodology changes were made for power uprate transient analysis. GE topical report NEDC-32071P, PP&L topical report NE-092-001 (when approved), and the NRC Safety Evaluation Report on the PP&L power uprate licensing topical (when issued) are proposed to be added as Topical Reports No. 15, 16, and 17, respectively.

1. **No.** These changes are editorial in nature in that only the references to documents are being changed. The methodology used to determine core limits have been previously reviewed and approved by the NRC.

2. **No.** See the response to Question 1 above.

3. **No.** See the response to Question 1 above.

OPEN ITEMS/COMMITMENTS

POWER UPRATE STARTUP TESTS

PP&L plans to perform a post-power uprate startup test program similar in nature to the original SSES startup test program described in FSAR Chapter 14, but with the scope of testing limited to those tests or portions of tests affected by power uprate or increased core flow. The test program will be conducted in five separate segments or test plateaus. Each test plateau will contain one or more test conditions which define uprate power levels and core flows at which the tests are to be performed. The test plateaus and test conditions are shown in Table 1 attached to this letter. Generally, all tests scheduled to be performed in one test condition are to be completed before proceeding to the next higher test condition. After all testing in each plateau is completed, the test results for all tests will be reviewed by the plant operating review committee (PORC) before authorization is given to proceed to the next test plateau.

The requirements for power uprate startup testing come from a review of Chapter 14 of the FSAR, the GE Power Uprate Startup Test Specification, the proposed Technical Specifications for Power Uprate and the SSES Licensing Topical Report NE-092-001. The tests which will be performed for the power uprate startup test program are given in Table 2-6 attached to this letter. Generally, the tests are the same as performed in the original startup test program except that only those portions of the original test which are affected by power uprate are performed; the same acceptance criteria will be used as in the original power ascension test except in those tests originally specified by GE where GE has provided updated acceptance criteria power uprate.

Testing to further evaluate four-pump operation in the river water makeup system will be conducted after the power uprate.

HIGH ENERGY LINE BREAK (HELB) PROGRAMS

The evaluations of the high energy piping systems under Power Uprate Conditions (Sections 3.5.1 and 3.5.2.1 of the Susquehanna SES FSAR) show that the existing postulated break locations were not affected and no new pipe break locations were identified. The effects of Power Uprate on the HELB programs for Compartment Pressurization, Pipe Whip Analyses and Restraints, Containment Isolation Valve Operability, and Jet Impingement were evaluated.

The effects of power uprate on HELB were found to be proportional to the increase in reactor vessel pressure which resulted in higher loads, stresses, and displacements on the piping, supports, and whip restraints. However, these increases were relatively small and the original design basis HELB commitments contained in Section 3.6 of the Susquehanna SES FSAR are still satisfied.

SUMMARY OF GE POWER UPRATE ATWS ANALYSIS FOR SUSQUEHANNA

The Susquehanna ATWS analysis for power uprate conditions is documented in GENE-637-024-0893. Seven limiting events were analyzed. All events were initiated at the Extended Load Line Limit, 100% of uprate power (3441 MWth) and 87% or rated core flow (87MLb/hr). Susquehanna's performance was evaluated against criteria specified in NUREG-0460. Specifically, the criteria are as follows:

- **Reactor Pressure Vessel (RPV) Integrity** - The peak RPV pressure must be less than 1500 psig (Service Level C).

- **Fuel Integrity** - The maximum fuel cladding temperature cannot exceed 220°F and the local cladding oxidation must be less than 17%.

- **Containment Integrity** - The peak containment pressure must remain below the design pressure of 53 psig, while the peak suppression pool bulk temperature must remain less than 190 °F.

As shown by the summary of results in Table 7, the most limiting transient is the MSIV closure. However, the peak vessel pressure, cladding temperature, and suppression pool temperature all remain below the NUREG-0460 limits.

EMERGENCY OPERATING PROCEDURES

Emergency Operating Procedures to support uprated power operation are under development with implementation, to include operator training, scheduled to take place prior to startup following the end of the Unit 2 outage to support Cycle 7 operation.

MINOR CORRECTIONS TO THE TOPICAL REPORT

Listed below are minor correction which have been made to the SSES Power Uprate Licensing Topical Report found during preparation of this proposed license amendment. Each change is accompanied by an evaluation relative to its effect upon safety. In no case do any of these corrections change a conclusion in our report. Therefore, no revisions to that report are being issued. These items are provided to complete the record.

1. In Table 1-1, change the Full Power Core Flow Range for the Uprated Value and the Uprated Value with Increased Core Flow to "87 to 100" and "87 to 108" respectively.

   Evaluation

   This change reflects the Uprate ELLLA analysis that was added to PP&L Licensing Topical Report NE-092-001, (LTR) by Revision 1. The corresponding change to Table 1-1 was inadvertently omitted from Revision 1. This change is consistent with all safety analyses reported in the LTR.

2. In Section 5.1.1, first paragraph, third sentence, delete the phrase "rod block monitor (RBM) rod block." Add the following sentences to the end of the first paragraph:

   > The setpoints for the flow biased rod block monitor (RBM) rod block will change. On an absolute megawatt vs. reactor flow basis, the flow biased rod block remains approximately the same in order to maintain the pre-power uprate flow biased rod block.

   Evaluation

   The power uprate RWE analyses, performed after the submittal of the Licensing Topical Report, require the flow biased rod block to remain approximately the same (in MW_t) in order to produce acceptable MCPR operating limits. Because the definition of Rated Thermal Power changes by 4.5%, the RBM rod block equations must be rescaled to maintain the same level of protection as previously provided. This change to the Licensing Topical Report is consistent with the proposed changes to the Susquehanna Steam Electric Station, Unit 2, Technical Specifications.

As noted in the Item 1 above, all analyses contained in the LTR were done for uprated ELLLA conditions which would allow full power operation at core flows between 87 million lbm/hr and 108 lbm/hr. The RWE analysis does not allow full use of this operating domain and has the effect of restricting rod movement at 100% power to within a core flow window of 93 million lbm/hr to 108 million lbm/hr. If future RWE analyses should allow full use of the uprated ELLLA domain, a separate Technical Specification change will be submitted to raise the flow biased RBM rod block by 4.5%.

3. Change the second paragraph of Section 5.1.2.5 to read as follows:

> For increased core flow, the APRM rod block is clamped at a nominal value of 108%. The APRM clamp provides an alarm to help the operator avoid inadvertent scrams in the ICF region by maintaining the same margin between the APRM scram and the APRM rod block.

Evaluation

As noted in Item 2 above, the RBM rod block was maintained at its present level of protection while the APRM scram and rod block were increased by 4.5%. The increased margin between the RBM rod block and the APRM scram and rod block, allowed the rod withdrawal error analysis to assume the RBM rod block was not clamped in the ICF region. The RWE analysis was able to maintain adequate MCPR operating margins without the clamp in the ICF region. This change to the LTR is consistent with the proposed changes to the Susquehanna Steam Electric Station, Unit 2, Technical Specifications.

4. In Table 5-1, change the Analytical Limit for the APRM Simulated Thermal Power Scram-Clamped, for Power Uprate and Power Uprate With ICF, to 118%.

Evaluation

The GE methodology for calculating Nuclear Instrumentation setpoints changed following the Licensing Topical Report submittal. The revised methodology now includes a 2% allowance for gain adjustment factor deviations. A 2% deviation is allowed by the Technical Specifications before adjustments are required. This additional calibration error required the analytical limit to increase by 1% in order to maintain the Technical Specification setpoint and allowable value at their current settings. Since the APRM Simulated Thermal Power Scram-Clamped function is not used in any Susquehanna safety analysis, the change does not change or alter the conclusions of the Licensing Topical Report. The fact that the APRM Simulated Thermal Power Scram-Clamped is not used in Susquehanna safety analysis was identified to the NRC in PLA-4010, dated August 5, 1993. This letter responded to power uprate questions from the Reactor Systems Branch.

5. In Table 5-1, change the Analytical Limit for the Turbine First Stage Scram and EOC RPT Bypass Pressure, for Power Uprate and Power Uprate With ICF to 147.7 psig.

   Evaluation

   The values of turbine first stage power at 30% reactor power, were taken from a preliminary heat balance. The final turbine heat balance, which was finalized after the LTR was submitted, resulted in a small change to the turbine first stage pressure at 30% reactor power. The Table is being changed to reflect this change. Since the safety analysis, which utilizes this function, uses reactor power not turbine first stage pressure, this change does not affect any safety analysis. This change to the LTR is consistent with the proposed changes to the Susquehanna Steam Electric Station, Unit 2, Technical Specifications.

6. In Section 10.6, paragraph 1, subitem 4, change the last sentence to the following:

   > The same performance criteria will be used as in the original power ascension test except in those tests originally specified by GE where GE has provided updated performance criteria for power uprate.

   Evaluation

   This change acknowledges the fact that GE has modified performance test criteria in the time since Susquehanna was completed and tested. The change identifies the fact that the latest GE performance test criteria will be used in the power uprate test program.

Copies of those sections of the SSES Unit 2 Technical Specifications which have the requested changes written in are attached to this letter as Attachment 2.

Should you have any questions on the above material, they should be directed to Mr. R.R. Sgarro at (215) 774-7914.

Very truly yours,

G. T. Jones

Enclosure

cc: NRC Document Control Desk (original)
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