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SUBJECT: Provides response to request for addl info re licensing topical rept for power uprate.

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SEP 29 1993

Robert G. Byram
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Director of Nuclear Reactor Regulation
Attention: Mr. M. L. Boyle, Acting Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION,
LICENSING TOPICAL REPORT ON POWER UPRATE — **Docket Nos. 50-387 and 50-388**
PLA-4028 FILE R41-2

Dear Mr. Boyle:

References: 1) *"Request for Additional Information, Licensing Topical Report on Power Uprate, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (Tac Nos. M83426 and M83427," Richard J. Clark to Robert G. Byram, dated September 21, 1993.*
2) *PLA-3788, H.W. Keiser to C.L. Miller, "Submittal of Licensing Topical Report on Power Uprate with Increased Core Flow" dated June 15, 1992.*

The purpose of this letter is to respond to your staff's request for additional information (Reference 1) on the Susquehanna SES licensing topical report on power uprate (Reference 2). PP&L's responses to your staff's questions are provided in Attachment 1 to this letter.

Any questions on these responses should be direct to Mr. R.R. Sgarro at (215) 774-7914.

Very truly yours,

R. G. Byram

Attachment

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector - SSES
Mr. R. J. Clark, NRC Sr. Project Manager - Rockville

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PENNSYLVANIA POWER & LIGHT COMPANY

RESPONSE TO NRC QUESTIONS

September 27, 1993

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 1 (Section 3.1) For the power uprate, the number of SRV valve groups is reduced from five to three to prevent an increase in the number of unnecessary SRV actuations. Provide a discussion on how the change in number of SRV groups from five to three affect the SRV dynamic loads regarding frequency content, amplitude and symmetry of load boundary.

Response a. Amplitude

Several SRV load cases were postulated to develop the original pool boundary load due to the SRV air clearing phenomena. The worst case load on the pool boundary is the all valve load case, which assumes the simultaneous actuation of all sixteen SRVs. All SRVs are assumed to actuate simultaneously with the air bubbles entering the pool at the same time and oscillating in-phase. No load reduction credit is taken for the fact that the SRVDL air bubbles would enter the pool at different times due to differences in SRV setpoints. This assumption represents a substantial conservatism in the original load methodology, since out of phase SRV air bubbles would significantly reduce the SRV load amplitude. Therefore, the original SRV pool boundary load amplitude is conservative for the change from five to three valve groups.

b. Symmetry

Symmetry effects for the original SRV load definition were considered by postulating asymmetric and symmetric load cases. The symmetric load case assumes the simultaneous actuation of all sixteen SRVs as described in a. above. The all valve load case results in the largest symmetric load because all SRVs are assumed to actuate and the quenchers are spaced symmetrically around the suppression pool. The change from five to three valve groups does not affect the symmetry of the original symmetric load case, since the original load conservatively assumes all valves actuate simultaneously without regard to differences in opening setpoints.

The asymmetric load case is postulated for two SRV loading conditions: 1) a single SRV actuation and 2) the simultaneous actuation of three SRVs whose quenchers are adjacent to each other in the suppression pool. The three valve load case is a more severe load than the one valve load case. The three valve case is non-mechanistically postulated, independent of the SRV valve groups setpoints, to define an extreme asymmetric SRV boundary load for design purposes. The two load cases are specified by a conservative asymmetric azimuthal pressure

Response to NRC Request for Additional Information
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Mechanical Engineering Branch

Question 1 Response Continued

b. Symmetry (Continued)

distribution on the containment walls that maximizes the pressure amplitude on one side of the containment, while minimizing pressure amplitude on the opposite side. The asymmetric pressure distribution is the worst case asymmetric load on the containment boundary for all postulated operating conditions. The change from five to three valve groups does not create a postulated asymmetric pressure distribution that is not already bounded by the original asymmetric pressure distribution.

It should be noted that the effects of the SRV boundary load are considered in the original plant design by taking the envelope of the symmetric and asymmetric load cases.

c. Frequency Content

Experimental data confirms that the frequency content of the SRV pool boundary load is primarily dictated by the initial air mass in the SRV discharge line (SRVDL) prior to SRV actuation and the number of SRVs that actuate. Power uprate does not affect the initial air mass in the SRVDL. Since the load cases are not changing, the number of SRVs that are postulated to actuate does not change for power uprate. Therefore, there is no effect on the frequency content of the SRV pool boundary load from power uprate.

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 2 (Section 3.3.2.1) Tables 3-1 and 3-2 show that the reactor internal pressure differential (RIPD) increase from 9% to 26% for the power uprate with increased core flow (ICF), in comparison to the original analysis. However, the topical report stated that recalculated core loads and RIPDs decrease for power uprate with ICF because of the improved current fuel design. Please clarify.

Response In the Section 3.3.2.1 text, it is stated, "...However, recalculated core loads and RIPDs for power uprate with ICF can decrease compared with the original analyses, because of the improved thermal hydraulic of the current fuel designs...".

The "original analyses" in the above paragraph include both "Original Analysis" and "Original Power with ICF Analysis" columns in Tables 3-1 and 3-2. In these tables, by comparing the first two columns, it can be seen that the increase in RIPDs is caused by the increase in the core flow. Comparing the last two columns of these tables also shows that the effect of fuel design thermal-hydraulic characteristics is more apparent in the RIPD calculation, and can result in lower calculated RIPDs. This is the basis for the statement in Section 3.3.2.1.

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 3 (Section 3.3.2.1) LOCA dynamic loads consist of pool swell, condensation oscillation and chugging loads. The licensee's evaluation concluded that all original LOCA dynamic loads, except pool swell, are bounding for power uprate. Provide a discussion on how the increase in pool swell dynamic loads for power uprate affect the design basis analysis of the reactor internals and the reactor coolant piping.

Response NUREG-0487 provides the NRC's review and acceptance of the generic Mark II pool swell load methodology adopted by PP&L. Pool swell produces loads on the submerged structure boundary and submerged structures (downcomer bracing, piping, etc.). NUREG-0487 describes each type of load and the methodology for calculating the specific load. Pool swell loads are restricted to only suppression pool structures and components located in the pool swell zone. The pool swell zone extends from the downcomer exit elevation to the maximum pool swell height. As described in NUREG-0487, pool swell loads on the suppression pool structure and components do not produce dynamic loads on components located outside of the pool swell zone.

The pool swell air bubble load on the suppression pool boundary is defined as a static load (NUREG-0487, page III-23) and, therefore, does not result in containment dynamic loads on the reactor internals and piping components. The remaining pool swell loads are drag, impact and fallback loads on the suppression pool components located in the pool swell zone. These loads produce stresses and deformations on the suppression pool components only, and do not result in global loads on the containment structure. Therefore, the pool swell loads on the suppression pool components do not transfer containment dynamic loads to the reactor internal piping components.

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 4 (Section 3.3.2.2) The licensee concluded that the reactor internals response to flow-induced vibration for the power uprate with ICF will remain within acceptable limits, based on the comparison of valid prototype vibration data and SSES test data. Provide a detailed discussion regarding the effects of power uprate with ICF on flow-induced vibration, and how the valid prototype vibration data and SSES test data were used for the SSES power uprate conditions. Also, provide locations of the critical low-frequency components, the maximum fluid-elastic instability ratio, and cumulative fatigue usage for the uprated conditions.

Response The vibration evaluation was performed by analyzing the data recorded during plant operation at the prototype plant. There were more than 40 test conditions during hot (power) operation at the prototype plant including normal balanced flow operations, unbalanced flow operations and increased core flow operations. Tests were conducted at 50%, 75% and 100% rod lines. The vibration evaluation was performed bas edon the extrapolation of these test results.

The following components were instrumented at the prototype plant with strain gages, accelerometers and displacement measuring devices: Upper bolt guide ring, shroud/vessel, jet pump elbows, diffuser, riser braces, feedwater spargers, in-core guide tubes, control rod guide tubes and shroud support legs. At SSES the steam dryer was instrumented and data was obtained up to 100% of rated power. Results were extrapolated for the power uprate.

The vibrations of reactor internals submerged underwater varied with the core flow and were not significantly affected by steam flow (power uprate). The dome components such as the separator and steam dryer were affected by the steam flow. All the above evaluated components except some jet pump sensing lines under increased core flow had less than 10,000 psi peak stress intensity under power uprate conditions. Clamps will be installed in the affected sensing lines for increased core flow operations which will lower their stress below 10,000 psi. Since the peak stress intensity is less than the endurance limit, there is no fatigue usage due to vibration in the normal or power uprate condition.

Components with potential instability problems have already been extensively tested at GE's facilities, before installation at the site. In addition tests up to 113% of rated core flow were made at the prototype plant during power operation and the measured data of reactor internal vibrations showed no instability problems.

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 5 (Section 3.5.1) The licensee's evaluation was rather conclusory and lack adequate technical basis. Provide a detailed discussion regarding the evaluation of pipe supports, equipment nozzles, and in-line components for the power uprate, and summary of results in comparison to the existing design allowable.

Response Power uprate affects the Class I reactor coolant pressure boundary (RCPB) piping components due to the changes in operating flow rates, temperatures and pressures. The design pressures and temperatures remain the same. A certified Design Specification was prepared to incorporate the power uprate flow, temperature and pressure changes for each system. A detailed assessment of all Class I piping components, including in-line components (equipment nozzles, valve moment loading, flange connections, etc.), pipe supports and flued heads was performed based on the Certified Design Specification requirements. The assessment was completed to the requirements in the original ASME Code of Record and approved Code Addendum, and was documented in design calculations and revised Class I Stress Reports. The piping assessment concluded that Class I piping components meet the ASME Code requirements for power uprate conditions as described below:

The existing thermal transient analysis was revised based on the new ΔT_1 and ΔT_2 and $T_a - T_b$ (as defined in Subsection NB-3653), and then combined with the power uprate pressures and thermal expansion piping moments to assess the Class I piping for fatigue. The primary plus secondary stresses were calculated using equations 10 through 14 and the cumulative usage factors were generated for each Class I weld/in-line component in accordance with the requirements described in Subsection NB-3650 of the ASME B&PV Section III Code. The Class I primary stresses were calculated using equation 9 for the Design, Normal/Upset, Emergency and Faulted conditions. All Nuclear Class I flued head reports were also revised to incorporate the power uprate parameters. Furthermore, the high energy line breaks were reviewed to the SRP 3.6 requirements based on the revised fatigue analysis and concluded that the previous postulated break configurations and locations remain unchanged.

The thermal modes and flexibility calculations for each Class I system were evaluated for power uprate temperature effects to assess the load increases on in-line components such as valve end moment loading (if applicable), flange connections, etc., equipment nozzles and pipe supports. Increases in loads on the components were calculated and the revised loads compared to the existing allowables to determine their acceptability. Systems which were subjected to fluid transient loads such as Main Steam

Response to NRC Request for Additional Information
Licensing Topical Report for Power Uprate
Susquehanna Steam Electric Station

Mechanical Engineering Branch

Question 5 Response Continued

(Inside Drywell) were reevaluated in detail using the revised fluid transient forces which were developed using original design basis methodology and power uprate pressures and flow rates.

The acceptability of the revised piping loads on pipe supports due to power uprate temperature changes was demonstrated by comparing the revised loads against the margins in the original design basis calculation. The change in pipe movements at supports due to changes in thermal expansion were revised against the margins available in spring and snubber settings, and binding checks were performed in unrestrained directions. Where required, detailed pipe support calculations were performed to demonstrate acceptability.

Response to NRC Request for Additional Information
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Mechanical Engineering Branch

Question 6 (Section 3.5.2.1) It appears that the licensee did not perform detailed evaluation, but rather, suggested that BOP piping systems will be acceptable by methods as stated in this section. Provide a description and results of evaluations that were performed, regarding compliance with the code of record. Please also discuss the SRV discharge line evaluation, considering an increase of pool swell velocity for the power uprate condition with increased core flow.

Response The balance of plant (BOP) piping systems have been evaluated in detail to assure that these systems still meet the requirements of the Code of Record and approved Code Addendum after accounting for power uprate changes in temperature, pressure and flow rate. The scope of (BOP) piping which is affected by power uprate changes was determined from the uprated reactor and BOP heat balances. A matrix of affected portions of piping systems, and the revised pressures, temperatures and flow rates for these systems was developed. A detailed review of the existing design basis calculation for these systems was performed. A vast majority of these systems were originally designed to maximum temperature and pressures that enveloped the increases in operating pressures and temperatures due to power uprate conditions and were therefore considered acceptable.

Portions of systems whose design pressure and temperatures were not enveloped by power uprate were reviewed in detail to identify the margins in piping stress levels when compared to Code of Record allowables. Upon review of those affected calculations, it was concluded that sufficient margin exists in the stress levels to accommodate the differences in temperature and/or pressure due to power uprate.

Systems which originally required evaluations for fluid transients such as the Main Steam bypass lines, Reactor Feed Pump Turbine lines, etc. were evaluated for the effects of the revised fluid transient loads. Transient loadings were calculated using the same methodologies and assumptions used for the original licensing basis, except that power uprate conditions were incorporated.

The results of these analyses were used to verify the acceptability of the system under power uprate conditions and documented either by comparison of actual piping stresses against the Code of Record allowable limits used in the original design; or by assuring that the uprated pressures, temperatures and flow rates were bounded by the pressures, temperatures and flow rates used in the original design.

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Mechanical Engineering Branch

Question 6 Response Continued

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 to 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 effect of power uprate conditions on the existing postulated pipe breaks in the portions of high energy systems was evaluated to the requirements of SRP 3.6. The conclusion of this review showed no change in the location of existing pipe breaks.