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U.S. Nuclear Regulatory Commission
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**SUSQUEHANNA STEAM ELECTRIC STATION
SUPPLEMENTAL INFORMATION APPLICABLE
TO PROPOSED AMENDMENT NO. 235 TO
LICENSE NPF-14 AND PROPOSED AMENDMENT
NO. 200 TO LICENSE NPF-22: POWER UPRATE
PLA-5321**

**Docket No. 50-387
and 50-388**

Reference: 1) *PLA-5276, R. G. Byram To USNRC, Revised Submittal of Proposed Amendment No. 235 to License NPF-14 and Proposed Amendment No. 200 to NPF-22: Power Uprate dated 02/08/2001*
2) *NRC RAI, R. G. Schaaf to R. G. Byram, "Request for Additional Information Regarding 1.4 – Percent Power Uprate (TAC NOS. MB0444 and MB0445) dated 4/30/2001*
3) *PLA-5300, R. G. Byram to USNRC, Revised Submittal of Proposed Amendment No. 235 to License NPF-14 and Proposed Amendment No. 200 to NPF-22: Power Uprate dated 5/22/2001*

The purpose of this letter is to provide supplemental information regarding our proposed amendment requests made in Reference (1). The need for this supplemental information was developed during a teleconference held May 21, 2001.

The questions and our responses are contained in Attachment 1.

If you have any questions, please contact Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

G. T. Jones

G. T. Jones

Attachment

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Project Manager
Mr. D. J. Allard, PA DEP

A001

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

SUPPLEMENTAL INFORMATION APPLICABLE TO PROPOSED AMENDMENT NO. 235 TO LICENSE NPF-14: POWER UPRATE UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files supplemental information in support of a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

G.T. Jones

G. T. Jones
Vice-President - Nuclear Engineering & Support

Sworn to and subscribed before me this 31st day of May, 2001.

Nancy J. Lannen

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

SUPPLEMENTAL INFORMATION APPLICABLE TO PROPOSED AMENDMENT NO. 200 TO LICENSE NPF-22: POWER UPRATE UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files supplemental information in support of a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
By:

G. T. Jones
Vice-President - Nuclear Engineering & Support

Sworn to and subscribed before me
this 31st day of May, 2001.

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

Attachment 1 to PLA-5321

Supplemental Information

Supplemental Information

1. In reference to Section 2.6.1 of Topical Report NE-2000-001P, provide a summary of the evaluation of the effects of the 1.4 percent power uprate on the design basis analysis of the Control Rod Drive Mechanisms (CRDM's). Confirm that the CRDM's structural integrity will be adequate for the 1.4 percent power uprate.

   Response

   Enclosure 4 of Reference 1 contained responses to NRC questions in support of the SSES "Power Uprate with Increase Core Flow" License Amendment requests. "Question 1" related directly to the CRDM's. A copy of the question and the response provided is included in Attachment 2. This response remains valid for the current proposed power uprate since the parameters analyzed for the previous power uprate remain bounding.

   Therefore the CRD system will perform its function at the power uprate conditions and the CRDM's structural integrity will be adequate for the 1.4% uprate conditions.

2. Provide a summary of evaluations for the reactor internals and the reactor coolant pressure piping. The evaluations should include the existing minimum margin in stress and cumulative usage factor (CUF) which will accommodate the changes as a result of the 1.4 percent power uprate or confirm that the component design basis pressure, temperature and flow rate are bounding for the power uprate condition. Also, confirm that there is no increase in the potential for flow induced vibration and that there are no changes in postulated high energy line break locations, jet impingement and thrust forces as a result of the power uprate.

   Response

   Reactor coolant system stresses and cumulative usage factors (CUF) are determined from design basis pressure, temperature and flow conditions. The design basis conditions are 3510 MWt, 1053 psia, the associated saturated temperature of 551°F, and a total core flow rate of 108 Mlb_m/hr. These design conditions do not change under operation at increased rated thermal power, therefore, no change to the analysis of reactor vessel and reactor internal stresses is necessary. Structural integrity of reactor vessel components was also addressed in Enclosure 4 of Reference 1. See "Question 3" which is included in Attachment 2. The response provided to "Question 3" remains valid. The current ASME Class 1 analysis for the Feedwater and Main Steam lines use temperatures and flows which bound the values associated with increased rated thermal power, therefore there is no effect on the existing stresses or CUF's for these systems. Structural integrity of reactor coolant pressure boundary piping systems was all also addressed in Enclosure 4 of Reference 1. See "Question 4" which is included in Attachment 2. The response provided to "Question 4" remains valid. In addition, since the design conditions do not change, there is no change to high energy line break locations, jet impingement or thrust forces as a result of operating at increased rated thermal power. The potential for flow induced vibration in the reactor vessel, reactor vessel internals and recirculation piping has been evaluated at the above design conditions, therefore, those evaluations remain bounding. Flow increases of over 5% were also evaluated during the previous power uprate effort for the Feedwater and Main Steam systems with respect to structural integrity of these systems and found to be acceptable.

3. Provide a summary of evaluations for safety-related mechanical components (i.e., all safety related valves and pumps, including air-operated valves (AOVs) and power-operated relief valves) affected by the power uprate to demonstrate that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirrn that the existing design basis analysis bounds the 1.4 percent power uprate condition associated with the system pressure, temperature, flow rate, and pressure and temperature differentials. Also, confirm that there will be no impact on the plant safety related valves including air-operated and motor-operated valves and GL 89-10 MOV program, and that there are no changes in the post-LOCA conditions associated with GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," following the 1.4 percent power uprate.

<u>Response</u>

Potential impacts of the proposed power uprate on PPL's Generic Letter 89-10, 95-07, and AOV Programs have been reviewed. In addition, the potential for impact to PPL's responses to the staff's concerns identified in Generic Letter 96-06 have been evaluated. The following summarizes the results of these reviews.

<u>Generic Letter 89-10</u>

The general assumptions and input parameters which PPL used to develop the G/L 89-10 design basis operating conditions were reviewed, with specific consideration given to the operational changes induced by the proposed power uprate. In addition, the changes to valve design bases, which were made in support of the first SSES Power Uprate Project (Reference 1) were also reviewed. It has been determined that the proposed power uprate does not affect any of the safety-related accident operating conditions specified for MOVs in PPL's G/L 89-10 Program.

The basic reason for this conclusion is that under the proposed power uprate, there are no changes to the input assumptions used to develop the safety related valve operational design bases. In determining the safety-related operating conditions, worst case parameters were obtained from the applicable accident analyses. Since these accident analyses are not being affected by the proposed power uprate, the peak parameters previously used are still applicable. In addition, there are no safety relief valve, instrument, or alarm setpoint changes. There are no changes to any maximum or minimum Technical Specification parameters (i.e., vessel pressure, suppression pool level/temperature, spray pond level/temperature, etc.). There are no hardware changes being implemented which would increase system pressures and flows. There are no changes to the ambient operating environment, and since the proposed power uprate does not affect the plants voltage study, the are no changes to the voltage profiles under which valves must operate. Also, there are no changes to the manner in which systems will be operated during the mitigation of accidents/transients.

There will be no significant changes to the normal RPV operating conditions (i.e., water level, dome pressure, core flow, etc.). Feedwater and Main Steam Flow will be increased. The increase is bounded by current analyses. Based on the review of the general assumptions and input parameters, it is apparent that these two parameters were not used in the development of any safety related operating conditions for valves in PPL's G/L 89-10 Program. It is therefore concluded that the proposed power uprate does not affect PPL's G/L 89-10, nor 96-05 Programs, as accepted by the staff in References 2 & 3.

Generic Letter 95-07

PPL's original review of G/L 95-07 ultimately concluded that nineteen flex wedge gate valves per unit, comprising eleven design applications, had safety-related functions to open. (Note that the difference between eleven and nineteen is that some design applications had redundant divisions; i.e., "A" and "B" loops.) Of these eleven applications, three were found to be not susceptible to pressure locking due to system design, or procedural considerations. Of the remaining eight applications, seven were modified by drilling their discs to eliminate the potential for pressure locking, and one was addressed via procedural guidance. These corrective actions, which the staff found to be acceptable as documented in Reference 4, are not affected by the proposed power uprate.

With respect to thermal binding, the majority of valves are not susceptible as a result of their service application and materials of construction. Only one valve, the RHR heat exchanger outlet valve, was identified as being potentially susceptible to thermal binding. However, precautions have been added to the applicable procedures to prevent closure of this valve when operating conditions pose the potential for thermal binding. As with the corrective actions taken for pressure locking, the staff found this approach to be acceptable as documented in Reference 4, and it is not affected by the proposed power uprate.

Air Operated Valve (AOV) Program

The scope of valves with active safety functions (or those with functions important to safety), which are included in PPL's evolving AOV Program, was reviewed for any potential impacts. For the reasons discussed above, most valves in the AOV Program are unaffected by the proposed increase in power. However, there are three design applications which are potentially affected by the increase in feed-flow and steam flow. The first two applications are the feedwater startup valve, and its low load bypass valve. However, these valves are only used during unit startups and shutdowns, and are closed during full power operation. Since they are only operated during periods of reduced feed-flow, they are unaffected by the increase in full rated feed-flow.

The final application which is potentially affected is the Main Steam Isolation Valves, which have been found to be acceptable for the uprated conditions, as documented in Section 3.7 of PPL's submittal for the proposed increased in power (Reference 5).

Generic Letter 96-06

In Generic Letter 96-06, the NRC identified three concerns regarding primary containment integrity during Design Basis Accidents (DBAs): 1) the potential for two-phase flow in drywell air cooling systems, which could result in reduced cooling capacity; 2) the potential for waterhammer resulting from draining of containment air cooling systems, which could threaten the function of affected components; and, 3) the potential for thermally-induced overpressurization of isolated containment penetrations, which could lead to the excessive release of fission products. PPL's position on each of these items, along with a brief discussion on any impacts due to the proposed power uprate, are provided below:

1) Potential For Two-Phase Flow In Containment Air Cooling Systems

As identified in previous PPL submittals (References 6 & 7), the SSES drywell cooling system is a non-safety-related system used to maintain containment temperatures during normal plant operations. The system automatically isolates under conditions indicative of a Loss of Coolant Accident (LOCA). The cooling function of this system is not credited in any DBA analyses, nor in the SSES Individual Plant Evaluation (IPE). Hence, this system is not a factor in determining overall plant safety or risk. As such, any impact which two-phase flow may have on the effectiveness of the drywell coolers is of no consequence.

No changes to the design, licensing, or operating bases of the SSES drywell cooling system are being proposed under the proposed power uprate. As such, PPL's position regarding the potential for two-phase flow in drywell cooling systems, as previously submitted to the NRC, is unaffected by the proposed increase in power.

2) Potential For Waterhammer Resulting From Draining Of Containment Air Cooling Systems

As previously identified, and discussed in Reference 7, the SSES drywell cooling system is a non-safety-related system which automatically isolates during a LOCA. However, during scenarios which are beyond the SSES design/licensing bases, the plant's Emergency Operating Procedures (EOPs) allow for the system to be restored, should it isolate on a "false-LOCA" signal. While an extremely remote possibility, under these conditions, a water hammer could occur during the emergency restoration of drywell cooling. However, as described in Reference 7, analyses have been performed which demonstrate that the worst case physical loads resulting from such a water hammer result in piping stresses which are well within ASME Code allowables. Therefore, the restoration of drywell cooling during scenarios which are beyond the SSES design/licensing bases will not threaten the integrity of primary containment. Finally, it should be noted that, in response to G/L 96-06, procedural controls have been established which prohibit the restoration of drywell cooling under "true-LOCA" conditions.

No changes to the design, licensing, or operating bases of the SSES drywell cooling system are being proposed under the proposed power uprate. As such, PPL's position regarding the potential for water hammer in drywell cooling systems, as previously submitted to the NRC, is unaffected by the proposed increase in power.

3) Potential For Thermally-Induced Overpressurization Of Isolated Containment Penetrations

PPL is currently pursuing further efforts to resolve the potential for containment penetration overpressurization. In general, the objective of these efforts is to provide an analytical, and/or a risk based disposition of the staffs concerns. It is PPL's position that this approach is in the best interest of plant safety and reliability.

Containment penetration heat-up rates are being modeled based on the peak drywell temperature for a small steam line break, which produces the worst case drywell temperature, as discussed in the SSES FSAR (Reference 8). This peak drywell temperature is determined by identifying the vessel pressure, and corresponding drywell pressure, which produce the maximum calculated drywell temperature. This worst case drywell temperature occurs when the vessel has depressurized to about 450 psia, and hence, is not a function of initial power level. Thus, ongoing analytical efforts are not affected by the proposed increase in power.

Since the worst case penetration heat-up scenario is not affected by the initial reactor power level, it is concluded that the proposed increase in reactor power will not affect PPL's disposition to concerns regarding the potential for penetration overpressurization.

From the discussions above, it is evident that PPL's responses to the potential for two-phase flow and water hammer in drywell air cooling systems are unaffected by the proposed increase in power. In addition, ongoing efforts to resolve concerns regarding the potential for containment penetration overpressurization are independent of initial reactor power level. Therefore, the proposed increase in reactor power will not affect PPL's resolution to Generic Letter 96-06.

Safety Related Pumps

All safety related pumps were evaluated as discussed in Section 4.0 of the PPL Topical Report NE-2000-001P. The conditions in which the pumps will be required to operate upon implementation of the proposed changes are bounded by the analyses.

References

1) Susquehanna Steam Electric Station, Submittal of Licensing Topical Report on Power Uprate With Increased Core Flow, PLA-3788, Keiser, Harold, W. to Miller, C. L., 06/15/92

2) NRC Combined Inspection Report 50-387/96-13, 50-388/96-13 and Notice of Violation, Pasciak, Walter, J. to Byram, Robert, G., 01/29/97

3) Safety Evaluation of Licensee Response to Generic Letter 96-05, Susquehanna Steam Electric Station Units 1 and 2 (TAC Nos. M97109 And M97110), Schaaf, Robert, G to Byram, Robert, G., 12/29/00

4) Safety Evaluation of Licensee Response to Generic Letter 95-07, Susquehanna Steam Electric Station Units 1 and 2 (TAC Nos. M93528 And M93529), Nerses, Victor to Byram, Robert, G., 11/01/99

5) Susquehanna Steam Electric Station, Revised Submittal of Proposed Amendment No. 235 to License NPF-14 and Proposed Amendment No. 200 To NPF-22:Power Uprate, PLA-5276, Byram, Robert, G. to NRC, 02/05/01

6) Susquehanna Steam Electric Station, 120-Day Response to Generic Letter 96-06, PLA-4551, Byram, Robert, G. to NRC, 01/29/97

7) Susquehanna Steam Electric Station, Response for Additional Information Related to Generic Letter 96-06, PLA-4999, Byram, Robert, G. to NRC, 11/09/98

8) SSES Final Safety Analysis Report, Section 6.2.1.1.3.3.5, "Small Size Breaks", Rev. 54, 10/99

Attachment 2 to PLA-5321

Selected Pages from Enclosure 4 of PLA-3788 dated 6/15/1992, "Submittal of Licensing Topical Report on Power Uprate with Increased Core Flow"

RESPONSE TO NRC QUESTIONS

MECHANICAL ENGINEERING BRANCH

QUESTION 1

(Section 2.5.1) - Discuss the effects of bottom head pressure increase on the structural and functional integrity of the control rod drive system (CRDS) due to power uprate. State the basis of determining the acceptability of the CRDS regarding compliance with the Code, to include not only the Code allowables, but the calculated maximum stresses, deformation, and fatigue for the uprated power conditions, and assumptions used in the calculations.

RESPONSE

The Susquehanna Units 1 and 2 Control Rod Drive (CRD) system was evaluated for a bounding reactor dome pressure to 1060 psig and an additional 35 psid for the vessel bottom head. The CRD mechanism structural and functional integrity was deemed acceptable for the vessel bottom head pressure of 1095 psig. This bounds the uprated maximum operational dome pressure. The components of the CRD mechanism designated as primary pressure boundary have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The applicable ASME Code effective date for Susquehanna 1&2 is 1971 Edition, up to Winter 1972 Addenda. The limiting component of the CRD mechanism is the indicator tube which has a calculated stress of 20,795 psi (allowable is 31,050 psi). The maximum stress is due to a maximum CRD internal hydraulic pressure of 1750 psig. The analysis for cyclic operation of a CRD mode of similar design was previously evaluated in accordance with ASME Code N-415.1. All requirements of N-415.1 are satisfied even when considering the increase in power uprate vessel bottom head pressure, thereby satisfying the peak stress intensity limits governed by fatigue. Furthermore, the maximum fatigue usage factor calculated per ASME Section III, Subsection NB3222.4 is 0.15. It should be noted that the CRD was analyzed and tested to a scram pressure which exceeds the power uprate pressure conditions. Deformation has not been specifically analyzed, however, the CRD has been successfully tested for all operational modes at simulated reactor vessel pressures up to 1250 psig saturated conditions which demonstrates that deformation is not a concern. The CRD system is capable of providing 250 psid differential pressure between the Hydraulic Control Unit and the reactor vessel for control rod insert and withdraw operation. At power operation, the primary scram pressure is provided by the reactor vessel pressure. Therefore, the CRD system will perform its function at the power uprate conditions.

RESPONSE TO NRC QUESTIONS

MECHANICAL ENGINEERING BRANCH

QUESTION 3

(Section 3.3.3) - 10 CFR Part 50, Appendix A, GDC 15 requires that the reactor coolant system be designed with sufficient margin to assure that the design considerations are not exceeded. For the core spray at the uprated power, the cumulative usage factor (CUF) was stated to be 0.99 which is nearly the limit of 1.0 set forth by Code. However, adequate technical basis was not given for the acceptance of 0.99. Provide detailed discussions regarding the critical location(s) of concern, analysis methodology and assumptions, vibrating inputs and thermal transients, and the edition of code used in the determination of the cumulative usage factor.

RESPONSE

The effects of power uprate for Susquehanna Units 1 and 2 were evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the 1968 Edition of ASME Boiler and Pressure Vessel Code with Addenda to and including Summer 1970. For Susquehanna, the limiting component was the Head Flange with a usage factor of 0.92 (see Table 3-4 of NE-092-001). Since the condition used in the original analysis bounded those for the power uprate conditions, the fatigue usage calculated in the original analysis is applicable to the power uprate conditions. Detailed analyses are available for NRC review.

RESPONSE TO NRC QUESTIONS

MECHANICAL ENGINEERING BRANCH

QUESTION 4

(Section 3.5) - It appears that no substantive evaluation regarding the acceptability of the reactor coolant pressure boundary (RCPB) piping systems including main steam, main steam drains, recirculation loop, core spray, standby liquid control, and CRD piping was provided for uprated conditions. Provide a discussion regarding analysis methods and assumptions and compliance with their Code of record. This includes not only the Code allowables, but the calculated maximum stresses and fatigue for normal, upset and faulted conditions.

RESPONSE

As discussed in Subsection 3.5.1 of NE-092-001 (with the exception of the CRD piping which is covered in Subsection 3.5.2 under the ASME Class II evaluation), a rigorous evaluation of all ASME Class I piping systems has been performed for the effects of power uprate using the ASME code of record. All stresses and usage factors will meet the ASME code criteria. Specific stresses and usage factors are available for NRC review.