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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
SUPPLEMENT TO PROPOSED AMENDMENT NO. 244
TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 208
TO LICENSE NPF-22: REVISE MAIN STEAM RELIEF VALVE
STEAM SETPOINT TOLERANCE AND REQUESTS FOR
RELIEF FROM IST AND ASME CODE REQUIREMENTS
PLA- 5430**

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

Reference: 1) PLA-5377, R. G. Byram (PPL) to USNRC Document Control Desk, "Proposed Amendment No. 244 to License NPF-14 and Proposed Amendment No. 208 to License NPF-22: Revise Main Steam Relief Valve Setpoint Tolerance and Requests for Relief from IST and ASME Code Requirements", dated October 18, 2001.

2) Letter, NRC to R. G. Byram (PPL), "Susquehanna Steam Electric Station, Units 1 and 2 - Request for Additional Information Re: Amendment Request to Revise Main Steam Relief Valve Setpoint Tolerance (TAC Nos. MB3273 and MB3274)", dated January 8, 2002.

The purpose of this letter is to provide supplemental information necessary for the NRC staff to complete its review of the license amendment proposed in PLA-5377 (Reference 1).

PLA-5377 proposed changes to the Units 1 and 2 Technical Specifications to revise the allowable tolerance for the main steam relief valve setpoints for as-found testing. The Nuclear Regulatory Commission staff has reviewed Reference 1 and has determined that additional information is required in order to complete the NRC review. The additional information requested is documented in a Request for Additional Information (RAI) dated January 8, 2002, (Reference 2).

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Attachment 1 to this letter contains PPL's responses to the NRC Request for Additional Information (Reference 2). Attachment 2 provides revised markups and camera ready copies of Technical Specification SR 3.4.3.1 for both Units 1 and 2. The Technical Specification SR 3.4.3.1 bases changes were provided for information in Attachment 3 to Reference 1.

There is no change to the No Significant Hazards Considerations provided in Reference 1 as a result of this supplemental information. This proposed amendment does not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
- Create the possibility of a new or different kind of accident from any previously analyzed; or
- Involve a significant reduction in Margin of Safety.

We trust that this information is sufficient for NRC to complete its review by March 1, 2002 in order that the new setpoint tolerances may be utilized during the upcoming Refueling and Inspection Outage. If you have any questions, please contact Mr. D. L. Filchner at (610) 774-7819.

Sincerely,

R. G. Byram

Attachment

cc: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. D. S. Collins, NRC Project Manager

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

SUPPLEMENT TO PROPOSED AMENDMENT NO. 244 TO LICENSE NPF-14: REVISE MAIN STEAM RELIEF VALVE SETPOINT TOLERANCE AND REQUESTS FOR RELIEF FROM IST AND ASME CODE REQUIREMENTS
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files a supplement to Proposed Amendment No. 244 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:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
This 5th day of February, 2002.

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

SUPPLEMENT TO PROPOSED AMENDMENT NO. 208
TO LICENSE NPF-22: REVISE MAIN STEAM RELIEF VALVE SETPOINT TOLERANCE AND REQUESTS FOR RELIEF FROM IST AND ASME CODE REQUIREMENTS
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files a supplement to Proposed Amendment No. 208 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:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 5th day of February, 2002.

Notary Public

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

Attachment 1 to PLA-5430

Response to RAI Questions

Attachment 1 – Response to RAI Questions

NRC Question 1:

Discuss any differences in the Susquehanna Steam Electric Station, Units 1 and 2 (SSES-1 and 2), core designs and their effects on the transient analysis of abnormal operational occurrences, and the analysis of the design-basis overpressurization event when the analyses are performed using a ± 3% setpoint tolerance for the main steam relief valves (MSRVs).

PPL Response:

The Susquehanna Unit 1 and 2 core designs and analyses of abnormal operational occurrences are performed using Framatome-ANP and PPL approved methods listed in Section 5.6.5 of the Technical Specifications. The Unit 1 and 2 cores are designed with a conventional scatter-loading pattern and are designed for 24 month operating cycles. It should be noted that the next cycle for Unit 1 starting in Spring 2002 is scheduled to contain 764 Framatome-ANP ATRIUM-10 Fuel Assemblies. The core composition for each unit follows:

Unit	Current Cycle	Loaded Fuel Designs
1	12	200 Framatome-ANP 9x9-2 Fuel Assemblies 564 Framatome-ANP ATRIUM-10 Fuel Assemblies
2	11	764 Framatome-ANP ATRIUM-10 Fuel Assemblies

Since the Unit 1 and 2 core designs are very similar (i.e., 24 month cycles, scatter loaded core, and majority of ATRIUM-10 fuel assemblies) and the analyses are performed using the same methods, the analysis response to abnormal operational occurrences is similar for Unit 1 and Unit 2. The small differences that are observed in the analysis response are due to small differences in the burn-up history of the two units.

For abnormal operational occurrences in which the safety function of the main steam relief valves (MSRVs) is important (e.g., ASME overpressurization and small break LOCA), a ±3% setpoint tolerance has been traditionally used at Susquehanna because it is conservative and it was recognized that the setpoint tolerance might be changed to ±3%.

NRC Question 2:

a. How many MSRV's are taken credit for in the analysis of the design-basis overpressurization event?

b. Discuss the results of the design-basis overpressurization event analysis when conducted using a ± 3% setpoint tolerance for the MSRVs.

c. Discuss/compare the determination of the limiting transient for 1% and 3% setpoint tolerances.

PPL Response:

a) 12 MSRVs are taken credit for in the analysis of the ASME overpressurization event. This is consistent with TS 3.4.3.

b) The ASME overpressurization event has been traditionally performed using a ±3% setpoint tolerance. The results for the current cycles are:

Unit	Cycle	Peak Vessel Pressure
1	12	1349 psia
2	11	1348 psia

The results show that with a ±3% MSRV setpoint tolerance, there is still significant margin to the design limit of 1389.7 psia.

c) For the ASME overpressurization transient, all abnormal operational occurrences that result in significant reactor pressurization were evaluated. The events evaluated were: failure of feedwater system to high demand, turbine trip without bypass, generator load reject without bypass, and closure of all 4 main steam isolation valves (MSIVs). The evaluation was performed using a ±3% setpoint tolerance for the MSRVs. The evaluation demonstrated that the closure of all 4 MSIVs results in the worst overpressurization transient. Since Susquehanna has traditionally assumed a ±3% setpoint tolerance for the MSRVs, an evaluation of the limiting ASME overpressurization transient using a ±1% setpoint tolerance has not been performed.

NRC Question 3:

Attachment 1 to the application, Section III. a notes that, "the fuel physics cycle calculation for Loss of Pressure Control for Rod Withdrawal Error performed in accordance with Nuclear Fuels Instruction ...[is] analyzed using the relief mode of the MSRVs". Discuss this analysis and its relevance to the requested amendment.

PPL Response:

The Rod Withdrawal Error event is defined as the erroneous withdrawal of a high worth control rod from the reactor. The withdrawal of this control rod leads to an increase in reactor power and steam flow. If the bypass valves are inoperable, a loss of pressure control may occur if the increase in steam flow from the rod withdrawal exceeds the steam relieving capability of the turbine control valves. If this situation occurs, reactor pressure will increase. The increase in reactor pressure introduces positive reactivity and reactor power will begin to increase. This results in a positive feedback loop with reactor pressure and power increasing. This will continue until either a new steady state is reached at a higher power and pressure, or the reactor scrams on high-pressure (1107.7 psia from TS 3.3.1.1). Since this process is slow, after the scram is received, reactor pressure and power decrease rapidly and the reactor pressure never reaches the MSRV setpoints from TS 3.4.3. Therefore, changing the MSRV setpoint tolerances to ±3% has no effect on the Rod Withdrawal Error analysis.

NRC Question 4:

Attachment 1 to the application, Section III.b summarizes the General Electric, NEDC-31753P, "BWROG In Service Pressure Relief Technical Specification Revision Licensing Topical Report, Loss-of-Coolant Accident (LOCA) evaluation. However no plant-specific discussion for SSES-1 or 2 is provided. Although the Nuclear Regulatory Commission (NRC) staff has previously reviewed and approved NEDC-31753P, the NRC notes that the topical report is based on General Electric (GE) fuel core designs whereas PPL Susquehanna, LLC (PPL), is not currently using GE fuel. Therefore, provide plant-specific LOCA evaluations for SSES-1 and 2 core designs, or, provide information to demonstrate that the NEDC-31753P LOCA evaluation is applicable to non-GE fuel core designs.

PPL Response:

Framatome-ANP performed LOCA analyses for the ATRIUM-10 fuel. These analyses were performed in accordance with the approved methods listed in Section 5.6.5 of the Technical Specifications. The ATRIUM-10 analyses were performed assuming a ±3% setpoint tolerance for the MSRVs. The results of the analyses demonstrate compliance with the acceptance criteria. Therefore, the LOCA analyses in place for Susquehanna

were performed per approved methodology and support a ±3% setpoint tolerance for the MSRVs.

NRC Question 5:

The application states that the requested ± 3% MSRV setpoint tolerance will apply only to as-found testing and that as-left testing would still be subject to the current ± 1% tolerance. The proposed technical specifications (TSs) do not make this distinction between the as-found and as-left acceptance criteria clear. Provide revised markups that clearly specify the differences between the allowable tolerances for the as-found and as-left testing.

PPL Response:

Revised markups of SR 3.4.3.1 for both Units 1 and 2 along with camera ready copies are provided in Attachment 2. This revision incorporates a footnote which establishes the as-left setting tolerance. The footnote is consistent with NUREG 1433 BWR /4 Standard Technical Specifications.

NRC Question 6:

As discussed in a conference call between the NRC staff and PPL staff on November 20, 2001, the American Society of Mechanical Engineers (ASME) Code Relief Request for high pressure coolant injection (HPCI) main pump discharge piping is not an appropriate mechanism for PPL to make the requested change. The requirements of Title 10 of the Code of Federal Regulations (10CFR), Section 50.55a, for the design of ASME Code Class 2 components including piping were incorporated into the regulations after the date that the SSES-1 and 2 Construction Permit applications were docketed and, thus, do not apply to the design of SSES-1 and 2. The proposed change must be evaluated under 10 CFR 50.59. If the proposed change is found to meet any of the criteria in 10 CFR 50.59 (c)(2), then a license amendment shall be obtained.

In light of the above discussion, the NRC staff notes that, with regard to the HPCI main pump discharge line, Attachment 1 to the application, Section III.f, merely references the relief request and provides no other evaluation or justification for the requested change. In order to complete our evaluation of the related proposed TS change, provide the results of PPL's 50.59 evaluations of the proposed change with respect to the HPCI main pump discharge line, and discuss whether a license amendment is required for the HPCI main pump discharge line.

PPL Response:

Based on discussions between PPL and the NRC staff, PPL concurs that the ASME Code Relief Request for HPCI Main Pump Discharge Piping, Attachment 6 to PLA-5377, (Reference 1) is not required and is hereby withdrawn. Accordingly, the last paragraph in Section III.f "HPCI and RCIC Systems" on page 6 of Attachment 1 to PLA-5377 (Reference 1) is revised as follows:

Calculations have been performed to determine the RCIC and HPCI Main and Booster pump discharge pressures for operational conditions with an MSRV setpoint tolerance of +3%. The RCIC and HPCI system discharge piping was evaluated against the requirements of the ASME Code. All sections of the HPCI Booster and RCIC pump discharge lines meet the requirements of the ASME Code when Certified Material Test Reports (CMTR) data is used to meet the Class 2 hoop stress equation for the HPCI main pump discharge line.

For the HPCI main pump discharge line, recalculation using allowable stress based on Certified Material Test Reports (CMTR), of the installed materials, to meet the ASME Class 2 hoop stress equation was required. Based on using the CMTR values to determine the higher calculated allowables, ASME Section III NC-3641.1 Equation 4 was met. All other applicable ASME Code equations were met using allowable values specified in the Code. The higher allowables calculated for the HPCI main pump discharge line were screened per the 50.59 process and determined to meet the ASME code requirements.

The following discussion provides the results of the 50.59 screening which determined that prior NRC approval is not required for this change:

During preparation of calculations supporting the Main Steam Safety Relief Valve setpoint tolerance change, it was determined that the HPCI main pump discharge piping could experience a maximum pressure of 1583 psig. The piping was qualified to 1360 psig. This piping is designed to ASME Section III Class 2, 1971 Edition through and including the Winter 1972 Addenda.

Design requirements for Class 2 components are specified in Subsection NC, Article NC-3000. Paragraph NC-3641.1 specifies the minimum wall thickness requirements for piping subjected to internal pressure. NC-3641.1 Equation 4 is used to calculate the maximum allowable design pressure using a defined minimum wall thickness. Equations 3 and 4 require the use of the code allowable stress (S) as specified in Tables I-7.1, I-7.2, and I-7.3 for the respective material and design temperature.

The HPCI main pump discharge piping, designated as EBB-102/202, is SA-106 Grade B material and the fittings are SA-234 Grade WPB material. The maximum allowable stress (S) for SA-106 Grade B piping and SA-234 Grade WPB fittings at a design temperature of 220 degrees F is 15000 psi, per Table I-7.1. However, as specified in NC-3612.3, if the maximum pressure occurs less than one percent of the time, the allowable stress may be increased by twenty percent. The maximum pump discharge pressure of 1583 psig, as identified above, will not occur more than one percent of the time as this maximum pressure will only occur in the event of a failure of the HPCI injection valve. Closure of the injection valve will dead-head the pump at its maximum speed. Thus, the maximum stress allowed by the code for the HPCI main pump discharge line is 18000 psi, i.e., (15000 x 1.2).

By using the code allowable stress of 18000 psi in Equation 4, the maximum design pressure for the 14" EBB-102/202 HPCI main pump discharge piping is 1532 psig; and the maximum design pressure for the 14"x10" reducers at the HPCI main pump discharge nozzles is 1519 psig. These are lower than the maximum pressure of 1583 psig thus additional analysis was needed.

Certified Material Test Reports (CMTR) for the HPCI main pump discharge EBB piping were retrieved from plant historical records. The lowest recorded values for the yield strength and ultimate strength of the material were 39000 psi and 70000 psi, respectively. Using Paragraph III-3210, an alternate allowable stress of 17500 psi, (rather than 15000 previously assumed) can be applied to the EBB piping.

With the twenty percent increase allowed by NC-3612.3, an allowable stress of 21000 psi, i.e. (18000 x 1.2), instead of 18000 psi, may be used in NC-3641.1 Equation 4. This new allowable stress results in the maximum design pressure for the 14" EBB-102/202 HPCI Main Pump discharge piping of 1788 psig.

Similarly, for the 14"x10" reducers, CMTR data provided values for the yield strength and ultimate strength of 35300 psi and 63420 psi, respectively. Using Paragraph III-3210, an alternate allowable stress of 15855 psi can be applied to the EBB 14"x10" reducers.

With the twenty percent increase allowed by NC-3612.3, an allowable stress of 19026 psi, i.e. (15855 x 1.2), instead of 18000 psi, may be used in NC-3641.1 Equation 4. This new allowable stress results in the maximum design pressure for the 14"x10" reducers, at the HPCI Main Pump discharge nozzles, of 1606 psig.

Thus, for the 14" EBB piping and the 14"x10" reducers, the use of the alternate allowable stress in NC-3641.1 Equation 4 results in a maximum design pressure greater than the increased value of 1583 psig for the HPCI Main Pump maximum discharge pressure.

The 50.59 review determined that this proposed change does not pose an adverse impact on the HPCI system and thus NRC approval of this aspect of the proposed change is not required. The 50.59 screen conclusions are as follows:

1. The proposed activity does not involve a change to a SSC that adversely affects an FSAR described design function. HPCI will perform its FSAR described design function.

2. The proposed activity does not affect any procedures as described in the FSAR for SSC design functions.

3. The proposed activity does not involve replacing an FSAR described evaluation methodology. The FSAR in Table 3.2-1 identifies that the HPCI piping beyond the outermost containment isolation valve used ASME Section III Class 2 as it's principle construction code. The methodology of the ASME Section III Class 2 piping hoop stress design is not described in the FSAR. Since the ASME methodology for Class 2 piping hoop stress is not described in the FSAR and the use of the 1.2 factor and use of CMTR data in conjunction with ASME Code Methology is within the restrictions of ASME Code, this change does not involve a change to a FSAR described evalution method.

4. The proposed activity does not involve any tests or experiments.

5. The proposed activity does not involve a change to the Technical Specifications.

Attachment 2 to PLA-5430
Revised Markups and Camera Ready Copies of SR 3.4.3.1 - Units 1 & 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required S/RVs are as follows: ------------------NOTE------------------ Up to two inoperable required S/RVs may be replaced with spare OPERABLE S/RVs having lower setpoints until the next refueling outage. --- See setpoint table below. Following testing, lift settings shall be within ±1%	In accordance with the Inservice Testing Program

Number of S/RVs	Setpoint (psig)
2	1175 (≥ ~~1164~~ 1140 and ≤ ~~1186~~ 1210)
6	1195 ~~(≥ 1184 and ≤ 1206)~~ (≥1160 and≤1230)
8	1205 (≥ ~~1193~~ 1169 and ≤ ~~1217~~ 1241)

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