March 7, 2002

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISE MAIN STEAM RELIEF VALVE STEAM SETPOINT TOLERANCE (TAC NOS. MB3273 AND MB3274)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. NPF-14 and Amendment No. 175 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 18, 2001, as supplemented February 5, 2002.

These amendments revise the TSs surveillance requirement 3.4.3.1 for testing of the main steam safety relief valves to permit the setpoint tolerance for "as-found" testing to be changed from ±1 percent to ±3 percent. An editorial change will also be made to remove a note regarding an associated relief request.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 201 to License No. NPF-14
2. Amendment No. 175 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PPL Susquehanna, LLC, dated October 18, 2001, as supplemented February 5, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 201 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented during the spring 2002 refueling and inspection outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 7, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.4-9	3.4-9

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the PPL Susquehanna, LLC, dated October 18, 2001, as supplemented February 5, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 175 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented during the spring 2003 refueling and inspection outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 7, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 175

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.4-9	3.4-9

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. NPF-22

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated October 18, 2001, as supplemented February 5, 2002, PPL Susquehanna, LLC (the licensee), submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TSs). The requested changes would revise the TS surveillance requirement (SR) 3.4.3.1 to permit the allowable tolerance for "as-found" testing of the main steam relief valves (MSRVs) safety function lift setpoints to be changed from ± 1 percent to a new tolerance value of ± 3 percent. The proposed changes would not alter the TS requirements on the nominal MSRV safety function lift setpoints, the MSRV relief function setpoints, the required frequency for the MSRV lift setpoint testing, nor the number of MSRV's currently required to be operable. The requirements for testing of the tolerances associated with "as-left" testing will remain unchanged. The February 5, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design," requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The SSES Units 1 and 2 MSRVs were originally purchased to the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section III, "Nuclear Vessels," up to and including the Summer 1970 Addenda. The designed setpoint tolerance for the safety setting was ±1 percent for construction based upon ASME Code, Section III, Article NB-7000, Summer 1971 Addenda. The SSES Units 1 and 2 TS testing requirements are based upon this design code.

The U.S. Nuclear Regulatory Commission (NRC) staff has previously approved for several licensees of boiling-water reactor (BWR) plants requests to increase the as-found SRV tolerance value to ±3 percent. The increased setpoint tolerance is based on the Licensing

Topical Report (NEDC-31753P), "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," developed by the BWR Owners' Group (BWROG). NEDC-31753P was developed as a result of a generic problem in the industry with the use of ±1 percent allowable as-found MSRV safety function lift setpoint tolerance in plant TSs. The BWROG developed NEDC-31753P to support the use of a ±3 percent setpoint tolerance which is consistent with ASME Code, Section XI, requirements. By letter to the BWROG, "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'" dated March 8, 1993, the NRC staff approved use of the licensing topical report with the following six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(1) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P should be performed utilizing a ±3 percent tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the NRC staff) should be used in this analysis.

(2) Analysis of the design-basis over-pressurization event using the ±3 percent tolerance limit is required to confirm that the vessel pressure does not exceed the ASME Code upset limit.

(3) The plant-specific analysis described in items (1) and (2) should assure that the number of SSVs and SRVs, and relief valves (RVs) included in the analysis corresponds to the number of valves required to be operable in the TSs.

(4) Reevaluation of the performance of high-pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping must be completed considering the ±3 percent tolerance limit.

(5) Evaluation of the ±3 percent tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(6) Evaluation of the effect of the ±3 percent tolerance limit on the containment response during loss-of-coolant accidents (LOCAs) and on the hydrodynamic loads associated with the SRV discharge lines and containment should be completed.

In the October 18, 2001, letter, as supplemented by the February 5, 2002, letter, the licensee stated that the ASME Code requires the reactor pressure vessel to be protected from overpressure during upset conditions by self-actuated safety valves. The purpose of the nuclear pressure relief valve system is to prevent over-pressurization of the reactor vessel boiler system during abnormal operational transients. The MSRVs provide a capability to reduce the reactor vessel pressure sufficiently to enable the use of the low-pressure core cooling systems to maintain water level and provide core cooling. As part of the nuclear pressure relief system, the size and number of MSRVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. This protects the primary system process barrier from failure which could result in the release of fission products.

The MSRVs are located in the main steam lines between the reactor pressure vessel and the first isolation valve within the drywell. There are a total of 16 MSRVs of which any 12 are required to be operable. The MSRVs provide three main protection functions, which are: the overpressure relief mode, the overpressure safety mode, and the automatic depressurization operation. In the safety mode (or the spring mode of operation), the valves open when steam pressure at the valve inlet overcomes the spring forces holding the valve closed. This mode satisfies the ASME Code requirement. It is this mode of operation for which the tolerance will be relaxed from ±1 percent to ± 3 percent. The relief and automatic depressurization modes rely upon solenoid actuation to open the valve and are not affected by this change.

In addition, the licensee stated that the proposed revision implements a higher MSRV setpoint tolerance to better match the TS performance requirements with the installed valve capabilities. The licensee also stated that the proposed revision will reduce the number of Licensee Event Reports written due to MSRV as-found setpoints. The proposed revision would increase the allowable MSRV safety function spring setpoint tolerance from ±1 percent to ± 3 percent. The proposed change does not alter the TS requirements on the nominal MSRV safety function lift setpoints, the MSRV relief function setpoints, the required frequency for the MSRV lift setpoint testing, nor the number of MSRVs currently required to be operable. This proposed revision does not change the requirement that the MSRVs be adjusted to within ±1 percent of their nominal lift setpoint following testing as required in the MSRV Inservice Inspection Program.

3.0 EVALUATION

The NRC staff reviewed the licensee's letter dated October 18, 2001, as supplemented by its letter dated February 5, 2002.

The safety function of the SRVs is, in part, to prevent overpressurization of the reactor coolant pressure boundary. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of the fission products. The pressure relief system at SSES Units 1 and 2 includes 16 MSRVs arranged into three setpoint groupings: one group consisting of 2 MSRVs set at 1175 psig, a second group consisting of 6 MSRVs set at 1195 psig, and a third group consisting of 8 MSRVs set at 1205 psig. The existing TS provides a ±1 percent as-found setpoint tolerance and a ±1 percent as-left setpoint tolerance. The proposed modification would provide a ±3 percent as-found setpoint tolerance and a ±1 percent as-left setpoint tolerance. The proposed change does not alter the TS requirements on the nominal MSRV safety function lift setpoints, the MSRV relief function setpoints, the required frequency for the MSRV lift setpoint testing, nor the number of MSRV's currently required to be operable. The licensee's submittals were evaluated for consistency with the six conditions discussed in the NRC staff's March 8, 1993, safety evaluation (SE) described above.

3.1 Transient Analysis/Reload Methodology

To address the first condition, the licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). Transient analysis of all abnormal operational occurrences as described in NEDC-31753P are not required for plants with dual-mode MSRVs because licensing events are evaluated by taking credit for the relief mode of MSRV actuation. The Nuclear Fuels Engineering fuel cycle design calculations/analysis and the SSES Units 1 and 2 Final Safety Analysis Report (FSAR) assume the functional relief mode of the MSRVs.

Therefore, the abnormal operational occurrences for SSES Units 1 and 2 are not affected by this proposed change in safety setpoint tolerance.

Fuel cycle calculations for the following design-basis transient events are described in Chapter 15 of the SSES Units 1 and 2 FSAR: Generator Load Rejection without Bypass, Feedwater Controller Failure, and Recirculation Flow Controller Failure are completed by Nuclear Fuels in accordance with Nuclear Fuels Instructions. All analyses are performed using the relief mode of the MSRVs and are not affected by the increased MSRV safety setpoint tolerance.

The SSES Unit 1 and 2 core designs and analysis of abnormal operational occurrences are performed using Framatome-ANP- and PPL-approved methods listed in Section 5.6.5 of the TSs. The Units 1 and 2 cores are loaded with a majority of ATRIUM-10 fuel assemblies and are designed for 24-month operating cycles. It should be noted that the next cycle for Unit 1 starting spring 2002 is scheduled to contain 764 Framatome-ANP ATRIUM-10 Fuel Assemblies. Because the Unit 1 and 2 core designs are similar (i.e., 24-month cycles, scatter-loaded core, and majority of ATRIUM-10 fuel assemblies) and the analyses were performed using the same methods, the analysis response to abnormal operational occurrences is also similar for Units 1 and 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 MSRVs is important (i.e., ASME overpressurization and small break LOCA), a ±3 percent setpoint tolerance has been used at SSES Units 1 and 2 because it is conservative, and it was recognized that setpoint tolerance might be changed to ±3 percent.

The fuel physics cycle calculation for loss of pressure control for a rod withdrawal error event was performed. 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). Because 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 tolerance to ±3 percent has no effect on the rod withdrawal error analysis.

3.2 Analysis of Design-Basis Over-Pressurization Event

To address the second condition, the licensee is required to reevaluate the limiting design pressurization transient using the ±3 percent tolerance limit to confirm that the vessel pressure does not exceed the ASME Code upset limit.

ASME Code, Section III, permits pressure transients up to 10 percent over the design pressure. The limiting pressurization AOT analyzed by the licensee is the simultaneous closure of all main steam isolation valves (MSIVs) followed by a scram on high neutron flux, i.e. a failure of the direct scram on MSIV closure. This transient is analyzed for each fuel reload using the

NRC-approved RETRAN System Model in accordance with Nuclear Fuels Procedures. The limiting design-basis pressurization transient has been evaluated for each refuel cycle since a power uprate using the ±3 percent MSRV setpoint tolerance to confirm that the vessel pressure does not exceed the ASME Code limit. The ASME Code over-pressurization event has been performed using a ±3 percent setpoint tolerance. The results are: (1) peak vessel pressure for Unit 1 is 1349 psia for the 12th fuel cycle, and (2) peak vessel pressure for Unit 2 is 1348 psia for the 11th fuel cycle. The results show that with a ±3 percent MSRV setpoint tolerance, there is still significant margin to the design limit of 1389.7 psia.

3.3 TS Operability Statement for SRVs

To address the third, the licensee is required to assure that the number of MSRVs included in the analysis of conditions (1) and (2) above corresponds to the number of MSRVs required to be operable in the SSES Units 1 and 2 TSs.

The MSRVs provide three main protection functions, which are: the overpressure relief mode, the overpressure safety mode, and the automatic depressurization operation. It is the overpressure safety mode that relies on the spring setting, for which the tolerance will be relaxed from ±1 percent to ±3 percent. The relief or automatic depressurization modes, which rely upon solenoid actuation, are unaffected by this change. The licensee has stated that all plant-specific analyses have been conducted with the number of MSRVs included in the analyses corresponding to the number of valves required to be operable in the TSs. The analysis took credit for only 12 of the 16 MSRVs, which is the required number of operable MSRVs in the TSs. This is acceptable to the NRC staff.

The required number of operable MSRVs has been determined for reactor operation using the MSRV setpoint tolerance of ±3 percent. The number of operable valves (12) as assumed in the analysis corresponds to the number of operable valves currently required in TS Limiting Conditions for Operation (LCO) 3.4.3. Therefore, no change is required to TS 3.4.3 to account for the increased MSRV setpoint tolerance of ±1 percent to ±3 percent.

3.4 Reevaluation of the Performance of High-Pressure Systems

3.4.1 System Performance

To address the fourth condition, the licensee must reevaluate performance of the high-pressure system (pump capacity, discharge pressure, etc.), considering the ±3 percent tolerance. Generic evaluations were performed by General Electric (GE) in BWROG Licensing Topical Report NEDC-31753P and reviewed by Brookhaven Nation Laboratory (BNL) for the NRC to study the effects of higher vessel pressure upon components and systems. Components and systems addressed in the evaluation included: MSIV timing specification, CRD pump injection, recirculation pump seal leakage and feedwater coolant injection. The GE and BNL evaluations determined that the effect of a higher MSRV opening pressure is insignificant to the operation of these systems and components. These evaluations were reviewed and determined to be applicable to SSES Units 1 and 2 systems.

High-Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems:

The HPCI and RCIC systems are designed to inject into the reactor pressure vessel at the lowest MSRV setpoint pressure plus the setpoint tolerance. The licensee evaluated systems' performance for increasing the reactor pressure from the lowest MSRV setpoint +1 percent to the lowest MSRV setpoint +3 percent. Evaluation and analyses have determined that the HPCI and RCIC systems are capable of delivering the required flow to the reactor pressure vessel at the higher steam-dome pressure. The increased turbine operating speeds will result in a slight reduction in margin to the mechanical overspeed trip setpoints. The RCIC system has an electronic overspeed trip that will be adjusted by a plant modification to 110 percent of the new turbine-rated speed. The HPCI system does not have an electronic overspeed trip. The reliability of both systems is maintained because of the relatively large speed margin which was originally specified for the turbines and also because of improvements in the turbine start-up control logic implemented through modifications in response to GE Service Information Letter 377. These modifications, refered to as “bypass start” modifications, limit the initial peak speed during system startup, thus reducing the possibility of turbine overspeed trip.

The licensee performed a calculation to determine the RCIC and HPCI main and booster pump discharge pressures for operational conditions with an MSRV setpoint tolerance of +3 percent. 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 report (CMTR) data is used to meet the ASME Code, Class 2, hoop stress equation for the HPCI main pump discharge line.

For the HPCI main pump discharge line, a recalculation was performed by using an allowable stress, based on the CMTR of the installed materials, to meet the ASME Code, Class 2, hoop stress equation. The licensee used CMTR values to establish an increased allowable stress limit for ASME Code, Class 2, piping in order to meet ASME Code, Section III, NC-3641.1, Equation 4. All other applicable ASME Code, Section III, equations were met using allowable values specified in the ASME Code. The higher allowable limits that were calculated for the HPCI main pump discharge line were evaluated by the licensee in accordance with the 10 CFR 50.59 process, and the licensee determined that the changes did not require a license amendment pursuant to 10 CFR 50.90.

Standby Liquid Control (SLC) System:

The current SSES Units 1 and 2 anticipated transient without scram (ATWS), analysis methodology includes evaluations for several scenarios, as required by NUREG-0460. With the exception of the loss-of-offsite power (LOOP) event, the pneumatically-assisted relief mode of the MSRVs is credited. For the LOOP event, the relief mode is not available because the loss of AC power causes the loss of the containment instrument gas system, which provides the pneumatic-motive force to open the MSRVs in the relief mode. Under these circumstances, the MSRVs are postulated to lift at higher nominal spring setpoints. The current analysis demonstrates that during this event, vessel pressure is controlled to the second MSRV setpoint of 1195 psig. Because the valves are postulated to lift at the nominal setpoint of 1195 psig, this is the SLC design-basis steam dome pressure, and is not affected by the proposed increase in the allowable MSRV tolerance.

The licensee states that the SLC system and pump design pressures were based on a vessel pressure corresponding to the lower setpoint MSRV relief mode of operation. Initially these pressures did not account for the ATWS-LOOP event described above. Therefore, this condition was entered into the SSES Units 1 and 2 Corrective Action Program, and design modifications with a discharge relief setpoint of 1500 psig have been already developed to increase the SLC system pressure. The licensee states that for Unit 2, this was modified during the spring 2001 Refueling Outage, whereas, for Unit 1, the installation of this modification shall be performed during the spring 2002 Refueling Outage.

Feedwater System:

The feedwater piping (ASME Code, Class 1) was evaluated against the ASME Code and other standards for the MSRV setpoint tolerance of ±3 percent. The additional pressure/loads induced by the proposed increase in the MSRV tolerance are bounded by the existing design capability of all potentially affected feedwater system piping and components.

3.4.2 Evaluation of Motor-Operated Valves

The impact of increased reactor pressure on HPCI and RCIC motor-operated valves (MOVs) was evaluated using the Generic Letter 89-10 program. The affected valves are HV-1(2)50F045 (RCIC Turbine Steam Admission valves), HV-1(2)49F013 (RCIC Injection Shutoff valves), HV-1(2)50F046 (RCIC Turbine Cooling Water Supply valves), FV-1(2)49F019 (RCIC Pump Minimum Flow Bypass valves), HV-1(2)55F001 (HPCI Turbine Steam Admission valves), HV-1(2)55F006 (HPCI Injection Shutoff valves), HV-1(2)56F059 (HPCI Lube Oil Cooler Cooling Water Supply valves), and HV-1(2)55F012 (HPCI Pump Minimum Flow Bypass valves). Calculations show that these valves will see an increased maximum differential pressure, and that with the increased differential pressure, the MOV operation will not be affected by the increase in MSRV setpoint tolerance.

3.4.3 Vessel Instrumentation

The vessel instrumentation was evaluated against ASME Code and other standards for the MSRV setpoint tolerance of ±3 percent. The additional pressure and loads induced by the proposed increase in the MSRV tolerance are bounded by the existing design capability of all potentially affected vessel instrumentation.

3.5 Alternate Operating Modes

To address the fifth condition, the licensee must evaluate the impact of the ±3 percent tolerance on any plant-specific operating modes.

The SSES Units 1 and 2 Increased Core Flow, Extended Load Line Limit, and Single Loop Operation operating modes have been reviewed and MSRV tolerance is not used as an input in any potentially affected analyses. There is one event in which the MSRVs are assumed to lift at their safety setpoints; the low power generator load reject without bypass. However, this analysis already accounts for the proposed increase in MSRV tolerance.

3.6 Containment Response during LOCA/Hydrodynamic Loads

Finally, to address the sixth condition, the licensee must evaluate the effect of the ±3 percent tolerance limit on the containment response during a LOCA and on hydrodynamic loads.

3.6.1 Loss of Coolant Accident

LOCA evaluations are detailed in NEDC-31753P and are summarized below.

In a large break LOCA, the reactor vessel de-pressurizes rapidly through the break. Because the vessel immediately de-pressurizes, no MSRV actuation will occur. Therefore, an increase in MSRV opening pressure has no impact on the limiting break LOCA analysis.

For small-break LOCAs, inventory loss and vessel depressurization occurs more slowly than the large-break LOCA. The vessel may remain within the normal operating pressure and, upon vessel isolation, might pressurize. An increase in the MSRV's setpoint tolerance may result in a slight delay in MSRV actuation. When the MSRV's actuate at a higher vessel pressure, the instantaneous flow rate out of the MSRV is increased due to higher critical flow rates. However, the total inventory lost from the vessel when the MSRV actuates at a higher pressure is not significantly changed from operation at lower MSRV setpoint tolerance. The impact of increased MSRV setpoint tolerance on the small LOCA is insignificant.

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 TSs. The ATRIUM-10 analyses were performed assuming a ±3 percent setpoint tolerance for the MSRVs. The results of the analyses meet the acceptance criteria. Therefore, the LOCA analyses for SSES Units 1 and 2 were performed using an approved methodology and support a ±3 percent setpoint tolerance for the MSRVs.

3.6.2 Hydrodynamic Loads

Fatigue analysis of the MSRV discharge lines between the flued head penetration at the diaphragm slab and the suppression pool quenchers was completed using a ±3 percent MSRV setpoint tolerance. The MSRV hydrodynamic loads were analyzed using the increased MSRV setpoint tolerance of ±3 percent. Both analyses determined that additional pressure and loads induced by the proposed increase in the MSRV tolerance are bounded by the existing design capability of the containment and MSRV discharge lines.

3.6.3 Conclusion

Based on its review of the information provided by the licensee, the NRC staff concludes that the plant will continue to satisfy the acceptance criteria for the limiting pressurization transient, AOTs and design-basis accident. In addition, the NRC staff finds the licensee's evaluation of the six conditions, which must be addressed on a plant-specific basis by licensees applying for increased SRV setpoint tolerance, acceptable. Therefore, the NRC staff concludes that the proposed MSRVs setpoint tolerance changes from ±1 percent to ±3 percent in the TSs and TS Bases are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 59511). Accordingly, the amendments meet eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: G. S. Bedi

Date: March 7, 2002

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

March 7, 2002

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISE MAIN STEAM RELIEF VALVE STEAM SETPOINT TOLERANCE (TAC NOS. MB3273 AND MB3274)

Dear Mr. Byram:

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A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 201 to License No. NPF-14
2. Amendment No. 175 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

PUBLIC	MO'Brien	DTerao	TTate
PDI-1 Reading	TColburn	WBeckner	BPlatchek, RGN-1
EAdensam	OGC	ACRS	GBedi
JMunday	G. Hill(4)	CCowgill, RGN-I	

** see previous concurrence

Package: ML020740610 **Amendment TSs: ML**020670007

ACCESSION NO.: ML020520018 *safety evaluation dated 2/15/02, no major changes made

OFFICE	PDI-2/PM	PDI-1/PM	PDI-2/LA	EMCB\SC*	OGC**	PDI-1/(A)SC
NAME	TTate	TColburn	MO'Brien	DTerao	RHoefling	JMunday
DATE	3/6/02	3/6/02	3/6/02	SE dtd 2/15/02	3/04/02	3/5/02

OFFICIAL RECORD COPY

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