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U.S. Nuclear Regulatory Commission
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Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 244 TO LICENSE
NFP-14 AND PROPOSED AMENDMENT NO. 208 TO
LICENSE NFP-22: REVISE MAIN STEAM RELIEF VALVE
SETPOINT TOLERANCE AND REQUESTS FOR RELIEF
FROM IST AND ASME CODE REQUIREMENTS
PLA-5377**

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

Reference: 1) *NRC Safety Evaluation by the Office of Nuclear Reactor Regulation Related to An Inservice Testing Request for Relief, Susquehanna Steam Electric Station Unit 1, dated April 7, 1998.*

2) *NRC Safety Evaluation by the Office of Nuclear Reactor Regulation Related to An Inservice Testing Request for Relief, Susquehanna Steam Electric Station Unit 2, dated December 16, 1998.*

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Units 1 and 2 Technical Specifications. These changes revise the allowable tolerances for the safety function lift setpoints of each group of the Main Steam Relief Valves (MSRV's) in Technical Specification SR 3.4.3.1. The allowable MSRV safety function spring setpoint tolerance is being increased from the present value of ± 1% to ±3%. This change does not alter the Technical Specification requirements on the nominal MSRV safety function lift setpoints, the MSRV relief function setpoints, the required frequency for the MSRV lift setpoint testing, or the number of MSRV's currently required to be operable. It merely changes the criteria for the "As Found" setpoint tolerance to be ± 3%, it does not affect the ± 1% "As Left" tolerance which follows testing. This change in MSRV setpoint tolerance has been approved at other BWR's which include Vermont Yankee, Duane Arnold, LaSalle, Limerick, and Hope Creek.

Also included with this letter is a revision to IST Relief Request Number 34 for each Susquehanna Unit. Relief Request 34 was originally submitted to the NRC to request approval of PPL's alternative proposal to Code test requirements to allow the MSRV's to be tested within three 24-month fuel cycles (or 6 years) rather than a 5 year period. This revision is necessary because the NRC originally established acceptance of Relief Request 34 based on a MSRV setpoint tolerance of ± 1% as documented in the Safety Evaluations dated April 7, 1998 and December 16, 1998 (References 1 & 2). These

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SER's required that any future change to the ± 1% tolerance would require resubmission of the relief request because the basis for staff acceptance would no longer be applicable.

Finally, a Relief Request from ASME Code Pressure Design requirements for a portion of the HPCI main pump discharge piping is also included. The purpose of this relief request is to allow the use of an alternate allowable stress, determined in accordance with ASME Section III Paragraph III-3210, to qualify that portion of the piping for postulated system transients associated with the +3% setpoint tolerance.

Attachment 1 to this letter is the "Safety Assessment" supporting this change. Attachment 2 to this letter contains the "No Significant Hazards Considerations Evaluation" performed in accordance with the criteria of 10CFR 50.92 and the categorical exclusion for an Environmental Assessment as specified in 10CFR 51.22. Attachment 3 to this letter contains the current pages of the Susquehanna SES Units 1 and 2 Technical Specifications and Technical Specification Bases marked to show the proposed changes. Attachment 4 to this letter is the "camera ready" version of the revised Technical Specification pages. Attachment 5 contains the revised IST Relief Request Number 34 for each Susquehanna unit. Attachment 6 to this letter is the ASME Code Relief Request applicable to the HPCI Main Pump Discharge Piping on both Units 1 and 2.

The Susquehanna SES Plant Operations Review Committee and the Susquehanna Review Committee have reviewed the proposed changes.

PPL plans to implement the proposed changes to support MSRV as-found testing during the Unit 1 Refueling and Inspection Outage scheduled to begin in the Spring of 2002 and the Unit 2 outage in the Spring of 2003. Therefore, we request NRC to complete the review of this change by March 1, 2002 to support implementation for Unit 1.

If you have any questions, please contact Mr. D. L. Filchner at (610) 774-7819.

Sincerely,

R. G. Byram

Attachment

copy: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Project Manager

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

**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 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 18th day of October, 2001.

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

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 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 19th day of October, 2001.

Notary Public

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

Attachment 1 to PLA-5377

Safety Assessment

SAFETY ASSESSMENT

Section I

Summary of Proposed Change

The action proposed revises SSES Unit 1 and Unit 2 Technical Specification (TS) Surveillance Requirements 3.4.3.1 and the associated bases. The proposed revision implements a higher Main Steam Safety/Relief Valve (MSRV) setpoint tolerance to better match the TS performance requirements with the installed valve capabilities. This will reduce the number of Licensee Event Reports written due to MSRV as-found setpoints. The intended change increases the allowable MSRV safety function spring setpoint tolerance from ±1% to ±3%. 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, or the number of MSRVs currently required to be operable. This proposed action does not change the requirement that the MSRVs be adjusted to within ±1% of their nominal lift setpoints following testing as required in the MSRV In-service Inspection Procedures.

The use of ±1% allowable as-found MSRV safety function lift setpoint tolerance in plant Technical Specifications was a generic problem in the industry. As a result, the BWR Owners' Group (BWROG) developed NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to support the use of ±3% setpoint tolerance which is consistent with ASME Section XI requirements. On March 8, 1993, the NRC Staff issued their Safety Evaluation (SER) of Licensing Topical Report NEDC-31753P. In the SER, the NRC stated that a generic change of setpoint tolerance to ±3% is acceptable provided that it is evaluated in the analytical bases. Specific analyses required to be provided are transient analysis, design basis overpressurization event, re-evaluation of high pressure systems, (Motor Operated Valves, Reactor Vessel instrumentation and piping), alternate operating modes, containment response during LOCA, and hydrodynamic loads on MSRV discharge lines.

The results of these plant specific analyses are discussed in Section III.

Section II

Description and Basis (Both Licensing and Design) of Current Requirements

The purpose of the nuclear pressure relief system is to prevent over-pressurization of the Reactor Vessel boiler system during abnormal operational transients. This protects the primary system process barrier from failure which could result in the release of fission products.

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel to be protected from overpressure during upset conditions by self-actuated safety valves. 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 (RCPB).

The MSRVs are located in the main steam lines between the reactor 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 force 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% to ±3%. The relief and automatic depressurization modes rely upon solenoid actuation to open the valve and are not affected by this change.

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.

The SSES MSRVs were originally purchased to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, up to and including Summer 1970 Addenda. The designed setpoint tolerance for the safety setting was ±1% for construction based upon ASME Section III, Summer 1971, Article NB-7000. The Technical Specification testing requirements were based upon this design code.

Section III

Safety Analysis

The NRC Staff issued a SER on the BWROG Licensing Topical Report NEDC-31753P. In the SER, the NRC stated that a generic change of setpoint tolerance to ±3% is acceptable provided that it is evaluated in the analytical bases. The NRC indicated in the SER that licensees planning to implement these Technical Specification changes to increase the MSRV setpoint tolerance should provide the following plant specific analyses:

1. Transient analysis of all abnormal operational occurrences as described in NEDC-31753P, should be performed utilizing a ±3% setpoint tolerance for the safety mode of MSRVs. In addition, the standard reload methodology (or other method approved by the Staff) should be used for this analysis.

2. Analysis of the design basis overpressurization event using the ±3% tolerance for the MSRV setpoint is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

3. The plant specific analysis described in Items 1 and 2 should assure that the number of MSRVs included in the analysis corresponds to the number of valves required to be operable in the Technical Specifications.

4. Re-evaluation of high pressure systems (pump capacity, discharge pressure, etc.) motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the ±3% setpoint tolerance.

5. Evaluation of the ±3% 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% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the MSRV discharge lines and containment should be completed.

This section describes the SSES specific technical evaluations of the above listed items focusing on the safety function of the affected structures, systems and components and the proposed change from the current requirements and design basis.

III.a ***Abnormal Operational Occurrences***

Transient analyses of all abnormal operational occurrences as described in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," are not required for plants with dual mode MSRVs, since licensing events are evaluated by taking credit for the relief mode of MSRV actuation. Nuclear Fuels Engineering fuel cycle design calculations/analyses and the SSES FSAR assume functional relief mode of the MSRVs, therefore, the abnormal operational occurrences for SSES are not affected by the change in safety setpoint tolerance.

Fuel cycle calculations for the following Design Basis Transients are described in Chapter 15 of the FSAR: Generator Load Reject 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 Fuel Physics cycle calculation for Loss of Pressure Control for Rod Withdrawal Error performed in accordance with Nuclear Fuels Instruction and FSAR Chapter 15 non-limiting events are analyzed using the relief mode of the MSRVs. Therefore, these analyses are not affected by the change in MSRV safety setpoint tolerance.

III.b ***LOCA***

LOCA evaluations are detailed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report" and are summarized below.

In a large break LOCA, the reactor vessel de-pressurizes very 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 near the normal operating pressure and upon vessel isolation may even pressurize. Increase in MSRV setpoint tolerance may result in a slight delay in MSRV actuation. When the MSRVs actuate at the 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 the lower MSRV setpoint tolerance. The impact of increased MSRV setpoint tolerance on the small break LOCA is insignificant.

III.c <u>Design Basis Overpressurization Event and Required Number of MSRVs</u>

The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10% over design pressure. The limiting pressurization abnormal operating transient analyzed is simultaneous closure of all Main Steam Isolation Valves (MSIV) 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 Power Uprate using the ±3% MSRV setpoint tolerance to confirm that the vessel pressure does not exceed the ASME pressure vessel code limit.

The required number of operable MSRVs has been determined for Reactor operation using the MSRV setpoint tolerance of ±3%. The required number of operable valves (12) as determined by this analysis corresponds to the number of operable valves required in Technical Specification LCO 3.4.3 currently. Therefore, no change is required to Technical Specification 3.4.3 to account for the increased SRV setpoint tolerance of ±1% to ±3%.

III.d <u>Main Steam Safety Relief Valves</u>

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% to ±3%. The relief or automatic depressurization modes, which rely upon solenoid actuation, are unaffected by this change. Section III.c details the design basis overpressurization analysis and the required number of operable MSRVs.

III.e <u>High Pressure Systems</u>

Generic evaluations were performed by General Electric 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 this evaluation included; MSIV timing specifications, 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 operation of these systems and components. These analyses were reviewed and determined to be applicable to SSES systems.

III.f HPCI and RCIC Systems

The HPCI and RCIC systems are designed to inject to the reactor vessel at the lowest MSRV setpoint pressure plus the setpoint tolerance. The systems' performances were evaluated for increasing the reactor pressure from the lowest MSRV setpoint pressure +1% to the lowest MSRV setpoint +3%. Evaluation and analyses have determined that the HPCI and RCIC systems are capable of delivering the required flow to the Reactor Vessel at the higher steam dome pressure. The increased turbine operating speeds will result in a slight reduction in the 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% 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 General Electric Service Information Letter 377. These modifications, referred to as "bypass start" modifications, limit the initial peak speed during system startup, thus reducing the chance of a turbine overspeed trip.

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 valve), HV-149F013 (RCIC Injection Shutoff valve), HV-1(2)50F046 (RCIC Turbine Cooling Water Supply valve), FV-1(2)49F019 (RCIC Pump Minimum Flow Bypass valve), HV-1(2)55F001 (HPCI Turbine Steam Admission valve), HV-1(2)55F006 (HPCI Injection Shutoff valve), HV-1(2)56F059 (HPCI Lube Oil Cooler Cooling Water Supply valve), and HV-1(2)55F012 (HPCI Pump Minimum Flow Bypass valve). Calculations show that these valves will see an increased maximum differential pressure. Calculations evaluate the increased differential pressure and document that MOV operation will not be affected by the increase in MSRV setpoint tolerance.

Calculations determine 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. The HPCI Booster and RCIC pump discharge lines meet the requirements of the ASME code. The HPCI Main pump discharge line, does not meet the requirements of the ASME code and a relief from the ASME code requirements is being requested with this T.S. submittal. Attachment 6 details the request.

III.g *Standby Liquid Control System*

The current SSES 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 since the loss of AC power causes the loss of the Containment Instrument Gas (CIG) 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 the higher nominal spring setpoints. The current analysis demonstrates that during this event, vessel pressure is controlled to the second MSRV setpoint of 1195 psi. Since the valves are postulated to lift at the nominal setpoint 1195 psi, this is the SBLC design basis steam dome pressure, and is not affected by the proposed increase in the allowable MSRV tolerance.

In February of 2001, PPL realized that the SBLC system/pump design pressures were based on a vessel pressure corresponding to the lower setpoint MSRV relief mode of operation. That is, these pressures did not account for the ATWS-LOOP event described above. In response, this condition was entered into the SSES Corrective Action Program, and design modifications have been developed to increase the SBLC system design pressures. These modifications, which involve re-working the pump discharge flange, and resetting the pump discharge relief valve setpoint to 1500 psi, have been installed in Unit 2 during its 10^{th} Refueling Outage in the Spring of 2001. It is PPL's intent to install them in Unit 1 during the Unit's 12 Refueling Outage in the Spring of 2002, or next unit outage of sufficient length.

As a final note, although not required by the SBLC design and licensing bases, if the +3% MSRV tolerance were, in fact, considered in the SSES ATWS-LOOP event analysis, the postulated vessel steam dome pressure would be 1231 (1195 x 1.03). Based on the upgraded design of the SSES SBLC systems (i.e., design pressure of 1500 psi), these system would still be fully capable of injecting their boron solution under such postulated conditions.

III.h *Feedwater System*

The Feedwater Steam Supply piping (ASME Class 1) was evaluated against ASME Class I (one inch and smaller), II, III and ANSI B31.1 standards for the MSRV setpoint tolerance of ±3%. 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.

III.i Vessel Instrumentation

The Vessel Instrumentation was evaluated using the increased MSRV setpoint tolerance of ±3% against ASME Class I (one inch and smaller), II, III and ANSI B31.1 standards for an MSRV setpoint tolerance of ±3%. The additional pressure/loads induced by the proposed increase in the MSRV tolerance are bounded by the existing design capability of all potentially affected vessel instrumentation.

III.j Alternate Operating Modes

The SSES 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.

III.k Containment Response during LOCA and Hydrodynamic Loads on the MSRV Discharge Lines

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

Section IV

Conclusions

All applicable analyses supporting the increase in SRV setpoint tolerance have been completed. The proposed increase in SRV setpoint tolerance meets the conditions of NEDC-31753P and the associated NRC SER and, therefore, does not endanger the health and safety of the public.

Attachment 2 to PLA-5377

No Significant Hazards Considerations Evaluation

and

Environmental Assessment

NO SIGNIFICANT HAZARDS CONSIDERATIONS EVALUATION AND ENVIRONMENTAL ASSESSMENT

In 10 CFR 50.92(c), the NRC provides the following standards to be used in determining the existence for a significant hazards consideration:

...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 for a testing facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident of a new or different kind from any previously evaluated; (2) Create a possibility of a new or different kind of accident than previously evaluated or (3) Involve a significant reduction in the margin of safety.

PPL Susquehanna, LLC (PPL) has evaluated the proposed Technical Specification change in accordance with the criteria specified in 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration. The criteria and conclusions of our evaluation are presented below.

1. **The proposed action does not involve a significant increase in the probability or consequences of an accident as previously evaluated.**

   The proposed change allows an increase in the as-found MSRV safety mode setpoint tolerance, determined by test after the valves have been removed from service, from ±1% to ±3%. The proposed change does not alter the TS 3.4.3 Surveillance Requirements on the nominal MSRV safety mode lift setpoints, the MSRV relief mode setpoints, the required frequency for the MSRV lift setpoint tests, or the number of MSRVs required to be operable.

   Consistent with current requirements, this change continues to require that these valves be adjusted to within ± 1% of their nominal lift setpoints following testing. The proposed action does not change any other behavior or operation of any MSRV, and, therefore, has no significant impact on reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Final Safety Analysis Report (FSAR).

The proposed action does not involve physical changes to the valves, nor does it change the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components. Therefore, these changes will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," and were reviewed and approved by the NRC in a Safety Evaluation Report (SER) dated March 8, 1993. The plant specific evaluations, required by the NRC's SER and performed to support this proposed change, show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a ±3% setpoint tolerance. These analyses also show that operation of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems are not adversely affected and the containment response from a loss of coolant accident is acceptable. The plant systems associated with these proposed changes are capable of meeting all applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the FSAR. Therefore, these changes do not involve an increase in the consequences of any accident previously evaluated.

Therefore, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

2. **The proposed action does not create a possibility of a new or different kind of accident than previously evaluated.**

The proposed change was developed in accordance with the provisions contained in the NRC SER, dated March 8, 1993, for the "BWR Owners Group Inservice Pressure Relief Technical Specification Revision Licensing Topical Report," NEDC-31753P. The revised MSRV setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Since the proposed action does not involve hardware changes, significant changes to the operation of any systems or components, nor changes to existing structures, systems, or components, there is no possibility that a new or different kind of accident is created.

The proposed change to allow an increase in the MSRV safety mode setpoint tolerance from ±1% to ±3% does not alter the nominal MSRV lift setpoints or the number of MSRVs currently required to be operable by SSES Technical

Specifications. The proposed action does not involve physical changes to the valves, nor does it change the safety function of the valves. The proposed action does not involve a physical alteration of any existing plant equipment. No new or different equipment is being installed. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the procedures governing normal plant operation, nor the procedures utilized to respond to plant transients.

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

3. The proposed action does not involve a significant reduction in a margin of safety.

The proposed action does not involve a significant reduction in a margin of safety. Establishment of the ±3% MSRV safety setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Engineering evaluations concluded that there are no significant impacts on fuel thermal limits, safety related systems, structures or components, and no significant impact on the accident analyses associated with the proposed changes.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the proposed change, because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. As discussed in the "No Significant Hazards Consideration Evaluation", the proposed change does not involve a significant consideration. The proposed change does not involve a change in the types or increase in the amounts of effluents that may be released off-site. In addition, the proposed change does not involve an increase in the individual or cumulative occupational radiation exposure.

Attachment 3 to PLA-5377

Technical Specification Markups
Technical Specification Bases Markups
(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. ---	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)

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. ---	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)

BASES

APPLICABLE SAFETY ANALYSES (continued)

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 2 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The safety function of 12 of the 16 S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the FSAR are based on these setpoints, but also include the additional uncertainty of ~~± 1%~~ ±3% of the nominal setpoint to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, all required S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4 reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed

(continued)

BASES

APPLICABILITY (continued)

operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more required S/RVs is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENT

SR 3.4.3.1

The Surveillance requires that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ~~±1%~~ ±3% of the nominal setpoint for OPERABILITY. ~~(NOTE: If this setpoint tolerance is revised, the relief request #34 for Inservice Testing of S/RVs must be revised and resubmitted to the NRC for reapproval.)~~ Requirements for accelerated testing are established in accordance with the Inservice Test Program. Any of the 16 S/RVs, identified in this Surveillance Requirement, with their associated setpoints, can be designated as the 12 required S/RVs. This maintains the assumptions in the overpressure analysis.

A Note is provided to allow up to two of the required 12 S/RVs to be physically replaced with S/RVs with lower setpoints until the next refueling outage. This provides operational flexibility which maintains the assumptions in the over-pressure analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 2 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The safety function of 12 of the 16 S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the FSAR are based on these setpoints, but also include the additional uncertainty of ~~± 1%~~ ±3% of the nominal setpoint to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, all required S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4 reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed

(continued)

BASES

APPLICABILITY (continued)

operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more required S/RVs is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENT

SR 3.4.3.1

The Surveillance requires that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ~~±1%~~ ±3% of the nominal setpoint for OPERABILITY. ~~(NOTE: If this setpoint tolerance is revised, the relief request #34 for Inservice Testing of S/RVs must be revised and resubmitted to the NRC for reapproval.)~~ Requirements for accelerated testing are established in accordance with the Inservice Test Program. Any of the 16 S/RVs, identified in this Surveillance Requirement, with their associated setpoints, can be designated as the 12 required S/RVs. This maintains the assumptions in the overpressure analysis.

A Note is provided to allow up to two of the required 12 S/RVs to be physically replaced with S/RVs with lower setpoints until the next refueling outage. This provides operational flexibility which maintains the assumptions in the over-pressure analysis.

(continued)

Attachment 4 to PLA-5377

"Camera Ready" Technical Specifications

(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. -- Number of S/RVs / Setpoint (psig) 2 / 1175 ($\geq$ 1140 and $\leq$ 1210) 6 / 1195 ($\geq$ 1160 and $\leq$ 1230) 8 / 1205 ($\geq$ 1169 and $\leq$ 1241)	In accordance with the Inservice Testing Program

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. ------------------------------ Number of S/RVs — Setpoint (psig) 2 — 1175 ($\geq$ 1140 and $\leq$ 1210) 6 — 1195 ($\geq$ 1160 and $\leq$ 1230) 8 — 1205 ($\geq$ 1169 and $\leq$ 1241)	In accordance with the Inservice Testing Program

Attachment 5 to PLA-5377

Revised IST Relief Request Number 34

(Units 1 & 2)

RELIEF REQUEST NUMBER 34

System: Nuclear Boiler

P&ID: 141

Valves	Category	Valves	Category
PSV-141F013A	C	PSV-141F013G	B,C
PSV-141F013B	C	PSV-141F013J	B,C
PSV-141F013C	C	PSV-141F013K	B,C
PSV-141F013D	C	PSV-141F013L	B,C
PSV-141F013E	C	PSV-141F013M	B,C
PSV-141F013F	C	PSV-141F013N	B,C
PSV-141F013H	C		
PSV-141F013P	C		
PSV-141F013R	C		
PSV-141F013S	C		

Class: 1

Function: Main Steam Safety/Relief Valve

Impractical Test Requirement:
OM-1987, Part 1, paragraph 1.3.3.1(b) requires all valves of each type and manufacture shall be tested within each subsequent 5 year period with a minimum of 20% of the valves tested within any 24 months. This 20% shall be previously untested valves, if they exist.

Basis for Relief: Pursuant to 10CFR50.55a(f)(6)(i), relief is requested from the requirements of ASME Code Section XI, OM-1987 Part 1, Paragraph 1.3.3.1(b). Due to Susquehanna's implementation of a 24-month fuel cycle, the requirements described above potentially compromise radiation safety and could jeopardize refuel outage schedule durations. The proposed alternative testing frequency will continue to provide an acceptable level of quality and safety pursuant to 10CFR50.55a(a)(3)(i).

RELIEF REQUEST NUMBER 34 (CONT'D)

Susquehanna currently removes and tests 8 of the 16 Main Steam Safety/Relief Valves during each refueling outage. This methodology meets the Code criteria of testing previously untested valves and permits the removal and replacement of weeping valves detected during the previous operating cycle. Weeping MSRV's are detected by monitoring tailpipe temperatures. If the tailpipe temperature exceeds 200 degrees F, then the relief valve is viewed as a weeper. With an 18-month fuel cycle, the completion of Code testing was accomplished over a period of 3 refuel cycles. This approach has resulted in maintenance and operational flexibility which has had the following benefits for Susquehanna:

- Provides the ability to both test the Code required valves out of the population not yet tested, and replace any weeping MSRV's.
- Maintains relatively leak-free MSRV's, thus minimizing the necessary run time of ECCS systems that provide suppression pool cooling.
- Consistent application of ALARA principles.
- Enhances equipment reliability.
- Results in minimal impact on outage durations.

Without Code relief for 24 month fuel cycles, strict Code compliance would restrict Susquehanna's operating philosophy to not operate with weeping MSRV's as Code testing would be required to be completed within 5 years. This testing strategy does not account for any leaking valves that may need to be refurbished. Since Susquehanna's philosophy is to share spare valves between both units, (the valves that are removed from one unit are installed in the other unit's next refueling outage), this testing strategy is less than adequate. This strategy could only be accomplished if a larger population of MSRV's are tested each outage or additional spare valves are purchased. More than 8 valves would need to be sent to the offsite testing facility during a refueling outage. The testing and return of these valves would have to be completed expeditiously in order to not impact the refuel outage schedule duration. For this reason, additional expenditures would be incurred to purchase and test a greater number of valves each outage. Without Code relief, the additional outage work would be

RELIEF REQUEST NUMBER 34 (CONT'D)

contrary to the principles of ALARA and could compromise radiation safety. Because of the location of certain MSRV's in the containment, interferences exist that would require the removal of more valves and piping to get to those valves that must be removed for the sample testing. This results in more radiation exposure to the maintenance personnel than is desirable.

With Code relief, the total of 16 MSRV's per unit and 8 spares that are shared between the two units can be tested within 6 years to complete the Code required testing for the total population and accommodate any weeping MSRVs.

The increased testing over only 2 refuel cycles will result in no additional safety benefit to the plant. Susquehanna has had excellent performance with MSRV's over the last 10 years. Since 1987, Susquehanna has imposed more conservative as-left leakage criteria on the testing facility than was specified in the General Electric Specification and incorporated in the PP&L Specification for testing Crosby style relief valves. The criterion imposed on the test lab is 0-ml/5 min (via the purchase order), compared to a GE Specification "as-left" leakage criteria of 38-ml/5 min.

Additionally, a review of the setpoint testing results (for both units) for the time period from initial operation to the present (June, 2001), which comprises 231 data points, shows that the average of the setpoint drift percentages is -0.687%. This indicates that, in general, the SRV's tend to drift slightly downward, not upward. The calculated standard deviation from the average for the data was determined to be 1.45%. The data indicates that a significant number of the as-found setpoints were outside the +/- 1% tolerance allowed by the plant Technical Specifications. However, most of the points outside the TS tolerance were below -1%, not above +1%, which results in a slightly downward setpoint drift trend over time. This indicates that for the longer test interval proposed, there is not expected to be a reduced capability of the SRV's to provide adequate system overpressure protection. Also, the testing history shows that since commercial operation we have had only two "as found" set pressure test acceptance criteria failures (+3%) of the tested valves, which required testing of additional MSRV's.

RELIEF REQUEST NUMBER 34 (CONT'D)

Alternative Testing: The Main Steam Safety/Relief Valves will be tested such that a minimum of 20% of the valves (previously untested, if they exist) are tested every 24 months, such that all the valves will be tested within 3 refuel cycles. This proposal utilizes the same maintenance and testing approach that was applied in 18-month refuel cycles. This alternative frequency will continue to provide assurance of the valve operational readiness, as required by OM-1987, Part 1, paragraph 1.3.1.2, and provides an acceptable level of quality and safety.

Additionally, any failures, either seat leakage or pressure set, occurring at the test facility, as well as weeping MSRV's that develop during the operating cycle, will be documented via the corrective action program, evaluated and dispositioned accordingly.

RELIEF REQUEST NUMBER 34

System: Nuclear Boiler

P&ID: 2141

Valves	Category	Valves	Category
PSV-241F013A	C	PSV-241F013G	B,C
PSV-241F013B	C	PSV-241F013J	B,C
PSV-241F013C	C	PSV-241F013K	B,C
PSV-241F013D	C	PSV-241F013L	B,C
PSV-241F013E	C	PSV-241F013M	B,C
PSV-241F013F	C	PSV-241F013N	B,C
PSV-241F013H	C		
PSV-241F013P	C		
PSV-241F013R	C		
PSV-241F013S	C		

Class: 1

Function: Main Steam Safety/Relief Valve

Impractical Test Requirement:
OM-1987, Part 1, paragraph 1.3.3.1(b) requires all valves of each type and manufacture shall be tested within each subsequent 5 year period with a minimum of 20% of the valves tested within any 24 months. This 20% shall be previously untested valves, if they exist.

Basis for Relief: Pursuant to 10CFR50.55a(f)(6)(i), relief is requested from the requirements of ASME Code Section XI, OM-1987 Part 1, Paragraph 1.3.3.1(b). Due to Susquehanna's implementation of a 24-month fuel cycle, the requirements described above potentially compromise radiation safety and could jeopardize refuel outage schedule durations. The proposed alternative testing frequency will continue to provide an acceptable level of quality and safety pursuant to 10CFR50.55a(a)(3)(i).

RELIEF REQUEST NUMBER 34 (CONT'D)

Susquehanna currently removes and tests 8 of the 16 Main Steam Safety/Relief Valves during each refueling outage. This methodology meets the Code criteria of testing previously untested valves and permits the removal and replacement of weeping valves detected during the previous operating cycle. Weeping MSRV's are detected by monitoring tailpipe temperatures. If the tailpipe temperature exceeds 200 degrees F, then the relief valve is viewed as a weeper. With an 18-month fuel cycle, the completion of Code testing was accomplished over a period of 3 refuel cycles. This approach has resulted in maintenance and operational flexibility which has had the following benefits for Susquehanna:

- Provides the ability to both test the Code required valves out of the population not yet tested, and replace any weeping MSRV's.
- Maintains relatively leak-free MSRV's, thus minimizing the necessary run time of ECCS systems that provide suppression pool cooling.
- Consistent application of ALARA principles.
- Enhances equipment reliability.
- Results in minimal impact on outage durations.

Without Code relief for 24 month fuel cycles, strict Code compliance would restrict Susquehanna's operating philosophy to not operate with weeping MSRV's as Code testing would be required to be completed within 5 years. This testing strategy does not account for any leaking valves that may need to be refurbished. Since Susquehanna's philosophy is to share spare valves between both units, (the valves that are removed from one unit are installed in the other unit's next refueling outage), this testing strategy is less than adequate. This strategy could only be accomplished if a larger population of MSRV's are tested each outage or additional spare valves are purchased. More than 8 valves would need to be sent to the offsite testing facility during a refueling outage. The testing and return of these valves would have to be completed expeditiously in order to not impact the refuel outage schedule duration. For this reason, additional expenditures would be incurred to purchase and test a greater number of valves each

RELIEF REQUEST NUMBER 34 (CONT'D)

outage. Without Code relief, the additional outage work would be contrary to the principles of ALARA and could compromise radiation safety. Because of the location of certain MSRV's in the containment, interferences exist that would require the removal of more valves and piping to get to those valves that must be removed for the sample testing. This results in more radiation exposure to the maintenance personnel than is desirable.

With Code relief, the total of 16 MSRV's per unit and 8 spares that are shared between the two units can be tested within 6 years to complete the Code required testing for the total population and accommodate any weeping MSRVs.

The increased testing over only 2 refuel cycles will result in no additional safety benefit to the plant. Susquehanna has had excellent performance with MSRV's over the last 10 years. Since 1987, Susquehanna has imposed more conservative as-left leakage criteria on the testing facility than was specified in the General Electric Specification and incorporated in the PP&L Specification for testing Crosby style relief valves. The criterion imposed on the test lab is 0-ml/5 min (via the purchase order), compared to a GE Specification "as-left" leakage criteria of 38-ml/5 min.

Additionally, a review of the setpoint testing results (for both units) for the time period from initial operation to the present (June, 2001), which comprises 231 data points, shows that the average of the setpoint drift percentages is -0.687%. This indicates that, in general, the SRV's tend to drift slightly downward, not upward. The calculated standard deviation from the average for the data was determined to be 1.45%. The data indicates that a significant number of the as-found setpoints were outside the +/- 1% tolerance allowed by the plant Technical Specifications. However, most of the points outside the TS tolerance were below -1%, not above +1%, which results in a slightly downward setpoint drift trend over time. This indicates that for the longer test interval proposed, there is not expected to be a reduced capability of the SRV's to provide adequate system

RELIEF REQUEST NUMBER 34 (CONT'D)

overpressure protection. Also, the testing history shows that since commercial operation we have had only two "as found" set pressure test acceptance criteria failures (+3%) of the tested valves, which required testing of additional MSRV's.

Alternative Testing: The Main Steam Safety/Relief Valves will be tested such that a minimum of 20% of the valves (previously untested, if they exist) are tested every 24 months, such that all the valves will be tested within 3 refuel cycles. This proposal utilizes the same maintenance and testing approach that was applied in 18-month refuel cycles. This alternative frequency will continue to provide assurance of the valve operational readiness, as required by OM-1987, Part 1, paragraph 1.3.1.2, and provides an acceptable level of quality and safety.

Additionally, any failures, either seat leakage or pressure set, occurring at the test facility, as well as weeping MSRV's that develop during the operating cycle, will be documented via the corrective action program, evaluated and dispositioned accordingly.

Attachment 6 to PLA-5377

ASME Code Relief Request for HPCI Main Pump Discharge Piping

(Units 1 & 2)

ASME CODE RELIEF REQUEST FOR HPCI MAIN PUMP DISCHARGE PIPING

Background

The Main Steam Safety Relief Valve (MSRV) setpoint tolerance is being increased from one percent to three percent. The increased setpoint tolerance will allow a higher steam inlet pressure at the HPCI Turbine. This results in a higher maximum turbine speed, and, ultimately, the HPCI Main Pump maximum discharge pressure will be increased to 1583 psig. This maximum pressure would only occur under pump deadhead conditions.

The affected piping must be qualified for the increased maximum pressure. A review of the applicable piping design analyses was performed. The pipe class EBB portion of the HPCI Main Pump discharge piping, designated as pipeline EBB-102/202, had been previously qualified to a maximum pressure of only 1360 psig. Thus, it was necessary to evaluate the class EBB portion of the discharge piping for the higher maximum pressure of 1583 psig.

The EBB portion of the discharge piping consists of a 14"x10" reducer, welded to the 10" pump discharge nozzle, approximately 70 linear feet of 14" piping, 60 linear feet of 4" piping, and various lengths of 1", ¾", and ½" piping for vents, drains, and instrumentation. Pipe class EBB is designated as ASME Section III Class 2. All piping is ASME SA-106, Grade B, seamless material. The 14"x10" reducer is a butt weld fitting per ASME SA-234, Grade WPB, with wall thickness to match the piping.

The evaluation considered all affected piping. Using the applicable design code, the evaluation determined that the higher pressure was not acceptable for the 14" EBB-102/202 HPCI Main Pump discharge piping and the 10" diameter section of the 14"x10" piping reducer welded to the pump discharge nozzle.

Code Design Requirements

The applicable design code for ASME piping at Susquehanna SES is the ASME Boiler and Pressure Vessel Code, Section III, 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.

Relief Requested

ASME Code Pressure Design

NC-3641.1 Equation 3 is used to calculate the required minimum wall thickness using a defined design 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.

Since the wall thickness is defined by the installed piping, Equation 3 is not applicable. Thus, Equation 4 is applied to determine the maximum allowable pressure under the design conditions.

Besides the nominal wall thickness, Equation 4 requires the use of defined values for the outside diameter of the pipe, the maximum allowable stress, joint efficiency, and an allowance for additional wall thickness to compensate for corrosion and/or erosion losses.

The maximum allowable stress (S) for SA-106 Grade B piping and SA-234 Grade WPB fittings at a design temperature of 220°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. Thus, the maximum stress allowed by the code is 18000 psi.

As determined by Equation 4 using the code allowable stress of 18000 psi:

- The maximum design pressure for the 14" EBB-102/202 HPCI Main Pump discharge piping is 1532 psig.

- The maximum design pressure for the 14"x10" reducers at the HPCI Main Pump discharge nozzles is 1519 psig.

Since the maximum design pressures for the 14" piping and the 14"x10" reducers are less than the increased value of 1583 psig for the HPCI Main Pump maximum discharge pressure, the ASME Code requirements are not met.

There are two options. The first option is to increase the wall thickness of the components to allow higher design pressures. This option is not practical. The second option is to use an alternate allowable stress in lieu of the maximum allowable stress defined in the applicable table in Appendix I, per NC-3641.1.

In order to qualify the affected piping for the HPCI Main Pump maximum discharge pressure of 1583 psig, relief is requested from the use of the code allowable stress (S) as specified in Table I-7.1 for SA-106 Grade B material at a design temperature of 220°F.

For the construction of the Susquehanna SES Units 1 and 2, vendors submitted Certified Material Test Reports (CMTR's) for all Quality-related piping materials. The CMTR's include test data for the actual yield and ultimate (tensile) stress values of the piping material. Article III-3000 of the ASME Code Section III discusses the basis for establishing allowable stress values. Paragraph III-3210 specifies that the maximum allowable stress (S) is the lowest of 1/4 of the tensile strength at room or design temperature, or 5/8 of the yield strength at room or design temperature. Accordingly, the CMTR data for yield and ultimate (tensile) strength may be used to develop an alternate allowable stress for the HPCI Main Pump discharge piping.

Therefore, this relief request is to allow the use of an alternate allowable stress, determined in accordance with Paragraph III-3210, instead of the allowable stress (S) as specified in Paragraph NC-3461.1.

Pressure Design with Requested Relief

CMTR's for the 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 can be applied to the EBB piping.

With the twenty percent increase allowed by NC-3612.3, an allowable stress of 21000 psi, instead of 18000 psi, may be used in NC-3641.1 Equation 4. This new allowable stress results in the following:

- The maximum design pressure for the 14" EBB-102/202 HPCI Main Pump discharge piping is 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, instead of 18000 psi, may be used in NC-3641.1 Equation 4. This new allowable stress results in the following:

- The maximum design pressure for the 14"x10" reducers at the HPCI Main Pump discharge nozzles is 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.

While the use of an alternate allowable stress is not strictly in accordance with NC-3641.1, the alternate allowable stress was determined in accordance with Paragraph III-3210. Therefore, the use of the alternate allowable stress provides an acceptable level of quality and safety since the as-built piping material data properties are being used to qualify the piping.

To assure the conservatism in this approach, sample wall thickness measurements were taken at various locations on the 14" EBB piping and the 14"x10" reducers. All sample measurements yielded values that were greater than the minimum wall thickness that would be allowed by NC-3641.1. Thus, the as-built wall thicknesses exceed the ASME Code requirement and provide additional conservatism in the design.

Alternate Provision

The 14" EBB-102/202 HPCI Main Pump discharge piping and 14"x10" reducers will be qualified for ASME Section III NC-3641.1 Equation 4 using actual CMTR material data rather than minimum allowable stress (S) values taken from Appendix I.

Applicable Time Period

This relief request will remain in effect for the duration of the original 40-year operating license of Susquehanna SES Units 1 and 2.