August 5, 2002

Mr. Bryce L. Shriver
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: HIGH PRESSURE COOLANT INJECTION PUMP AUTOMATIC TRANSFER TO SUPPRESSION POOL LOGIC ELIMINATION (TAC NOS. MB2190 AND MB2191)

Dear Mr. Shriver:

The Commission has issued the enclosed Amendment No. 204 to Facility Operating License No. NPF-14 and Amendment No. 178 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 8, 2001 and supplements dated February 4, April 8, May 7, June 6, and June 28, 2002.

These amendments revise TS 3.3.5.1, "Emergency Core Cooling System Instrumentation," by deleting Function 3e, thus preventing the automatic swap of the suction source for the high pressure coolant injection pump from the condensate storage tank to the suppression pool on high suppression pool level. This change, and its associated plant modifications, eliminates a vulnerability identified by the SSES-1 and 2 Individual Plant Examination; an anticipated transient without scram event combined with a failure of the standby liquid control system.

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 by TColburn for/

Eric M. Thomas, 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. 204 to License No. NPF-14
2. Amendment No. 178 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. 204
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 June 8, 2001, as supplemented by letters dated February 4, April 8, May 7, June 6 and June 28, 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. 204 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 within 30 days of its associated plant modifications, and no later than December 31, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by PTam for/

Richard J. Laufer, 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: August 5, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 204

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-44	3.3-44
3.3-45	3.3-45
3.3-46	3.3-46

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
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 June 8, 2001, as supplemented by letters dated February 4, April 8, May 7, June 6 and June 28, 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. 178 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 within 30 days of its associated plant modifications, and no later than December 31, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by PTam for/

Richard J. Laufer, 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: August 5, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 178

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
3.3-45	3.3-45

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 178 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 application dated June 8, 2001, as supplemented by letters dated February 4, April 8, May 7, June 6 and June 28, 2002, PPL Susquehanna, LLC (the licensee), requested changes to the Technical Specifications (TSs) for Susquehanna Steam Electric Station, Units 1 and 2 (SSES-1and 2). The supplements dated February 4, April 8, May 7, June 6, and June 28, 2002, were in response to the staff's requests for additional information (RAI's) dated December 18, 2001, and April 22, 2002, and questions resulting from teleconferences between the staff and the licensee on March 25 and 26, April 9, and May 14, 15, 17, and 22, 2002. These supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 3, 2001 (66 FR 50471).

The proposed changes would revise TS 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," by deleting Function 3e from TS Table 3.3.5.1-1, "HPCI [high-pressure coolant injection] System Suppression Pool Water Level - High," for both units. Specifically, the proposed changes prevent the automatic swap of the suction source for the HPCI pump from the condensate storage tank (CST) to the suppression pool on high suppression pool level. The proposed changes, and the implementation of associated proposed plant modifications, are needed to eliminate a vulnerability identified by the SSES-1 and 2 Individual Plant Examination (IPE); an anticipated transient without scram (ATWS) event combined with a failure of the standby liquid control system (SLCS).

2.0 REGULATORY EVALUATION

In an ATWS event with SLCS failure, the operator can reduce reactor power by manually inserting control rods, one at a time, with the rod drive control system. This capability is the result of a previous plant modification to the rod sequence control system (RSCS). As recommended by the IPE, an RSCS keylock bypass switch was installed on the Unit Operating Benchboard 1(2)C651. Use of the bypass switch in an ATWS event inhibits rod insert blocks which allows the operator to manually insert control rods to reduce reactor power. Manual insertion of control rods also requires bypass of the rod worth minimizer, an action which can also be performed from the control room.

A reactor shutdown by manual control rod insertion is possible only if the reactor can be maintained at high pressure where the core is not susceptible to power/flow instabilities. Maintaining the reactor at high pressure requires operability of HPCI. This is the only HPCI system available to maintain vessel inventory in an isolation ATWS.

The safety function of HPCI is to provide core cooling for a wide range of reactor pressures. HPCI is primarily used to maintain the reactor inventory after small break loss-of-coolant accidents (LOCAs) that do not result in the depressurization of the reactor. The suppression pool water level-high function is provided to eliminate the possibility of HPCI continuing to provide additional water from the CST to the suppression pool. Reactor blowdown with a high pool level could result in loads that exceed the suppression pool design values. The transfer function on high suppression pool level exists to preclude excessively high suppression pool levels (SSES TS Bases Section 3.3.5.1).

The current plant design includes an automatic transfer of HPCI suction from the CST to the suppression pool on elevated suppression pool level. The set point for the suction transfer is 23 ft 10 in. The suppression pool water level, as required in TS 3.6.2.2, is to be greater than 22 ft but less than 24 ft. The 24-ft limit prevents excessive clearing loads from safety-relief valve (SRV) discharge and excessive pool swell loads during a postulated design-basis LOCA.

The suppression pool level is monitored by alarm window AR-114-F01 (AR-214-F01), “Suppression Pool Hi Level,” which has a setpoint of 23.75 ft. The operator also has level indicator LI-15775B (LI-25775B), “Suppression Pool Level.” The alarm window and the level indicator are both located on the HPCI section of Panel 1C601 (2C601).

Susquehanna’s current emergency operating procedures (EOPs) instruct the operator to manually bypass the HPCI automatic suction transfer on high suppression pool level and realign the HPCI suction back to the CST in an ATWS event. The manual bypass is performed outside the control room and completion is expected to take in excess of 30 minutes. Analysis shows that an ATWS event could cause a rapid rise in suppression pool temperature and a failure of HPCI on loss of lube oil cooling within the first 10 minutes of the event, which is considerably less time than it takes the operator to manually bypass the suction transfer to the suppression pool. With the present plant configuration, the operator has a means of shutting down the reactor in an ATWS with SLCS failure, however a loss of the HPCI system prior to an appreciable decrease in power level would preclude the operator’s ability to achieve a reactor shutdown at high-pressure conditions where the reactor core is stable.

Operation of the HPCI system is assured with suction source water temperatures up to 140 °F. Sustained HPCI pump operation is assured if the pump’s discharge temperature is limited to 140 °F, since the discharge flow cools the HPCI pump lube oil and HPCI pump failure may occur if the lube oil temperature becomes excessive. Sustained HPCI pump operation is also assured if short-duration temperatures do not exceed 190 °F. Suppression pool temperatures for design-basis accidents (DBAs) are not expected to exceed the 140 °F limit for the time frame the HPCI pump is expected to operate. Beyond design-basis events may result in suppression pool temperatures that exceed the 140 °F limit, and the short duration 190 °F limit, while HPCI operation is desired. General Electric Design and Performance Specification

386HA817, "Anticipated Transient Without Scram (ATWS)," indicates that procedural controls are necessary to preclude the transfer from the CST to the suppression pool when the pool temperature is too high for HPCI pump and reactor core isolation cooling (RCIC) pump operation.

This proposed change deletes from TS Table 3.3.5.1-1 the "High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level - High," (Function 3e) in both the Unit 1 and 2 TSs and prevents the automatic HPCI pump suction transfer from the CST to the suppression pool on high level. Procedures are to be revised to replace this automatic action with operator instructions to manually transfer the HPCI suction source to the suppression pool when the suppression pool level reaches 25 ft as long as the suppression pool water temperature remains less than or equal to 140 °F.

HPCI and RCIC are the only ECCS pumps available in a station blackout (SBO) event, and manual bypass of the automatic transfer from the CST to the suppression pool is necessary to prevent damage to the HPCI system from hot suppression pool water (> 140 °F). Procedures are currently in place to prevent HPCI system damage. Emergency Operating Procedure (EOP) EO-100/200-032, “HPCI Operating Guidelines During Station Blackout," instructs the operator to prevent the automatic transfer from the CST to the suppression pool on high pool level. A manual bypass of the suction transfer logic is carried out in accordance with Emergency Support Procedure ES-152/252-002. Therefore, the removal of the automatic HPCI suction transfer on high pool level will also reduce the operator burden during an SBO event.

The licensee believes the proposed change is justified because the safety assessment shows that with the elimination of the suppression pool “Water Level - High” function:

- The plant response to an ATWS event is enhanced, increasing the probability of a safe shutdown.
- The probability that HPCI and RCIC will be available in an SBO event is increased.
- The ability of HPCI to respond and perform its design-basis function will not be affected.
- The suppression ability of the suppression pool and plant response to DBAs and postulated transient events will not be affected.
- Operator burden is reduced.

This proposed change also corrects a typographical error for function 5a, “ADS Trip System B Reactor Vessel Water Level,” in the Unit 1 TS Table 3.3.5.1-1 to indicate "Low, Low, Low Level 1" instead of "Low Level 1." The Unit 2 TS does not contain this same typographical error.

3.0 TECHNICAL EVALUATION

The SSES primary containment utilizes a Mark II over/under containment design consisting of a drywell and wetwell. The wetwell suppression pool is designed to absorb the energy associated with the reactor core decay heat and the sensible heat from the reactor coolant and structures released during a reactor blowdown from the SRVs, the automatic depressurization system (ADS) or from a DBA. The suppression pool must quench all the steam released through the downcomer lines during a LOCA. This ensures that the peak containment pressure remains below the maximum allowable. The suppression pool must also condense the steam ejected from the HPCI and RCIC turbines.

Manual HPCI suction transfer is not needed to maintain containment hydrodynamic loads within design limits. LOCA blowdown loads are dependent only on the initial suppression pool level at the onset of the accident. This is unaffected by the proposed modification. While a rising suppression pool level by itself intensifies hydrodynamic loads associated with SRV/ADS blowdown, the pool level increase during a LOCA is accompanied by a decrease in reactor pressure. The reduction in reactor pressure offsets the adverse effects associated with the increase in pool level.

The staff reviewed the licensee's assessment of the effects of the proposed change to remove the automatic HPCI transfer from the CST to the suppression pool on containment hydrodynamic loads during DBAs. Additionally, the staff reviewed the licensee's assessment of the effects of the proposed change on the severity of the anticipated plant transients that may result in HPCI system initiation. Finally, the staff reviewed the licensee's assessment of the effects of the proposed change on the operation of several other plant components and systems associated with the suppression pool. The staff concluded that there is a narrow range of small liquid break LOCAs during which the manual HPCI suction swap would be required by EOPs. The resultant operator actions to realign the HPCI pump suction as necessary result in a decrease in overall plant risk, and the proposed changes do not increase the severity of any anticipated transients or DBAs. For these reasons, the staff finds the licensee's analyses acceptable.

3.1 Loss-of-Coolant Accidents - Inside Containment

3.1.1 Large-Break LOCA

The proposed HPCI suction transfer logic should have no appreciable influence on the rate of the suppression pool level increase because of the very short time period of HPCI operation for this DBA event, a full double-ended guillotine break of the recirculation suction line. Elimination of the HPCI suction transfer logic does not affect the requirement to maintain the suppression pool level less than 24 ft in accordance with TS 3.6.2.2. The initial pool level assumed in the LOCA analysis corresponds to 24 ft and remains unchanged after the proposed suction transfer modification. The licensee concluded that the proposed modification has no adverse impact on the containment response during the large-break LOCA.

The staff finds the licensee's evaluation of a large-break LOCA acceptable to support the proposed HPCI modification. For the DBA event, HPCI would operate for a very short time period, and changing the suction transfer logic does not adversely impact containment hydrodynamic loads.

3.1.2 1.0 ft^2 Recirculation Discharge Line Break

HPCI is assumed to be inoperable, as a result of the single failure criterion, for this DBA LOCA analysis. The 1.0 ft^2 break of the recirculation line causes rapid loss of vessel inventory, which results in a depressurization of the reactor vessel. ADS automatically initiates on low reactor water level, but the reactor is depressurized by the time the ADS valves open (326 psig at 121 seconds). This DBA is not impacted by the proposed HPCI suction transfer modification.

The break flow for this size LOCA is an order of magnitude larger than the HPCI injection rate. The licensee concluded that if HPCI were operational (with suction from the CST) there would be no significant impact on reactor and containment response during the early part of the transient that would impact the hydrodynamic loads. The suppression pool level and reactor pressure at the time of ADS actuation should be essentially the same as in the case where HPCI is assumed to be inoperable and the containment hydrodynamic loads should be essentially the same. These loads are bounded by the design-basis SRV/LOCA load definitions, which are based on a higher reactor pressure for ADS initiation.

The staff finds the licensee's evaluation of a 1.0 ft^2 break LOCA acceptable to support the proposed HPCI modification. For the DBA, HPCI is assumed to be inoperable. With HPCI, the resultant containment hydrodynamic loads should be essentially the same and are bounded by the current design-basis loads definition.

3.1.3 0.1 ft^2 Intermediate-Break LOCA

This DBA, with HPCI unavailable as a result of the single failure criterion, is not impacted by the proposed HPCI suction transfer modification. A representative intermediate size break (0.1 ft^2) with HPCI operable was analyzed by the licensee with the Simulation of ATWS in Boiling Water Reactors (SABRE) computer program to assess the proposed HPCI modification. The licensee concluded that the SABRE computer program was adequate for the evaluation of LOCAs to assess the HPCI modification.

SABRE was developed to study anticipated transient without scram (ATWS) events and used for the development of emergency procedure guidelines to address reactor core instabilities. The review of the use of SABRE in support of the proposed HPCI modification is provided in Section 3.16 of this evaluation. The staff has determined that SABRE is acceptable for the qualitative evaluation of the plant response to the scenarios of interest for this license amendment.

RCIC is assumed to be inoperable because it is not a safety system. A loss-of-offsite power (LOOP) is also assumed to occur coincidentally with the break to be consistent with the design-basis LOCA analysis. Assuming a LOOP maximizes the operating time of HPCI during the accident, since feedwater is lost within a few seconds of event initiation, and maximizes the effect of the proposed HPCI modification on the containment response.

The LOCA simulation was carried out until the onset of low-pressure core injection (LPCI) to the reactor vessel was predicted. Core spray initiation occurs when the reactor pressure drops to about 300 psig. The predicted difference in drywell and wetwell pressures indicates that the downcomer vents are cleared throughout the entire transient. Consistent with the design-basis function of the HPCI system, the operator will use the core spray to provide coolant make-up to the vessel, and HPCI operation will no longer be required.

Suppression pool level rises during this event due to the steam discharged from the HPCI turbine and steam discharged to the suppression pool through the downcomer vents. This mass addition is unaffected by the HPCI suction source. The water level in the drywell is not predicted to reach 18 inches were it would begin to overflow from the drywell to the wetwell through the downcomer vents (the downcomer vents extend 18 inches above the floor of the drywell).

For a 0.1 ft^2 break with HPCI operable, there is no concern of HPCI causing the suppression pool level to exceed 24 ft prior to initiation of SRV/ADS blowdown because the reactor pressure would decrease. Elimination of the automatic HPCI suction transfer on high suppression pool level does not affect the TS 3.6.2.2 requirement to maintain pool level less than 24 ft prior to the occurrence of a break. Containment loads due to the LOCA are based on the initial suppression pool level. The DBA LOCA produces bounding loads based on an initial level of 24 ft and all break flow going to the suppression pool.

The staff finds the licensee's evaluation of an 0.1 ft^2 break LOCA acceptable to support the proposed HPCI modification. For the DBA, HPCI is assumed to be inoperable. With HPCI, the resultant containment hydrodynamic loads should be essentially the same and are bounded by the current design-basis loads definition.

The licensee examined the impact of HPCI failure with suppression pool level greater than 24 ft for this intermediate break LOCA. However, it is not a licensing requirement to consider a single failure at times other than the initiation of the accident. When the reactor pressure drops to 300 psig, which is the shutoff head of the low-pressure ECCS, at about 600 seconds into the event, the suppression pool level has only risen 4 inches. The water level inside the SRV tailpipe is depressed when the downcomer vents are cleared and the larger air volume within the line is the most significant factor that affects the SRV loads relative to SRV loads under non-LOCA conditions. The proposed change has no effect on the discharge line air volume when the downcomer vents are cleared. The modification only affects the back pressure on the line as a result of the slightly higher pool level. The licensee concluded that this has no effect on the amplitude of SRV loads and negligible effect on the load frequency.

Since HPCI is running at full flow in this transient, and reactor pressure vessel (RPV) water level is significantly below the high-level trip of 54 inches, a HPCI trip (on high level) and a restart is very unlikely. The consequence of a HPCI trip and restart is addressed in the section on small-break LOCA.

3.1.4 Small-Break LOCA

The HPCI design-basis document states that "It [HPCI] is designed to be capable of making up inventory losses for liquid breaks below about 0.02 sqft, thus maintaining reactor level." With regard to the 1-in line break, Final Safety Analysis Report (FSAR), Section 6.3.1.1.1 states "One high pressure cooling system is provided which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 1 inch nominal diameter."

SABRE calculations were carried out by the licensee for two small breaks to evaluate the impact of eliminating the HPCI high-pool-level suction transfer on a small-break LOCA. A 0.02 ft^2 break and a 1-in line break (0.00545 ft^2) were evaluated by the licensee. Additional SABRE calculations were performed in response to a staff RAI. SABRE calculations were performed for 0.0375 ft^2, 0.04 ft^2, 0.045 ft^2, and 0.06 ft^2 breaks. The SABRE results indicate the

0.02 ft^2 break to be limiting with respect to the maximum predicted suppression pool temperature, 141 °F at the time of core spray initiation at 105.3 minutes into the accident. The suppression pool level reaches 25 ft in 22.5 minutes with this break size.

The LOCAs were simulated up to the point where reactor pressure drops below the shutoff head of the core spray system (~300 psig). For the 1-in line break (0.00545 ft^2), suppression pool level increases by only 4 in. The 0.02 ft^2 line break is the more limiting case and was evaluated by the licensee to assess the proposed elimination of the automatic HPCI suction transfer.

RCIC is assumed inoperable in this event because it is not a safety system. The initial suppression pool level is specified as the TS limit of 24 ft. A LOOP is also assumed to occur coincidentally with the break to be consistent with the design-basis LOCA analysis to maximize the operating time of HPCI during the accident and maximizes the effect of the proposed HPCI modification on the containment response.

With the proposed change implemented, HPCI takes suction from the CST until the suppression pool reaches 25 ft, at which point the operator manually transfers HPCI suction from the CST to the suppression pool.

The proposed modification includes a new operator action to perform the CST to suppression pool HPCI suction transfer on high suppression pool level. The review of the operator's action and the risk significance of this operator's action to support the proposed HPCI modification are provided in Section 5 of this evaluation.

Based on current procedures for a small-break LOCA, the operator performs a controlled cooldown of the reactor, at 90 °F/hr, by occasionally opening an SRV to depressurize the reactor. The cooldown is initiated at 10 minutes into the event. One loop of suppression pool cooling becomes effective at 15 minutes into the event and suppression pool letdown via the RHR system to liquid radwaste is initiated at 30 minutes. Suppression pool letdown is used to control the level below 25 ft. The licensee's SABRE analyses predict the suppression pool temperature to remain near or below 140 °F, and the operator will transfer the suction to the CST when the level reaches 25 ft. EO-000-103, "Primary Containment Control," is being revised to instruct the operator to maintain suppression pool level less than 25 ft with the suppression pool letdown, which is part of the residual heat removal system. To account for uncertainty in the evaluation of the suppression pool temperature, if the suppression pool temperature exceeds 140 °F after the manual swap of HPCI pump suction to the suppression pool, the revisions to EO-000-103 will instruct the operator to realign the HPCI suction back to the CST.

If HPCI suction remains aligned to the CST, the suppression pool water level continues to increase during this event. Although it is not a licensing requirement to examine the consequences of a single equipment failure (for example, HPCI failure) which occurs during the long-term part of an accident, this concern was addressed by the licensee as discussed below.

Containment loads associated with a small-break LOCA combined with ADS actuation are within the design limits. Whenever the downcomer pipes are cleared, the air volume inside the SRV tailpipe is independent of the suppression pool level. This parameter is not affected by the proposed modification. The higher pool level associated with the modification only results in a higher back pressure on the SRV discharge line and has no effect on the amplitude of the SRV loads. For the 0.02 ft^2 break, the downcomer vents are predicted to be cleared for the first 970 seconds of the event. After 970 seconds, the downcomer vents begin to refill with water because the cold HPCI injection from the CST decreases the break enthalpy to the point where the coolant discharged to the drywell starts to have a cooling effect.

The state of the downcomer vents (open or closed) leads to two distinct situations for consideration when evaluating ADS loads with elevated suppression pool level. If the downcomer vents are cleared, the proposed modification has no effect on ADS hydrodynamic loads as discussed above. If the downcomer vents are sealed with water, there are no dynamic-pressure LOCA loads (for example, condensation oscillations or chugging) within the suppression chamber, but the ADS loads become dependent on suppression pool water level. In this case, the SRV loads associated with ADS actuation are acceptable as long as suppression pool level is below the Load-Limit curve.

The licensee's evaluation results show that the suppression pool water level is always below the Load-Limit curve and demonstrates that ADS actuation, necessitated by HPCI failure at any time during the event, would result in acceptable loads. At a reactor pressure of 1175 psig, the Load-Limit curve gives a suppression pool level of 26.1 ft. That is, the containment design allows for simultaneous actuations of SRVs with suppression pool water level up to 26.1 ft. During DBAs, the operator will maintain the level in the suppression pool at less than 25 ft.

During a small break LOCA, the revised EOPs will require the operator to shift HPCI pump suction between the CST and suppression pool in order to maintain pool level below 25 ft, and to prevent excessive suppression pool temperatures of > 140 ^{0}F from damaging the pump. Based on the above analyses, the staff finds the licensee's evaluation of a small-break LOCA acceptable to support the proposed HPCI modification.

3.2 Inadvertent Safety/Relief Valve Opening (IORV)

This event is discussed in Section 15.1.4 of the FSAR. The opening of an SRV will cause a mild depressurization transient and the pressure regulator will adjust the turbine control valves to stabilize pressure. When suppression pool temperature exceeds 90 °F, the operator will enter the Primary Containment Control EOP (EO-000-103). The procedure instructs the operator to initiate suppression pool cooling to restore pool temperature less than 90 °F. If the level exceeds 24 ft, the EOP also requires the operator to reduce suppression pool level with the suppression pool letdown systems. EO-000-103, "Primary Containment Control," is being revised to instruct the operator to maintain suppression pool level less than 25 ft with the suppression pool letdown, which is part of the residual heat removal (RHR) system.

If the SRV remains open, the suppression pool temperature will continue to increase and will reach 100 °F at about nine minutes into the event. Before the pool reaches this temperature, the operator will initiate a reactor scram in accordance with the EOPs. Following the reactor scram, the stuck open SRV will begin to depressurize the reactor. The reactor scram may cause a HPCI initiation on low reactor vessel water level (-38 inches).

Prior to the event, HPCI is not operating and it has no adverse effect on the air clearing load due to the IORV. Following the scram when HPCI is operating, the IORV has the potential for only producing steam condensation loads on submerged structures. Air clearing loads cannot be produced since this requires the SRV to close and then reopen. Steam condensation loads are bounded by SRV air clearing loads, as long as the suppression pool temperature response is maintained within the limits identified in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," U.S. Nuclear Regulatory Commission, November 1981 (SRP Section 6.2.1.1.c, “Pressure-Suppression Type BWR Containments”). The design-basis IORV transient analysis verifies that the pool temperature response to an IORV event remains within the limits of NUREG-0783. Therefore, SRV steam condensation loads when HPCI is operating do not adversely affect the SRV containment hydrodynamic loads. Therefore, the staff finds the licensee’s evaluation of an IORV event acceptable to support the proposed HPCI modification.

3.3 Primary System Pipe Break Outside Containment

Coolant injected by HPCI will not end up in the suppression pool for a break external to the primary containment. It exits the break within the secondary containment. HPCI injection also does not cause a rise in pool level. Steam would be added to the pool from the HPCI turbine exhaust, but this steam would also be present without the proposed modification. While the addition of steam to the suppression pool from the HPCI turbine exhaust would cause a slow rise in the pool level compared to a liquid break inside containment, there will be an ample margin to the Load-Limit curve.

Because the proposed change does not increase the severity of this event, the staff finds the licensee’s evaluation for breaks outside containment acceptable to support the proposed HPCI modification.

3.4 Inadvertent HPCI Initiation

This event is discussed in Section 15.5.1 of the FSAR. Only small changes in plant conditions are expected in this event because of the pressure regulator and water level controller response. Since no SRV actuations are expected, SRV/ADS hydrodynamic loads are not an issue.

Because the proposed change does not increase the severity of this anticipated transient, the staff finds the licensee’s evaluation for an inadvertent HPCI initiation event acceptable to support the proposed HPCI modification.

3.5 Loss-of-Feedwater (LOF) Flow

The reactor will scram when reactor vessel level drops to +13 in for a LOF flow event. The void collapse caused by the scram will generate a HPCI initiation on low level (-38 in). No SRV actuations are expected in this scenario because main steam isolation valves (MSIVs) remain open. Therefore, SRV/ADS hydrodynamic loads are not an issue.

Because the proposed change does not increase the severity of this anticipated transient, the staff finds the licensee’s evaluation for a LOF flow event acceptable to support the proposed HPCI modification.

3.6 Loss of Offsite AC Power

A LOOP initiates a reactor scram, recirculation pump trip, and MSIV closure. The effect of HPCI operation on containment hydrodynamic loads is the same as in the case of an inadvertent MSIV closure which is discussed in Section 3.8

3.7 Loss of Main Condenser Vacuum

A loss of main condenser vacuum leads to closure of the MSIVs. The relationship between HPCI operation and containment hydrodynamic loads for an MSIV closure is discussed in Section 3.8.

3.8 Inadvertent MSIV Closure

Closure of the MSIVs generates a reactor scram. HPCI will initiate on low reactor water level. The HPCI suction transfer logic has no impact on containment loads generated by SRV actuations during the pressurization event because HPCI is not operating prior to the MSIV closure. Following the MSIV closure, some cycling of SRVs will occur as decay heat is transferred to the suppression pool, but only the first group of valves (two valves) will open. With only a small number of SRVs cycling, minor suppression pool level transients are not of much concern with respect to containment hydrodynamic loads.

The safety setpoint for the first group of SRVs is 1175 psig. The design-basis event for SRV hydrodynamic loads is the American Society of Mechanical Engineers (ASME) Overpressurization Event that results in the maximum steam dome pressure, which envelopes the 1175 psig SRV opening pressure. A SABRE calculation estimates that the pool level will rise only about 1 in in the first 10 minutes following an MSIV closure. The margin in peak steam dome pressure overwhelms any negative effects associated with the small increase in suppression pool level. This conclusion can also be arrived at through consideration of the Load-Limit curve. At a reactor pressure of 1175 psig, the Load-Limit curve gives a suppression pool level of 26.1 ft. That is, the containment design allows for simultaneous actuations of SRVs with suppression pool water level up to 26.1 ft.

In the long-term (>10 minutes), it is assumed that the operator will initiate a controlled cooldown of the reactor in accordance with the EOPs. The suppression pool level response during the cooldown is bounded by the response for the small-break LOCA. Therefore, the suppression pool level is always well below the Load-Limit curve and there are no adverse consequences associated with SRV actuations during the cooldown.

Because the proposed change does not increase the severity of this anticipated transient, the staff finds the licensee's evaluation for an inadvertent MSIV closure event acceptable to support the proposed HPCI modification. This conclusion also applies to the licensee's evaluation of, (a) a loss of offsite AC power event - Section 3.6, (b) a loss of main condenser vacuum event - Section 3.7, (c) a turbine trip (with and without bypass) - Section 3.9, (d) a generator load rejection (with and without bypass) - Section 3.10, (e) a pressure regulator failed-closed event - Section 3.11, and (f) a pressure regulator failed-open event - Section 3.12.

3.9 Turbine Trip (with and without Bypass)

The most severe case with respect to containment hydrodynamic loads involves failure of the bypass valves because it results in a higher reactor pressure and a larger number of open SRVs. HPCI suction transfer logic has no influence on containment loads generated by SRV actuations during the pressurization event because HPCI is not operating prior to the turbine trip. Following a turbine trip event, it is unlikely that HPCI would be used for vessel make-up because feedwater would be available. If for some reason HPCI is used for vessel make-up following the vessel pressurization transient, its impact on containment loads is no different than that already discussed in Section 3.8.

3.10 Generator Load Rejection (with and without Bypass)

For purposes of evaluating the impact of the proposed plant modification on the containment loads, this transient is the same as the turbine trip with and without bypass (Section 3.9).

3.11 Pressure Regulator Failure - Closed

This transient is discussed in Section 15.2.1 of the FSAR. If the backup pressure regulator is also assumed to fail, then a reactor pressurization will result and the reactor will scram on high vessel pressure or high neutron flux. This pressurization event is less severe than the turbine trip, as discussed in Section 3.9.

3.12 Pressure Regulator Failure - Open

This event is discussed in Section 15.1.3 of the FSAR (Rev. 54, 10/99). Failure of the pressure regulator causes a reactor depressurization that initiates closure of the MSIVs. The MSIV closure generates a reactor scram. Here MSIV closure occurs at reduced reactor pressure so SRV actuations do not occur. Later in the transient, SRV cycling will occur as decay heat is removed from the RPV. SRV cycling following an MSIV closure with HPCI injecting to the vessel has already been addressed in Section 3.8.

3.13 Diaphragm-Slab Differential Pressure

The higher suppression pool water level associated with the proposed modification does not impact the diaphragm-slab differential pressure or drywell-negative-pressure analyses (FSAR Section 6.2.1.1.4). These analyses assume that drywell sprays are initiated during a small break accident and that all noncondensible gases are contained within the wetwell air space at the time of the spray actuation. In addition, the wetwell temperature is non-mechanistically set to 50 °F. With the HPCI auto suction transfer elimination, the suppression pool level can rise to 25 ft during a small-break LOCA. This causes a reduction in the wetwell air space volume. If a smaller wetwell air space volume is considered in the diaphragm-slab-differential-pressure and drywell-negative-pressure analyses, the results will be more favorable because the wetwell will exhibit a faster pressure response upon opening of the vacuum breakers. That is, the wetwell pressure will more closely follow the drywell pressure. Therefore, it is conservative to neglect the reduction in the suppression chamber-free volume when computing the diaphragm slab differential pressure and the drywell peak negative pressure.

Because the proposed change would result in more favorable conditions on the diaphragm-slab differential pressure, the staff finds the licensee's evaluation for this component acceptable to support the proposed HPCI modification.

3.14 Safety-Related Valves on Piping Connected to Suppression Chamber

Safety-related valves on piping connected to the suppression chamber provide flow paths for ECCS and suppression pool cooling. Other valves on piping connected to the suppression chamber include SRVs and vacuum breakers on the downcomer vents. SRVs prevent overpressurization of the reactor vessel and the downcomer-vent vacuum breakers equalize pressure across the drywell floor in the event of a LOCA. The licensee evaluated the potential effects of the elimination of the automatic HCPI suction transfers on these components.

3.14.1 Motor-Operated Valves (MOVs)

Suppression pool level could potentially increase by 1 ft during a design-basis small-break accident as a result of this modification (i.e., 24 ft to 25 ft). The increase in pressure due to the additional 1 ft is 0.43 psi (based on a suppression pool temperature of 90 $^\circ$F). This small increase in pressure will not adversely affect the pressure retaining capability of any valve on piping connected to the suppression pool. In addition, the ΔP across these valves could increase by 0.43 psi, depending on valve function. The ability to open or close these valves in accordance with applicable design criteria is not affected by this change in ΔP as documented in MOV design-basis calculations. These calculations conclude that there are no adverse effects to the operation or performance of any valve on piping connected to the suppression pool as a result of this small pressure increase.

The HPCI automatic suction transfer logic elimination does not increase the severity of the suppression pool temperature transient in a small-break accident. In fact, the suppression pool temperature rise (for a small-break LOCA) would be larger under the current plant configuration than it would be with the proposed modification installed. A smaller suppression pool temperature rise would result because of the additional mass added to the suppression pool when HPCI suction is maintained on the CST until pool level reaches 25 ft. Since the energy deposited in the suppression pool is unchanged by the proposed modification, the additional mass leads to a smaller increase in pool temperature. Thermal locking effects due to a suppression pool temperature increase are already considered in the Generic Letter 95-07 operability evaluation and, since the suppression pool temperature response for the proposed modification is bounded by the current response, there is no impact on valve thermal locking.

3.14.2 Vacuum Breakers

Allowing suppression pool level to potentially increase to 25 ft in a design-basis accident does not impact operation of downcomer-vent vacuum breakers because the vacuum breakers are located 42 ft above the bottom of the suppression pool.

3.14.3 SRVs/Tailpipes

The increased suppression pool level associated with the proposed change has no effect on SRV operation because flow through the SRVs is choked. SRV flow is decoupled from downstream conditions when the flow through the valves is choked. The higher suppression pool level does, however, lead to a higher peak pressure within the SRV tailpipe upon valve actuation. When an SRV opens, the SRV tailpipe rapidly pressurizes as the slug of water within the pipe is expelled. As suppression pool water level increases, there is an equivalent increase in the water column height within the tailpipe, and consequently, the maximum pressure buildup within the tailpipe increases with pool level. Although higher suppression pool levels are expected under accident conditions with the proposed change, the magnitudes are such that SRV tailpipe failure is not expected to occur. With the proposed change, suppression pool level could potentially increase to 25 ft. Design-basis loads on the SRV system are conservatively based on an initial pool level of 35 ft. This level is 10 ft above the maximum suppression pool level that could occur with the proposed change. For a LOCA, the available margin is much greater than 10 ft because the reactor pressure continually decreases during the event. For beyond-design-basis conditions, the EOPs require a reactor depressurization before pool level reaches the point where SRV tailpipe integrity is threatened.

Because the proposed change will not adversely affect the operation of these components, the staff finds the licensee’s evaluation for the MOVs, vacuum breakers and the SRVs/tailpipes connected to suppression chamber acceptable to support the proposed HPCI modification.

3.15 RCIC Turbine

The RCIC system functions to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of feedwater. RCIC does not perform a safety-related function, except for containment isolation. The RCIC system is, however, classified as an Appendix R Safe Shutdown System and may be used for vessel injection in the event of a fire on the site.

RCIC is used to provide coolant make-up following a reactor vessel isolation event. In an MSIV closure event, HPCI and RCIC would initiate on low RPV water level. These systems would inject into the vessel until they automatically trip when the RPV level reaches 54 in. There would be no additional HPCI/RCIC initiations within the first 10 minutes of the event. After 10 minutes, it can be assumed that the operator will use RCIC for RPV make-up and HPCI will be used for pressure control (in the CST-to-CST mode). Therefore, the long-term part of the scenario, greater than 10 minutes, is unaffected by the proposed change.

RCIC may be used for Appendix R Safe Shutdown, however, the shutdown scenario assumes vessel isolation and the effects are similar to the isolation event described above.

If the initial suppression pool level is at the TS limit of 24 ft, then the suppression pool level will be about 2 in higher with the proposed change. With the current design, the suction transfer would occur at a level of 23 ft 10 in. This small level change has negligible effect on RCIC turbine exhaust pressure and a large margin remains to the turbine exhaust line elevation of 25.1 ft. In the more realistic situation where the initial suppression pool level is at a nominal value of 23 ft, the proposed change has no effect on containment response.

Although RCIC is not designed for vessel make-up in a small-break accident, the licensee examined the impact of the proposed change on RCIC operation under LOCA conditions. Flooding of the exhaust line is not an issue for the RCIC turbine. The bottom of the horizontal section of turbine exhaust piping corresponds to 25.83 ft and the horizontal run of piping cannot become flooded because the suppression pool level will not exceed 25 ft in a DBA. Continued steady-state operation is not applicable to the RCIC system. The maximum turbine exhaust pressure for continuous operation is 25 psia (DBDO4 1, Rev. 0, Requirement 2.3.2.1.4) and this value would be exceeded in a small-break LOCA even if there is no increase in the suppression pool level. The maximum increase in the suppression pool level is one foot (24 ft to 25 ft) which has negligible effect on RCIC turbine operation as it corresponds to a pressure increase of only 0.43 psi at the turbine exhaust.

Because of the negligible effect of the proposed change on RCIC operation, the staff finds the licensee's evaluation for the RCIC turbine acceptable to support the proposed HPCI modification.

3.16 SABRE Computer Program Evaluation

The SABRE computer program was used by the licensee to evaluate the effects of the HPCI modification on the MSIV closure event and for intermediate and small break LOCAs.

Three system evaluation criteria were used to evaluate the SABRE code results for these accident scenarios. First, the operator needs to maintain the suppression pool water level less than or equal to 25 ft, which is just below the elevation of the HPCI and RCIC turbine exhaust lines. Second, for DBAs the suppression pool level should be low enough to maintain containment hydrodynamic loads within design limits. Third, the suppression pool water temperature should be less than 140 °F when the HPCI is manually switched from the CST to the suppression pool on high water level.

Based on these criteria and the accident scenarios, the dominant physical phenomena are:

1. Steady state in-vessel forced circulation and vapor generation
2. Steady state and transient heat conduction
3. Containment drywell liquid retention
4. Suppression pool steam condensation and mixing
5. HPCI pump flow and injection

3.16.1 Assessment of the SABRE Computer Program

To model these physical phenomenons, a computer program should have the capability to calculate the initial coolant mass and energy inventory, sensible heat in the solid heat structure and preserve the mass and energy during the transient. Transient heat conduction models should be incorporated into the program to model the transient heat transfer between the heat structure surface and the coolant. Conservative or realistic models should be included in the program to calculate the drywell and suppression pool inventory build-up and temperature responses. A transient HPCI injection model needs to be provided to calculate the HPCI injection mass flow rate based on the changing downstream vessel pressure.

The staff reviewed the SABRE computer program to check the following items with the emphasis on analyzing the impact of the HPCI modification:

- Mass and energy conservation equations
- Fuel rod conduction calculation
- Core power calculation for LOCA scenarios

- Containment model
- HPCI flow model
- Time step synchronization
- Quality assurance

3.16.1.1 Mass and Energy Conservation Equations

A 1-D approach is used in the SABRE code to model the fluid flow in the jet pump, the lower plenum, the core, the bypass, the upper plenum, the riser and the separator regions. The total two-phase mixture mass is conserved. The fluid kinetics energy dissipation is neglected. This is considered acceptable since the magnitude of the kinetics energy dissipation is usually small. The drift-flux model is employed to model the slip between the liquid phase and the vapor phase. An equilibrium assumption is used to calculate the liquid and vapor properties in the sub-cooled region, saturated region and superheated vapor region. This assumption can introduce significant vapor or liquid temperature errors when ECCS injection, core spray, is initiated. However, for the specific scenarios of interest, most of the vessel internal fluid regions experience a slow depressurization process and an equilibrium assumption should not introduce significant mass and energy errors.

The SABRE reactor core model lumps all the fuel bundles into a 1-D average thermal-hydraulic channel. An evaluation performed by the licensee using a 3-D core simulator (SIMULATE-3) concluded that the SABRE code can calculate the core average void fraction reasonably well. Therefore, the initial steady-state in-core coolant and energy inventory calculated with SABRE 1-D core model is considered reasonable. The pressure distribution in the upper-plenum and lower plenum is usually uniform in the radial direction. Therefore, the pressure drop across the 1-D average fuel channel should reasonably represent the scenarios of interest. The calculated average pressure drop across the core is considered reasonable.

Variable volumes are used to model the downcomer and the steam dome regions. Momentum transfer is not considered in these regions. A lumped parameter approach is used for the mass and energy balance. The potential and kinetic energy effects are neglected as their magnitudes are small. Although the approach may not be accurate enough to model pressure wave propagation or other transients, it is considered applicable to the mass and energy calculation needed to support the HPCI modification. The main steam line beyond the MSIV is modeled with SABRE using a controlled volume approach. A simple mass balance equation is used for the steam flow through the main steam line isolation valve, the bypass valve, and the flow to the main turbine and feedwater turbines. The pressure is calculated assuming that the steam behaves as an ideal gas.

Each reactor has two recirculation lines. For the HPCI modification, proper modeling of the fluid and the metal mass of the recirculation lines is important during a small-break LOCA as the fluid mass and metal mass contain significant portions of the total stored energy in the reactor system. The original SABRE calculations did not model the fluid and metal mass of the recirculation lines and was considered by the staff to be inadequate for the evaluation of the HPCI modification. The staff raised the issue to the licensee regarding the impact on the suppression pool temperature and the time to reach the 25 ft suppression pool level. In response, the licensee revised the SABRE code and recalculated the key parameters. The staff finds the revisions to SABRE acceptable for the HPCI modification evaluation.

Because the SABRE code is able to accurately model the parameters required for this analysis, the staff finds the mass and energy conservation equations as modeled in SABRE acceptable for the HPCI modification evaluation.

3.16.1.2 Fuel Rod Conduction Calculation

The fuel and cladding transient heat conduction is modeled using a radially-lumped parameter model. The axial conduction is neglected and the fuel pin is radially divided into two fuel conduction zones and the cladding zone. The transient fuel pin temperature versus reactor power and the heat transfer coefficients are calculated with this model. The calculated transient heat transfer process is reasonable for predicting the energy removal from the core for the HPCI modification evaluation. Therefore, the staff finds the fuel rod conduction calculation as modeled in SABRE acceptable for the HPCI modification evaluation.

3.16.1.3 Core Power Calculation for LOCA Scenarios

A 1-D kinetics model based on the two-group, one-dimensional neutron diffusion equation is used in SABRE to compute the core power. Although the core kinetics behavior is very important for ATWS simulation, the current application is focused on the reactor and containment response during a small-break LOCA. Therefore, the 1-D kinetics model and its numerical solution scheme were not reviewed. The decay heat, however, is the dominant power source to heat the core during a LOCA. In SABRE, decay heat power is computed from the eleven-equation decay heat model used in the RETRAN code. Since the RETRAN methodology for transient calculation has been approved by the NRC (Reference: Letter, U.S. Nuclear Regulatory Commission to Mr. Harold W. Keiser, PPL Susquehanna, Subject: Topical Report PL-NF-89-005, “Qualification of Transient Analysis Methods for BWR Design and Analysis,” Susquehanna Steam Electric Station, Units 1 and 2 (TAC Nos. M82371 and M82372), the decay heat model was not examined further.

SABRE lumps all the vessel metal mass into one lumped metal mass and all the vessel internal metal mass into another. Only one average heat structure temperature is assumed for each lumped metal mass. The summation of the heat flux from the surface of the vessel wall and the vessel internals is added into the core decay heat equation in order to simplify the heat transfer model. Although this approach ignores the local effect of vessel and vessel internal heat structure temperature distribution, the overall metal sensible heat effect is modeled and its impact on the suppression pool temperature and level is considered in SABRE. Therefore, the staff finds the core power calculation for LOCA scenarios as modeled in SABRE acceptable for the HPCI modification evaluation.

3.16.1.4 Containment Model

SABRE models the suppression pool and drywell pressure and temperature response to an ATWS, a small-break LOCA, or a normal plant transient. The drywell is modeled as one control volume. The suppression pool free volume has one temperature and pressure. Simple ordinary differential equations are used to model the mass and energy conservation equations. SABRE also models the vacuum breakers between the wetwell air space and the drywell. The thermal response of the structural material in both the drywell and the wetwell is also modeled. The drywell floor flooding is modeled so that the liquid will flow through the downcomer when the depth of the water layer on the drywell floor reaches 18 in. A simple heat removal model is used to calculate the suppression pool cooling system heat removal capability.

The SABRE containment model was benchmarked against the CONTAIN computer program by the licensee (Reference: Licensee calculation EC-ATWS-0505 Rev.8. "SABRE: A Computer Code for Simulation of Boiling Water Reactor Dynamics Under Failure-to-Scram Conditions"). The calculated suppression pool temperature during a postulated ATWS and a small-break LOCA was compared to CONTAIN results. Good agreement was obtained between the SABRE and the CONTAIN codes. The suppression pool heat-up process is an important physical process associated with the HPCI modification.

Because of its modeling techniques for mass and energy, and its close agreement with results obtained by the CONTAIN computer program, the staff finds the containment model in SABRE acceptable for the HPCI modification evaluation.

3.16.1.5 HPCI Flow Model

Based on the specified HPCI capability, a correlation between the mass flow rate and the vessel pressure was developed for use in SABRE. The correlation for the steam flow through the HPCI turbine to the suppression pool was also developed. Since the system pressure remains close to the rated pressure during the initial 1000 seconds of a small-break LOCA, SABRE calculates a constant HPCI mass flow rate. This HPCI model predicts the correct HPCI flow rate for the period of time important to the calculation of the time to reach the 25 ft suppression pool level. Since the mass and energy are either discharged through the SRV or the break, the predicted HPCI mass flow rate does not have a significant impact on the suppression pool temperature. Therefore, the staff finds the HPCI flow model in SABRE acceptable for the HPCI modification evaluation.

3.16.1.6 Time Step Synchronization

The SABRE core power calculation and the thermal-hydraulic calculation do not always use the same time step size. A question was raised by the staff about the synchronization between these two solvers and the impact on the calculated core power. In SABRE, the hydraulics and neutronics calculations are performed sequentially. The neutronics calculation lags the thermal-hydraulics calculation by at least one time step. The thermal-hydraulics time step size is usually larger than that of the neutronics and is set by the user as part of the input. The neutronics solver chooses a time step size based on a user specified error criteria. If the neutronics time step size is greater than the hydraulics step, the solver integrates beyond the thermal-hydraulics time step; however, a solution at the end of thermal-hydraulics time step is obtained by interpolation. If the neutronics step size is smaller than the hydraulics step size, multiple kinetics steps are taken for each hydraulics step and interpolation is used to obtain the kinetics output at the end of the thermal-hydraulic time step. Sensitivity cases have been run by the licensee to evaluate the impact of different time step size on the calculated ATWS power history. A small change in the reactor peak power was observed. For LOCA cases, however, the coupling scheme between the neutronics and the thermal-hydraulics gives higher step-wise power data and the accumulated total energy release from the fuel rod to the coolant should be conservatively higher. This time step synchronization scheme model in SABRE is considered reasonable for the HPCI modification evaluation. Therefore, the staff finds it acceptable for the HPCI modification evaluation.

3.16.1.7 Quality Assurance

The licensee developed a version control and quality assurance procedure to maintain SABRE on a Hewlett Packard workstation (HP-j2240-c) in licensee calculation EC-SATH-1007 “Installation of SABRE Computer Code in Systems Analysis.”

The staff finds the quality assurance program for SABRE to be in compliance with the criteria contained in the Standard Review Plan (NUREG-0800) and, therefore, acceptable for the HPCI modification evaluation.

3.16.2 Susquehanna Transient Reactor Analysis Code (TRAC) Model Development

A Susquehanna TRAC model was developed by the staff based on a standard BWR-4 model to perform a confirmatory analysis of a selected small-break LOCA accident to assess the performance of SABRE for the HPCI modification evaluation.
The 0.02 ft^2 break is considered a typical small-break LOCA case and it was analyzed by the licensee. This case was analyzed using the TRAC model. A steady-state calculation was performed to calculate the initial in-vessel mass and energy inventory. The initial total reactor power is 3510 MW(t) and the system pressure is 1050 psia. The drywell water retention factor is assumed to be 0.45 before the drywell floor water level reaches the 18-in clearance point based on the results obtained by the licensee with SABRE. The TRAC calculated suppression pool level reaches 25 ft at about 21 minutes, as compared to the licensee’s result of 22.5 minutes.

3.16.2.1 Sensitivity Analyses

Although the licensee’s calculation for the time to reach 25 ft is close to the TRAC result, the drywell liquid retention factor is an important parameter. In order to conservatively calculate the time when the pool reaches 25 ft, another TRAC run was made assuming that no water mass is retained in the drywell during the transient. The suppression pool reaches 25 ft at 11.7 minutes. The drywell liquid retention factor has a significant impact on the time when the suppression pool level reaches 25 ft.

The break size also affects the calculated time. The larger the break size, the faster the suppression pool floods. A simple hand calculation, based on the maximum HPCI flow rate without retaining water in the drywell indicates that the 25 ft level could be reached in about 8 minutes. The licensee’s SABRE results indicate that the minimum time to each 25 ft is 21.0 minutes for a 0.04 ft^2 break.

3.16.2.2 Summary

The staff has reviewed the SABRE computer program for its applicability to the HPCI modification evaluation. In addition, the staff developed a TRAC model for Susquehanna and performed a confirmatory analysis of the 0.02 ft^2 small-break LOCA. The staff finds the SABRE computer program acceptable for the qualitative evaluation of the proposed HPCI modification.

The licensee’s analysis of primary containment response to small breaks with the proposed HPCI modification has been reviewed. The following conclusions were made:

- During a small-break LOCA, the earliest time the suppression pool reaches 25 ft could be about 8 minutes as compared to the 21 minutes calculated by the licensee. The revised EOP will instruct the operator to transfer HPCI suction from the CST to the suppression pool when the level reaches 25 ft, as long as the suppression pool temperature remains less than or equal to 140 °F.

- The original SABRE code underestimated the suppression pool temperature as a result of not modeling the recirculation line fluid and metal mass and fuel bundle in-channel metal mass. The calculated maximum suppression pool temperature was 135 °F. Based on the revised SABRE code, the calculated maximum suppression pool temperature is 141 °F. Therefore, during a small-break LOCA, the suppression pool may exceed the HPCI pump operating limit for a short period of time prior to the initiation of the core spray and the operator may need to switch the HPCI pump suction from suppression pool back to the CST.

3.17 Risk Assessment

The disadvantage of the HPCI modification is that, without the automatic suction transfer, the suppression pool water level will increase during a small-break LOCA and eventually reach the elevation of the horizontal portion of the HPCI turbine exhaust line (25.1 ft). While the licensee’s evaluation indicates that it would take about 21 minutes to reach the transfer setpoint of 25 ft, it may take less time for this level to be reached during a small-break LOCA. Tripping the HPCI pump with the water level in the suppression pool higher than the HPCI turbine exhaust line would allow water to enter the exhaust line and threaten the ability of the HPCI system to successfully restart. To minimize the risk associated with the tripping of the HPCI pump, the licensee proposes to add an action to their EOPs to ensure that the HPCI suction is manually transferred to the suppression pool whenever the suppression pool level is above 25 ft, as long as the suppression pool temperature remains less than or equal to 140 °F. In addition, as already directed by the EOPs, the operators have to gain and maintain control of RPV water level below the HPCI high level trip setpoint (level 8) to avoid tripping the HPCI pump.

3.17.1 Assessment of Plant-Specific PRA

To evaluate the risk implications of the proposed license amendment, the licensee assessed their plant-specific Probabilistic Risk Assessment (PRA) for internal events. The staff reviewed the information provided in the licensee’s June 8, 2001, submittal and the licensee’s associated supplemental information and responses to the staff’s RAIs. In particular, the staff reviewed the licensee’s risk evaluation of the HPCI modification presented in licensee calculation EC-RISK-1083, Revision 1.

The staff used the guidance provided in Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” The staff’s evaluation of the licensee’s submittal focused on the capability of the licensee’s analysis to evaluate the risk impacts stemming from both the current, pre-modification plant operations, and the proposed, post-modification, conditions. The staff’s evaluation did not involve an in-depth review of the licensee's PRA, but rather focused on the aspects of the risk analysis potentially impacted by the proposed modification. Further, since the proposed modification was not expected to significantly impact the risks associated with external events or shutdown operations, the staff did not evaluate these conditions.

Based on the information provided by the licensee and the staff and expert reviews of the licensee's SSES individual plant examination (IPE), the staff concludes that there is sufficient evidence that the quality of the risk analysis is adequate for this specific license application.

In the licensee's risk analysis, the current core damage frequency (CDF) and large early release frequency (LERF) were calculated to be 5.29×10^{-7}/year and 1.45×10^{-8}/year, respectively. As a result of the HPCI modification, the mean CDF and LERF values were calculated to be reduced to 4.86×10^{-7}/year and 1.39×10^{-8}/year, respectively. The licensee's risk analysis indicates that the HPCI modification would result in a reduction in CDF of 4.3×10^{-8}/year; a decrease of about 8 percent. It would also result in a reduction in LERF of 6.0×10^{-10}/year; a decrease of about 4 percent.

The staff evaluated the licensee's modeling assumptions and approach to ensure that they were reasonable and appropriately considered the main risk implications of the HPCI modification, including the potentially negative risk impacts for small-break LOCAs. In particular, for small-break LOCA sequences, the operators will have to control the RPV level to avoid tripping the HPCI pump at its high-level trip setpoint (level 8). The licensee did not explicitly model all potential causes for a HPCI pump trip, but they did address the need for the operators to gain control of the RPV level and provided the results if this action always failed. These results would bound the analysis of other means of losing or causing a trip of the HPCI pump. Under these conditions, the CDF value is increased to 4.92×10^{-7}/year, which is still less than the current plant CDF. The LERF value remains unchanged as the small-break LOCA does not contribute appreciably to LERF.

The calculated risk reduction is driven almost exclusively by the fact that the HPCI modification will allow the continued operability of the HPCI system during all ATWS events, including high-powered ATWS events that involve failure of the SLCS. The only calculated risk increase associated with the modification is from small-break LOCA sequences, whose CDF increases by about 47 percent. However, since small-break LOCAs contribute less than one-tenth of one percent to the plant's risk and ATWS events contribute over 15 percent to plant CDF and over 45 percent to plant LERF, the proposed HPCI modification results in the cited overall risk reduction.

Based on the staff's evaluation of the risk-related information provided by the licensee, the staff finds that there is reasonable assurance that the overall impact of the proposed modification will be a risk reduction and therefore is acceptable.

3.18 Technical Summary

The licensee proposed to amend the SSES-1 and 2 TSs. These changes would delete Function 3e from TS Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation," for both units. Implementation of this proposed change would eliminate the automatic transfer of the HPCI pump suction source from the CST to the suppression pool on a high suppression pool level. This change, and the implementation of its associated plant modifications, is needed to eliminate the vulnerability identified by the SSES IPE - an ATWS event combined with a failure of the SLCS.

The staff reviewed the licensee’s assessment of the effects of the proposed change to remove the automatic HPCI transfer from the CST to the suppression pool on containment hydrodynamic loads during DBAs.

Manual HPCI suction transfer is not needed to maintain containment hydrodynamic loads within design limits. LOCA blowdown loads are dependent only on the suppression pool level at the initiation of the accident and this is unaffected by the proposed modification. Although a rising suppression pool level by itself intensifies hydrodynamic loads associated with SRV/ADS blowdown, the pool level increase during a LOCA is accompanied by a decrease in reactor pressure. The reduction in reactor pressure offsets the adverse effects associated with the increase in pool level.

The licensee also assessed the effects of the proposed HPCI modification on the diaphragm-slab differential pressure, on the safety-related valves and piping connected to the suppression chamber, and on the RCIC turbine.

The staff finds the licensee’s assessment of the effects of the proposed HPCI modification acceptable for reasons explained in sections 3.13, 3.14, and 3.15 of this safety evaluation.

Implementation of the proposed change will involve removal of contacts in the open logic of the HPCI F042 suction swap valve so that the F042 valve will only be automatically opened due to low CST level. The HPCI injection valve permissive logic that is solely associated with the suppression pool high level automatic transfer will also be removed. The F042 valve can also be manually operated from the control room.

Applicable SSES procedures (including the EOP's) will also be revised to include actions to manually open the F042 valve when appropriate. Procedures will be revised to replace this automatic action with operator instructions to manually swap the HPCI suction source to the suppression pool when suppression pool level reaches 25 ft as long as suppression pool water temperature remains less than or equal to 140 °F. To account for the possibility of the suppression pool temperatures exceeding 140 °F, the procedures will also be revised to instruct the operator to realign the HPCI suction to the CST if the suppression pool temperature cannot be maintained below 140 °F after the manual transfer has been made. The procedures will also be revised to instruct the operator to maintain suppression pool level less than 25 ft with the suppression pool letdown, which is part of the RHR system.

The manual HPCI suction swap would be performed for a narrow range of small liquid break LOCAs to prevent the suppression pool level from exceeding the elevation of the HPCI turbine exhaust piping. Without the manual transfer, the suppression pool level could reach the HPCI turbine exhaust line elevation and a potential for a loss or failure of the HPCI system could arise, but only if the operator allows the HPCI to trip on high reactor water level and subsequently fails to prevent a HPCI restart on low reactor level. Current procedures direct the operator to gain and maintain control of the RPV level below the HPCI high level trip setpoint to avoid tripping of the HPCI pump. The licensee assessed the risk implications of the loss of the HPCI system.

The licensee provided quantitative core damage frequency and large early release frequency results for both the current, pre-modification plant operations and the post-modification conditions. Based on the staff's evaluation of the risk-related information provided by the licensee, the staff finds that there is reasonable assurance that the licensee's risk analysis used to support this license application is adequate and that the overall impact of the proposed modification will be a risk reduction. In accordance with RG 1.174 , when an application clearly can be shown to result in a decrease in risk, it is considered to have satisfied the relevant principles of risk-informed regulation. Therefore, the staff concludes that the licensee's application is acceptable from a risk perspective.

The staff determined that the proposed TS modification to delete Function 3e from TS Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation," for SSES-1 and 2 is acceptable. For any of the narrow range of small liquid break LOCAs during which the HPCI suction swap would be required by EOPs, the resultant operator actions to realign HPCI pump suction as necessary result in a decrease in overall plant risk. Additionally, the proposed changes do not increase the severity of any anticipated transients or design basis accidents.

The proposed change also corrects a typographical error for function 5a, "ADS Trip System B Reactor Vessel Water Level," in the Unit 1 TSs, Table 3.3.5.1-1, to indicate "Low, Low, Low Level 1" instead of "Low Level 1." The Unit 2 TS does not contain this error.

The staff determined that there is reasonable assurance that SSES-1 and 2 will continue to be in compliance with General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," and GDC 50, "Containment design basis," following the proposed HPCI modification.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 50471). Accordingly, the amendments meet the 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.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: E. Throm, S. Lu, D. Harrison

Date: August 5, 2002

Mr. Bryce L. Shriver
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603-0467

August 5, 2002

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: HIGH PRESSURE COOLANT INJECTION PUMP AUTOMATIC TRANSFER TO SUPPRESSION POOL LOGIC ELIMINATION (TAC NOS. MB2190 AND MB2191)

Dear Mr. Shriver:

The Commission has issued the enclosed Amendment No. 204 to Facility Operating License No. NPF-14 and Amendment No. 178 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 8, 2001 and supplements dated February 4, April 8, May 7, June 6, and June 28, 2002.

These amendments revise TS 3.3.5.1, "Emergency Core Cooling System Instrumentation," by deleting Function 3e, thus preventing the automatic swap of the suction source for the high pressure coolant injection pump from the condensate storage tank to the suppression pool on high suppression pool level. This change, and its associated plant modifications, eliminates a vulnerability identified by the SSES-1 and 2 Individual Plant Examination; an anticipated transient without scram event combined with a failure of the standby liquid control system.

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 by TColburn for/

Eric M. Thomas, 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. 204 to License No. NPF-14
2. Amendment No. 178 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION</u>

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TColburn	SDWeerakkody			

Package: ML022180418

ACCESSION NO.: ML022000064 **TSs: ML022180186** *SE provided, no major changes made

OFFICE	PDI-1/PM	PDI-2/LA	SPLB	SRXB	SPSB
NAME	TColburn for EThomas	MO'Brien	SDWeerakkody*	RCaruso*	MPRubin*
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NAME	JHeck	PTam for RLaufer			
DATE	7/30/02	8/1/02			

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