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

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 239 TO LICENSE
NFP-14 AND PROPOSED AMENDMENT NO. 204 TO
LICENSE NFP-22: HPCI PUMP AUTOMATIC TRANSFER
TO SUPPRESSION POOL LOGIC ELIMINATION
PLA-5322**

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

Reference: 1) PLA-4948, R. G. Byram to US NRC, "Withdrawal of Proposed Amendment No. 157 to License No. NPF-14 and Proposed Amendment No. 107 to License NPF-22: Changes to HPCI Suction Transfer Logic", dated October 23, 1998.

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Units 1 and 2 Technical Specifications. The change deletes from Technical Specification Table 3.3.5.1-1 the "High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level – High" (Function 3e) for both units. Implementation of this proposed change eliminates automatic transfer of the HPCI pump suction source from the Condensate Storage Tank to the Suppression Pool for a high Suppression Pool level. This change and implementation of the associated plant modifications is essential to elimination of a vulnerability identified by the Susquehanna SES IPE.

Reference 1 withdrew a similar proposed change to the SSES Technical Specifications. Reference 1 indicated that PPL was evaluating options that could result in a re-submittal of the proposed change as a risk-based submittal. PPL has completed the evaluation and determined that the withdrawn submittal supplemented by additional supporting analyses would be the most appropriate means to justify the change to eliminate this plant vulnerability.

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

A001

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 provides the "camera ready" version of the revised Technical Specification pages.

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

PPL plans to implement the proposed changes during the Unit 1 Refueling and Inspection Outage scheduled to begin in March 2002 and the Unit 2 outage in March 2003. Therefore, we request NRC to complete the review of this change request by December 31, 2001 to support our scheduled implementation dates.

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

Sincerely,

R. G. Byram

Attachments

copy: NRC Region I
Mr. R. G. Schaaf, NRC Sr. Project Manager
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. W. P. Dornsife, PA DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

PROPOSED AMENDMENT NO. 239 TO LICENSE NPF-14:
HPCI PUMP AUTOMATIC TRANSFER
TO SUPPRESSION POOL LOGIC ELIMINATION
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files Proposed Amendment No. 239 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 8th day of June, 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. 204 TO LICENSE NPF-22:
HPCI PUMP AUTOMATIC TRANSFER
TO SUPPRESSION POOL LOGIC ELIMINATION
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files Proposed Amendment No. 204 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 8th day of June, 2001.

Nancy J. Lannen
Notary Public

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

Attachment 1 to PLA-5322

Safety Assessment

SECTION I

SUMMARY OF PROPOSED CHANGE

In an Anticipated Transient Without Scram (ATWS) event with Standby Liquid Control System (SLCS) failure, the operator can reduce reactor power by manually inserting control rods, one at a time, via the Rod Drive Control System. This event, which is evaluated in the Susquehanna IPE (Individual Plant Evaluation), is conservative with respect to the requirements of the ATWS Rule, 10 CFR 50.62. The capability for manual control rod insertion is the result of a 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 RWM (Rod Worth Minimizer) which can also be performed from the Control Room.

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 the High Pressure Coolant Injection system (HPCI) as this is the only high-pressure injection system which can maintain vessel inventory in an isolation ATWS. Current plant design includes an auto-transfer of HPCI suction from the Condensate Storage Tank (CST) to the suppression pool on elevated pool level. The set point for the suction transfer, 23′-10″, is reached in the first few minutes of the ATWS. Since suppression pool temperature increases rapidly, and HPCI is cooled by suction water, HPCI failure on loss of lube oil cooling is expected early in the event (~10 minutes). Loss of HPCI before power is appreciably reduced by manual rod insertion requires rapid depressurization of the reactor in order to obtain coolant makeup from low-pressure injection systems. Core damage from unstable operation is expected upon depressurization of a critical core.

Although Susquehanna EOPs (Emergency Operating Procedures) instruct the operator to manually bypass the HPCI auto-suction transfer on high suppression pool level and realign HPCI suction back to the CST in an ATWS event, the 10 minutes available to the operator is insufficient to perform the bypass. The manual bypass is performed outside the Control Room and completion is expected to take in excess of 30 minutes. With the present plant configuration, the operator has a means of shutting down the reactor in an ATWS with SLCS failure, but there is insufficient time available to achieve reactor shutdown at high pressure conditions where the reactor core is stable.

This proposed change deletes from Technical Specification 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 Technical Specifications. The allowable value for this function as delineated in the Table is 23 feet and 11 inches. Elimination of Function 3e will increase the availability of the HPCI system during a postulated ATWS with SLCS failure and provide the operator with sufficient time to manually insert control rods and shutdown the reactor at high pressure. In addition the change reduces operator burden during a postulated Station Blackout Event (SBO). This change does not adversely affect the ability of the HPCI system or other plant systems to perform their design basis functions. This change was identified by the SSES Individual Plant Evaluation (IPE).

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 feet as long as suppression pool water temperature is less than or equal to 140°F. The manual HPCI suction swap would be performed in a narrow range of small liquid break LOCAs in order to prevent suppression pool level from exceeding the elevation of the HPCI turbine exhaust piping. This operator action eliminates any potential for waterhammer in the HPCI turbine exhaust piping should the system trip and need to be restarted. Without the manual transfer, suppression pool level could reach the HPCI turbine exhaust line elevation, and a potential for water hammer could arise, but only if the operator allows HPCI to trip on high reactor water level and subsequently fails to prevent a HPCI restart on low reactor level.

The manual HPCI suction transfer is not needed to maintain containment hydrodynamic loads within design limits. LOCA blowdown loads are dependent only on 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 more than offsets the adverse effects associated with the increase in pool level.

Implementation of this proposed change will allow safe shutdown of the reactor in a postulated ATWS with SLCS failure. In addition, the proposed change will also increase HPCI reliability in an ATWS event in which SLCS is operable. Currently, to achieve safe shutdown for the postulated ATWS with SLCS failure, a bypass needs to be installed by the operators during the event. The purpose of the bypass is to override the Function 3e automatic HPCI suppression pool water level – high suction source swap (the function

to be eliminated). The bypass is needed as HPCI would be rendered inoperable due to the high suppression pool temperatures that result from the ATWS with SLCS failure. With HPCI inoperable, rapid depressurization of the reactor would be required in order to obtain vessel makeup from the low-pressure ECCS systems. Operation of a critical reactor at low pressure is highly undesirable due to the potential for reactivity–induced core damage caused by high-flow-rate low-pressure injection systems (LPCI and condensate). Core damage from an unstable core could result.

The proposed change will increase the availability of the HPCI system, thereby increasing the probability of a safe shutdown during a postulated ATWS event.

During an SBO event, HPCI and RCIC are the only pumps available. Thus it is crucial to prevent damage to the HPCI system. High suppression pool temperatures expected during a postulated SBO event also necessitate removal of the automatic transfer function (Function 3e) of the HPCI pump suction on Suppression Pool High Level from the CST to the Suppression Pool when the HPCI pump discharge valve is open. Implementation will reduce operator burden during an SBO event.

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 is also 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.

This safety assessment shows that with elimination of the suppression pool water level – high function:

- The plant response to an ATWS event is enhanced increasing the probability of 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 it's design basis function will not be affected.
- The suppression ability of the suppression pool and plant response during Design Basis Accident and postulated transient events will not be affected.
- Operator burden is reduced.

This change also corrects a typographical error for function 5a ADS Trip System B Reactor Vessel Water Level in the Unit 1 Technical Specification Table 3.3.5.1-1 (page 5 of 6). The Unit 2 Technical Specifications do not contain this same typographical error. It should indicate "Low, Low, Low Level 1" instead of "Low Level 1."

SECTION II

DESCRIPTION AND BASIS (BOTH LICENSING AND DESIGN) OF THE CURRENT REQUIREMENTS

HPCI System

The safety function of HPCI is to provide core cooling for a wide range of reactor pressures (SSES Technical Specification Bases Background Section 3.5.1). Primarily, HPCI is to maintain reactor inventory after small break LOCA's that do not result in depressurization of the reactor.

HPCI has two water sources from which it can draw to fulfill it's safety function. Suction is normally aligned to the CST since the CST contains reactor grade quality water. However, the suppression pool is the suction source assumed in accident analyses since the CST is a non-safety related system. The F042 suppression pool suction valve is normally closed and the F004 CST suction valve is normally open assuring that the CST water is initially injected to the core. The pump suction automatically transfers from the CST to the suppression pool on high suppression pool level or low CST level (FSAR, Section 6.3.2.2.1). The transfer is accomplished by the automatic opening of F042 valve and automatic closing of the F004 valve. The F004 valve closes once the F042 valve is in the open position to assure a suction source for the HPCI pump.

As described in the SSES Technical Specification Bases Section 3.3.5.1, the transfer function on high suppression pool level exists to preclude excessively high suppression pool levels. Reactor blowdown loads with a high pool level could result in blowdown loads that exceed suppression pool design values.

The safety function of the suppression pool water level – high transfer logic as described in the HPCI Design Basis Document is as follows:

"The basis for the suction transfer on high suppression pool level is to prevent the HPCI System from contributing to the further increase in the suppression pool level. The maximum suppression pool water level is dictated by the need to maintain sufficient air space to accommodate the non-condensable gases that are blown down to the suppression chamber during an accident. If the suppression pool water level were too high, the non-condensable gases would cause the containment pressure to exceed design values. The water level would also be a factor in the calculation of pool swell loads which would arise from the gaseous discharge from the containment drywell to the wetwell during the early stages of a postulated Design Basis Accident, and from the blowdown loads generated by an ADS depressurization event."

No regulatory requirements specify the means (i.e. automatic vs. manual) or the plant conditions that require HPCI system suction source swap.

HPCI's operation is assured with suction source water temperatures up to 140°F. Suppression pool temperatures expected for DBA events do not exceed 140°F in the time frame in which HPCI would be expected to operate. Beyond design basis events do however result in suppression pool temperatures that exceed 140°F while HPCI operation is desired.

General Electric Design and Performance Specification 386HA817 "Anticipated Transient Without Scram (ATWS)" identifies that as a result of this limitation, procedural controls are necessary to preclude the swap from the condensate storage tank to the suppression pool when the pool temperature is to high for HPCI and RCIC pump operation.

The F042 suppression pool suction valve also functions as a containment isolation valve. This valve is listed in Technical Specification Bases Table B 3.6.1.3-1 "Primary Containment Isolation Valve".

<u>Suppression Pool</u>

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 decay and sensible heat released during a reactor blowdown from the SRV's or from a DBA. The suppression pool must quench all the steam released through the downcomer lines during a LOCA. This is the essential mitigative function of the pressure suppression containment that ensures the peak containment pressure remains below the maximum allowable. The suppression pool also must condense the steam ejected from the HPCI and RCIC turbines.

The suppression pool is the water source credited in accident analyses for all of the ECCS and the RCIC system.

Technical Specification Bases Section 3.3.5.1 identifies that excessively high suppression pool water level could result in the loads on the suppression chamber exceeding design values should a blowdown occur. 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. This function is implicitly assumed in accident and transient analyses (which take credit for HPCI) since the analyses assume that the suction source is the suppression pool.

Suppression pool water level as required in Technical Specification 3.6.2.2 is to be greater than 22 feet but less than 24 feet. The limit of 24 feet specified by the Technical Specification Bases is specified to preclude excessive clearing loads from SRV discharge and excessive pool swell loads during a postulated DBA LOCA.

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'. 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).

Previous related PPL Submittals

In October 1998, PPL withdrew from NRC consideration this same proposed Technical Specification change. It was withdrawn so that PPL could evaluate re-submittal as a risk-based submittal. It has been determined that supplementing the original submittal with additional supporting evaluations is the most appropriate means to justify the change. Therefore this proposed change is based largely on deterministic analyses.

SECTION III

EVALUATION OF PROPOSED CHANGE AND BASIS

The proposed change is justified because this safety assessment shows that with elimination of the suppression pool water level – high function:

- The plant response to an ATWS event is enhanced increasing the probability of 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 Design Basis Accident and postulated transient events will not be affected.
- Operator Burden is reduced.

This section describes the PPL technical evaluation of the proposed change focusing on the safety function of the affected structures, systems, and components and the proposed change from the current requirements and design basis. The following contains a description of applicable analytical methods, data and analysis results.

The plant response to an ATWS event is enhanced increasing the probability of safe shutdown.

This change increases HPCI system reliability in an ATWS event. Currently in an ATWS event, high suppression pool temperatures necessitate the manual bypass of the HPCI suction transfer logic allowing the operator to align suction to the cooler CST water. The HPCI system is designed for continuous operation with suppression pool water temperatures up to 140°F and for short-term operation with suction water temperature up to 170°F (DBD004, Section 1.4.2.2). Pool temperature is monitored by SPOTMOS (Suppression Pool Temperature Monitoring System) which is located within 8 feet of the HPCI control room panel section allowing one operator to easily read SPOTMOS and operate HPCI.

In an ATWS event with no additional failures, suppression pool temperature is expected to exceed the maximum temperature (140°F) considered acceptable for continuous operation of the HPCI system. The short-term operating limit (170°F) is also expected to be exceeded, but by only a small amount. In the FSAR ATWS analysis, it is assumed that HPCI remains operable because peak suppression pool temperature is only slightly greater than the 170°F short-term operating limit. HPCI operation with pool temperatures in the range of 170°F is required for about 20 minutes in the ATWS event with no additional failures. For ATWS events that involve additional equipment failures, much higher pool temperatures are expected. In particular, the MSIV-closure ATWS with SLCS failure results in a peak pool temperature substantially greater than 200°F. This is well beyond the HPCI design temperature.

Due to the excessive suppression pool temperatures expected for postulated ATWS events, the Susquehanna EOP for ATWS mitigation instructs the operator to maintain HPCI suction on the CST and to bypass the high suppression pool suction transfer logic, if suppression pool temperature is expected to exceed 140°F.

For the postulated ATWS with SLCS failure event, the bypass cannot be carried out in time to assure continued operation of HPCI. Should HPCI fail, rapid depressurization of the reactor is required in order to obtain vessel makeup from low-pressure sources. Operation of a critical reactor at low pressure is highly undesirable.

The proposed change eliminates the need for the installation of the bypass. Elimination of the automatic suction transfer on high pool level increases the probability that HPCI will be available to mitigate ATWS events and increases the probability that the plant will be able to be safely shutdown. The proposed change also eliminates an operator action and eliminates a control room diversion from the primary focus of monitoring, assessing, and taking the necessary mitigating actions needed to achieve safe shutdown.

Based on the above, it is concluded that by implementation of the proposed change, the plant response to an ATWS event is enhanced increasing the probability of achieving safe shutdown since the probability that HPCI will be available is increased and since required operator actions are lessened.

<u>The probability that HPCI and RCIC will be available in an SBO event is increased.</u>

EO-100/200-032, "HPCI Operating Guidelines During Station Blackout," instructs the operator to prevent the auto swap over from the CST to the suppression pool on high pool level. Manual bypass of the suction transfer logic is carried out in accordance with Emergency Support Procedure ES-152/252-002. Since HPCI and RCIC are the only ECCS pumps available in a SBO, it is crucial to prevent damage to the HPCI system from injection of hot suppression pool water. Removal of the HPCI suction transfer on high pool level will reduce operator burden during a SBO event.

<u>The ability of HPCI to respond and perform its design basis function will not be affected.</u>

The proposed change will impact the suppression pool level. It has a negligible impact on suppression pool temperature. The impact on suppression pool level is evaluated in order to determine if it adversely affects HPCI operation during DBA and beyond design basis events.

The operator can monitor suppression pool water level with control room level indicators. The range of these safety-related instruments is 18-26.5 feet. These indicators are located on panel 1C601/2C601. Suppression Pool level is monitored by alarm window AR-114-F01, Suppression Pool Hi Level, which currently has a setpoint of 23.75'. The operator also has level indicator LI-15775B, Suppression Pool Level. The alarm window and the level indicator are both located on the HPCI section of Panel 1C601.

The impact of high suppression pool level on the HPCI operation and components was evaluated including water hammer concerns and the effects of increased pool level on turbine exhaust pressure. These evaluations are described below.

HV-155/255-F042 & HV-155/255-F004 Potential Effects

With the proposed change, HPCI will take suction from the CST until the water source is depleted (auto transfer), or until reactor pressure drops below the shutoff head of low-pressure ECCS (at which point HPCI operation is terminated), or until pool level reaches 25 feet (manual transfer), depending on the accident conditions.

A consideration in formulating the station battery load profile is the time line associated with the addition of various loads on the batteries. For the F042 valve, however, it cannot be determined when a low CST tank level may occur because the CST is not seismically qualified. Therefore, the design-basis evaluation for the battery loads assumes that the F042 valve starts to open and F004 starts to close during the heaviest loaded segment of the Battery loading sequence. As a result, no changes are required to the Battery load profile because of the proposed change.

HPCI suction valve HV-155/255-F042 is also a containment isolation valve. The containment isolation function and logic are not affected by the proposed change.

HPCI Pump and Turbine Potential Effects

Pump - Eliminating the HPCI suction transfer on high suppression pool level will increase HPCI reliability in beyond design basis events that result in suppression pool temperatures greater than 140 F. Since HPCI lube oil is cooled by the pumped fluid, failure of the pump from overheating is precluded if suction is maintained on the CST whose temperature will always be below 140 F. The proposed change has no other potential effects on HPCI pump operation.

Turbine - There are four potential concerns that need to be addressed related to operating, tripping, and restarting of the HPCI turbine with high suppression pool level. These concerns all relate to suppression pool level. Suppression pool letdown and the manual HPCI suction transfer can be used to lower and maintain the pool level such that the concerns addressed below are avoided. These potential concerns are addressed below.

1. The potential was identified for HPCI turbine exhaust line flooding in a small-break accident if HPCI trips with pool level above the exhaust line containment penetration (25.6 feet above the bottom of the pool). It was postulated that water would leak through turbine exhaust line check valve F049, and a water hammer would then occur upon restart of the HPCI turbine possibly damaging the turbine and associated piping.

   Based on expected leakage rates through the F049 valve, evaluation has concluded that leakage will be contained within the turbine exhaust line drain pot. The evaluation shows that even if the initial drain pot level is at the high-level alarm set point (75% full), there is sufficient capacity to allow for a leakage rate which is 50 times the measured value. Therefore, a water hammer is not expected to occur upon restart of the turbine.

2. With the proposed change, water level, in a small break accident, may reach 27.2 feet and completely submerge the horizontal section of the turbine exhaust line that penetrates the containment.[1] If HPCI trips with pool level ≥ 27.2 feet, water will flood the horizontal section of piping up to isolation check valve F049. When this occurs, the column length of water in the exhaust line increases by about 25 feet. Due to inertial effects, a higher turbine exhaust pressure will develop as this column of water is expelled upon auto restart of the turbine. This raises a potential that the HPCI pressure-relief diaphragms will rupture upon turbine restart performed in accordance with the plant procedures and render the system inoperable.

   As a result, the EOPs will be modified to include operator action to manually transfer HPCI suction from the CST to the suppression pool if pool level reaches 25 feet with pool temperature less than 140°F. If HPCI trips with pool level at 25 feet, there will be no suppression pool water contained within the horizontal section of exhaust piping (20 inch pipe). In a small break accident, pool level can reach 25 feet for a narrow range of break sizes. Moreover, the operator action to manually transfer

[1] 25.6 feet corresponds to the bottom of the horizontal piping at isolation check valve F049; the top of this piping is at an elevation of 27.2 feet (piping ID is 19.2 inches). The minimum inside elevation of the 20" horizontal HPCI turbine exhaust piping is 25.1 feet (piping slopes away from check valve F049 to prevent water accumulation).

HPCI suction from the CST to the suppression pool is not required in the early part of the accident. The earliest the manual transfer could be required is 21 minutes into the accident with pool level initially at 24 feet.

If pool temperature is greater than 140°F when pool level reaches 25 feet, HPCI suction will not be transferred to the pool because adequate cooling of the HPCI pump cannot be assured. However, if the pump suction auto transfers to the suppression pool on low condensate storage tank level when suppression pool temperature is greater than 140°F, the operator will continue to use HPCI as necessary. With HPCI injecting to the vessel, suppression pool temperature is expected to exceed 140°F only for beyond-design-basis events. It is appropriate to continue using HPCI with suction temperature greater than 140°F in a beyond-design-basis event because the system may be required to prevent actuation of ADS, or it may be required to prevent core damage in accident scenarios where low pressure injection systems or the depressurization capability are unavailable.

3. If, in a small-break accident, suppression pool level reaches 28.5 feet, the air intake for the HPCI turbine exhaust-line vacuum breakers (F076 and F077 on the HPCI turbine exhaust line) would be submerged. This would disable the vacuum breakers and could subsequently cause a water hammer on the turbine exhaust-line check valve (F049) if the system trips.

   This concern is eliminated by the resolution to item 2 above.

4. The HPCI turbine exhaust pressure for continuous steady-state operation may exceed the design limit.

   The HPCI turbine is designed to operate at a maximum continuous exhaust pressure of 65 psia (HPCI DBD004, Requirement 2.3.3.1.4). Analysis shows that there is ample margin to the design exhaust pressure limit of 65 psia.

Based on the above, the proposed change will have no impact on the ability of HPCI to perform it's design basis function.

The suppression ability of the suppression pool and plant response to Design Basis Accident and postulated transient events will not be affected.

The proposed change was evaluated for impacts on containment hydrodynamic loads for LOCAs, plant transients, and beyond design basis events (ATWS and SBO).

Removal of the HPCI auto transfer on high Suppression Pool level will allow slightly higher pool levels during plant accidents and transients; however, it has been concluded that the higher pool level does not lead to violation of design limits for hydrodynamic loads.

LOCA and SRV hydrodynamic loads potentially affect all primary containment safety-related structures, systems, and components. Dynamic loads on the primary containment indirectly affect the reactor building, and all safety-related equipment in the reactor building, because the structures are interconnected. The primary containment and reactor building safety-related structures, systems, and components perform numerous safety functions.

1. Primary Containment Design-Basis Loads

The Susquehanna primary containment is designed to accommodate loads generated by a LOCA and/or SRV discharge. The SRV and LOCA load definitions were reviewed in order to determine the impact of the proposed modification on the containment hydrodynamic loads.

1.1 SRV Load Definition

Loads associated with SRV discharge can be divided into two categories:

- Loads on submerged suppression pool structures, and
- Loads on the SRV system.

Both of these loads are discussed in the following subsections.

1.1.1 Loads on Suppression Pool Structures Due to SRV Actuation

SRV steam condensation loads on wetted portions of the suppression pool boundary and submerged structures are bounded by SRV air clearing loads (DBD046, Rev. 1, pp. 3, 42). A conservative SRV load definition for SSES was developed from examination of SRV test results for KWU (Kraftwerk Union) BWRs.[2] Out of the extensive KWU data base, three pressure-versus-time traces (so called KKB traces)[3], which were expected to result in conservatively high loadings, were chosen to define the suppression pool wetted-boundary and

[2] DBD046, Rev. 1, p. 42.
[3] Traces were obtained from SRV in-plant tests conducted at KKB power plant (Germany).

submerged-structure design basis loads (DBD046, Rev. 1, pp. 42-43). Frequency and amplitude adjustments were carried out on the data to add further conservatism.

Since the SSES SRV load specifications (based on KKB traces) were derived from test data for a similar, but not identical quencher design, it was deemed necessary to carry out testing with a prototype of the SSES quencher. The purpose of the prototypical testing was to ensure that SRV loads were bounded by the design load specification, and to further verify the steam quenching capability of the KWU quencher.[4] This testing was carried out by KWU at the Karlstein test facility (Germany).

SRV Actuation Under LOCA Conditions

The Karlstein tests used to verify SRV loads resulting from ADS actuation were carried out with depressed water level inside the SRV tail pipe. Owing to the SRV tailpipe vacuum breakers and the pressure differential between the drywell and wetwell, the water level inside the SRV tailpipe is independent of suppression pool level during a LOCA. The level coincides with the bottom of the downcomer pipes.[5][6] When level inside the SRV tailpipe is depressed as a result of the drywell/wetwell pressure differential, there is a larger volume of air within the line. The larger air volume during LOCA conditions is the most significant factor that affects the SRV loads relative to the SRV loads during non-LOCA conditions when the water level inside the SRV line is equal to the water level outside the line. The larger air volume results in a decrease in SRV load frequency and an increase in load amplitude.[7]

Previous KWU testing has shown that the increase in wetwell airspace pressure during LOCA conditions (up to 30 psig) has no effect on the amplitude of the SRV loads.[8] A 30 psig wetwell airspace pressure is equivalent to the hydrostatic pressure due to a pool level increase of approximately 69 feet. It then follows that an increase in pool level has no effect on the amplitude of the SRV loads during LOCA conditions.

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[4] DBD046, Rev. 1, pp. 45-46

[5] DBD046, Rev. 1, pp. 45-46.

[6] The bottom of the downcomer pipes are 12 feet above the bottom of the suppression pool.

[7] SSES DAR, Sections 4.1.1.e and Figure 8-169

[8] SSES DAR Section 8.5.3.3.3.4.

As discussed later in §2.1.3, the maximum expected suppression pool level increase in a design-basis accident is 1 foot (maximum pool level is 25 feet). For SRV performance, this is equivalent to an increase in wetwell pressure of 0.43 psi (0.030 bar). This small pressure change has negligible effect on the SRV load frequency.[9]

Table 1 shows the range of parameters considered in the ADS loading verification tests which were carried out with depressed water level inside the SRV discharge line. Test 11.1 was used to verify the conservatism of the ADS containment load definition since it produced the most severe boundary loads. Notice that the test resulting in the smallest containment hydrodynamic loads corresponds to the lowest reactor pressure (318 psig). Another important point concerning the SRV test conditions is the trending in suppression pool temperature, and accumulator pressure. Tests corresponding to reduced reactor pressure have higher initial pool temperatures. This is consistent with conditions expected in the plant: Low reactor pressure implies that significant reactor inventory has been discharged to the containment resulting in a rise in pool temperature.

Table 1
Initial Conditions and Pressure Amplitudes for SSES ADS Load Verification Tests Conducted at Karlstein Test Facility[10]

Test No.	Pool Level (ft)	Accumulator Pressure (psig)[11]	Suppression Pool Temp. (°F)	Discharge Line Level (ft)	Discharge Line Air Temp. (°F)	Pool Boundary Over-Pressure Amplitude (bar)
10.3	22.6	1160	73	11.7	126	0.40
11.1	24.3	1168	111	12.0	120	0.60
12.1	24.6	647	149	12.4	122	0.48
13.1	24.6	318	174	11.7	120	0.28

[9] SSES DAR Figure 8-175.
[10] SSES DAR, Tables 8.4 and 8.9. Pressure amplitude value corresponds to wall pressure (point 5.10).
[11] Accumulator pressure is equivalent to reactor pressure.

SRV Actuation Under Non-LOCA Conditions

Under non-LOCA conditions, water level inside the SRV tailpipe is approximately equal to the suppression pool level. Consequently, a rising pool level will result in increased loading on submerged containment structures because of the higher vent clearing pressure. However, a rising pool level has a negligible effect on the SRV load amplitude relative to other more significant parameters such as initial SRV discharge line volume, number of quenchers firing, etc.[12] The increase in containment loading associated with higher discharge line water levels can, however, be offset by decreasing reactor pressure. This relationship has been evaluated quantitatively by KWU using the Susquehanna-specific SRV discharge test results obtained at the Karlstein test facility. The following load-limit curve has been developed for Susquehanna.[13]

$L = -0.01662P_R + 45.6$ where (1)

L = suppression pool water level (ft), and

P_R = reactor pressure (psig).

If suppression pool water level is maintained below the curve defined by Equation (1), then containment loads for SRV actuation will remain within the design-basis envelope.

In developing the load limit curve, the most limiting component (downcomer bracing) was evaluated to ensure that adequate stress margin was available to accommodate the change in SRV loads anticipated along the load-limit line. The stress margin was conservatively based on the simultaneous occurrence of the following loads:

SRV + SSE + LOCA

where

SRV = loads due to SRV actuations,

SSE = loads due to Safe Shutdown Earthquake, and

LOCA = loads due to LOCA steam condensation (condensation oscillation and chugging).

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[12] SSES DAR Section 4.1.1.

[13] PLI-29888, "Suppression Pool Load Limit Curve", File 172-17, 835-02, December 1983.

1.1.2 Loads on SRV System

For purposes of calculating loads on the SRV system due to valve actuation, a very conservative initial level of 35.33 ft was assumed for the discharge line.[14] This value is conservative because piping forces and discharge line back pressure both increase with the initial height of the water column within the line. This initial level inside the tailpipe was based on Bechtel calculations of the reflood height within the discharge line subsequent to a valve actuation. In calculating the reflood height, it was assumed that one vacuum relief valve failed to operate. This calculation of the reflood height was recognized to be very conservative because of known computer code limitations. DAR Figure 8-103 shows with one vacuum breaker locked closed that level does not come back to suppression pool level confirming the conservatism in the Bechtel calculation. For comparison, the KWU Karlstein tests confirmed that in only two instances did the reflood height exceed the pool level outside the SRV discharge line. The exceedance was less than 1.5 ft.[15]

1.2 LOCA Load Definition

Dynamic pressure loads generated during a LOCA are attributed to two steam condensation phenomenon, condensation oscillations which occur in the early part of the transient and "chugging" which occurs later in the blowdown. The design basis LOCA loads are based on full-scale steam condensation tests conducted by KWU at the GKM II-M test facility in Mannheim, Germany. Single cell tests were carried out at the test facility which consisted of a downcomer pipe and proportionate drywell and wetwell volumes. Downcomer submergence in the testing was 12 feet[16] which corresponds to a suppression pool level of 24 feet.[17]

Four different breaks were considered as part of the testing:[18]

- Complete break of a recirculation loop,
- Complete break of a main steam line,
- 1/3 main steam line break, and
- 1/6 main steam line break.

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[14] Bechtel Calculations PUP-15598-S2, PUP-15598-S6, and PLE-15315 (March 2, 1992).
[15] SSES DAR Section 8.4.2.2.4 and Figure 8-101.
[16] SSES DAR, Section 9.1.2.2.3.
[17] SSES FSAR, Table 6.2-1.
[18] SSES DAR, Section 9.3.

In carrying out the LOCA tests, a rupture disk is broken and steam flows through a discharge line into the drywell section of the test tank.[19] No water was removed from the suppression chamber section of the facility during the course of the test to account for ECCS suction from the suppression pool. The pool level was allowed to increase based on the blowdown rate into the pool. Therefore, the rising pool level realized during these tests is proto-typical of the pool level increase expected at Susquehanna with the proposed modification installed.

2. Review of Design-Basis Accident Sequences Against Proposed Modification

This section examines the impact of the proposed modification on the plant response to relevant accidents and transients. Events that are considered consist of all design basis events which involve loss of coolant inventory and any other event, within the plant design basis, which may result in HPCI initiation either automatically or by manual operator action. These events are listed below:

- Loss of Coolant Accidents inside containment,
- Inadvertent Safety/Relief valve opening,
- Primary system break outside containment,
- Inadvertent HPCI initiation,
- Loss of feedwater flow,
- Loss of Offsite AC Power,
- Loss of Main Condenser vacuum,
- Inadvertent MSIV closure,
- Turbine trip (with and without bypass),
- Generator Load Rejection (with and without bypass),
- Pressure regulator failure-closed/open,

Two beyond design basis events, ATWS and Station Blackout, are also considered in the evaluation. Each of these events are discussed in detail below.

2.1 Loss of Coolant Accidents Inside Containment

Large, intermediate, and small break LOCAs are addressed separately in the following subsections.

[19] SSES DAR, Section 9.4.1.

2.1.1 Large Break LOCA

With respect to break area, the spectrum of large breaks is bounded by the full recirculation suction line break (4.16 ft^2)[20] and a 1 ft^2 break in the recirculation discharge line. Both of these breaks were analyzed in the Susquehanna LOCA analysis. The results are summarized below.

Full Recirculation Suction Line Break

The effective flow area for a suction side break of the recirculation line (DBA) is 4.16 ft^2. For the DBA suction line break, HPCI initiation signal (high drywell pressure) is generated at 0.3 seconds. HPCI begins to inject to the vessel at 30.3 seconds and stops at 43.9 seconds due to the rapid rate of vessel depressurization. During this event, the HPCI suction transfer logic has no appreciable influence on the rate of suppression pool level increase because of the very short time period of HPCI operation.

Elimination of the HPCI suction transfer logic does not affect the requirement to maintain the suppression pool level less than 24 feet in accordance with Technical Specification 3.6.2.2. Therefore, the initial pool level assumed in the LOCA analysis corresponds to 24 feet allowed by Tech. Spec. 3.6.2.2, and remains unchanged after the suction transfer modification. Consequently, the proposed modification has no adverse impact on containment response during the large-break LOCA.

1.0 ft^2 Recirculation Discharge Line Break

In the LOCA analysis, HPCI is assumed inoperable, and the 1.0 ft^2 break of the recirculation line causes rapid loss of vessel inventory which results in depressurization of the reactor vessel. ADS automatically initiates on low reactor water level, but the reactor is substantially depressurized (326 psig) by the time (121 seconds) the ADS valves open.

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[20] SSES FSAR Table 6.2-3.

Break flows in this event are an order of magnitude larger than the HPCI injection rate. Therefore, HPCI operation (with suction from the CST) is not expected to have a significant impact on reactor and containment response during the early part of the transient. The scenario presented above, for HPCI inoperable, should accurately describe the rate of vessel depressurization and level decrease with HPCI injecting to the reactor. As a result, suppression pool level and reactor pressure at the time of ADS actuation will be essentially the same as in the case where HPCI is inoperable. Therefore, the containment hydrodynamic loads will be essentially the same as in the case where HPCI is inoperable. These loads are bounded by the design-basis SRV/LOCA load definitions which are based on a higher reactor pressure for ADS initiation.

2.1.2 Intermediate Break LOCA

A 0.1 ft^2 break area is considered representative of an intermediate break.[21] In the Susquehanna analysis for the 0.1 ft^2 intermediate break, HPCI was unavailable as a result of the single failure. Obviously this scenario is not of much interest in evaluating the impact of eliminating the HPCI suction transfer logic on containment loads. Therefore, the case of an intermediate break (0.1 ft^2) with HPCI operable will be analyzed.

A SABRE[22] calculation was carried out to determine the reactor response to a 0.1 ft^2 break in the recirculation line with the HPCI system operable (RCIC is assumed to be inoperable because it is not a safety system[23]). A LOOP is also assumed to occur coincidentally with the break to be consistent with the design-basis LOCA analysis. HPCI initiates on high drywell pressure at about 1 second into the event. It is assumed that HPCI always takes suction from the CST, i.e., the automatic suction transfer on high suppression pool level has been eliminated. The LOOP causes a reactor scram, recirculation pump trip, loss of feedwater, and MSIV closure early in the event. Assuming a LOOP maximizes the operating time of HPCI during the accident (feedwater is lost within a few seconds of event initiation). This in turn maximizes the effect of the proposed modification on containment response.

[21] SSES FSAR, Section 6.2.1.1.3.3.4.

[22] Inputs and results of SABRE calculations discussed in this study are documented in PPL Calc. EC-052-1025, Rev. 1, "SABRE Calculations for IPE HPCI Modification." A summary level description of the SABRE code can be found in Attachment 1 to this safety assessment.

[23] Assuming RCIC inoperable is conservative with respect to this analysis. With RCIC operating, reactor pressure is reduced more quickly (more steam condensation on cold makeup flow) and ADS actuation is delayed slightly because of greater makeup flow. Therefore, in cases where ADS initiates, it does so at a lower reactor pressure which results in reduced containment loads.

HPCI prevents level from dropping to the ADS initiation set point, but injection flow is not sufficient to maintain reactor level above the feedwater spargers. Steam condensation on the subcooled liquid injected by HPCI causes the reactor to depressurize. The difference in Drywell and Wetwell pressures indicates that the downcomer vents are cleared throughout the entire transient. The LOCA is simulated up to the point where reactor pressure drops below the shutoff head of the core spray system. This corresponds to a ΔP of 292 psi between the water source and the reactor vessel.[24] For the LOCA scenarios of interest, Core Spray initiation occurs when reactor pressure drops to about 300 psig. The actual time of Core Spray injection is computed by the SABRE code based on reactor and containment pressures. LOCA simulations are carried out until the code predicts the onset of Core Spray injection to the vessel. At this point it is assumed that the operator will use core spray to provide coolant makeup to the vessel, and HPCI operation will no longer be required. This assumption is consistent with the design-basis function of the HPCI system given in the FSAR. Section 6.3.2.2.1 of the FSAR states "The HPCI system continues to operate until the reactor vessel pressure is below the pressure at which LPCI operation or core spray system operation can maintain core cooling."

Suppression Pool level rises during the event due to steam discharged from the HPCI turbine (~50 Lb_m/sec) and steam discharged to the Suppression Pool through the downcomer vents. This mass addition is unaffected by the HPCI suction source. Water level in the drywell does not reach 18" where it would begin to overflow from the drywell to the wetwell through the downcomer vents (the downcomer vents extend 18" above the floor of the drywell).

Since ADS would not be initiated for a 0.1 ft^2 break with HPCI operable, the concern about HPCI causing suppression pool level to exceed 24 feet prior to initiation of SRV/ADS blowdown is not valid. Moreover, as discussed for the large-break LOCA (§ 2.1.1), elimination of the automatic HPCI suction transfer on high suppression pool level does not affect the Technical Specification requirement to maintain pool level less than 24 feet 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 which were derived from an initial level of 24 feet and all break flow going to the pool.

[24] "Susquehanna Steam Electric Station Individual Plant Evaluation," NPE-91-001, Vol. 2, p. A-104, December 1991

Although it is not a licensing requirement to consider a single failure at times other than the initiation of the accident, it is prudent to examine the impact of HPCI failure with Suppression Pool level greater than 24 feet. In this event, the reactor vessel depressurizes below the shutoff head of the low pressure ECCS (~300 psig) before there is any substantial rise in suppression pool level. When reactor pressure drops to 300 psig at about 600 seconds into the event, suppression pool level has only risen to 24′-4″. 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 affect 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. This has no affect on the amplitude of SRV loads and negligible affect on the load frequency.

Since HPCI is running at full flow in this transient, and RPV water level is significantly below the high-level trip of 54", a HPCI trip (on high level) and restart is very unlikely. Therefore, it will not be considered here. The consequences of a HPCI trip and restart are addressed below in the section on small break LOCA.

2.1.3 Small Break LOCA

In order to evaluate the impact of eliminating the HPCI high-pool-level suction transfer on a small break LOCA, SABRE calculations were carried out for two small breaks: a 0.02 ft^2 break and a 1" line break (0.00545 ft^2). The HPCI DBD states that "It [HPCI] is designed to be capable of making up inventory losses for liquid breaks below about 0.02 sq ft, thus maintaining reactor level." With regard to the 1" line break, FSAR Section 6.3.1.1.1 states "One high pressure cooling system is provided which is capable of maintaining water level above top of core and preventing ADS actuation for breaks of lines less than 1 inch nominal diameter." The LOCAs are simulated up to the point where reactor pressure drops below the shutoff head of the core spray system (~300 psig).

For the 1" line break (0.00545 ft^2), suppression pool level increases by only 4 inches. The 0.02 ft^2 line break is much more limiting and thus is discussed below.

With the proposed change implemented, HPCI takes suction from the CST until Suppression Pool reaches 25 feet. When level reaches 25 feet, the operator manually transfers HPCI suction from the CST to the Suppression Pool. The rationale for the manual transfer is discussed later in this section. RCIC is assumed inoperable in this event because it is not a safety system. The initial suppression pool level is specified as the Technical Specification limit (24 feet). A LOOP is also assumed to occur coincidentally with the break. As mentioned earlier, assuming a LOOP maximizes the operating time of HPCI during the accident (feedwater is lost early in the event), which in turn maximizes the effect of the proposed modification on containment response. A controlled cooldown of the reactor, at 90 F/hr, is initiated at 10 minutes into the event[25]. 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. The Suppression Pool letdown flowrate is 120 Lb_m/sec.[26]

Cold water injected by HPCI quickly increases the core-inlet subcooling which lowers the vapor generation rate within the core. Condensation on HPCI injection flow, while the feedwater spargers are uncovered, and steam extraction by the HPCI turbine act to slowly depressurize the vessel. After 10 minutes, the operator occasionally opens a SRV to depressurize the reactor at 90 F/hr.

With HPCI suction aligned to the CST, Suppression Pool water level continues to increase during the event. Although it is not licensing requirement to examine the consequences of a single equipment failure (HPCI failure) which occurs during the long-term part of an accident, it is prudent to do so, and therefore, this concern is addressed in the following discussion.

HPCI Failure with High Suppression Pool Level

Containment loads associated with a small break LOCA combined with ADS actuation are within design limits.[27] Whenever the downcomer pipes are cleared, the air volume inside the SRV tailpipe is independent of suppression pool level. Thus, 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, but this has no effect on the amplitude of the SRV loads. For the 0.02 ft^2 break, the downcomer vents are cleared for the first 970

25 EO-000-102.

26 T.S. Yih, "Suppression Pool Let-Down Flow Rate In Suppression Pool Cooling Mode," Calc. EC-THYD-1007, Rev. 0.

27 Susquehanna FSAR Table 3.9-2, Rev. 40, 09/88.

seconds of the event. After 970 seconds, the downcomer vents begin to refill with water.[28] The downcomer vents refill because the cold HPCI injection 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 to consider when evaluating ADS loads with elevated Suppression Pool level. If the downcomer vents are cleared, the level inside the SRV tailpipe is not influenced by pool level, and as discussed above, the proposed modification has no influence on ADS hydrodynamic loads.

On the other hand, if the downcomer vents are sealed with water, there are no dynamic-pressure LOCA loads (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 evaluation results show that the Suppression Pool water level is always well below the Load Limit curve which demonstrates that ADS actuation, necessitated by HPCI failure at any time during the event, is acceptable.

In developing the Load Limit Curve, the most limiting component (downcomer bracing) was evaluated to ensure that adequate stress margin was available to accommodate the change in SRV loads along the Load-Limit Curve. The stress margin was conservatively based on the simultaneous occurrence of the following design-basis loads:

SRV + SSE + LOCA

where

SRV = loads due to SRV actuations,

SSE = loads due to Safe Shutdown Earthquake, and

LOCA= loads due to LOCA steam condensation (condensation oscillation and chugging).

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[28] PPL Calc. EC-052-1025, Rev. 1.

For this event, ADS actuation may occur when the downcomer vents are not cleared and so the LOCA steam condensation loads cannot occur. In addition, it is improbable that the SSE would occur simultaneously with ADS actuation. The LOCA and SSE loads comprise a significant portion of the total component stress in developing the Load-Limit Curve. Removing the LOCA and SSE loads increases the stress margin and would allow the Load-Limit Curve to be moved upward. Comparing the ADS loads for this event to the Load-Limit Curve is extremely conservative.

2.2 Inadvertent Safety/Relief Valve Opening (IORV)

This event is discussed in Section 15.1.4 of the FSAR. Opening of a SRV will cause a mild depressurization transient, but 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. The procedure instructs the operator to initiate suppression pool cooling to restore pool temperature less than 90°F. If level exceeds 24 feet, the EOP also requires the operator to reduce suppression pool level below 24 feet using suppression pool letdown systems.

If the SRV remains open, pool temperature will continue to increase and will reach 110°F at about 9 minutes into the event.[29] 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 water level (-38").

Prior to the event, HPCI is not operating; therefore, it has no adverse effect on the air clearing load due to the actuation of the IORV. Following the scram when HPCI is operating, the IORV has the potential for producing only 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 enveloped by SRV air clearing loads, as long as the suppression pool temperature response is maintained within the limits of NUREG-0783. 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.

[29] This time was estimated from the suppression pool heat up curve presented in Calc. SE-B-NA-128 (EC-059-0532).

2.3 Primary System Pipe Break Outside Containment

For a break external to the primary containment, any coolant injected by HPCI will not end up in the suppression pool; it exits the break within the secondary containment. Therefore, in this situation HPCI injection 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. The addition of steam to the suppression pool from HPCI turbine exhaust would cause a slow rise in pool level compared to a liquid break inside containment, and therefore, there will be ample margin to the Load Limit Curve.

2.4 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.

2.5 Loss of Feedwater Flow

On a loss of feedwater flow, the reactor will scram when level drops to +13". The void collapse caused by the scram will generate a HPCI initiation on low level (-38"). No SRV actuations are expected in this scenario because MSIVs remain open. Therefore, SRV/ADS hydrodynamic loads are not an issue.

2.6 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 §2.8.

2.7 Loss of Main Condenser Vacuum

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 below.

2.8 Inadvertent MSIV Closure

Closure of the MSIVs generates a reactor scram, and 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 (2 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[30]. The design basis event for SRV hydrodynamic loads is the ASME Overpressurization Event which 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 inch in the first 10 minutes following a MSIV closure.[31] The margin in peak steam dome pressure overwhelms any negative effects associated with the very small increase in suppression pool level. Note that 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 feet. That is, the containment design allows for simultaneous actuations of SRVs with suppression pool water level up to 26.1 feet.

In the long-term part of the event (>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 certainly bounded by the response for the small-break LOCA (§ 2.1.3). Therefore, pool level is always well below the Load Limit curve, and there are no adverse consequences associated with SRV actuations during the cooldown.

Following a transient such as a MSIV closure, it is not necessary to postulate a LOCA. Consideration of a LOCA following a transient is beyond the plant design basis.[32]

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[30] SSES Technical Specifications, Section 3.4.3, Amendment 178.
[31] PPL Calculation EC-052-1025.
[32] DBD035, Section 2.2.2.1.7.

2.9 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. As discussed in the previous section, the 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 makeup because feedwater would be available. If for some reason HPCI is used for vessel makeup following the vessel pressurization transient, its impact on containment loads is no different that that already discussed in §2.8.

2.10 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/without bypass.

2.11 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 that the turbine trip which was discussed in §2.9 (FSAR Section 15.2.1.2.3, Rev. 54, 10/99).

2.12 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 reactor depressurization which initiates closure of the MSIVs. The MSIV closure generates a reactor scram. Here MSIV closure occurs at reduced reactor pressure so SRVs actuations do not occur. Later in the transient, SRV cycling will occur as decay heat is removed from the RPV. SRV cycling following a MSIV closure with HPCI injecting to the vessel has already been addressed in §2.8.

2.13 Diaphragm – Slab – Differential - Pressure

Note that 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, Suppression Pool level can rise to 25 feet in a small break accident. 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 suppression chamber free volume when computing the diaphragm slab differential pressure and the drywell peak negative pressure.

2.14 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.

2.15 Safety-Related Valves on Piping Connected to Suppression Chamber Potential Effects

MOVs - Suppression pool level could potentially increase by 1 foot during a design basis small break accident, as a result of this modification (i.e., 24 feet to 25 feet). The increase in pressure due to the additional 1 foot is 0.43 psi (assuming suppression pool temperature of 90 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 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 condensate storage tank until pool level reaches 25 feet. 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 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.

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

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 a 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 there is no threat of SRV tailpipe failure. With the proposed change, suppression pool level could potentially increase to 25 feet. Design basis loads on the SRV system are conservatively based on an initial pool level of 35 feet. This level is ten feet above the maximum suppression pool level

that could occur with the proposed change. Actually, for a loss-of-coolant accident, the available margin is much greater than ten feet because reactor pressure continually decreases during the event. For beyond-design-basis conditions, the EOPs require reactor depressurization before pool level reaches the point where SRV tailpipe integrity is threatened.

2.16 RCIC Turbine

In accordance with its design basis, 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 site.

As discussed above, RCIC is used to provide coolant makeup following a reactor vessel isolation event. In a MSIV closure event, HPCI and RCIC would initiate on low RPV water level. These systems would inject to the vessel until they automatically trip when level reaches 54 inches. 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 makeup, and HPCI will be used for pressure control (CST-to-CST mode). Therefore, the long-term part of the scenario (t > 10 minutes) is completely 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 Technical Specification limit of 24 feet, then suppression pool level will be about 2 inches higher with the proposed change.[33] 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 feet. In the more realistic situation where the initial Suppression Pool level is at a nominal value of 23 feet, the proposed change has no effect on containment response.

33 With current plant design, HPCI would take suction from the suppression pool if the initial pool level > 23'-11", and the volume of coolant injected to the RPV prior to the HPCI trip on 54" corresponds to a 2" level decrease in the suppression pool.

Although RCIC is not designed for vessel makeup in a small break accident, it is prudent to examine the impact of the proposed change on RCIC operation under LOCA conditions. The issues and resolutions for RCIC turbine operation with elevated suppression pool level are essentially the same as those presented above for the HPCI turbine. The first concern is not an issue for the RCIC turbine, however, because the bottom of the horizontal section of turbine exhaust piping corresponds to 25.83 feet, and therefore the horizontal run of piping cannot become flooded because pool level will not exceed 25 feet in a design basis accident. Also, Concern #4 is not applicable to the RCIC system as its maximum turbine exhaust pressure for continuous operation is only 25 psia (DBD041, Rev. 0, Requirement 2.3.2.1.4), and this value would be exceeded in a small break accident even if there is no increase in suppression pool level.[34] Moreover, the maximum increase in suppression pool level is one foot (24 feet to 25 feet) which has negligible effect on RCIC turbine operation as it corresponds to a pressure increase of only 0.43 psi at the turbine exhaust.

OPERATOR BURDEN REDUCTION

Currently an operator action outside the Control Room is required to bypass HPCI suction transfer logic during the early stages of an ATWS or SBO event. With the proposed modification, a manual action performed from the Control Room will only be required in the long-term part (> 20 minutes) of the small break LOCA event. Thus, the proposed change allows the operator to better focus his attention on the event without unnecessarily having to be concerned with the HPCI suction source.

PROCEDURE IMPACTS

The SSES EOP's will be revised to require a manual HPCI suction transfer for situations where suppression pool level reaches 25 feet and pool temperature is less than 140 F, as it is necessary for the operator to manually transfer HPCI suction to the suppression pool in order to mitigate the rise in pool level and avoid potential malfunction of the HPCI system.

If the operator swaps the suction source to the suppression pool when the suppression pool level reaches 25 feet (pool temperature < 140°F), no adverse consequences would result and HPCI will function as designed.

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34 Peak wetwell atmosphere pressure is calculated to be about 35 psia for a small break accident.

If the operator delays action and swaps the suction source to the suppression pool after the pool level reaches 25 feet (pool temperature <140°F) there would be no adverse impact associated with delaying the manual transfer. However, if HPCI trips because the operator fails to control RPV water level less than 54" or because of an equipment malfunction, and the operator attempts to restart HPCI, there may be difficulty in starting the system with suppression pool level above 25 feet.

The bottom of the horizontal section of the HPCI turbine exhaust line corresponds to an elevation of 25.1 feet (above the bottom of the pool), with the top of this horizontal section of piping located at 27.2 feet. When suppression pool water level exceeds 25.1 feet, the horizontal section of the HPCI turbine exhaust line is partially filled with suppression pool water. If the HPCI system is started with pool level greater than 25.1 feet, the high-velocity steam could cause the formation of waves at the vapor-liquid surface. If waves contact the upper surface of the pipe, steam pockets can become entrapped within subcooled water. Rapid condensation of entrapped steam can result in a large differential pressure across the water slug formed by the wave on the liquid surface. Due to the large pressure differential, these liquid waves can be propelled through the piping system at high velocity. As the entrapped steam void vanishes due to condensation, the high velocity liquid wave may strike the pipe wall where the horizontal pipe bends downward into the suppression pool. As the slug of water is arrested, large amplitude pressure waves may be generated with potential for damage to piping[35]. In addition, if HPCI trips with pool level at 27.2 feet, the turbine exhaust line, up to isolation check valve F049, will completely fill with suppression pool water. When HPCI is restarted, the water column within the horizontal section of piping (about 25 feet) must be expelled as the line is cleared. Due to inertial effects, a higher turbine exhaust pressure is expected as this column of water is discharged to the pool. If the discharge line pressure becomes high enough, the HPCI pressure relief diaphragms will rupture and render the system inoperable.

Because of the uncertainty associated with restarting the HPCI system under conditions of high suppression pool level, the system would not be restarted if suppression pool level is greater than 25 feet. For instance, if the HPCI system trips (a trip could occur because the operator failed to control RPV water level less than 54 inches (operator error) or because of equipment malfunction) during a small break LOCA, the operator would rely on ADS and low-pressure injection systems (LPCI and Core Spray) to maintain sufficient coolant inventory.

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[35] Izenson, M.G., Rothe, P.H., and Wallis, G.B., "Diagnosis of Condensation-Induced Waterhammer," NUREG/CR-5220, Vol. 1, October 1988.

Operator Training will be conducted on the new procedures to explain the modification. Operation's staff will issue procedural changes to EOPs, OPs, and ESs. Operations staff will also issue a written 'Hotbox' which is required reading for all licensed operators.

SECTION IV

CONCLUSIONS

The proposed change is justified because this safety assessment shows that with elimination of the suppression pool water level – high function:

- The plant response to an ATWS event is enhanced increasing the probability of 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 Design Basis Accident and postulated transient events will not be affected.
- Operator Burden is reduced.

SECTION V

REFERENCES

- Susquehanna Steam Electric Station FSAR
- DBD 004 Rev. 3, "Design Basis Document for High Pressure Coolant Injection System"
- Design Assessment Report
- Generic Letter 95-07

SAFETY IMPACT ASSESSMENT

ATTACHMENT 1

SABRE DESCRIPTION

SABRE DESCRIPTION

Overview of SABRE Code

In an ATWS (Anticipated Transient Without Scram) event, a BWR (boiling water reactor) may be operated over a wide range of conditions far removed from those encountered during normal operation. In a non-isolation ATWS (main steam isolation valves open) for instance, the reactor operates in natural circulation with large reductions in feedwater enthalpy caused by loss of turbine extraction steam flow to feedwater heaters. Decreased make-up-flow enthalpy results in increased core-inlet subcooling which can lead to unstable power oscillations.[36]

If the ATWS event involves closure of the main steam isolation valves (MSIVs), a drop in water level occurs due to loss of feedwater flow. Vessel makeup is then provided by the HPCI (high pressure coolant injection) system which supplies highly subcooled water to the vessel at a reduced flow rate. At the low reactor water levels which result with HPCI injection, the feedwater spargers become uncovered and cold make-up coolant is injected directly into a region occupied by saturated steam. The development of condensation on the injection flow has a significant modulating influence on increasing core-inlet subcooling.

An ATWS may also involve depressurization of the reactor vessel as a mitigative response to failure of certain equipment. For example, should HPCI fail to function, the reactor could be depressurized below ~600 psia to allow coolant injection by intermediate-pressure-range condensate pumps.

In order to properly formulate a mitigative strategy for response to an ATWS event, an understanding of the reactor dynamic behavior over the wide range of conditions described above is required. Consequently, the SABRE (Simulation of ATWS in Boiling-Water Reactors) computer code was developed by PPL to simulate BWR transient behavior under natural circulation conditions with failure to scram. SABRE contains thermal-hydraulic models of the reactor jet pump, lower plenum, core, bypass, upper plenum, riser, separator, steam dome, and downcomer regions of the reactor vessel. A three-node, radially-lumped parameter model describes fuel-to-coolant heat transfer in each of the axial nodes within the core region. Nuclear heating effects are simulated using a two-group, one-dimensional kinetics model. 1-D cross section files are developed for U2C7, U2C9, U2C10, and U1C12 cores.

[36] Wulff, W., Cheng, H.S., and Mallen, A.N., "Causes of Instability at LaSalle and Consequences from Postulated Scram Failure", in Proceedings of International Workshop on Boiling Water Reactor Stability, Holtsville, New York, October 17-19, 1990.

The SABRE code also includes a model of the primary containment. In an isolation ATWS, reactor steam is discharged to the primary containment where it is condensed within the suppression pool. As pool temperature rises, the containment begins to pressurize, and alternate methods of reactor shutdown (boron injection or manual insertion of control rods) are required to maintain containment structural integrity.

It is important to emphasize, in the context of emergency operating procedure development, that SABRE results are not used to justify reactor operation under conditions where core dynamics are poorly behaved by demonstrating that fuel integrity can be maintained. On the contrary, at PPL, the SABRE calculations are used to identify operating regimes where core/containment integrity is likely to be threatened. The mitigative strategy for ATWS is then constructed to avoid these severe operating regimes whenever possible. Consequently, the reactor is maintained within an operating domain where there is confidence that core and containment integrity will be preserved.

Although SABRE was developed to study reactor behavior under ATWS conditions, it can also be used to investigate reactor and containment response to small break LOCAs or anticipated reactor transients such as an MSIV closure with scram.

The NRC has examined SABRE results for ATWS scenarios as part of their review of proposed changes to the BWR Owners' Group Emergency Procedure Guidelines.[37] ATWS simulations were performed by the NRC using the TRAC-BF1 and RAMONA-4B computer codes. When discussing their results in the Safety Evaluation Report, the NRC makes the following statement, "Results obtained with these two codes were reasonably consistent and were also comparable to PPL findings for Susquehanna BWRs using SABRE, after making adjustments to compensate for different procedural assumptions." It is concluded, based on this Safety Evaluation, and subsequent conversations with the NRC[38], that SABRE can be used for ATWS analysis.

The basic assumptions and modeling features used in the SABRE code are summarized below:

<u>Reactor Model</u>

1. The flow is one-dimensional, and each flow region (core, bypass, upper plenum etc.) has an axially-uniform flow area.

2. Acoustic waves travel at infinite speed.

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[37] Nuclear Regulatory Commission, "Safety Evaluation Report Modifications to the Boiling Water Reactor (BWR) Emergency Procedure Guidelines to Address Reactor Core Instabilities," June 6, 1996.

[38] PLA-4480, "Unit 2 Cycle 9 ATWS Evaluation," File R41-2, Docket No. 50-388, July 23, 1996.

3. Potential and kinetic energy effects are neglected.

4. Slip between phases is governed by a Drift-Flux model.

5. Complete vapor-liquid separation occurs at exit of steam separator (steam dryer not explicitly modeled).

6. Flow regime can be co-current or counter-current.

7. The axial power shape is non-uniform and time varying.

8. The 764 fuel bundles are averaged into a single channel.

9. Gamma heating within fuel bundles is specified as part of input data.

10. Gamma heating in bypass channel is specified as part of input data.

11. Efficiency of steam condensation is 95% when downcomer level is a meter or more below the feedwater nozzles.

12. Twenty-seven axial hydraulic nodes are used for the core region.

13. Twenty-five axial and two radial nodes are used to model the fuel. The cladding is modeled with twenty-five axial nodes and one radial node. The number of fuel pins can vary axially along the core.

14. In the outer radial node of the fuel, the volumetric heat generation rate is 10% greater than the radially-averaged heat generation rate to account for self-shielding effects.

15. Core power is computed using a two-group 1-D kinetics model with six delayed-neutron groups.

16. The kinetics model includes reactivity contributions from control rod insertion, boron injection, and variations in moderator density and fuel temperature,.

17. Axial power shape for fission power changes with core conditions based on the 1-D kinetics model, but the power shape for decay power remains equal to the initial axial power shape.

18. Injected boron stagnates in lower plenum if total core flow is less than 5 MLb/hr.

19. Stagnated boron re-mixes if total core flow exceeds 15 MLb/hr.

20. Perfect mixing occurs in each boron mixing node.

21. Dissolved boron does not affect coolant density. The density of injected boron solution is assumed to be the same as water at the same temperature.

22. Boron volatility is neglected.

23. Boron transport is calculated from coolant flow.

24. In general, a boron mixing node consists of several hydraulic nodes.

25. The thermal absorption cross section for B-10 is assumed to be a linear function of the boron concentration in the fuel channels and the bypass region.

26. Model includes thermal capacitance of reactor vessel and vessel internal structures. Sensible heat given off by these structures is superimposed on core power.

27. Pump power decays exponentially following a recirculation pump trip.

28. For a recirculation pump runback, pump power decays exponentially to a lower value which maintains the desired total core flow.

29. An inertial flow model describes steam flow through closing MSIVs.

30. Pressure wave phenomena in steam lines is neglected.

31. HPCI/RCIC extraction steam flow is included in steam dome mass/ energy balances.

32. SRV flow and break flow during a LOCA are computed with homogeneous critical flow model.

33. A proportional controller approximates the feedwater controller response.

34. Upon loss of feedwater heating, the decrease in feedwater temperature is approximated by a delayed exponential decay.

35. CRD flow reaches thermal equilibrium with coolant in lower plenum; flow enters reactor in lower node of bypass region.

36. RPV injection systems consist of feedwater, HPCI, RCIC, condensate, Core Spray, SLCS, and CRD.

Containment Model

37. The drywell and wetwell are each modeled with a single control volume.

38. Model includes thermal capacitance of structural steel and liner plate, but thermal capacitance of concrete structures is neglected.

39. Pressure within drywell and wetwell is computed with Ideal Gas equation of state.

40. Vacuum breakers and downcomer vents are modeled.

41. Drywell coolers and suppression pool cooling system are modeled.

42. Suppression pool letdown can be described by specifying a constant letdown flow rate.

43. Model includes heat transfer from SRV tailpipes to suppression chamber atmosphere.

44. Heat structure model includes natural convection, condensation, and radiation heat transfer effects.

45. Drywell heat load includes dissipation from reactor vessel.

46. Wetwell air space varies with suppression pool level.

47. Pool layer can form on drywell floor. Depth of pool layer can reach top of downcomer pipes.

48. Condensation on liner plate and structural steel drains into drywell/wetwell pool.

49. Effect of pool formation on drywell free volume is negligible.

50. Model includes heat transfer/evaporation effects at surface of drywell/ wetwell pool.

51. For LOCA, break flow comes to pressure equilibrium with drywell.

Attachment 2 to PLA-5322

No Significant Hazards Considerations Evaluation

and

Environmental Assessment

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

PPL Susquehanna, LLC has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10CFR50.92 (c) a proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility with the propose amendment would not:

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

PPL Susquehanna, LLC proposes to delete the "High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level – High" (Function 3e) in both Unit 1 and Unit 2 Technical Specifications Table 3.3.5.1-1. The allowable value for this function as delineated in the Table is 23 feet and 11 inches. Elimination of Function 3e increases the availability of the HPCI system during a postulated Anticipated Transient Without Scram (ATWS) event and reduces operator burden during a postulated Station Blackout Event (SBO). This change does not adversely affect the ability of the HPCI system or other plant systems to perform their design basis functions. This change was determined to be necessary by the SSES Individual Plant Evaluation (IPE) to meet the defense in depth criteria set forth in the IPE.

As indicated below, the criteria set forth in 10CFR50.92 are met for this amendment:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Deletion of the automatic HPCI suction transfer from the CST to the suppression pool for a high suppression pool level condition was analyzed for impacts against all previously evaluated accidents and transients. Eliminating the automatic transfer increases the availability of the HPCI system during an ATWS event and operator burden is reduced during a postulated Station Blackout Event (SBO). There are no adverse effects, consequences, or changes in the probability of an accident occurring as a result of this change. HPCI operation is improved and all other plant systems remain unaffected in their ability to perform their design basis functions as a result of this change.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously analyzed?

   Implementation of this change increases the availability of the HPCI system during a postulated ATWS with Standby Liquid Control System (SLCS) failure. The change only affects the HPCI suction source and whether the source is automatically transferred from the preferred CST to the suppression pool for a high suppression pool level. Continued HPCI operation utilizing the CST as a suction source does not create a new or different type of accident from those previously analyzed. The primary effect of this change is to the suppression pool level which has been evaluated and found to be acceptable for all relevent accidents and transients. Therefore a new or different accident is not created and all other accident analyses are unaffected by the change.

3. Does the proposed change involve a significant reduction in a margin of safety.

   This change does not reduce any margin of safety. The increase in suppression pool water level does not cause containment hydrodynamic loads to exceed design limits under accident conditions. Overall, HPCI reliability is increased as it would remain operable during the ATWS with Loss of SLCS event. This increased availability of the HPCI system provides for additional defense in depth which reduces the probability of core damage.

Based upon the above, the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the proposed change to Technical Specifications Table 3.3.5.1-1 because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). This change deletes the "High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level – High" (Function 3e) in both Unit 1 and Unit 2 Technical Specifications Table 3.3.5.1-1. The requested change will have no impact on the environment because HPCI operation is improved and all other plant systems remain unaffected in their ability to perform their design basis functions. As discussed in the "No Significant Hazards Consideration Evaluation", the proposed change does not involve a significant hazard 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-5322

Technical Specification Markups

Technical Specification Bases Markups

(Units 1&2)

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Manual Initiation	1,2,3, 4(a), 5(a)	2 1 per subsystem	C	SR 3.3.5.1.5	NA
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level – Low, Level 2	1, 2(e), 3(e)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -45 inches
b. Drywell Pressure – High	1, 2(e),3(e)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level – High, Level 8	1, 2(e), 3(e)	2	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 55.5 inches
d. Condensate Storage Tank Level – Low	1, 2(e), 3(e)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 36.0 inches above tank bottom
~~e. Suppression Pool Water Level – High~~	~~1, 2(e), 3(e)~~	~~2~~	~~D~~	~~SR 3.3.5.1.2~~ ~~SR 3.3.5.1.3~~ ~~SR 3.3.5.1.5~~	~~≤ 23 feet 11 inches~~

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(e) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
e ~~f.~~ Manual Initiation	1, 2(e), 3(e)	1	C	SR 3.3.5.1.5	NA
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2(e), 3(e)	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, 2(e), 3(e)	2	E	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Automatic Depressurization System Initiation Timer	1, 2(e), 3(e)	1	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 114 seconds
d. Reactor Vessel Water Level – Low, Level 3 (Confirmatory)	1, 2(e), 3(e)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 11.5 inches
e. Core Spray Pump Discharge Pressure – High	1, 2(e), 3(e)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 125 psig and ≤ 165 psig

(continued)

(e) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
f. Low Pressure Coolant Injection Pump Discharge Pressure – High	1, $2^{(e)}$, $3^{(e)}$	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 115 psig and ≤ 135 psig
g. Automatic Depressurization System Drywell Pressure Bypass Actuation Timer	1, $2^{(e)}$, $3^{(e)}$	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 450 sec
h. Manual Initiation	1, $2^{(e)}$, $3^{(e)}$	2	F	SR 3.3.5.1.5	NA
5. ADS Trip System B					
a. Reactor Vessel Water Level – Low, Level 1 Low Low	1, $2^{(e)}$, $3^{(e)}$	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, $2^{(e)}$, $3^{(e)}$	2	E	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Automatic Depressurization System Initiation Timer	1, $2^{(e)}$, $3^{(e)}$	1	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 114 sec
d. Reactor Vessel Water Level – Low, Level 3 (Confirmatory)	1, $2^{(e)}$, $3^{(e)}$	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 11.5 inches
e. Core Spray Pump Discharge Pressure – High	1, $2^{(e)}$, $3^{(e)}$	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 125 psig and ≤ 165 psig

(continued)

(e) With reactor steam dome pressure > 150 psig.

BASES

BACKGROUND (continued)

Full flow test lines are provided for each LPCI subsystem to route water from the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. Whenever ~~the HPCI injection valve is open and the suppression pool level is high or if~~ the CST water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (165 psia to 1225 psia). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The HPCI, LPCI and CS System discharge lines are kept full of water using a "keep fill" system that is supplied using the condensate transfer system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for by 10 CFR 50.49) are accounted for.

An exception to the methodology described to derive the Allowable Value is the methodology used to determine the Allowable Values for the ECCS pump start time delays and HPCI CST Level 1 - Low. ~~HPCI Suppression Pool Water Level - High~~. These Allowable Values are based on system calculations and/or engineering judgement which establishes a conservative limit at which the function should occur.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals. The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level—Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 2. In addition, the Reactor Vessel Water Level—Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low, Level 2 or Drywell Pressure—High. Each of these variables is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

HPCI suction is normally maintained on the CST until it transfers to the suppression pool on low CST level or is manually transferred by the operator.

The HPCI System also monitors the water level~~s~~ in the condensate storage tank (CST). ~~and the suppression pool because these are the two sources of water for HPCI operation~~. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless the suppression pool suction valve is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valve automatically opens, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valve to open and the CST suction valve to close. ~~The suppression pool suction valve also automatically opens and the CST suction valve closes if high water level is detected in the suppression pool when the HPCI injection valve is open~~. ~~Suppression Pool Water Level—High signals are initiated from two level instruments~~. ~~The logic is arranged such that either switch can cause the function~~. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level—High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve, minimum flow valve, the cooling water isolation valve, and the injectionvalve to close. The logic is two-out-of-two to provide high reliability of the HPCI System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.d. Condensate Storage Tank Level—Low (continued)

Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valve automatically opens, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Condensate Storage Tank Level—Low signals are initiated from two level instruments. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level—Low Function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST.

Two channels of the Condensate Storage Tank Level—Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.e. Suppression Pool Water Level—High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~3.e. Suppression Pool Water Level—High (continued)~~

~~This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.~~

~~Suppression Pool Water Level—High signals are initiated from two level instruments. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close provided the HPCI injection valve is open. The Allowable Value for the Suppression Pool Water Level—High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.~~

~~Two channels of Suppression Pool Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.~~

3.~~f~~e. Manual Initiation

The Manual Initiation push button channel introduces signals into the HPCI logic to provide manual initiation capability and is redundant to the automatic protective instrumentation. There is one push button for the HPCI System.

The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the HPCI function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of the Manual Initiation Function is required to be OPERABLE only when the HPCI System is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 3) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition G must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic component initiation capability for the HPCI System. Automatic component initiation capability is lost if two Function 3.d channels ~~or two Function 3.e channels~~ are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate and the HPCI System must be declared inoperable within 1 hour after discovery of loss of HPCI initiation capability. A Note identifies that Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed. This allows the HPCI pump suction to be realigned to the Suppression Pool within 1 hour, if desired.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the HPCI System cannot be automatically

(continued)

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level – Low Low, Level 2	1, 2(e), 3(e)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -45 inches
b. Drywell Pressure – High	1, 2(e), 3(e)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level – High, Level 8	1, 2(e), 3(e)	2	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 55.5 inches
d. Condensate Storage Tank Level – Low	1, 2(e), 3(e)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 36.0 inches above tank bottom
~~e. Suppression Pool Water Level-High~~	~~1, 2(e), 3(e)~~	~~2~~	~~D~~	~~SR 3.3.5.1.2~~ ~~SR 3.3.5.1.3~~ ~~SR 3.3.5.1.5~~	~~≤ 23 feet, 11 inches~~
e. ~~f.~~ Manual Initiation	1, 2(e), 3(e)	1	D	SR 3.3.5.1.5	NA

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(e) With reactor steam dome pressure > 150 psig.

BASES

BACKGROUND (continued)

Full flow test lines are provided for each LPCI subsystem to route water from the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. Whenever ~~the HPCI injection valve is open and the suppression pool level is high or if~~ the CST water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (165 psia to 1225 psia). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The HPCI, LPCI and CS System discharge lines are kept full of water using a "keep fill" system that is supplied using the condensate transfer system.

(continued)

BACKGROUND (continued)

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low, Level 2 or Drywell Pressure – High. Each of these variables is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

normally HPCI suction is maintained on the CST until it transfers to the suppression Pool on low CST level or is manually transferred by the operator.

The HPCI System also monitors the water level~~s~~ in the condensate storage tank (CST)~~, and the suppression pool because these are the two sources of water for HPCI operation~~. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless the suppression pool suction valve is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valve automatically opens, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valve to open and the CST suction valve to close. ~~The suppression pool suction valve also automatically opens and the CST suction valve closes if high water level is detected in the suppression pool when the HPCI injection valve is open. Suppression Pool Water Level – High signals are initiated from two level instruments. The logic is arranged such that either switch can cause the function~~. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level – High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve, minimum flow valve, the cooling water isolation valve, and the injection valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level – Low Low, Level 2 signal is subsequently received.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

An exception to the methodology described to derive the Allowable Value is the methodology used to determine ~~the~~ Allowable Values for the ECCS pump start time delays, and HPCI CST Level 1 - Low. ~~HPCI Suppression Pool Water Level - High~~. These Allowable Values are based on system calculations and/or engineering judgement which establishes a conservative limit at which the function should occur.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals. The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a. 2.a. Reactor Vessel Water Level – Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level – Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 2. In addition, the Reactor Vessel Water Level – Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

3.e. Suppression Pool Water Level – High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level – High signals are initiated from two level instruments. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close provided the HPCI injection valve is open. The Allowable Value for the Suppression Pool Water Level – High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

Two channels of Suppression Pool Water Level – High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.~~f~~e. Manual Initiation

The Manual Initiation push button channel introduces signals into the HPCI logic to provide manual initiation capability and is redundant to the automatic protective instrumentation. There is one push button for the HPCI System.

(continued)

Attachment 4 to PLA-5322

“Camera Ready” Technical Specifications

(Units 1&2)

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Manual Initiation	1,2,3, 4(a), 5(a)	2 1 per subsystem	C	SR 3.3.5.1.5	NA
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level—Low, Level 2	1, 2(e), 3(e)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -45 inches
b. Drywell Pressure—High	1, 2(e),3(e)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level—High, Level 8	1, 2(e), 3(e)	2	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 55.5 inches
d. Condensate Storage Tank Level—Low	1, 2(e), 3(e)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 36.0 inches above tank bottom

(a) When the associated subsystem(s) are required to be OPERABLE.

(e) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
e. Manual Initiation	1, 2(e), 3(e)	1	C	SR 3.3.5.1.5	NA
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level—Low Low Low, Level 1	1, 2(e), 3(e)	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure—High	1, 2(e), 3(e)	2	E	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Automatic Depressurization System Initiation Timer	1, 2(e), 3(e)	1	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 114 seconds
d. Reactor Vessel Water Level—Low, Level 3 (Confirmatory)	1, 2(e), 3(e)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 11.5 inches
e. Core Spray Pump Discharge Pressure—High	1, 2(e), 3(e)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 125 psig and ≤ 165 psig

(continued)

(e) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
f. Low Pressure Coolant Injection Pump Discharge Pressure—High	1, 2(e), 3(e)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 115 psig and ≤ 135 psig
g. Automatic Depressurization System Drywell Pressure Bypass Actuation Timer	1, 2(e), 3(e)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 450 sec
h. Manual Initiation	1, 2(e), 3(e)	2	F	SR 3.3.5.1.5	NA
5. ADS Trip System B					
a. Reactor Vessel Water Level—Low Low Low, Level 1	1, 2(e), 3(e)	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure—High	1, 2(e), 3(e)	2	E	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Automatic Depressurization System Initiation Timer	1, 2(e), 3(e)	1	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 114 sec
d. Reactor Vessel Water Level—Low, Level 3 (Confirmatory)	1, 2(e), 3(e)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 11.5 inches
e. Core Spray Pump Discharge Pressure—High	1, 2(e), 3(e)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 125 psig and ≤ 165 psig

(continued)

(e) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level—Low Low, Level 2	1, 2(e), 3(e)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -45 inches
b. Drywell Pressure—High	1, 2(e),3(e)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level—High, Level 8	1 2(e), 3(e)	2	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 55.5 inches
d. Condensate Storage Tank Level—Low	1, 2(e), 3(e)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 36.0 inches above tank bottom
e. Manual Initiation	1, 2(e), 3(e)	1	D	SR 3.3.5.1.5	NA

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(e) With reactor steam dome pressure > 150 psig.