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U.S. Nuclear Regulatory Commission
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**SUSQUEHANNA STEAM ELECTRIC STATION**
**SUPPLEMENT 3 TO PROPOSED AMENDMENT NO. 239 TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 204 TO LICENSE NPF-22: HPCI AUTOMATIC TRANSFER TO SUPPRESSION POOL LOGIC ELIMINATION**
**PLA-5470**

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

*Reference: 1) PLA-5322, R. G. Byram (PPL) to USNRC, "Proposed Amendment No. 239 to License NPF-14 and Proposed Amendment No. 204 to License NPF-22: HPCI Automatic Transfer to Suppression Pool Logic Elimination", dated June 8, 2001.*

*2) Letter, NRC to R. G. Byram (PPL), "Susquehanna Steam Electric Station, Units 1 and 2 - Request for Additional Information Re: Elimination of Automatic Transfer of High-Pressure Coolant Injection Pump Suction Source (TAC Nos. MB2190 and MB2191)", dated December 18, 2001.*

*3) PLA-5425, R. G. Byram (PPL) to USNRC, "Supplement to Proposed Amendment No. 239 to License NPF-14 and Proposed Amendment No. 204 to License NPF-22: HPCI Automatic Transfer to Suppression Pool Logic Elimination", dated February 4, 2002.*

*4) PLA-5456, R. G. Byram (PPL) to USNRC, "Supplement 2 to Proposed Amendment No. 239 to License NPF-14 and Proposed Amendment No. 204 to License NPF-22: HPCI Automatic Transfer to Suppression Pool Logic Elimination", dated April 8, 2002.*

*5) Letter, NRC to R. G. Byram (PPL), "Susquehanna Steam Electric Station, Units 1 and 2 - Request for Additional Information Re: High-Pressure Coolant Injection (HPCI) Pump Automatic Suction (TAC Nos. MB2190 and MB2191)", dated April 22, 2002.*

The purpose of this letter is to provide supplemental information as contained in Attachments 1 and 2, necessary for the NRC staff to continue its review of the license amendment originally proposed by Reference 1 and later supplemented with additional information in References 3 and 4.

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The need for this supplemental information was identified during teleconferences held between NRC and PPL on March 25, 2002 and April 9, 2002. Attachment 1 provides responses to the eighteen specific questions which resulted from those discussions as documented in the April 22, 2002 letter from the NRC to PPL (Reference 5). Attachment 2 contains a non-confidential version of the EC-Risk-1083 Revision 1 calculation which provides the results of the risk analysis for removal of the automatic HPCI suction transfer on a high Suppression Pool level condition.

In June, 2001, PLA-5322, (Reference 1), proposed deletion from the Unit 1 and Unit 2 Technical Specification Table 3.3.5.1-1 the "High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level – High" (Function 3e). 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. Implementation of the proposed change and the associated plant modifications are essential to eliminate a vulnerability identified by the PPL Susquehanna, LLC (PPL) Individual Plant Evaluation (IPE).

The Nuclear Regulatory Commission staff reviewed Reference 1 and determined that additional information was required in order to complete the NRC review. The additional information requested was documented in a Request for Additional Information (RAI) dated December 18, 2001, (Reference 2). PLA-5425 (Reference 3) and PLA-5456 (Reference 4) each provided additional information related to this NRC RAI.

Subsequently eighteen additional questions were documented in the April 22, 2002 letter from the NRC to PPL (Reference 5).

If you have any questions related to this submittal, please contact Mr. Duane L. Filchner at (610) 774-7819.

Sincerely,

R. G. Byram

Attachment

cc: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. T. G. Colburn, NRC Sr. Project Manager

**BEFORE THE**
**UNITED STATES NUCLEAR REGULATORY COMMISSION**

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

**SUPPLEMENT 3 TO PROPOSED AMENDMENT NO. 239**
**TO LICENSE NPF-14: HPCI AUTOMATIC TRANSFER TO SUPPRESSION**
**POOL LOGIC ELIMINATION**
**UNIT NO. 1**

Licensee, PPL Susquehanna, LLC, hereby files Supplement 3 to 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 7th day of May, 2002.

Nancy J. Lannen

Notary Public

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

**BEFORE THE**
**UNITED STATES NUCLEAR REGULATORY COMMISSION**

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

**SUPPLEMENT 3 TO PROPOSED AMENDMENT NO. 204 TO LICENSE NPF-22: HPCI AUTOMATIC TRANSFER TO SUPPRESSION POOL LOGIC ELIMINATION UNIT NO. 2**

Licensee, PPL Susquehanna, LLC, hereby files Supplement 3 to 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 7th day of May, 2002.

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-5470

# Responses to April 22, 2002 RAI

# Attachment 1 – Response to RAI

## REQUESTED INFORMATION ABOUT CALCULATION EC-ATWS-0505, REVISION 8

### NRC Question 1

A lot of SABRE computer code input deck data in Appendix D came from the document, PL-NF-89-005, Revision 0, and another RETRAN computer code calculation. It was indicated that these references have been approved by the Nuclear Regulatory Commission (NRC) through previous licensing submittals. Please provide relevant documents that verify NRC's approvals.

### PPL Response

PL-NF-89-005, Rev. 0 has been submitted to the NRC. The RETRAN input data was included in the submittal. The document was reissued as PL-NF-89-005-A in July of 1992. A copy of the Safety Evaluation for topical report PL-NF-89-005 is included within this attachment. The other RETRAN computer code calculation that is used extensively is PPL Calculation EC-FUEL-1375, Rev. 0, "RETRAN System Model/ ATRIUM-10 Core Model," 10/20/98. This package describes and documents the inputs for the RETRAN system model for an ATRIUM-10 core. The data in EC-FUEL-1375 provides inputs to the NRC approved methodology described in topical report PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design And Analysis." A copy of the Safety Evaluation for PL-NF-90-001-A is also included within this attachment.

### NRC Question 2

On Page 235, the loss coefficient of the fuel spacer is calculated by the correlation for ANF9x9 fuel. Does this correlation still apply to the current cycle? If not, what is the impact? It is found that the entire core is modeled by one 1-D hydraulic component. Please describe the modeling approach about lumping peripheral region bundles with central region bundles, which have different inlet orifice loss coefficients.

**PPL Response**

Per discussion on page 338, the flow-dependent friction factor and flow-dependent spacer loss coefficient correlations were not updated for ATRIUM-10 fuel although the hydraulic diameters and flow areas were updated. Therefore, engineering judgement was used and the difference between the 9x9 and 10x10 friction factor and spacer losses would have an insignificant effect on core flow behavior especially when the 10x10 hydraulic diameters and flow areas are used. This judgement was determined as follows:

(1) In both cases, the friction factors are for flow along a rod bundle, the liquid contact area is accounted for in the hydraulic diameter which was updated to 10x10 fuel.

(2) Spacer construction for a 9x9 and 10x10 bundle is expected to be similar. Therefore the local losses are expected to be similar when the difference in hydraulic diameters is taken into account.

The use of this engineering judgement is valid for current and future cycles of Atrium 10 fuel.

Recognizing that there is some uncertainty in the calculated fluid friction, the effect of this uncertainty is evaluated in the Susquehanna cycle-specific ATWS calculations. Under ATWS conditions, the reactor operates at natural circulation conditions so fluid friction affects core flow, which affects boron mixing, which in turn affects core power and peak suppression pool temperature. The fluid friction uncertainty analysis for ATWS examines the change in peak suppression pool temperature as the frictional resistance is varied. The fuel bundle orifice loss coefficient *K* is the parameter selected for the sensitivity study because it is the dominant contributor to frictional pressure drop across the core. The orifice *K* is not necessarily the parameter with the greatest degree of uncertainty, but relatively large variations in this parameter are expected to encompass the uncertainty in all other hydraulic resistance parameters within the core region. Sensitivity calculations for U1C13 and U2C11 (most recent fuel cycles) showed that the rise in suppression pool temperature under ATWS conditions could vary by 1.6% and 0.4%, respectively, when variations in *K* of ±25% are considered. These uncertainties are relatively small and are accounted for in the final reported value of peak suppression pool temperature.

SABRE is a single core channel model. With this approach, thermal-hydraulic conditions in the core are represented by an average core channel and a separate bypass channel. This averaging process does not introduce any significant error into the calculation of coolant inventory in the core region. Using a 3-D core simulator (3-D kinetics and all fuel bundles modeled), a core-average void fraction of 0.414 was calculated for the reactor operating at 3489 MWt, 87 MLb/hr total core flow, and rated reactor pressure.[1] The core-average void fraction computed with SABRE for the same operating conditions is 0.429 which shows good agreement with the SIMULATE result. The SABRE result is documented in PPL Calculation EC-ATWS-1007, Rev. 7, SABRE Run# 01-180.

Core conditions computed with the single channel model used in SABRE are representative of conditions in an "average" channel, i.e., a channel operating at the average power and the average flow. All of the 764 channels in the reactor core have the same pressure drop since they are connected to common inlet and outlet plenums. The 92 channels on the peripheral region of the core have a more restrictive inlet orifice than channels in the interior region of the core. Fuel bundles on the periphery of the core operate at flows and power levels that are considerably lower than the average flow and power. Thus an "average" channel is not at all representative of a peripheral channel. Although the power and flow conditions within the interior channels vary from channel to channel, there are channels with flows/power levels that are somewhat higher than the average and somewhat lower than the average. Thus, an "average" channel is much more representative of an interior channel than it is of a peripheral channel. Therefore, a loss coefficient corresponding to a central region orifice is used in the SABRE model. The core pressure drop calculated with SABRE shows reasonably good agreement with the design pressure drop for the core. From Table 4.4-1 of the SSES FSAR, Rev. 53, the core pressure drop at design thermal power and total core flow rate of 108 MLb/hr is 21.2 psi for ATRIUM-10 fuel. For the same operating conditions, SABRE calculates a core pressure drop of 22.3 psi which differs from the design value by only ~1 psi.

## NRC Question 3

What kind of post-processing package has been used to extract graphical data from SABRE computer code output? Please provide the package to the NRC staff.

## PPL Response

A FORTRAN-77 post-processing program was written in-house to extract selected variables from the SABRE output file and put them in a file which is compatible with the Tecplot plotting package. The program was forwarded to NRC by PLA-5456 dated April 18, 2002.

[1] This result was obtained using the EPRI SIMULATE computer program. Ref. EPRI NP-2792-CCM, "SIMULATE-E: A Nodal Core Analysis Program for Light Water Reactors," Electric Power Research Institute, March 1983. Results are documented in PPL Calculation EC-ATWS-1007, Rev. 0; SIMULATE Run# 0105021.

**NRC Question 4**

It is observed that SABRE computer code uses different time step sizes for thermal hydraulic calculations and neutronics calculations. Please explain how the core power calculation is synchronized with the fluid and heat transfer calculation. The impact on accuracy needs to be discussed. Please provide a comparison between the unsynchronized and synchronized results.

**PPL Response**

In SABRE, the hydraulic and neutronic calculations are performed sequentially. A hydraulic time step is taken first. This advances the solution from $t_1$ to $t_2$. The hydraulic step size is specified by the user in the SABRE input deck. After the hydraulic step is complete, the neutronics solver (LSODES) is called in order to update the flux and power. The solver will return a solution at a specified time, i.e., at exactly $t=t_2$. LSODES chooses a time step in order to satisfy specified error criteria. Therefore, the step size selected by LSODES will be different from the hydraulic step size. If the neutronics step is greater than the hydraulics step, LSODES will integrate beyond $t=t_2$; however, a solution at $t=t_2$ is obtained by interpolation (interpolation is performed by LSODES). If the neutronics step size is smaller then the hydraulics step size, multiple kinetics steps are taken for each hydraulics step and interpolation is used to get kinetics output at exactly $t=t_2$ if LSODES integrates beyond $t=t_2$.

Generally the kinetics time steps are similar to or smaller than the hydraulic time steps when flow conditions are changing rapidly so the difference in step sizes does not lead to significant accumulation of error. This is evident in the benchmark studies in Sections 5.7 and 5.8 of EC-ATWS-0505, Rev. 8 where fission power calculated with SABRE shows very good agreement with GE results for the PREGO (Pressure Regulator Failure Open) and MSIV Closure ATWS scenarios. In events where the reactor is scrammed or shutdown on boron and power is changing very slowly, the numerical solver allows kinetics time steps which are significantly larger than the hydraulics time steps. Benchmark problem 10, "Suppression Pool Heatup from Decay Heat" demonstrates that there is no significant error associated with large kinetics time steps under shutdown conditions. If the user wishes to override the automatic time step control performed by the numerical solver LSODES, the maximum kinetics step size can be limited to any desired value since this parameter is part of the SABRE code input data. A maximum kinetics step size of 1 second was used in performing the benchmark studies in EC-ATWS-0505, Rev 8.

Sensitivity studies on hydraulic and neutronic time steps have been carried out in §5.8 (MSIVC ATWS) of EC-ATWS-0505, Rev. 8. Figures 5.8-3 and 5.8-4 show the effect of reducing the hydraulics step size while keeping error criteria constant for LSODES. There is a small change in peak power but no change in peak suppression pool temperature. Figures 5.8-6 and 5.8-7 show the effect of reducing error limits for neutronics calculation by an order of magnitude while the hydraulics step size is held constant. There is no noticeable change in the results. Therefore, it is concluded that the sequential solution procedure gives accurate results.

Results of an additional sensitivity study are included along with this response to demonstrate that the automatic time step control algorithm used in LSODES gives accurate results for neutron power. Case 8 in Calculation EC-ATWS-0505, Rev. 8 is an MSIV Closure ATWS with boron injection. This case was run with a maximum allowable kinetics time step of 1 second (1000 msec). With this restriction, the LSODES solver will select a time step size based on specified error criteria which are supplied as part of the SABRE input data. If the step size is less than or equal to 1 second, it is used in integrating the two-group diffusion equations, if it is greater than 1 second, the step size is limited to 1 second. This case was rerun using a much smaller maximum kinetics time step size of 5 msec. This maximum step size corresponds to the hydraulic time step size for the first 30 seconds of the transient. Fission power results for the two calculations are plotted in the attached Figure 1, (Page 18 of Attachment 1). It can be seen that there is no noticeable difference in the calculated fission power. Detailed numerical results for the two calculations are included on the CD included within Attachment 1 of this submittal. Output for SABRE Case 8 of EC-ATWS-0505, Rev. 8 with maximum kinetics time step size of 5 msec is contained in file sabre3v1.02-25.out, and the output with maximum kinetics step size of 1 second is in file sabre3v1.02-26.out.

## REQUESTED INFORMATION ABOUT CALCULATION EC-052-1018

### NRC Question 5

If the proposed change is made to the plant, will the high pressure coolant injection (HPCI) pump suction auto-swap from the condensate storage tank (CST) to the suppression pool triggered by low CST water level be unaffected? If so, is there a concern that the HPCI system may fail during an anticipated transient without scram (ATWS). Has this been considered or modeled in the risk evaluation? It has been indicated that manual rod insertion (MRI) can be initiated within 10 minutes into the event. Please provide justification for the 10-minute assumption.

**PPL Response**

The HPCI suction transfer on low CST level is not removed as part of the proposed modification, and therefore HPCI suction transfer is not affected and it will automatically occur on low CST level.

For a full ATWS with initial operation on the highest possible rod line (87 MLb/hr initial core flow and rated core thermal power) and with complete failure of SLCS, the reactor can be brought to hot shutdown without running out of water in the CST if MRI is initiated at 10 min into the event and if the control rod insertion time is not greater than 60 sec/rod. In order to manually insert control rods with the Reactor Manual Control System, the operator must bypass the RWM (Rod Worth Minimizer) and the RSCS (Rod Sequence Control System). Both of these systems can be bypassed by means of Control Room keylock switches. Typical control rod insertion times range from 45-55 seconds (Calc EC-EOPC-0519, Rev. 4, p. 162). If there is no makeup to the CST, the CST will run out of water at about 1hr 40 min and then HPCI suction will transfer over to the suppression pool (EC-EOPC-0519, Rev. 4, p. 162). At that time HPCI could fail, however enough control rods are inserted so that the reactor core is not susceptible to nuclear-coupled, density-wave instabilities when the reactor is depressurized to obtain coolant makeup from low-pressure injection systems.

Initiation of MRI at 10 minutes for a full ATWS is not an assumption. Rather, it represents the success criteria for the scenario. That is, in order for there to be a successful outcome in the full ATWS with SLCS failure, the operator must initiate MRI within 10 minutes. There is no expectation that the operator will initiate MRI within 10 minutes 100% of the time, and consequently an operator error rate for the action is included in the risk model. In the Susquehanna risk model, the probability that the operator will fail to initiate MRI within 10 minutes is 0.126.

In an ATWS event, manual control rod insertion and initiation of SLCS are performed by two different control room operators based on different plant symptoms. The cue for the operator to initiate SLCS is power > 5% or high suppression pool temperature. More than one control rod not fully inserted is the cue for the operator to initiate MRI.

**NRC Question 6**

The proposed new emergency operating procedure (EOP) requires a manual HPCI pump suction swap from the CST to the suppression pool if the pool level reaches 25 feet and the suppression pool temperature is less than 140 °F. Technical Specifications state a 24-foot maximum suppression pool limit. Please explain the magnitude of the level difference. In addition, please provide the suppression pool water level instrumentation accuracy.

**PPL Response**

The Technical Specification limit of 24 feet sets the initial suppression pool level for the accident analysis. Containment design is based on LOCA blowdown loads with pool level starting at 24 feet. Analysis in Calculation EC-052-1018 shows that during the accident, suppression pool level can be significantly higher than 25 feet without exceeding design load limits on the containment. The purpose of the suction transfer at 25 feet is to limit the rise in pool level so that suppression pool water does not flood the HPCI turbine exhaust piping in the event the system trips. Note that the risk analysis does not take credit for the manual transfer of HPCI suction from the CST to the suppression pool at 25 feet. If HPCI trips because of operator error in controlling RPV water level less than Level 8, it is conservatively assumed that HPCI will not restart.

The operator can monitor suppression pool water level on level indicator LI-1(2)5775B which is located on the HPCI panel in the control room. The tolerance of this level indicator is ±4 inches.

**NRC Question 7**

Has the containment load-limit curve described in Equation (1) on Page 7 been previously approved by the NRC? If not, what is the justification for using it?

**PPL Response**

The Load Limit Curve was developed by Kraftwerk Union (KWU) based on SRV load data obtained from the Karlstein T-quencher blowdown tests (Section 8 of SSES Design Assessment Report). These tests were designed to simulate SRV blowdown at SSES, and provided the necessary data to confirm the conservatism of the SSES SRV load definition. The data used in developing the Load Limit Curve is documented in the SSES Design Assessment Report. The NRC's review of the Design Assessment Report and acceptance in the Susquehanna Safety Evaluation Report (NUREG-0776) formed the basis for issuing the Susquehanna Operating License.

**NRC Question 8**

What is the elevation difference between the HPCI turbine outlet (not the exhaust line) and the suppression pool normal water level?

**PPL Response**

The elevation of the HPCI turbine outlet is (651'-6")-(21")= 649.75 feet. The elevation at the water surface in the suppression pool is 672 feet with water level at 24 feet. The elevation difference is 672-649.75=22.25 feet.

**NRC Question 9**

Are all the safety relief valve (SRV) discharge line vacuum breakers located in the drywell? If they are, do we expect that the water level in the SRV discharge lines is lower than the suppression pool level during a postulated loss-of-cooling accident (LOCA)?

**PPL Response**

The SRV discharge line vacuum breakers are located in the drywell. During a small break LOCA, e.g., 0.02 $ft^2$ break, the downcomer vents are cleared in the early part of the event (t<970 seconds per §5.1.3 of EC-052-1018), but they later refill because the cold water injected by HPCI decreases the enthalpy of the coolant exiting the break and this starts to have a cooling effect on the drywell atmosphere. Thus the SRV tailpipe water level is lower in the early part of the event but not in the latter part.

**NRC Question 10**

It is stated in the calculation that the suppression pool letdown system will be used to lower the suppression pool water level during a small-break LOCA with the assumption of loss-of-offsite power (LOOP). Please provide the letdown system flow path drawings, relevant portions of the applicable EOP and documentation to demonstrate that the letdown pump motor can be powered during a concurrent LOCA and LOOP event.

**PPL Response**

EO-000-103 is being revised to instruct the operator to maintain suppression pool level less than 25 feet by means of pool letdown. One of the available pool letdown paths utilizes the RHR suppression pool cooling flow path to letdown water to Liquid Radwaste. A safety-related RHR pump provides motive force for suppression pool letdown. The letdown capability associated with this path is 119.7 Lbm/sec for water density at 62.1 Lbm/$ft^3$.[2] This letdown rate corresponds to a volumetric flow of 865 gpm.

Included within Attachment 1 are selected pages from the procedures that instruct the operator to reduce suppression pool level by letdown of water to Liquid Radwaste or to the main condenser. Procedure pages that are included consist of:

[2] Calculation EC-THYD-1007, Rev. 0, p. 5

p. 19 of EO-000-103, "Primary Containment Control," Rev. 2 (Draft),

pp. 7 and 8 of ES-159-002, "Suppression Pool Letdown/Containment Venting Isolation Bypass," Rev. 4, and

pp. 13, 14, and 15 of OP-149-005, "RHR Suppression Pool Cooling," Rev. 19.

Also included is P&ID M-151, "Residual Heat Removal," Rev. 53, Sheet 1 which shows the suppression pool letdown path to Liquid Radwaste.

Note that Steps 4.3.1 and 4.3.2 of ES-1(2)59-002 do not allow suppression pool letdown until it is confirmed that primary coolant activity is within Technical Specification limits (T.S. 3.4.7). Currently, coolant activity less than Technical Specification limits is verified by performing Chemistry Surveillance SC-1(2)76-102 which takes several hours to complete. The requirement to perform a chemistry surveillance prior to initiating suppression pool letdown was added to the procedures after the analysis supporting the proposed modification to HPCI was already complete. Consequently, the delay associated with analyzing primary coolant activity was not factored into the safety analysis. A change to ES-1(2)59-002 is currently being pursued to replace the requirement to perform chemistry surveillance SC-1(2)76-102 with actions that are based on data readily available to the operator and which provide a high degree of assurance that primary coolant activity is less than the Technical Specification limit, but require much less time to perform.

## QUESTIONS ABOUT CALCULATION EC-RISK-1083

### NRC Question 11

In Section 2.5, two operator actions are identified to prevent water hammer damage to HPCI. Both actions are tied to the 26-foot level of the suppression pool. However, on page 32 in Attachment 1 of the June 8, 2001, submittal, it states that "[b]ecause 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." Based on the submittal, these actions would not be taken and should not be credited in the analysis, as the level would exceed 25 feet. Did the probability risk assessment evaluation include credit for either of these two operator actions? If so, please explain the apparent inconsistency between the submittal and the risk calculation and identify what the impact would be on the results if these two operator actions were not credited?

**PPL Response**

In the current EOPs, the actions to start HPCI when suppression pool level reaches 26 feet and to maintain suppression pool level less than 26 feet by using suppression pool cleanup, or RHR suppression pool cooling letdown, are both tied to the 26-foot level of the pool. However, the EOPs are being revised to change the pool level associated with these actions from 26 feet to 25 feet. In addition, the revised EOP will allow restarting of HPCI with suppression pool level greater than 25 feet if the system is needed for adequate core cooling or pressure control. The small liquid break accident is the only design-basis event where HPCI operability can be threatened by high suppression pool level. In the risk study which supports the proposed modification, CDF and LERF for the small liquid break accident are calculated conservatively by assuming that HPCI will always be inoperable if the system trips as a result of operator error in controlling reactor water level less than Level 8 (HPCI high level trip setpoint).

**NRC Question 12**

Based on statements contained on page 32 of Attachment 1 of the June 8, 2001, submittal, the exhaust line will begin to fill at 25.1 feet and be completely filled when the suppression pool level reaches 27.2 feet. The potential for failing HPCI on a restart is stated to be of concern if the suppression pool level is greater than the 25.1-foot level. This is why there is the restriction on the HPCI pump restart if the level is above 25 feet. Section 2.8 identifies a credible error in implementing the manual transfer that would cause the HPCI pump to trip, but then states this potential error has no consequences due to its brevity. It is not clear how long after the alarm signal is received that the operators will begin to execute the manual transfer. If there are procedural delays/confirmations or other factors that impact the initiation of the manual transfer, the transfer may occur approximately at the time the suppression pool level is actually reaching the 25-foot level. If the HPCI pump trips during the manual transfer at this time, then in accordance with the original submittal, a restart of the HPCI pump would not be allowed. Therefore, the identified operator error may have a direct impact on HPCI success and should be modeled as a potential failure mode of the system. Please explain the timing and associated factors leading up to the operator taking the steps to perform this manual transfer. If there is the potential for this operator error to result in a trip of HPCI at about the 25-foot level, please revise the model to reflect this potential failure mode of HPCI during the manual transfer and provide the revised results.

**PPL Response**

The risk model has been revised to not take credit for the manual HPCI suction transfer. In the small liquid break accident, if HPCI trips as a result of operator error in controlling reactor water level less than the HPCI high level trip setpoint, it is conservatively assumed that HPCI will not restart. If HPCI trips with suppression pool level greater than 25 feet, restarting the system would be allowed by procedure if HPCI is needed for adequate core cooling or pressure control.

**NRC Question 13**

Section 4.1.1 indicates that HPCI success is conditioned on standby liquid control (SBLC) operability. However, the event tree reverses these two top events. For the current condition, based on Section 4.1.1, sequences ATWS_8 and ATWS_9 are not possible because SBLC is failed, which should actually guarantee failure of HPCI and thus MRI. The event tree logic resulting in these sequences is not precisely correct. Further, it is not clear from the event tree if different results would be achieved if credit was given for the potential to use reactor core isolation cooling (RCIC), control rod drive (CRD), and SBLC, as identified in this section. Finally, it appears that the licensee has performed the analysis using a "one-top" model quantification process, which could result in the subsumming of valid event tree sequences.

**PPL Response**

In the ATWS_9 sequence, HPCI can start and run with out SBLC being successful. However, it will not run long enough to drive a sufficient number of control rods to shut down the reactor before core damage occurs. HPCI fails during the ATWS because the suppression pool level will rise above the automatic transfer point and temperature will increase above 190°F.

PPL is using a single-top model for each CDF and LERF. Using a single top model will subsume cutsets from sequences that, in the final result, are non-minimal. It is true that quantifying on a sequence basis each cutset is valid for that sequence. However, to arrive at the total CDF or LERF after quantifying by sequence; the sequences should be grouped together and subsumed to delete the non-minimal cutsets. PPL has modeled each sequence with sequence "flags" unique to the sequence in the fault tree model. As such, the cutsets are reported with sequence flags making them unique and not subsummable in other sequences.

**NRC Question 13A**

Please expand upon the discussion in Section 4.1.1 of using RCIC, CRD, and SBLC specifically identifying the conditions under which these systems can be or cannot be credited, state if these systems were credited in the analysis, and provide the revised results pre- and post- modification if it is appropriate to credit these systems.

**PPL Response**

The context in which credit is given to RCIC, CRD, and SBLC is for high-pressure makeup *after* SBLC has succeeded in injecting enough boron to make the reactor subcritical. RCIC, CRD, and SBLC are not a substitute for HPCI operation during the boron addition by SBLC. This statement was made as a matter of information. No model changes are necessary.

**NRC Question 13B**

By switching the event tree top logic so that the SBLC top event comes before the HPCI top event for the current plant conditions, correct sequencing would include cutset results for sequences ATWS_4, ATWS_6, ATWS_9, ATWS_11, ATWS_12, and ATWS_13, but not for sequences ATWS_8 and ATWS_14. However, using the calculation's ATWS event tree, sequence ATWS_11 could have been inappropriately eliminated if a "one-top" model quantification process was employed. Please provide on a sequence-specific basis the core damage frequency/large early release fraction (CDF/LERF) results pre- and post-modification for the ATWS event.

**PPL Response**

A sequence by sequence basis for CDF and LERF is included in Attachment 2 to PLA-5470, EC-RISK-1083 Revision 1 for the pre and post modification ATWS event.

**NRC Question 13C**

For the current plant, based on the switched event tree top logic, the end state class for Sequence ATWS_9 should be the same as that currently identified in the calculation for Sequence ATWS_14 (i.e., PDS-2), since in both sequences HPCI cannot be successful with SBLC failed for the current plant. Please describe and quantify the impact on LERF from switching the end state class for Sequence ATWS_9 to PDS-2 for the current plant.

**PPL Response**

In the ATWS_9 sequence, HPCI can start and run with out SBLC being successful. However, it will not run long enough to drive a sufficient number of control rods to shut down the reactor before core damage occurs. HPCI fails during the ATWS because the suppression pool level will rise above the automatic transfer point and temperature will increase above 190°F. In this situation, HPCI failed, SBLC failed, and core damage occurs the operator will continue to drive rods with MRI to save containment. If MRI does not fail, containment failure can be avoided. Hence, ATWS_9 does go to core damage due to limit cycle operation, PDS-2. (It was previously labeled as PDS-12 but it was correctly added into the CDF number, PDS-2 is also added to the CDF number.) If the operator is not successful in driving rods with MRI after core damage, the containment will fail. This sequence results in plant damage state PDS-5L and goes into the LERF number. PDS-5L has been added to Revision 1 of EC-RISK-1083, (Attachment 2 to PLA-5470).

A similar situation exists for ATWS_14. In an ATWS_14, HPCI fails immediately and SBLC fails. With HPCI initially failed, MRI is not tested for success or failure since without HPCI, MRI will not prevent core damage. ATWS_14 results in plant damage state PDS-2. Revision 1 of EC-RISK-1083 continues to test this sequence for driving control rods (MRI) even though core damage has occurred. Containment failure will be avoided if the operator successfully drives control rods. If rods are successfully driven, the sequence ends in plant damage state PDS-2 otherwise it ends in plant damage state PDS-5L. PDS-5L has been added to Revision 1 of EC-RISK-1083.

Note the operator error failing to drive rods for LERF is the same error as not driving rods to prevent core damage. Driving rods to prevent LERF is preformed by the same operator that would be driving rods after the ATWS occurred to prevent core damage. The same timing is used for both saving the core and saving the containment. Core damage will be avoided with successful HPCI operation (no automatic suction transfer is required for HPCI to succeed).

**NRC Question 13D**

The "one-top" model quantification process could affect other event tree results, in addition to the ATWS event tree. For this application, impacts are also expected in the small-break LOCA (SBLOCA) analysis. Therefore, the staff will also need to review the SBLOCA event tree and its results on a sequence-specific basis. Please provide the SBLOCA event tree and please provide on a sequence-specific basis the CDF/LERF results pre- and post-modification for the SBLOCA event tree.

**PPL Response**

The small liquid and steam break LOCA event trees are included in EC-RISK-1083 Revision 1.

A sequence by sequence basis for CDF and LERF is also included in the subject calculation for the pre and post modification small break LOCA event.

Please note that the sequence by sequence results compare favorably with the one top results if the same truncation level is used in the sequence by sequence case as was used in the one top model.

**NRC Question 14**

Section 4.1.2 identifies that two operator errors must fail for HPCI to fail. The first is for the operator control of reactor pressure vessel (RPV) water level, which is described further in Section 4.1.3a. The second operator error involves the failure to actually perform the manual transfer, which is described further in Section 4.1.3b. However, the first error analyzed is only for the operators to gain control of the RPV water level and does not address the potential for the operators to fail to maintain control of RPV water level. The second operator action would be highly dependent on this unanalyzed operator error of not maintaining RPV water level, especially since this error could occur very near the time needed to perform the transfer, which would result in the operators not restarting the HPCI pump and thus failing the system. In addition, the two identified operator actions may also be highly dependent as both actions use the same timing window, especially if performed by the same operator. Also, if the operator fails to gain control of RPV level, the HPCI pump will trip at RPV Level 8 and not restart until RPV Level 2 is reached, but the times associated with reaching RPV Level 8 and then reaching RPV Level 2 have not been provided. Again, this could put the HPCI being in the tripped state at the time the level in the suppression pool reaches the 25-foot level and would make the two identified operator actions essentially fully dependent. Please revise the model to reflect the potential for the operator error to maintain control of RPV water level to result in the direct failure of HPCI, without any other operator errors needed, discuss and revise the model accordingly to address the potential dependency between the identified operator actions, and provide the revised results.

**PPL Response**

The calculation has been revised to only credit one operator action to prevent HPCI failure, operator controlling RPV level with HPCI. No credit is taken for manual HPCI suction transfer to prevent HPCI failure on restart with high suppression pool level. If the operator is successful at initially taking control of the RPV level with HPCI, RPV level control is deemed successful. It is the opinion of a SSES simulator instructor that the dominant error would be to initially fail to take level control and once the operator takes control he will maintain control. Hence, an additional error to maintain control of RPV level after initially taking control is not necessary.

**NRC Question 15**

Section 4.1.3b indicates that the alarm is actuated by level switches LSHE411(2)N015A or LSHE411(2)N015B. Was the potential for the failure on demand and pre-initiator time-based failure of both switches and associated signal/relay logics modeled in the SSES PRA evaluation, including the potential for common cause failures? If so, please provide the associated demand and time-related failure probabilities used in the model and their bases. If not, please revise the model to reflect the potential for these failures to fail the associated operator action to perform the manual transfer and provide the revised results.

**PPL Response**

Section 4.1.3b has been deleted. No credit is taken for the operators preventing a HPCI failure by transferring the suction source to the suppression pool from the CST. Therefore the operator being cued by the high suppression pool level alarm is no longer relevant.

**NRC Question 16**

The estimated CDF/LERF results indicate no differences between using the mean, the 95 percentile human error probability (HEP), and the no operator error results (i.e., HEP=0). Also, the LERF results don't even change when the operator error is assumed certain (i.e., HEP=1). Please explain why there are no differences in these results, though the HEP value is changed, and please provide the subject HEP value(s) used in each of these quantifications.

**PPL Response**

The calculated total CDF indicates no change for the 95th percentile operator error, the mean operator error and no operator error. Note the only operator error that is being varied is the operator error to control RPV level after a small liquid break LOCA. The reason the total CDF does not change between these cases is that the small liquid break LOCAs CDF contribution is on the order of E-10. With the total CDF being 4.86E-7, the small liquid break LOCA contribution is lost in the number of significant figures reported.

The calculated total LERF results indicates no change for all post modification cases. All the LOCA sequences that contribute to LERF were quantified separately from the other LERF sequences. This effort revealed the largest LOCA LERF cutset having a frequency of 6.95E-14. The total of all the LOCA LERF cutsets was 1.04E-12. Since the truncation on all the LERF contributors was 1E-12 the LOCA LERF contributors don't make the truncation limit.

The HEP values used are discussed in Section 4.1.3 of calculation EC-RISK-1083 Revision 1.

**NRC Question 17**

The results for the post-modification using the mean and 95 percentile HEP actually indicates a relatively large CDF reduction for small LOCAs (both steam and liquid), which is counter-intuitive to what is expected. A relatively large CDF increase is identified for small liquid LOCAs, if the operator error is assumed to occur, which is expected. The evaluation also indicates a relatively large CDF reduction for the reactor building closed cooling water initiator and for the small steam LOCAs, even when assuming the certainty of the operator failure to perform the manual transfer. These events dominate the risk reduction, though they appear to be either unrelated to the proposed modification and/or are counter-intuitive results. Similarly, there are many reductions in LERF that are counter-intuitive and many initiators go from a contribution pre-modification to zero contribution post-modification. Please describe why and how each of the initiators that change in contribution (by absolute value) are impacted by the proposed modification. In addition, please explain why using the mean and 95 percentile HEP values result in a relatively large CDF reduction (factor of 2) for SBLOCAs, but assuming certain failure results in an even larger relative CDF increase (factor of 15) for small liquid LOCAs. Also, please explain why the modification has an impact on small liquid LOCAs, but not small steam LOCAs when the operator failure is assumed.

**PPL Response**

The post modification risk reduction for small break LOCAs (both steam and liquid) in EC-RISK-1083 Revision 0 is due to a modeling error. The fault tree for closing the MSIVs included LOCA initiators, which in reality would cause the MSIVs to close, but the fault tree was used in some ATWS sequences. Hence, the CDF number reported for these LOCAs included LOCA/ATWSs. We have no thermal hydraulic basis for a LOCA concurrent with an ATWS therefore the result of using an all inclusive fault tree yielded situations that went beyond our calculational basis. Revision 1 of EC-RISK-1083 corrects this problem.

Revision 1 of EC-RISK-1083 reports increases in CDF for small break liquid LOCAs post modification. The CDF contribution from small liquid break LOCAs is the same pre modification to that of the post modification case with no operator error. It is expected that these two cases would be the same. Prior to the modification HPCI automatically transferred to the suppression pool to preclude failing on restart with high suppression pool level and post modification (no operator error) the operator always succeeds in controlling RPV level with HPCI hence, HPCI does not trip. In both cases HPCI does not fail due to suppression pool level concerns. The post modification CDF increases from the no operator error to the certain operator error case for the small liquid break LOCAs, which is expected.

The modification has an impact on small liquid LOCAs but not small steam LOCAs when the operator failure is assumed because there is ample time to letdown the suppression pool inventory for the small steam LOCA and avoid HPCI failure on restart with high pool level.

**NRC Question 18**

Given the extremely low CDF/LERF results calculated, what quantification cutoff/truncation CDF/LERF values were used in requantifying the model? Please describe how the selected cutoff values assure that potentially important contributors have not been discarded. If the cutoff value was less than 4 orders of magnitude below the total CDF/LERF, please requantify the model using a cutoff value at least at these values (e.g., 1 E-11/year for CDF and 5E-13/year for LERF) and provide the revised results.

**PPL Response**

The model was quantified at 4 or more orders of magnitude below the total CDF/LERF values. The truncation limit used for both was 1E-12.

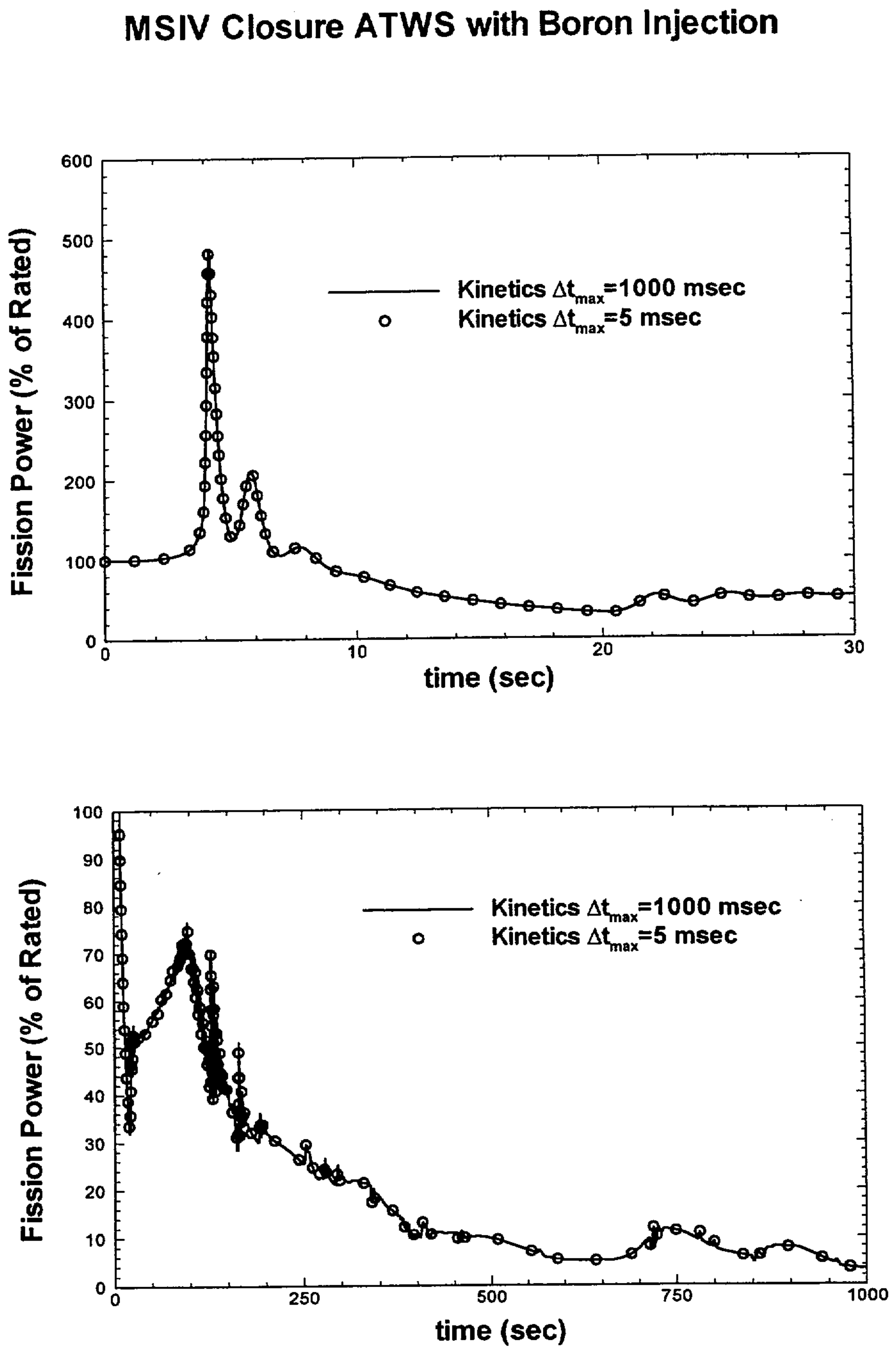



**Figure 1 Effect of reducing maximum kinetics time step size from 1 second to 5 msec for SABRE Case 8 of EC-ATWS-0505, Rev. 8.**

# Attachment 2 to PLA-5470

# Calculation EC-RISK-1083, Revision 1

SP/L-10 MAINTAIN SUPP POOL LVL < 25'
USING:

SUPP POOL CLEANUP SYSTEM
BYPASSING ISO AS NECESSARY
IAW ES-159-002(ES-259-002)
<u>OR</u>
RHR SUPP POOL COOLING LETDOWN
BYPASSING ISO AS NECESSARY
IAW ES-159-002(ES-259-002)

Water level is maintained below the elevation of the bottom of the HPCI turbine exhaust line which begins to flood at a suppression pool water level of 25' 1".

Since removal of water from the suppression pool may be prevented by isolation signals, permission is given in ES-159-002(ES-259-002), Primary Containment Letdown Isolation Bypass, to bypass these isolations.

(Reference: SSES-EPG SP/L-3.2)

4.3 To Bypass Isolation Signals to Support Suppression Pool Letdown Using RHR Suppression Pool Letdown Using RHR Suppression Pool Cooling:

4.3.1 NOTIFY Chemistry to perform SC-176-102.

NOTE: Results take several hours to obtain.

4.3.2 CONFIRM <u>minimal</u> or <u>no</u> fuel damage exists as indicated by the following:

a. Primary coolant activity within limits established in SC-176-102.

________________________
Confirmed By

<u>AND</u>

b. Offsite release less than "Alert" limits as stated in EP-PS-100, Emergency Director (as defined in Emergency Classification section).

________________________
Confirmed By

4.3.3 If desired to use RHR System operating in Suppression Pool Cooling Mode to lower level, PERFORM the following:

a. ENSURE valves/components are in their isolation position, and can be maintained in their isolation position for the duration of this procedure in accordance with Attachment B.

NOTE: RHR sampling valves may be opened for sampling as required per OP-149-005, and then closed when sampling is complete.

________________________
Confirmed By

b. In Panel 1C611 (LRR) (rear), RPS Trip Sys B1/B2 NSS Shutoff Sys Panel, INSTALL jumper from CCC7-5 to CCC7-6 to defeat Low Level 3/High Drywell Pressure isolation signal to valve HV-151F040.

NOTE: Terminal strip CCC7 is in RPS B1 panel right door on right side, approximately 5 feet off floor.

____________________
Confirmed By

c. In Panel 1C611 (LRR) (rear), RPS Trip Sys B1/B2 NSS Shutoff Sys Panel, INSTALL jumper from CCC9-1 to CCC9-2 to defeat Low Level 3/High Drywell Pressure isolation signal to valve HV-151F049.

NOTE: Terminal strip CCC9 is in RPS B2 panel left door on right side, approximately 5 feet off floor.

____________________
Confirmed By

d. At Panel 1C601 DEPRESS MN STM LINE DIV 1 and DIV 2 ISO RESET pushbuttons (HS-B21-1S32 and HS-B21-1S33).

____________________
Confirmed By

e. To lower level, PERFORM IAW OP-149-005 section Alternate Suppression Pool Level Control with RHR in Normal Suppression Pool Cooling Mode Operation.

____________________
Confirmed By

## 3.3 ALTERNATE SUPPRESSION POOL LEVEL CONTROL WITH RHR IN NORMAL SUPPRESSION POOL COOLING MODE OPERATION

3.3.1 Prerequisites

a. RHR Loop A(B) operating in Suppression Pool Cooling Mode in accordance with section 3.1 of this procedure.

b. If water to be routed to condenser hotwell, Suppression Pool Cleanup System shutdown in accordance with OP-159-001.

c. If water to be routed to condenser hotwell, condenser hotwell available to receive water from Suppression Pool.

d. If water to be routed to Radwaste, SUPP POOL WTR FILT PUMP or RWCU not rejecting water to liquid radwaste.

e. If water to be routed to Radwaste, Radwaste available to receive water.

f. CL-159-0012 complete.

3.3.2 Precautions

a. If off gas treatment system is not in service and mechanical vacuum pump is operating, transfer of suppression pool water to hotwell may result in increased activity levels at turbine building vent stack.

b. If off gas treatment system is not in service and vacuum is broken, transfer of suppression pool water to hotwell may result in increased activity levels in turbine building.

3.3.3 If water to be routed to condenser hotwell and Off Gas System is not in service, either:

a. ENSURE mechanical vacuum pump not in operation,

OR

b. NOTIFY Chemistry to:

(1) OBTAIN grab sample of suppression pool for radioactive iodine analysis.

(2) PROVIDE concurrence for continued operation of mechanical vacuum pump based on suppression pool radioactive iodine < 1E-3μCi/g.

3.3.4 PLACE RHR LOOP A(B) MOV OL BYPS HS-E11-1S62A(B) to TEST.

3.3.5 PLACE AC MOV OL BYPS HS-B21-1S37A to TEST.

3.3.6 PLACE DC MOV OL BYPS HS-B21-1S37B to TEST.

3.3.7 To allow piping between cross-tie valves to pressurize:

a. MANUALLY CRACK OPEN RHR Loop A(B) Cross-Tie HV-151-F010A(B) ≥ 15% (~ 285 turns on handwheel).

b. As directed by Shift Supervision, RESTORE RHR LOOP A(B) CROSS-TIE HV-151-F010A(B) to operable status by closing Loop A(B) Crosstie Vlv HV-151-F010A(B) supply breaker 1B216022(1B226064).

3.3.8 OPEN RHR LOOP A(B) CROSSTIE HV-151-F010A(B).

3.3.9 If water to be routed to condenser hotwell:

a. CLOSE Liquid RW Iso 151088.

b. OPEN Supp Pool Clnup to Cdsr Iso 157310.

NOTE: If water to be routed to Radwaste, no manual valve manipulations are required since Liquid RW Iso 151088 is normally open.

3.3.10 OPEN RADWASTE OB ISO HV-151-F049.

3.3.11 To establish desired letdown flow, THROTTLE OPEN RADWASTE IB ISO HV-151-F040.

3.3.12 After 2 minutes:

a. PLACE AC MOV OL BYPS HS-B21-1S37A to NORM.

b. PLACE DC MOV OL BYPS HS-B21-1S37B to NORM.

c. PLACE RHR LOOP A(B) MOV OL BYPS HS-E11-1S62A(B) to NORM.

3.3.13 MONITOR Suppression Pool level to ensure maintained in compliance with TS 3.6.2.

3.3.14 PLACE AC MOV OL BYPS HS-B21-1S37A to TEST.

3.3.15 PLACE DC MOV OL BYPS HS-B21-1S37B to TEST.

3.3.16 PLACE RHR LOOP A(B) MOV OL BYPS HS-E11-1S62A(B) to TEST.

3.3.17 When desired level reached or as required by plant condition <u>and</u> if water being routed to condenser PERFORM the following:

a. CLOSE Supp Pool Clnup to Cdsr Iso 157310.

b. OPEN Liquid RW Iso 151088.

3.3.18 CLOSE RADWASTE IB ISO HV-151-F040.

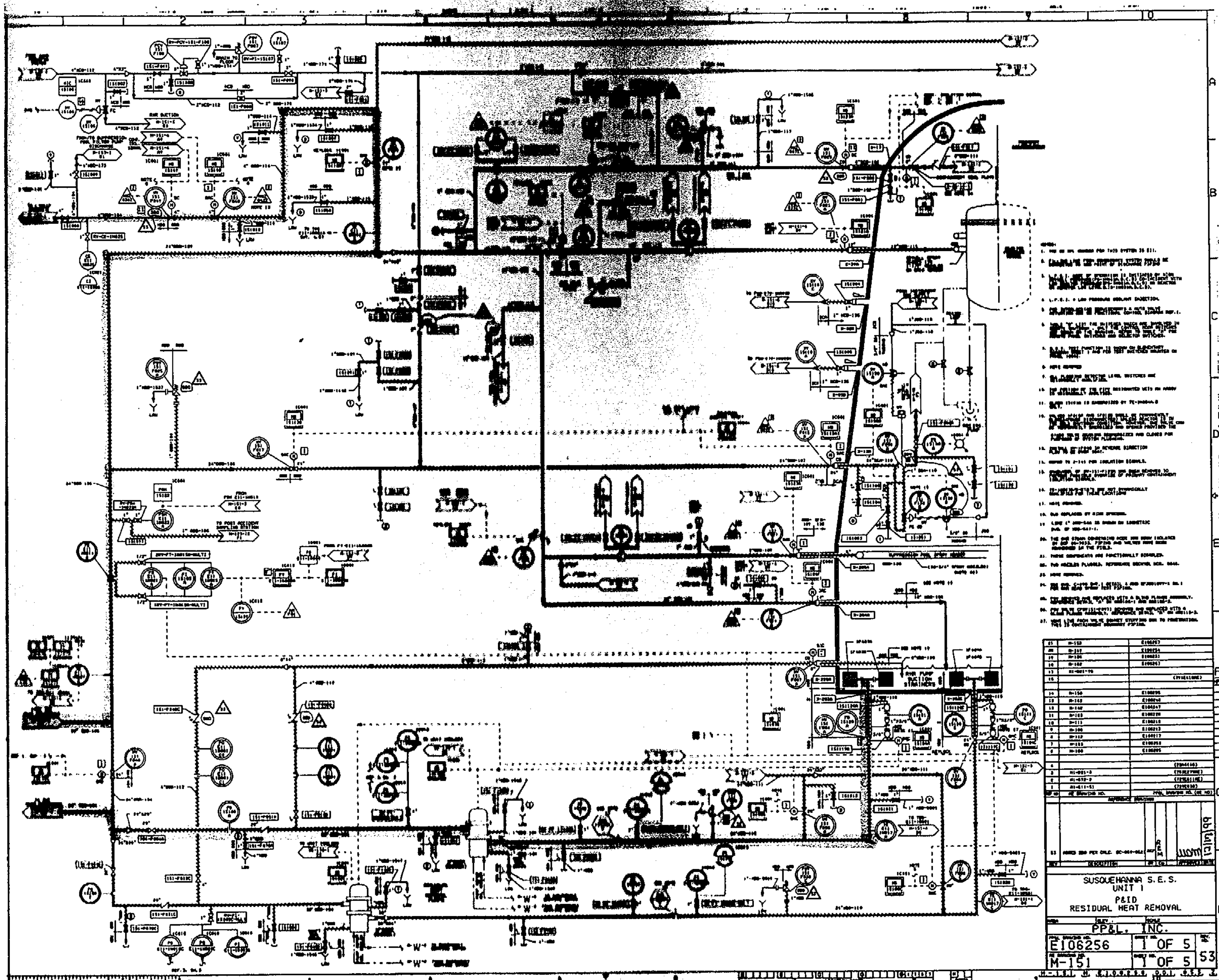
3.3.19 CLOSE RADWASTE OB ISO HV-151-F049.

3.3.20 CLOSE RHR LOOP A(B) CROSSTIE HV-151-F010A(B).

3.3.21 As directed by Shift Supervision, RESTORE RHR LOOP A(B) CROSS-TIE HV-151-F010A(B) to standby alignment by opening Loop A(B) Crosstie Vlv HV-151-F010A(B) supply breaker 1B216022(1B226064).

3.3.22 If operation of RHR Loop A(B) in Suppression Pool Cooling Mode no longer required, SHUT DOWN in accordance with section 3.1 of this procedure.

FOR INFORMATION ONLY



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 24, 1992

Docket Nos. 50-387
and 50-388

Mr. Harold W. Keiser
Senior Vice President - Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: TOPICAL REPORT PL-NF-89-005, "QUALIFICATION OF TRANSIENT ANALYSIS METHODS FOR BWR DESIGN AND ANALYSIS," SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M82371 AND M82372)

By letter dated January 22, 1990, Pennsylvania Power and Light Company (the licensee) requested NRC review and approval of Topical Report PL-NF-89-005, "Qualification of Transient Analysis Methods for BWR Design and Analysis" for Susquehanna Steam Electric Station (SSES), Units 1 and 2. Based upon the staff's review, as given in the enclosed NRC Safety Evaluation, we find the application of PL-NF-89-005 acceptable for use in the SSES, Unit 1 Cycle 7 reload analysis under the limitations delineated in the associated NRC technical evaluation.

A revised methodology which explicitly models a time-varying axial power distribution in the hot fuel bundle has also been reviewed and approved with the limitations delineated in the enclosed NRC Safety Evaluation. This revised methodology will be used by the Pennsylvania Power and Light Company for future reload analyses beginning with SSES, Unit 2 Cycle 6.

Sincerely,

James J. Raleigh

James J. Raleigh, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation and
Technical Evaluation Report

cc w/enclosure:
See next page

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MAR 27 1992

SENIOR VP NUCLEAR

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station,
Units 1 & 2

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ENCLOSURE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
# RELATING TO TOPICAL REPORT PL-NF-89-005
# "QUALIFICATION OF TRANSIENT ANALYSIS METHODS
# FOR BWR DESIGN AND ANALYSIS"
# PENNSYLVANIA POWER & LIGHT COMPANY
# SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2
# DOCKET NOS. 50-387 AND 50-388

## 1.0 INTRODUCTION

By letter from H. W. Keiser to W. R. Butler (NRC), dated January 22, 1990, Pennsylvania Power and Light Company (PP&L) submitted topical report PL-NF-89-005, "Qualification of Transient Analysis Methods for BWR Design and Analysis," for NRC review. The methodology described in the report was intended as a technical basis for the PP&L qualification to perform transient analyses for the two Susquehanna Steam Electric Station GE BWR-4 reactors. Subsequently, PP&L modified the proposed methodology to explicitly model a time-varying axial power distribution in the hot fuel bundle. PP&L intends to use this modified methodology (Method 2) starting with Susquehanna Unit 2 Cycle 6. The original methodology (Method 1), which assumes a constant axial power distribution in the hot bundle model, will only be used for the Susquehanna Unit 1 Cycle 7 reload analysis.

The NRC staff was supported in this review by our consultant, Brookhaven National Laboratory. The staff has adopted the findings recommended in our consultant's technical evaluation report (TER) which is attached. In addition, the staff's safety evaluation of the modified methodology (Method 2) which incorporates a time-varying axial power distribution follows.

## 2.0 EVALUATION

The attached TER provides the evaluation of the original methodology which assumes a constant axial power distribution in the hot bundle and will be used by PP&L only for the Susquehanna Unit 1 Cycle 7 reload analysis (Method 1). Calculations of limiting transients for Susquehanna Unit 1 Cycle 7 were performed with both the new approach (explicit treatment of time-varying axial power distribution) and the original method (constant axial power distribution) and have demonstrated the conservatism of the minimum critical power ratio (MCPR) operating limits generated with the original method. Therefore, based on the attached TER, the staff finds the original method (Method 1) acceptable for the Susquehanna Unit 1 Cycle 7 reload analysis.

For the revised methodology, the NRC-approved RETRAN02 MOD4 (Ref. 1) computer code was modified to explicitly model a time-varying axial power distribution in the hot bundle. In addition, a revised gap conductance methodology was used to model the hot bundle with the NRC approved ESCORE code (Ref. 2). As described in Reference 3, the axial power distribution and bundle power history used as input to ESCORE are derived from a SIMULATE-E (Ref. 4) cycle step-out calculation for the cycle being analyzed. This results in a power history of 6 kw/ft or less for most of the cycle.

During the NRC review of ESCORE, emphasis was on its application in LOCA analyses (e. g., conservatism in predicting fuel temperature during a transient) and benchmark data for operation below 6 kw/ft were not assessed. The staff, therefore, questioned the validity of ESCORE gap conductance predictions for the low power levels associated with the Susquehanna 9x9 fuel design. Although PP&L has indicated their predicted hot bundle fuel rod gap conductance is higher and, therefore, conservative relative to that calculated using a method previously approved by the NRC, comparisons with independent calculations and with benchmark cases presented for other codes resulted in values on the order of 10% to 20% higher than those obtained with ESCORE. The Safety Evaluation Report for ESCORE (Ref. 5) requires a calculational uncertainty to be determined in plant-specific applications and included explicitly as a conservative adjustment or used to confirm the adequacy of existing conservatism in fuel limits. Since no uncertainty estimates were provided for the ESCORE gap conductance, a 10% uncertainty multiplier (1.10) will be imposed on the calculated gap conductance. If appropriate benchmark information which validates the ESCORE calculated gap conductance at these lower powers is obtained at a later date, the staff will consider removing or revising this 10% uncertainty factor.

The staff also believes that the use of a best estimate power history in the transient analysis hot bundle gap conductance method may tend to underestimate the predicted gap conductance. If the actual hot bundle power exceeds the maximum bundle power assumed in the gap conductance analysis, more permanent pellet relocation would probably occur causing a higher hot bundle gap conductance than assumed. A hot bundle power 10% higher than the maximum power assumed in the gap conductance calculations would produce a gap conductance that is also approximately 10% higher. However, the net effect of a less than 10% increase in hot bundle gap conductance in conjunction with a similar increase in core average gap conductance is not expected to have a significant effect on the calculated change in critical power ratio (delta-CPR) for limiting events. Therefore, changes in hot bundle power which do not have peak powers greater than 110% of the maximum value used in the gap conductance calculation will not have a significant impact on minimum critical power ratio (MCPR) operating limits. PP&L has committed to reevaluate the MCPR operating limits in the event of occurrences which could potentially increase the hot bundle power by at least 10% above the value assumed in the licensing analysis of hot bundle gap conductance (Ref. 6). Those events which would require an evaluation are divided into three categories; core wide events, local power events, and changes in planned operation.

For core wide events, any plant event which increases reactor power to a value greater than 110% of rated power will require an evaluation of the MCPR operating limits. Examples of potential events which could cause this type of core wide power change are the generator load rejection, feedwater controller failure, and loss of feedwater heating events.

For local power events, any plant transient which produces a bundle power greater than 110% of the maximum bundle power assumed in the hot bundle gap conductance licensing analyses will require an evaluation of the MCPR operating limits. Examples of potential events which could cause this type of local power change are the rod withdrawal error, rod drop event, and rod drift.

Any change to the planned operation of the cycle which would result in bundle powers greater than 110% of the maximum bundle power assumed in the hot bundle gap conductance licensing analyses will require an evaluation of MCPR operating limits.

Based on this, the staff finds the revised PP&L transient methodology which incorporates an explicit modelling of the time dependent hot bundle axial power distribution (Method 2) acceptable for analysis of future Susquehanna reloads.

## 3.0 CONCLUSIONS

The staff has reviewed the PP&L transient methods topical report PL-NF-89-005 and the supporting documentation provided in response to our requests for additional information. Based on this review, the staff concludes that the PP&L transient methods and uncertainty estimates (Method 1) are acceptable for use in the Susquehanna Unit 1 Cycle 7 reload licensing analyses under the conditions stated in the attached TER.

The staff has also reviewed the revised methodology which incorporates an explicit modelling of the time dependent hot bundle axial power distribution (Method 2) and finds it acceptable for analysis of future Susquehanna reloads with the following provisions:

(1) The calculated value of gap conductance shall be increased by a 10% uncertainty factor. The staff will consider removing or revising this uncertainty at a later date if appropriate data becomes available to validate ESCORE calculated gap conductance values at these lower powers.

(2) The MCPR operating limits would require a reevaluation for any core wide event which increases reactor power to a value greater than 110% of

rated power or for any local power event or change to planned operation which produces bundle powers greater than 110% of the maximum bundle power assumed in the licensing analyses of gap conductance.

## 4.0 REFERENCES


1. "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A Revisions 2 and 3 Regarding RETRAN-02/MOD-003 and MOD-004", Letter from A. C. Thadani (NRC) to R. Furia (GPU Nuclear), October 19, 1988.

2. EPRI NP-5100-L-A, "ESCORE-The EPRI Steady State Core Reload Evaluator Code: General Description", April 1991.

3. PLA-3729, "Susquehanna Steam Electric Station, Response to RAI on Transient Analysis Methods", Letter from H. W. Keiser (PP&L) to C. L. Miller (NRC), February 12, 1992.

4. EPRI-NP-2792-CCM, "SIMULATE-E Computer Code Manual", March 1983.

5. "Acceptance for Referencing of Licensing Topical Report EPRI-NP-5100, 'ESCORE-The EPRI Steady-State Core Reload Evaluator Code: General Description'", Letter from A. C. Thadani (NRC) to C. R. Lehmann (PP&L), May 1990.

6. PLA-3748, "Susquehanna Steam Electric Station, Response to Question on Gap Conductance Methodology", Letter from H. W. Keiser (PP&L) to C. L. Miller (NRC), March 16, 1992.

ATTACHMENT

TECHNICAL EVALUATION REPORT

| | |
|---|---|
| Topical Report Title: | Qualification of Transient Analysis Methods for BWR Design and Analysis |
| Topical Report Number: | PL-NF-89-005 |
| Report Issue Date: | December 1989 |
| Originating Organization: | Pennsylvania Power and Light Company |

## 1.0 INTRODUCTION

By letter (Reference-1), the Pennsylvania Power and Light Company (PP&L) has submitted the Topical Report PL-NF-89-005, "Qualification of Transient Analysis Methods for BWR Design and Analysis." The methodology described in the report is intended as a technical basis for the PP&L qualification to perform transient analyses for the two Susquehanna Steam Electric Station (SES) GE BWR-4 Reactors.

The methodology is based on the EPRI computer codes SIMTRAN-E (Reference -2), ESCORE (Reference-3), and RETRAN-02 MOD-004 (Reference-4). The steady-state core physics input to these codes is provided by SIMULATE-E (Reference-5). The thermal margin evaluation is performed with the Advanced Nuclear Fuels (ANF) XN-3 critical power correlation (Reference-6). The topical report includes a description of the Susquehanna models, and the qualification benchmarking against the Susquehanna SES Units 1 and 2 startup tests and the Peach Bottom-2 turbine trip tests. The calculations and models are intended as best-estimates in order to determine the code and model uncertainty and their adequacy for performing transient analyses. The conservative licensing analyses and models are described in the PP&L reactor analysis methods applications Topical Report PL-NF-90-001 (Reference-7).

The review of the PL-NF-89-005 topical report is summarized in the following sections. The topical report is outlined in Section-2 and the evaluation of the PP&L transient analysis methods is summarized in Section-3. The technical position is given in Section-4.

## 2.0 SUMMARY OF THE TOPICAL REPORT

The topical report provides (1) a detailed description of the Susquehanna SES RETRAN-02 system model, (2) the benchmarking comparisons of this model versus reactor test data and (3) the determination of the code and model uncertainty based on these comparisons.

### 2.1 Susquehanna RETRAN-02 Model

The RETRAN-02 model includes a detailed nodalization and geometry description of the Susquehanna Reactor System. The reactor core and bypass regions are modeled with 27 axial nodes, 25 of which are in the active core. The power in the active zones is determined by the one-dimensional kinetics model using the same 27-zone axial representation. In this model the void and doppler feedback are determined using the local moderator density and average fuel pellet temperature. The moderator density calculation accounts for subcooled voids in the neutronics feedback. The fuel pellet temperature is calculated with a three region (pellet, gap and clad) thirteen mesh model. Both conduction and direct moderator heating of the bypass region are included.

The two recirculation loops are modeled explicitly including volumes for the suction piping, recirculation pump and discharge piping. The recirculation model is based on a detailed model which has been compared to vendor data.

The steam line is modeled with nine volumes. The model was validated by a series of sensitivity calculations in which the number of volumes was systematically increased. The steam line is connected to the vessel steam dome and the steam line valves (HPCI and RCIC supply valves, and safety/relief valves) are included as negative fill junctions. A signal for the pressure regulator control system and for MSIV closure are taken from steam line volumes. The main steam bypass system includes a junction representing the bypass valves and a volume for the bypass header and steam chest. Heat conduction through the bypass piping has been included in order to provide improved agreement with the test data. The pressure reducers, spargers and condenser in the bypass line are all modeled with individual volumes. The loss coefficient in the bypass line were determined by comparison to measured bypass flow.

The upper plenum is modeled as a single volume connected to a standpipe region which empties into the separators. The separator carryunder and dryer carryover are based on vendor data. An upper downcomer, middle downcomer and lower downcomer region are included. The separator and upper downcomer models provide good agreement between measurements and calculations of upper plenum pressure and dome pressure. The lower plenum is modeled as a single volume and the lower plenum to core bypass loss coefficient has been adjusted to preserve the core pressure drop as determined by SIMULATE-E.

The jet pump model used in the RETRAN-02 analysis is a collapsed simplified version of a detailed (53-volume) model which PP&L has shown to give good agreement with vendor supplied jet pump performance data. The recirculation pumps are modeled using vendor pump characteristic curves.

The Susquehanna RETRAN-02 model includes five safety/relief valves (SRVs). Each valve represents a composite of up to four valves having a common pressure setpoint. The specific SRV modeling is based on the FSAR data of Reference-8. The four inboard main steam isolation valves (MSIVs) are represented by a single valve. The form loss coefficient of this valve is increased as the valve closes to provide an accurate calculation of the pressure increase during an MSIV event.

The Susquehanna RETRAN-02 model includes an extensive set of trips based on calculated variables including core power, pressure, water level and flow. The trips include insertion of control rods, activation of the SRVs, recirculation pump trip and runback, turbine trip, feedwater trip and HPCI and RCIC trip. In addition, a set of special trips on elapsed time have been included to analyze special events such as loss of feedwater heating and generator load rejection.

The RETRAN-02 core neutronics analysis is performed using a one-dimensional axial model. The kinetics parameters including two-group cross-sections, diffusion coefficients and delayed neutron parameters are calculated with SIMULATE-E in three dimensions via a set of perturbation calculations in which the moderator density and fuel temperature independent variables are varied. The kinetics parameters are collapsed radially using adjoint or volume weighting at the required transient initial statepoint conditions and as a function of rod insertion if scram occurs during the transient.

In order to establish the adequacy of the steam line nodalization, an additional calculation was performed in which the number of steam line volumes was increased from eight to fifteen. The Peach Bottom-2 turbine trip test was then calculated with both nodalizations and the core peak power and peak reactivity was found to agree to within ~1%.

## 2.2 Susquehanna Model Benchmarking

In order to validate the Susquehanna RETRAN-02 model, PP&L has made detailed comparisons of the RETRAN-02 predictions with the Susquehanna Units 1 and 2 Cycle-1 startup test data, the Peach Bottom-2 turbine trip measurements and the licensing basis transient (LBT) calculations of General Electric (GE) and Brookhaven National Laboratory (BNL), References 9 and 10, respectively.

The Cycle-1 startup tests at Susquehanna Unit-2 are at close to operating conditions and include the feedwater system water level and pressure regulator setpoint tests, a loss of feedwater heating test and the recirculation pump trip tests. The setpoint tests provide validation for the controller models and the calculation of the overall system response. The feedwater heater transient resulted in a gradual increase in power (over ~300 seconds) which RETRAN-02 predicted to within ~10% and provides validation of the neutronics temperature feedback models at close to rated conditions. The recirculation pump trip tests result in a substantial reduction in core power which RETRAN-02 predicted to within ~5%. These tests provide validation of the RETRAN-02 calculation of the pump coastdown and system response.

PP&L has also provided a RETRAN-02 benchmark comparison for a Susquehanna-1 generator load rejection event at the end of Cycle-1. This was a rapid pressurization event similar to (although much milder than) the licensing basis overpressurization transient. The

RETRAN-02 calculated power increase of 29% compares reasonably well with the measured power increase of 34%.

The Peach Bottom-2 turbine trip tests were performed to provide benchmark data for BWR transient analyses. In fact, these transients are similar to the BWR licensing basis overpressurization transient. The three tests (TT1, TT2, and TT3) were performed at close to rated flow (81 to 91%) and over a range of core powers (47 to 69% of rated). The PP&L simulation of these transients is based on the same codes and best estimate methods used in the analysis of the Susquehanna units. The comparison of the RETRAN-02 calculated core power indicates a conservative overprediction of TT1 and good agreement for the TT2 and TT3 tests. The core pressure increase calculated by RETRAN-02 for all tests agreed with the measured values to within $<10\%$.

The licensing basis transient consists of a turbine trip without bypass from 104.5% power and rated flow. The initial conditions are for the Peach Bottom-2 end-of-Cycle-2 statepoint. The PP&L evaluation of this transient was based on the standard methods and included a Cycle-1 and Cycle-2 depletion and neutronics feedback calculation with SIMULATE-E. The initial axial power shape compares well with the GE and BNL results. The transient peak power was conservatively overpredicted by RETRAN-02 by $\sim30\%$.

## 2.3 Susquehanna Model Code Uncertainty

In order to estimate the RETRAN-02 calculational uncertainty in predicting the fractional change in critical power ratio, RCPR, PP&L has made calculation-to-measurement comparisons for the three Peach Bottom-2 turbine trip tests. The comparisons assume the fuel is similar to

ANF 9x9 fuel, and are not valid for fuel bundles that are significantly different. Based on these comparisons a 95/95 upper bound of 26.2% for internal rods and 32.2% for peripheral rods is determined for RETRAN-02 calculations of RCPR.

## 3.0 TECHNICAL EVALUATION

The PP&L BWR transient analysis methods qualification topical report describes the Susquehanna RETRAN-02 model and the benchmarking comparisons used to validate the model for reload licensing applications. The initial review of the topical report resulted in a request for additional information (RAI) which was transmitted to PP&L in Reference-11. This evaluation included both the model and benchmarking description included in the report, as well as the PP&L response to the RAI provided in References 12 and 13. The major issues raised during this review are summarized in the following.

### 3.1 Susquehanna RETRAN-02 Model

The core neutronics statepoint and transient feedback data are determined by a three-dimensional SIMULATE-E core calculation. The licensing transients are generally sensitive to both the initial statepoint conditions and precalculated reactivity feedback coefficients. In Response-1 (Reference-12) PP&L has indicated that a SIMULATE-E calculation is performed for each initial statepoint exposure distribution, power level, rod pattern and core flow, and that the one-dimensional RETRAN-02 cross-sections and feedback will include this detailed statepoint dependence.

The SIMULATE-E/SIMTRAN-E one-dimensional cross-sections include an adjustment to account for differences between the SIMULATE-E three-dimensional thermal hydraulics and the RETRAN-02 one-dimensional average channel thermal hydraulics. The adjustment is only required for the overpressurization transients: generator load rejection and feedwater controller failure events. For these applications the adjustment has been validated by the RETRAN-02 comparisons to the Peach Bottom-2 turbine trip tests and to the LBT calculations. The calculation of peak pressures in licensing analyses is performed without the adjustment which results in a conservative overprediction of the limiting pressure (Response-3, Reference-12).

The fuel rod gap conductance used to determine core response in the RETRAN-02 model is calculated with ESCORE using a core-average power history and axial power shape. In order to provide a bounding hot-bundle calculation, a separate conservative gap conductance is used for each potentially limiting hot-bundle fuel type (Response-26, Reference-12). The ESCORE calculation requires a resonance escape probability (REP) to determine the fuel rod parameters. PP&L uses a high value for REP to insure a conservatively high gap conductance. (Response 30, Reference-12).

The PP&L application of the RETRAN-02 model is consistent with the limitations of the RETRAN-02 SER (Response-31). For the relevant portions of the benchmarking calculations and the licensing basis transients, only pre-CHF heat transfer is required, the jet pump flow remains in the forward direction and a dominant flow direction exists in the volumes where the temperature delay model is used. As required, the upper downcomer volume will neither be completely full or empty due to a water level scram. PP&L executes a 10-second null transient to insure proper model initialization, and time step sensitivity calculations have been performed to demonstrate that the time step selection is adequate. If future licensing analyses require the application of the Susquehanna model outside the limitations of the RETRAN-02 SER additional justification will be provided.

The PP&L hot-bundle calculation for pressurization transients includes several modeling assumptions which result in an overprediction of the transient RCPR. The time-dependence of the radial bundle power used in the CPR calculation is assumed to be the same as the core thermal power (Response-22, Reference-12). However, since the limiting bundles at EOC are typically more bottom-peaked than the core-average axial power distribution, the scram in the limiting locations occurs earlier in the transient and the relative bundle power increase is less than that inferred from the core thermal power. This approximation results in an overprediction of the transient RCPR.

In the hot-bundle RCPR calculation the time-dependence of the axial power distribution is neglected. The hot-bundle axial power shape is taken to be the same as the initial core-average axial power distribution and independent of time. The void collapse and rod insertion during the generator load rejection without bypass (GLRWOB) and feedwater controller failure (FWCF) transients result in a shift of the axial power distribution toward the top of the core. To account for the neglect of the time-dependence of the axial power shape the PP&L methodology employs a hot-bundle gap conductance determined assuming a conservative fuel rod power history. In Reference-13, PP&L has evaluated the adequacy of this approach by reanalyzing both the GLRWOB and FWCF transients using a more realistic hot-bundle gap conductance together with a time-dependent axial power shape. The more realistic gap conductance was determined for Susquehanna-1 Cycle-7, using a SIMULATE-E cycle step-out analysis in which the hot-bundle power history was determined. Using the realistic fuel rod power history determined with SIMULATE-E (rather than the conservative power history) reduced the gap conductance from 1462 to 924 BTU/hr-ft$^2$-°F. PP&L has indicated that this reduced conductance is conservative relative to the value calculated using NRC approved methods.

The Reference-13 analysis indicates that for Susquehanna-1 Cycle-7, the PP&L methodology predicts GLRWOB and FWCF transient RCPRs that are equal to, or larger than, those predicted by the more realistic analyses using a time-dependent axial power shape. We therefore conclude that the PP&L hot-bundle model is acceptable for Susquehanna-1 Cycle-7 GLRWOB and FWCF transient RCPR calculations. In applications of the methodology to future reload cores, the conservatism in the hot-bundle gap conductance must be shown to be sufficient to compensate for the neglect of the time-dependence in the hot-bundle axial power shape.

## 3.2 Susquehanna Model Benchmarking

The Susquehanna model benchmarking comparisons to the startup and turbine trip test data and to the LBT calculations provide the validation of the RETRAN-02 model and procedures. PP&L has indicated that the model described in PL-NF-89-OO5 is based on best estimate input and procedures, rather than the conservative methods that will be used in the licensing analyses described in the applications Topical Report PL-NF-90-001. These benchmark comparisons will therefore allow the determination of the code/model calculational uncertainty. PP&L has indicated that the methods and procedures used in these benchmark comparisons are the same as will be used in the Susquehanna licensing analyses, except for conservatism that will be added in PL-NF-90-001. In particular, the moderator density adjustment to the SIMULATE-E/SIMTRAN-E cross-sections was only made for the Peach Bottom-2 turbine trip tests and the LBT calculation, where the moderator density effects are significant. This is consistent with the licensing application, since the adjustment will only be made to the generator load rejection and the feedwater controller failure overpressurization events (Responses 3 and 20, Reference-12).

The PP&L calculations of the Peach Bottom-2 turbine trip tests indicate generally good agreement between the RETRAN-02 predictions and measurements. However, the comparisons for the TT1 test indicate a 24% overprediction of the peak core power and an overprediction of the transient ΔCPR by 14% in the TT3 test. PP&L attributes the overprediction of the TT1 power to conservatism in the prediction of the increase in core pressure and to uncertainties in the time of the turbine trip (Response-18, Reference-12). Since these overpredictions are in the conservative direction and result in larger transient ΔCPRs, they are acceptable.

Based on the evaluation of the Susquehanna model and procedures and the benchmarking comparisons, it is concluded that the Susquehanna RETRAN-02 model is acceptable.

### 3.3 Susquehanna Model Code Uncertainty

The Susquehanna model code uncertainty was determined by comparing the predicted transient ΔCPR with values inferred from the Peach Bottom-2 measurements. There are significant differences between the Peach Bottom-2 and Susquehanna steam lines and in order to insure a consistent comparison, a special RETRAN-02 calculation, in which the measured dome pressure was imposed as an external boundary condition, was used for predicting the transient ΔCPR. As a result, the calculation-to-measurement ΔCPR differences do not include the effect of the uncertainty in the steamline modeling. In Response-13 (Reference-12) PP&L

has indicated that these uncertainties have been evaluated and will be applied in licensing calculations as described in the applications Topical Report PL-NF-90-001.

The hot-bundle calculation used in the uncertainty analysis assumes ANF 9x9 fuel, and the CPR calculation is carried out with the XN-3 correlation. If a significantly different fuel type is used in a future Susquehanna reload a new code uncertainty will be required.

The Peach Bottom-2 ΔCPR comparisons to measurement indicate a substantial ~9% conservative overprediction of the transient RCPR. This average bias is based on three calculation-to-measurement differences ranging from ~3% to 14%. The topical report does not provide any discussion of the uncertainty in the prediction of the peak transient pressure. In Response-11 (Reference-12) PP&L has indicated that the measured and calculated peak dome pressures for the three Peach Bottom-2 tests agreed to within ~5 psi. In addition, a ~57 psi conservatism is included in the licensing overpressure analysis described in PL-NF-90-001.

With the limitations discussed above it is concluded that the Susquehanna model uncertainty analysis is acceptable.

## 4.0 TECHNICAL POSITION

The PP&L transient methods Topical Report PL-NF-89-005 and supporting documentation provided in the PP&L responses of Reference-12 and Reference-13 have been reviewed in detail. The topical report provides the description of the core and system model to be used in the transient analyses of the Susquehanna Units 1 and 2, the code/model validation, and an uncertainty analysis for the prediction of transient ΔCPR. Based on this review it is concluded that the PP&L transient methods and uncertainty estimates are acceptable for Susquehanna reload licensing analyses under the conditions stated in Section-3 of the evaluation and summarized in the following.

(1) RETRAN-02 Model Limitations

If future licensing analyses result in conditions that are outside the RETRAN-02 model limitations, as specified in the RETRAN-02 SER, additional model justification will be required (Section-3.1).

(2) Application to New Fuel Designs

The uncertainty estimates, $E_{95/95}$ upper tolerance factors, and hot-bundle ΔCPR calculation are based on the assumption that the core is loaded with ANF 9x9 fuel. Consequently, the methodology and results are acceptable for cores loaded with ANF 9x9 or similar fuel. If a significantly different fuel type is introduced in a future Susquehanna reload, the methods will require further justification and a new ΔCPR uncertainty estimate will be required (Sections 3.1 and 3.3).

(3) Hot-Bundle Fuel Rod Gap Conductance

In applications of the transient methodology to reload cores other than Susquehanna-1 Cycle-7, the conservatism in the fuel rod gap conductance must be shown to be sufficient to compensate for the neglect of the time-dependence in the hot-bundle axial power shape (Section-3.1).

## REFERENCES


1. "Submittal of Topical Report PL-NF-89-005", Letter, Keiser, H.W. (PP&L) to Butler, W.R. (NRC), dated January 22, 1990.

2. "SIMTRAN-E: A SIMULATE-E to RETRAN-02 Data Link", EPRI NP-5509-CCM, December 1987.

3. "ESCORE - The EPRI Steady-State Core Reload Evaluator Code: General Description", EPRI-NP-5100, February 1987.

4. "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A Revisions 2 and 3 Regarding RETRAN-02/MOD-003 and MOD-OO4", Letter from A.C. Thadani (NRC) to R. Furia (GPU Nuclear), October 19, 1988.

5. "Qualification of Steady-State Core Physics Methods for BWR Design and Analysis", PL-NF-87-001-A, Pennsylvania Power & Light, July 1988.

6. "XN-3 Critical Power Correlation", XN-NF-512-P-A, Revision 1, October 1982.

7. "Application of Reactor Analysis Methods for BWR Design and Analysis", PL-NF-90-001, Pennsylvania Power and Light Company, August 1990.

8. "Susquehanna Steam Electric Station Units 1 & 2, Final Safety Analysis Report", Revision 41, May 1989.

9. "NRC Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P, Volumes I, II, and III", June 1980.

10. "Analysis of Licensing Basis Transients for a BWR/4", BNL-NUREG-26684, September 1979.

11. "Request for Additional Information-Susquehanna Steam Electric Stations, Units 1 and 2 (TAC Nos. 75999/76000)", Letter, M.C. Thadani (NRC) to H.W. Keiser (PP&L), dated February 15, 1991.

12. "Susquehanna Steam Electric Station, Response to RAI on PL-NF-89-005", Letter, H.W. Keiser (PP&L) to Director of NRR, dated March 13, 1991.

13. "Susquehanna Steam Electric Station, Response to RAI on Transient Analysis Methods", Letter, H. W. Keiser (PP&L) to Director of NRR, dated February 12, 1992.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 21, 1991

Docket Nos. 50-387
and 50-388

Mr. Harold W. Keiser
Senior Vice President - Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: TOPICAL REPORT PL-NF-90-001, "APPLICATION OF REACTOR ANALYSIS METHODS FOR BWR DESIGN AND ANALYSIS," SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2, (TAC NOS. M75999 AND M76000)

By letters dated August 8, 1990 and August 29, 1991, Pennylvania Power and Light Company, requested NRC review and approval of Topical Report PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis" for Susquehanna Steam Electric Station (SSES), Units 1 and 2. Based upon the staff's review, as given in the enclosed Safety Evaluation, we find the application of PL-NF-90-001 acceptable for use in reload analyses for the SSES, Units 1 and 2 under the limitations delineated in the associated technical evaluation.

This completes the staff effort on this issue and closed TAC NOs. M75999 and M76000.

Sincerely,

James J. Raleigh

James J. Raleigh, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT PL-NF-90-001
"APPLICATION OF REACTOR ANALYSIS METHODS FOR BWR DESIGN AND ANALYSIS"
PENNSYLVANIA POWER & LIGHT COMPANY
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2
DOCKET NOS. 50-387 AND 50-388

## 1.0 INTRODUCTION

By letter from H. W. Keiser to W. R. Butler (NRC), dated August 8, 1990, Pennsylvania Power and Light Company (PP&L) submitted topical report PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis," for NRC review. These methods will be used to determine the Susquehanna 1 and 2 operating limit minimum critical power ratio, demonstrate compliance with the ASME overpressurization criteria and provide physics input to fuel vendor reload safety analyses.

The NRC staff was supported in this review by our consultant, Brookhaven National Laboratory. The staff has adopted the findings recommended in our consultant's technical evaluation report (TER) which is attached.

## 2.0 EVALUATION

The attached TER provides the evaluation.

## 3.0 CONCLUSIONS

The staff has reviewed the PP&L application topical report PL-NF-90-001 and the supporting documentation provided in response to our request for additional information. Based on this review, the staff concludes that the proposed statistical combination of uncertainties (SCU) method is not acceptable for the reasons stated in TER Sections 3.1.2 and 3.2.1. Instead, the alternate method proposed in the August 29, 1991, letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), PLA-3641, "Susquehanna Steam Electric Station Licensing Methods: Plan for U1C7," should be used to determine the operating limit minimum critical power ratio for the rod withdrawal error, the generator load rejection without bypass, and the feedwater controller failure events. In addition, the presently approved POWERPLEX power distribution uncertainties should be retained and should not be reduced.

Attachment:
Technical Evaluation Report

Principal Contributor:
J. Carew

Date:

# ATTACHMENT

## TECHNICAL EVALUATION REPORT

| | |
|---|---|
| Topical Report Title: | Application of Reactor Analysis Methods for BWR Design and Analysis |
| Topical Report Number: | PL-NF-90-001 |
| Report Issue Date: | August 1990 |
| Originating Organization: | Pennsylvania Power and Light Company |

## 1.0 INTRODUCTION

In Reference-1, the Pennsylvania Power and Light (PP&L) Company has submitted the PP&L BWR reactor design and analysis methods. These methods are based on the PP&L transient analysis methods described in the PL-NF-89-005 topical report (Reference-2), and the PP&L steady-state core physics methods given in the PL-NF-87-001 topical report (Reference-3). While these steady-state physics methods and reactor transient methods provide best-estimate predictions, the proposed reactor applications methodology generally includes conservative adjustments to insure the required margin to fuel thermal and mechanical limits and to system performance criteria. The proposed reactor analysis methods are intended for application to reload licensing evaluations for the Susquehanna Units 1 and 2.

The PL-NF-90-001 methods will be used to determine the Susquehanna 1 and 2 operating limit minimum critical power ratio (OLMCPR), demonstrate compliance with the ASME overpressurization criteria and provide physics input to fuel vendor reload safety analyses. The primary codes used in the PP&L methodology are the CPM-2 lattice physics code (Reference-4), the SIMULATE-E (Reference-5) three dimensional steady-state core analysis code, the

RETRAN-02 (Reference-6) systems transient code, and the ESCORE (Reference-7) fuel performance code. The Siemens Nuclear Power Corporation -SNPC (formerly Advanced Nuclear Fuels) POWERPLEX code (Reference-8) is used for on-line core monitoring.

For typical Susquehanna 1 and 2 reload cores the potentially limiting events are identified as the fuel bundle misloading error (FBME), loss of feedwater heating (LFWH), rod withdrawal error (RWE), generator load rejection without bypass (GLRWOB), feedwater controller failure (FWCF), recirculation flow controller failure (RFCF) and the main steam isolation valve closure (MSIV) events. The reload application of the PL-NF-90-001 methods will typically be limited to the analysis of these transients. The topical report includes a sensitivity analysis of each of these events to key input parameters, as well as a detailed sample licensing analysis. The proposed methods include a statistical combination of uncertainty (SCU) approach in which the CPR monitoring uncertainties are statistically combined with the transient ΔCPR calculational uncertainties to determine an OLMCPR. This method is applied to the GLRWOB and FWCF transients, and to the analysis of the rod withdrawal event.

This review focused on the degree of conservatism included in the PL-NF-90-001 licensing methodology and the adequacy of the steady-state and transient methods for the specific events being analyzed. The review included two meetings and extensive discussions with PP&L, and a detailed review of the topical report and PP&L response to the request for additional information. The PP&L licensing methodology is summarized in the following section, the technical evaluation is given in Section-3, and the technical position is given in Section-4.

## 2.0 SUMMARY OF THE TOPICAL REPORT

The PP&L reactor analysis methodology includes both steady-state and transient analyses. The steady-state methods are applied to licensing events in which the final steady-state is limiting or the event is sufficiently slow that quasi-static methods apply. The transient methods are used to calculate RCPR (defined as ΔCPR/ICPR where ICPR is the initial CPR) for anticipated operational occurrences (AOOs), and for the MSIV overpressurization analysis. The Susquehanna FSAR AOOs are evaluated and the GLRWOB, FWCF and RFCF events are shown to be the limiting transients for determining the OLMCPR.

### 2.1 Steady-State Analyses

#### 2.1.1. Rod Withdrawal Error

The rod withdrawal event results from the erroneous selection and withdrawal of a control rod with a neighboring bundle at the MCPR operating limit. This rod withdrawal introduces a substantial amount of positive reactivity and causes a large increase in the bundle power and a reduction in MCPR. The transient RCPR is determined by the (flow-biased) rod block monitor (RBM) setpoint.

The RWE event is considered a quasi-static event and is calculated with SIMULATE-E. The reduction in MCPR, RCPR, and the associated LPRM responses are calculated with SIMULATE-E (for a particular selected rod) and are input to the PP&L program RBM which calculates the RCPR as a function of RBM setpoint. The RBM setpoint uncertainty, RBM response and RCPR calculational uncertainty and LPRM failure probability are combined statistically with RBMSTAT to determine the RCPR distribution resulting from an RWE. Using the statistical combination of uncertainties approach STATOL combines the RWE RCPR distribution with the standard safety limit uncertainties to determine an OLMCPR.

In order to insure that all the important features are correctly modeled in the RWE analysis, PP&L has performed a series of sensitivity calculations. These analyses indicate that the RBM setpoint for the RWE is relatively sensitive to the control rod pattern, error rod location, and the assumed LPRM failure rate and location.

### 2.1.2 Fuel Loading Error

Both the misloading of a fuel bundle into an incorrect core location and the rotation of a fuel bundle (by 90 or 180°) in its intended location are analyzed as part of the Susquehanna Units 1 and 2 reload evaluation. The fuel mislocation analysis is performed with SIMULATE-E and the largest RCPR is determined considering all potentially limiting mislocations and all cycle exposure points. An uncertainty allowance for the SIMULATE-E calculated RCPR is included. The Susquehanna Units are C-lattice plants (i.e., with equal water gaps) and, consequently, the increase in RCPR resulting from a rotated bundle is relatively small. The RCPR resulting from a rotated fuel bundle is determined with SIMULATE-E. A worst-case analysis has been performed which is expected to bound future reloads, and the applicability of the bounding analysis to a specific reload bundle design will be determined by comparing CPM-2 calculated peaking factors and S-factors.

### 2.1.3 Loss of Feedwater Heating

The PP&L methodology treats the loss of feedwater heating event as quasi-static and calculates the transient RCPR using SIMULATE-E. A bounding feedwater temperature decrease of 100°F together with a 5 psi pressure increase is assumed in the LFWH analysis. Sensitivity calculations for changes in pressure, rod pattern, cycle exposure and assembly reactivity were performed and used to determine a generic 95/95 transient RCPR correlation. This correlation will be verified and applied in reload licensing evaluations of the LFWH event.

### 2.1.4 Core Physics Parameters

Core physics parameters are required for the reload evaluation to demonstrate compliance with the technical specifications, provide nuclear cross-section input to RETRAN transient analyses, and to provide input to accident analyses which SNPC will perform (LOCA, safety limit MCPR, control rod drop, and fuel storage criticality analyses). These steady-state parameters are calculated with CPM-2 and SIMULATE-E. CPM-2 is used to calculate the pin-wise local peaking factors for the SNPC LOCA and SLMCPR analyses, and the data required for the POWERPLEX core monitoring system. SIMULATE-E is used to calculate (1) the core reactivity for shutdown and standby liquid control system analyses, (2) the scram and dropped rod reactivities and (3) the feedback coefficients for LOCA analyses. SIMTRAN-E (Reference-9) uses the SIMULATE-E radial flux solution to collapse the cross-sections to one-dimension for input to RETRAN-02. PP&L has performed sensitivity calculations for the SIMULATE-E analyses, and either determines conservative physics parameters or includes an explicit 95/95 uncertainty allowance.

## 2.2 Transient Analyses

### 2.2.1 Generator Load Rejection Without Bypass

The generator load rejection without bypass is calculated with the Reference-2 RETRAN-02 core and hot-bundle models. The calculation is performed for the case in which the transient is initiated by a fast turbine control valve closure, and for the case in which both the turbine control valves (TCV) and turbine stop valves (TSV) close. Conservative assumptions based on PP&L GLRWOB sensitivity studies are employed. These include neglect of the end-of-cycle (EOC) recirculation pump trip (EOC-RPT), technical specification mode operation of the safety relief valves and EOC all-rods-out conditions.

When the SCU method is not used in the GLRWOB analysis, the initial core power is conservatively increased to 104.4% (of rated) and the Reference-2 values of the 95/95 upper tolerance limits on RCPR are used to account for calculational uncertainties. When the SCU method is used, the effect of a 2% standard deviation in core power is combined with the effect of a 0.2 ft/sec (plant-specific) standard deviation in scram time using a two-dimensional RCPR response surface. The resulting RCPR variation is then combined with the Reference-2 code uncertainty and POWERPLEX MCPR monitoring uncertainties using a Monte Carlo approach. This statistical combination of uncertainties method yields a OLMCPR of 1.30 for the GLRWOB for Susquehanna Unit-2, Cycle-2 (U2C2).

### 2.2.2 Feedwater Controller Failure

The FWCF is analyzed at EOC with all-rods-out and with a maximum allowed flow rate failure. The event is analyzed with either the turbine bypass or EOC-RPT inoperable, and may be analyzed at earlier cycle exposures if exposure-dependent OLMCPR limits are required. PP&L has performed a series of sensitivity calculations to quantify the effect of the important transient parameters. Based on these studies a conservative methodology has been determined and assumes: (1) the technical specification minimum scram insertion rate, (2) 100% (of rated) core flow and (3) 85% of the best-estimate TCV closure time.

The steam line uncertainties are combined with the RCPR calculational uncertainty of Reference-2 and a 95/95 upper tolerance limit on the transient RCPR is determined. The SCU method may also be applied to the FWCF event and, in this case, a RCPR response surface will be constructed and used to determine the OLMCPR in a manner similar to that used for the GLRWOB transient.

### 2.2.3 Recirculation Flow Controller Failure

The recirculation flow controller failure event results in increased core flow and is a potentially limiting MCPR event. PP&L has evaluated both the master controller and single loop controller failure events, and has determined the master controller failure to be limiting. The licensing calculations are performed on the 100% rod line since these have a larger RCPR and bound the lower powered (higher MCPR) statepoints. An event-specific RCPR uncertainty has been determined, based on the doppler and void coefficient uncertainties, and is applied to the calculated RFCF RCPR. A limiting flow run-up rate will be calculated for the event and used in the Susquehanna reload licensing analyses.

### 2.2.4 Overpressurization Analysis

PP&L has evaluated both the GLRWOB and MSIV closure transients, and has determined that the MSIV closure is the limiting overpressurization event. The MSIV transient is analyzed with the RETRAN-02 model of Reference-2 with improved MSIV and safety/relief valve models. The analysis assumes that (1) the relief mode actuation of the SRVs is inoperable, (2) the six inoperable SRVs have the lowest pressure setpoints and (3) the SRVs have maximum opening times.

Sensitivity calculations were performed and indicate that the most important parameters are the initial core power, control rod insertion rate and the MSIV closure time. In the PP&L analysis the core power is taken to be 104.4% (of rated) and the MSIV closure and control rod insertion time are based on the plant technical specifications.

## 3.0 TECHNICAL EVALUATION

The reactor analysis methods Topical Report PL-NF-90-001 describes the methods that will be employed in the PP&L reload licensing evaluations for Susquehanna Units 1 and 2. The initial review of this report resulted in a request for additional information (RAI) which was transmitted to PP&L in References 10-11. This review included an evaluation of the proposed licensing analysis methods described in the topical report, as well as the PP&L responses to the RAI included in References 12-14. The major issues and concerns raised during the review are summarized in the following sections.

## 3.1 Steady-State Analyses

### 3.1.1 Rod Withdrawal Error

In the analysis of the RWE event, the maximum RCPR results when the selected rod yields the worst combination of control rod worth and RBM response. Consequently, the limiting control rod location depends on the cycle-specific core loading. For a given core reload, PP&L calculates the RWE transient RCPR for all control rods within a 5x5 control-cell region in the center of the core. PP&L has evaluated the control rod locations outside this central region, including control rods having only two or three LPRM strings available, and has determined that these locations are not limiting. In Response-6 (Reference-13) PP&L indicates that in this procedure the worst-case combination of rod worth and RBM response yielding the limiting transient RCPR is determined.

### 3.1.2 Application of the SCU Method to the RWE Event

In the application of the SCU method to the RWE, the POWERPLEX safety limit monitoring uncertainties[1] and the rod block monitor (RBM) response uncertainties are considered to be independent and are combined statistically to determine the MCPR operating limit. Since the POWERPLEX and RBM systems use the same LPRM input and make use of similar neutronics solutions, the uncertainties associated with these systems are not believed to be independent and, therefore, should not be combined using the proposed SCU method. In

[1] The POWERPLEX safety limit uncertainties are the POWERPLEX monitoring uncertainties (e.g., on bundle power) that are used in the statistical determination of the CPR safety limit.

addition, in the SCU approach the ΔCPR resulting from an RWE is calculated as a statistical average over all allowable (within technical specifications) LPRM failure states. Consequently, the calculated average ΔCPR is conservative for reactor states with a small number of LPRM failures and non-conservative for reactor states with many LPRM failures. This approach, therefore, does not provide protection for reactor states (with many LPRM failures) that are expected during normal operation and is considered unacceptable.

In Reference-14, PP&L has provided an alternate method for determining the transient ΔCPR for the RWE event. This method assumes the worst-case combination of LPRM detector failures together with the worst-case RBM channel failure, and is consistent with NRC approved methods. The proposed PP&L alternate method of Reference-14 for determining the ΔCPR resulting from an RWE event is therefore acceptable.

### 3.1.3 Fuel Loading Error

The RCPR resulting from a fuel bundle mislocation depends on the specific core location and the insertion of the neighboring control rods. In the calculation of the limiting RCPR, PP&L assumes that all control rods are withdrawn. In order to evaluate this approximation PP&L has calculated ninety-three combinations of misloading location and control rod pattern and finds only a slight (0.0017) underprediction of the limiting RCPR (Response-11, Reference-13). This underprediction is considered to be negligible. In order to account for the variation due to fuel bundle location and control rod pattern, a 95/95 upper tolerance factor is applied to the RCPR calculation for the fuel bundle mislocation.

### 3.1.4 Loss of Feedwater Heating

The transient RCPR resulting from a loss of feedwater heating event depends on the time-dependence of the core power, pressure and flow. PP&L has measured these variables during a loss of feedwater heating transient and has found that the core pressure and flow are essentially unchanged during the transient while the core power increased monotonically. This confirms the PP&L assumption that the final statepoint is RCPR limiting during the LFWH event.

The local power and linear heat generation rate (LHGR) increase during the LFWH transient due to the increased core power and axial (bottom) peaking. This increase is $\leq$ 20% and is bounded by the generic LHGR transients.

### 3.1.5 Core Physics Parameters

In the evaluation of the shutdown capability of the standby liquid control system, one of two methods are used to determine the boron reactivity worth. The first approximate method includes a substantial margin of conservatism and typically overpredicts the boron-worth by a factor of two. In the second method the cross-sections are adjusted to include the dependence on the soluble boron. In certain cases, the conservatism of this cross-section adjustment procedure is small (~5%). In order to provide additional conservatism in this calculation, PP&L has indicated (Response-18, Reference-13) that an additional 0.01 $\Delta k$ uncertainty allowance will be included when this method is used to determine boron reactivity worth.

The core shutdown margin is determined assuming the highest worth rod does not insert. The control rod worth depends on the specific core loading and the location of the stuck rod, and, to reduce the number of calculations required to identify the strongest rod, an approximate calculation is performed with RODDK-E (Reference-15). The uncertainty introduced by this approximation was determined by comparing RODDK-E to SIMULATE-E. Seventy comparisons were made, including variations in core loading, cycle exposure and control and void histories, and indicated a rod worth discrepancy of less than 0.002 $\Delta k$ (Response-5, Reference-13).

The SNPC LOCA analyses require reload-specific limiting core void and doppler feedback reactivities. In Response-19 (Reference-13) PP&L has indicated that the cycle-dependent variations in the feedback reactivities are small compared to the EOC reduction in scram reactivity, and the EOC statepoint is limiting. RETRAN-02 transient LOCA calculations at BOC, MOC and EOC were performed and demonstrated that EOC is limiting. PP&L intends to provide the core physics input for the SNPC reload LOCA, SLMCPR, control rod drop, and fuel storage criticality analyses. SNPC has indicated (Response-7, Reference-13) that the input data provided is determined in a manner consistent with the approved SNPC methods and uncertainty treatment.

PP&L intends to use CPM-2 rather than the SNPC XFYRE lattice physics code to determine the neutronics input data for the POWERPLEX core monitoring system. In order to determine the effect of this neutronics data change on the POWERPLEX power distribution uncertainties, PP&L has compared POWERPLEX predicted and measured TIP responses using both the CPM-2 and XFYRE data. These comparisons indicate that POWERPLEX/CPM-2 provides comparable or improved agreement with the TIP measurements, relative to POWERPLEX/XFYRE. PP&L proposed in Response-22 of Reference-13 to use reduced POWERPLEX/CPM-2 power distribution uncertainties, inferred from these comparisons, in the

SLMCPR analyses. However, in the thermal margin licensing basis, the adequacy of all the SLMCPR uncertainties (including bundle power, feedwater flow and temperature, core flow, etc.) has been demonstrated as a group, and any reduction in an individual uncertainty will invalidate the set of uncertainties. It is, therefore, concluded that the SLMCPR POWERPLEX power distribution uncertainties should not be reduced (as proposed in Response-22), but should remain at their presently approved values of References 16 and 17 as originally proposed in Section-2.9.2 of the topical report.

Based on the above and the information provided in References 12-14, it is concluded that the steady-state analyses are acceptable with the limitations indicated in Section-3.1.1 concerning the SCU method, and in Section-3.1.5 concerning the POWERPLEX SLMCPR uncertainties.

## 3.2 Transient Analyses

### 3.2.1. Application of the SCU Method to the GLRWOB and FWCF Events

The PL-NF-90-001 method for the analysis of the generator load rejection without bypass and feedwater controller failure events employs the SCU statistical combination of uncertainties method for including the calculational uncertainties in the transient ΔCPR. In this approach the transient ΔCPR calculational uncertainties are statistically combined with the POWERPLEX safety limit measurement uncertainties. This SCU approach differs from presently approved methods and results in a substantial nonconservative reduction in CPR margin relative to the approved methods where the uncertainties are added separately. The application of the SCU method to the GLRWOB and FWCF transients requires the mean, standard deviation and the distribution of the calculation uncertainty in the ΔCPR that occurs during the transient. The mean and standard deviation are determined with three calculation-to-measurement data points derived from the Peach Bottom-2 turbine trip tests. It is noteworthy that in typical statistical analyses the mean and standard deviation are determined from a relatively large data set. The minimum number of data points required to determine a mean and standard deviation is three (3) and, therefore, in the present application a data base with the minimum number of data points is used. Since the available data is not sufficient to determine the ΔCPR uncertainty distribution, the ΔCPR are taken to be distributed normally about the mean value ΔCPR. This assumption is important since the MCPR operating limit is sensitive to the details of the statistical representation used for the uncertainty in ΔCPR. It is concluded that the three Peach Bottom-2 data points and the normality assumption, when used to combine the safety limit monitoring and transient ΔCPR calculational uncertainties using the proposed SCU method, do not provide the high confidence required to protect the specified acceptable fuel design limits.

In the PP&L application of the SCU method, the value of the safety limit MCPR is not calculated as part of the determination of the operating limit MCPR. The value of the operating limit MCPR is determined by the condition that 99.9% of the fuel rods are not expected to experience boiling transition. PP&L intends to use this condition as the safety limit in the Susquehanna Units 1 and 2 technical specifications. This definition of the safety limit, as a condition, does not conform to the technical specification requirements of 10 CFR 50.36 (Reference-18) which states that the safety limits be "limits upon important process variables."

In Reference-14, PP&L has provided an alternate method for including the uncertainties in the calculated transient ΔCPR for the GLRWOB and the FWCF transients and determining the OLMCPR. The proposed method is the same as the Siemens Nuclear Power Corporation approach which has been approved by the NRC. In this method the ΔCPR uncertainties are accommodated by increasing the transient integral power calculated by RETRAN-02 by 10%. The transient ΔCPR resulting from this calculation is added algebraically to the SLMCPR to determine the OLMCPR. The SLMCPR is determined using the approved SNPC uncertainties and statistical methods. This method provides adequate allowance for the transient ΔCPR calculation uncertainties and is based on NRC approved methods. The proposed alternate method is therefore acceptable for calculating the OLMCPR for the GLRWOB and FWCF events.

In Reference-14 PP&L has indicated that certain cycle-specific evaluations will be performed. The initial power used in the GLRWOB will be determined for each reload cycle via a parametric evaluation. The GLRWOB is identified as the limiting ΔCPR transient resulting from core pressurization, however, PP&L has indicated (Response-1, Reference-14) that it will also evaluate the turbine trip without bypass for each cycle to determine the maximum transient ΔCPR.

### 3.2.2 Recirculation Flow Controller Failure

The recirculation flow controller failure is identified as one of the three limiting transient ΔCPR events. The event involves a two pump runup which results in an increase in core power and a reduction in MCPR. When the transient is initiated from a high power rod-line, the maximum combined (steam) flow limit (MCFL) may be reached (depending on the MCFL setting) resulting in a loss of pressure control. However, the resulting pressure increase is slow and the moderator density changes are relatively small. While the cross-section moderator density correction is not included in the calculation of this transient, it is indicated in Response-5 (Reference-14) that the neglect of this correction results in a conservative overprediction of ΔCPR.

The RFCF event is relatively slow ($\lesssim$ 100 seconds) and the ΔCPR uncertainties of Reference-2 (for overpressurization transients) are not applicable. PP&L has used Doppler and void reactivity uncertainties of 20% and 50%, respectively, and a flow runup rate which gives the maximum increase in ΔCPR. In addition, in Response-9 (Reference-14) PP&L has indicated that a conservative gap conductance corresponding to a value of 125% of rated power will be used.

### 3.2.3 Overpressurization Analysis

PP&L has performed extensive sensitivity calculations for the MSIV closure overpressurization transient which indicate that the most important parameters are the core power, scram reactivity and MSIV closure time. Conservative values of these input are assumed; (1) the core power is taken to be 104.4% (of rated), (2) the technical specification (maximum) scram time is assumed and (3) the technical specification minimum MSIV closure time is assumed. PP&L uses a flat axial power shape to determine the core average gap conductance, since this results in a minimum gap conductance and maximum transient pressure. The moderator density correction is not included in the neutronic cross-sections used in the

MSIV analysis, since this also results in a conservative overprediction of the transient pressure (Response-7, Reference-14). PP&L intends to perform a MSIV ovepressurization analysis for each reload cycle.

Based on the above and the information provided in References 12-14, it is concluded that the transient methods are acceptable with the limitation indicated in Section-3.2.1 concerning the SCU methods.

## 4.0 TECHNICAL POSITION

The PP&L methods described in PL-NF-90-001 and the additional information provided in References 12-14 has been reviewed in detail. The proposed methods are intended for the analysis of the limiting ΔCPR quasi-static and transient analyses, the overpressurization analysis, and for providing input to the SNPC reload LOCA, criticality and SLMCPR analyses. Based on this review it is concluded that the PP&L methods are acceptable for performing reload licensing analyses for Susquehanna Units 1 and 2, subject to the conditions given in Section-3 of this evaluation and summarized in the following.

(a) The proposed statistical combination of uncertainties SCU method is not approved (Sections-3.1.2 and 3.2.1). Instead, the alternate method proposed in Reference-14 should be used to determine the OLMCPR for the rod withdrawal error, the generator load rejection without bypass, and the feedwater controller failure events (Sections-3.1.1 and 3.2.1).

b) The presently approved POWERPLEX power distribution uncertainties given in References 16 and 17 should be used in the SNPC SLMCPR analyses (Section-3.1.5).

## References


1. "Submittal of Topical Report PL-NF-90-001", Letter, Keiser, H.W. (PP&L) to Butler, W.R. (NRC), dated August 8, 1990.

2. "Qualification of Transient Analysis Methods for BWR Design and Analysis", PL-NF-89-005, Pennsylvania Power & Light, December 1989.

3. "Qualification of Steady-State Core Physics Methods for BWR Design and Analysis", PL-NF-87-001-A, Pennsylvania Power & Light, July 1988.

4. D.B. Jones, "CPM-2 Computer Code User's Manual", Part II, Chapter 6 of EPRI-NP-4574-CCM, February 1987.

5. D.M. VerPlanck, "SIMULATE-E: A Nodal Core Analysis Program for Light Water Reactors", EPRI NP-2792-CCM, March 1983.

6. "RETRAN 02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", EPRI-NP-1850-CCM-A, Computer Code Manual, June 1987.

7. "ESCORE - The EPRI Steady-State Core Reload Evaluator Code: General Description", EPRI-NP-5100, February 1987.

8. S.W. Jones, et. al., "POWERPLEX Core Monitoring System Software Specification for the Susquehanna Steam Electric Station Units 1 and 2", XN-NF-83-35(P), Revision 1, August 1986.

9. "SIMTRAN-E: A SIMULATE-E TO RETRAN-02 Data link", EPRI NP-5509-CCM, December 1987.

10. "Request for Additional Information - Susquehanna Steam Electric Station Units 1 and 2," Letter, Thadani, M.C. (NRC) to Keiser, H.W. (PP&L), dated February 15, 1991.

11. "Request for Additional Information on PP&L Topical Report PL-NF-90-001," Letter, Raleigh, J.J.(NRC) to Keiser, H.W. (PP&L), dated August 1, 1991.

12. "Initial Response to RAI on PL-NF-90-001 (SCU Questions)", Letter, Keiser, H.W. (PP&L) to Butler, W.R. (NRC), PLA-3566, dated April 23, 1991.

13. "Final Response to RAI on PL-NF-90-001", Letter, Keiser, H.W. (PP&L) to Butler, W.R. (NRC), PLA-3578, dated June 4, 1991.

References (Cont'd)


14. "Susquehanna Steam Electric Station Licensing Methods: Plan for U1C7", Letter, Keiser, H.W. (PP&L) to Butler, W.R. (NRC), PLA-3641, dated August 29, 1991.

15. "ARMP-02 Documentation: Part II, Chapter 8 SIMULATE-E (Mod. 3) Computer Code Manual", EPRI NP-4574-CCM, Part II, Chapter 8, September 1987.

16. "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", XN-NF-524-P-A, Revision 1, November 1983.

17. "Exxon Nuclear Methodology for Boiling Water Reactor, THERMEX: Thermal Limits Methodology Summary Description", XN-NF-80-19-P-A, Volume 3, Revision 2, January 1987.

18. "Code of Federal Regulations", Office of the Federal Register, Part-10, January 1991.

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Units 1 & 2

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# Attachment 2 to PLA-5470

# Calculation EC-RISK-1083, Revision 1

# NUCLEAR ENGINEERING
# CALCULATION / STUDY COVER SHEET and
# DCS TRANSMITTAL SHEET



➢2. TYPE: Calc ➢3. NUMBER: EC-RISK-1083 ➢4. REVISION: 1

5. TRANSMITTAL#: ______ *➢6. UNIT: 3 *➢7. QUALITY CLASS: N

➢9. DESCRIPTION: Risk Associated with the Removal of the HPCI Automatic Suction Swap from the CST to the Suppression Pool *➢8. DISCIPLINE:

SUPERSEDED BY:

10. Alternate Number: ______ 11. Cycle: ______

12: Computer Code or Model used: CAFTA Fichee ☐ Dis x Am't 1

13. Application:

*➢14 *Affected Systems:* 152, 252, 156, 256, ,

* If N/A then line 15 is mandatory

**➢15. NON-SYSTEM DESIGNATOR:

**If N/A then line 14 is mandatory

16. Affected Documents:

☐ Lic. Doc Change Req'd

17. References:

18. Equipment / Component #:

19. DBD Number:

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|---|---|---|
| ➢20. PREPARED BY | Frank J. Czysz<br>*Print Name* | Frank J Czysz<br>*Signature* |
| ➢21. REVIEWED BY | Mark A. Chaiko<br>*Print Name* | Mark A. Chaiko<br>*Signature* |
| ➢21A. VERIFIED BY | *Print Name* | *Signature* |
| ➢22. APPROVED BY | K.W. Brinckman<br>*Print Name* | KW Brinckman 4/29/02<br>*Signature* |
| ➢23. ACCEPTED BY PP&L / DATE | *Print Name* | *Signature / DATE* |

**TO BE COMPLETED BY DCS**

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| NR-DCS SIGNATURE/DATE |

ADD A NEW COVER PAGE FOR EACH REVISION
FORM NEPM-QA-0221-1, Revision 5, Page 1 of 2, ELECTRONIC FORM

* Verified Fields
➢ REQUIRED FIELDS

| | ENGINEERING CALCULATION STUDY<br>REVISION DESCRIPTION SHEET |
|---|---|
| **REVISION NO:** 1 | **CALCULATION NUMBER:** EC-RISK-1083 |

This form shall be used to record the purpose or reason for the revision, indicate the revised pages and / or affected sections and give a short description of the revision. Check ( x ) the appropriate function to add, replace or remove the affected pages.

| Revised Pages | Affected Sections | Add | Rpl | Rmv | Description / Purpose of Revision |
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Model changes:

1. Deleted dual recovery of off site power. It is inappropriate to apply more than one LOOP recovery factor. The LOOP recovery factor is a multiplier to the cutset frequency, which accounts for the probability of not recovering from a LOOP in a certain amount of time. The time is derived from knowing what sequence the cutset is from and the timing provided in the IPE volume 4, section F.
2. Revised fault tree for "loss of condenser" (closure of MSIVs) to eliminate LOCAs. The LOCAs were getting into the ATWS sequences and there is no calculation for LOCA/ATWS.
3. Added sequence flags to all CDF and LERF sequences. This will identify the sequence the cutset is from and enable the use of recovery factors if appropriate.
4. Added recoveries for FLAG-TR-2-16 and FLAG-TR-2-20 (LERF). A LOOP recovery was added to the cutsets with an initiating event of loss of off site power and the subject sequence flags. Without the recovery factor the cumulative LERF was too high based on previous work.
5. Added successes to TR-2-7. These successes were added to eliminate illogical cutsets given the successes and failures shown on the event tree.
6. Deleted ATWS_9 from LERF. (see # 7 below) ATWS_9 was contributing to both CDF and LERF. A further review of this sequence indicated that if HPCI initially operated but failed due to high suppression pool temperature (HPCI automatic suction swap not removed) core damage would occur (PDS-12). After HPCI failure, the RPV level can not be maintained and the operators are instructed to ADS to allow for low-pressure makeup. Blowing down a critical core is predicted to cause core damage.
7. Added ATWS_9-LERF and ATWS_14-LERF to LERF. Added MRILERF to each for saving the containment. Given that an ATWS_9 sequence has occurred, the containment can be saved if the operator continues to drive rods via MRI. The fault tree MRILERF evaluates MRI without HPCI (used for saving the containment). ATWS_14 is similar to ATWS_9 except that HPCI initially fails. Again MRILERF is used to save containment.
8. Added logic to fail MRI during a LOOP. With a LOOP there is no power to the condensate pumps or the condensate transfer pump and the CST will drop below the standpipe before MRI succeeds. The CRD pumps take suction from the CST standpipe.
9. Deleted credit for operator action for manual HPCI suction transfer. If RPV level is not controlled and HPCI's suction source is transferred from the CST to SP the SP level will continue to rise due to RCIC running. The procedures do not instruct the operator to shut off RCIC. If HPCI restarts with high SP level it is assumed to trip due to high exhaust pressure.
10. Revised the success criteria for SLCS during an ATWS_4-LERF from one SLC PP success to 2 pump success. This is in accordance with EC-EOPC-0519 revision 4 page 97.
11. Combined initiators %LOACBUS, %LODCBUS_624 and %RBCCW with an operator failure to crosstie CIG to IA. These three initiators do not directly cause a plant trip. However, their failure will cause CIG to fail and the unit will be lost if the crosstie to IA is not successfull.
12. Changed the truncation in PRAQUANT to 1E-12 for both CDF and LERF calculations. Previously the cumulative CDF and LERF for each decade was not showing a decrease in frequency for the last decade quantified compared to the next higher decade. Decreasing the truncation limit to these values will remedy this condition.

13. Added Plant Damage States PDS-10 and PDS-11 to the LERF gate. Both states have core damage and containment isolation failure. PDS-11 also has vessel failure.

Calculation changes:

1. Revised calculation to account for HPCI potentially failing to restart with high suppression pool level due to potentially tripping on high exhaust pressure.
2. Added histogram for CDF by decade for the pre modification case and the post modification case with certain operator.
3. Added Small LOCA event trees and sequence quantification for all cases going to CDF or LERF.
4. Added ATWS sequence quantification for case going to CDF or LERF.

# 1.0 OBJECTIVE

## 1.1 Modification Evaluation

This calculation evaluates the change in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for a plant modification which would make the High Pressure Coolant Injection (HPCI) suction transfer from the Condensate Storage Tank (CST) to the suppression pool, on high suppression pool level, a manual operator action. Currently, this transfer is automatic. The motivation for this modification is to ensure availability of the HPCI system during all Anticipated Transient Without SCRAM (ATWS) events. Emergency Procedures Guidelines (EPG) revision 4 identified the need for operating HPCI from the CST and authorized defeating the auto suction transfer on high suppression pool level. The NRC approved EPG revision 4 by their SER dated September 12, 1988. The Susquehanna guidance to implement this recommendation is to boot a relay contact to defeat the transfer per ES-1/252-002. Implementing this procedure entails installing a rubber boot on a relay finger in an energized cabinet. For ATWS sequences with SLCS failure, the implementation process is too long for manual insertion of control rods to be successful. Hence, a modification is necessary to defeat the automatic suction transfer. Defeating the automatic suction transfer also requires a Technical Specification change. This calculation also addresses the risk-related questions in NRC's Request for Additional Information.

HPCI lube oil is cooled by a portion of the HPCI pump's discharge flow. As such, sustained operation with high process water temperatures may cause HPCI to fail as the lube oil temperature increases. Continued operation of HPCI is assured if sustained pump suction temperatures are limited to 140°F, reference 16. HPCI is relied on to operate during an ATWS event. As documented in Ref. 2, operation of HPCI is assured if short-duration temperature excursions do not exceed 190°F. If the Standby Liquid Control system (SBLC) is operable during an ATWS event, HPCI is capable of operating throughout the time frame required to bring the reactor to Hot Shutdown. For a high-powered ATWS event with failure of the SBLC, suppression pool temperatures are expected to exceed the short-term HPCI operating limit of 190°F by a large margin. Since with the present plant configuration, HPCI suction will transfer from the CST to the suppression pool in an ATWS event, failure of the SBLC will quickly lead to failure of HPCI on loss of lube oil cooling for high-powered ATWS events. Failure of HPCI will require the operator to rapidly depressurize the reactor pressure vessel (RPV) to obtain coolant makeup from low-pressure injection systems. Core damage from unstable operation is expected upon depressurization of a critical reactor. With the proposed modification installed, HPCI suction will remain aligned to the CST in all ATWS events. If the failure of SBLC occurs in the ATWS, HPCI will continue to inject with suction from the CST, and the operator can bring the reactor to Hot Shutdown by manually driving control rods.

The disadvantage of this modification is that during a small liquid break LOCA, it can not be guaranteed that HPCI will successfully re-start from a tripped condition with suppression pool level above 25'. Without the automatic HPCI suction transfer, suppression pool water level will increase during a small break LOCA and eventually reach the elevation of the horizontal portion of the HPCI turbine exhaust line (elevation 25.1'). This condition is not a problem unless HPCI trips. If the trip is due to high RPV water level, HPCI will automatically restart when the RPV level drops to level 2. The restart is assumed to fail since the exhaust line will contain water, which is assumed to cause HPCI to trip on high exhaust pressure as it tries to clear the water from the exhaust line. To minimize the risk of tripping HPCI after an automatic restart an operator action will be added to the emergency operating procedures (EOPs) to ensure that the HPCI suction is transferred to the suppression pool whenever pool level is above 25 feet as long as pool temperature can be maintained less than 140°F. Note the risk model does not credit this action since RCIC, by procedure, can still be running which will add more CST water to the Suppression Pool. Hence, the successful operator transfer of the HPCI suction from the CST to the Suppression Pool will not keep the Suppression Pool water level below the HPCI exhaust. However, credit is taken for the Operator's ability to control RPV water level below the HPCI high level trip (level 8). This requirement is presently contained in the EOPs.

For the one operator action discussed above, the calculation will be performed with mean operator error rate, upper 95% confidence limit, no operator error and with certain operator error.

## 1.2 Evaluation of HPCI Failure at 140°F

In the original IPE, 140°F was taken as the HPCI suction temperature operating limit. This assumption would cause HPCI to fail for all high-powered ATWS sequences. Since then we have received information that indicates that for short-term events HPCI can run with suction water temperatures up to 190°F. The CDF for the assumption of HPCI failure at 140°F will be evaluated with the automatic suction transfer and for the manual transfer. Only the ATWS sequences sensitive to HPCI failure will be evaluated. The two evaluations will use random HPCI failure and certain HPCI failure.

## 1.3 Taxonomy of Initiating Events

The calculation also will list the contribution of each initiating event and express the initiator's contribution as a percentage of the total CDF and LERF.

## 1.4 CDF and LERF Contribution from a Station Blackout

The calculation will identify the LOOP sequences with all diesel generators failed, loss of Emergency Service Water or loss of the 125V batteries, a Station Blackout (SBO). The CDF and LERF contribution of these SBO sequences will be totaled and compared to that of a LOOP.

1.5 Operator Actions

The specific operator actions required will be outlined.

1.6 Procedural Guidance

The procedural guidance for the operator actions will be discussed.

1.7 Training and Qualifications

The specific operator training/qualifications necessary to carry out the actions will be addressed.

1.8 Additional Support Personnel/Equipment and Instrumentation Required

The calculation will also discuss any additional support personnel and/or equipment required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and verify that the required action has successfully been taken.

1.9 Credible Errors

A discussion of the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery will be addressed in the calculation.

1.10 Risk Significance of Operator Actions

An evaluation of the risk significance of the proposed operator actions will be provided.

1.11 Histogram for CDF

A histogram for the CDF and LERF by decade is provided. The histogram shows that the truncation level is low enough to capture the major contributors to CDF.

1.12 LOCA Sequences

The calculation provides the LOCA event trees for a small break LOCA and quantifies each sequence that results in core damage or a large early release.

1.13 ATWS Sequences

The sequence quantification for the ATWS resulting in core damage or a large early release is provided.

# 2.0 CONCLUSIONS AND RECOMMENDATIONS

## 2.1 Modification Conclusions

The CDF and LERF are reduced by 8% and 4% respectively if the automatic suction swap is changed to a manual suction swap for mean and the upper 95% confidence level operator error rates. If the operator error is assumed to be certain for controlling RPV water level with HPCI for small break LOCA, the CDF and LERF reductions are 7% and 4% respectively for changing the automatic HPCI suction transfer to manual.

## 2.2 Conclusions for HPCI Failure at 140°F

The ATWS contribution to CDF will increase eleven fold if HPCI is assumed to fail at 140°F suction temperature. This conclusion was arrived at by failing HPCI for all ATWS sequences sensitive to HPCI failures.

## 2.3 Taxonomy of Initiating Events

### 2.3.1 CDF

The Loss of DC bus D624 is the largest contributor to CDF with and without the HPCI automatic suction swap available.

The reduction in CDF from pre-modification to post modification (mean operator error) is due the success of HPCI in the ATWS sequences. The reduction in the CDF due to ATWS is 4.3E-8 and the total change in CDF pre-modification to post modification is 4.3E-8. The contribution from the small liquid break LOCA, as expected increases post modification but the increase is small compared to the magnitude of the reduction obtained for ATWS.

The CDF contribution from a small liquid break LOCA increases as the operator error rate (for controlling reactor water level) increases from zero to 1.

### 2.3.2 LERF

A Loss of Offsite Power Initiator is the largest contributor to LERF with and without the HPCI automatic suction swap available. The post modification LERF results are insensitive to the operator error rates. This apparent insensitivity results because the fact that the most probable LOCA cutsets quantify at 6.9E-14 with certain operator error. This is over five orders of magnitude lower than the total LERF and is a casualty of significant figures.

## 2.4 CDF and LERF Contribution from a Station Black Out

The contribution of SBO to CDF and LERF is relatively constant in each case pre and post modification, see matrix in section 5.3.

## 2.5 Specific Operator Actions

### 2.5.1 Actions Credited in Risk Model

RPV level control after a small break liquid LOCA is one operator action credited in the risk model after the removal of the HPCI automatic suction transfer. RPV level control is part of the existing emergency operating procedures, step RC/L4 (reference 8).

### 2.5.2 Actions Not Credited in Risk Model

The manual suction transfer is not currently part of the emergency operating procedures and is not credited in the risk model. This new action will be added to the procedures in accordance with the administrative program that governs EOP changes. It is anticipated that the step will read as follows: When suppression pool level reaches 25' ensure HPCI and RCIC are running. If HPCI injecting into RPV and suppression pool temperature can be maintained less than 140°F transfer HPCI suction from CST to suppression pool.

Two other actions are taken to prevent HPCI failing on restart with high suppression pool level. Neither of these actions are credited in the risk model. One is an operator action to start HPCI when the suppression pool level reaches 26'(25' in next procedure revision), step SP/L-11 (reference 15). This is currently part of our Emergency Operating procedures therefore the operators are trained on this action. The second is SP/L-10 (reference 15) which instructs the operator to maintain the suppression pool less than 26 (25' in next procedure revision) feet by using suppression pool cleanup or by using RHR suppression pool cooling letdown.

## 2.6 Procedural Guidance for Required Actions

Guidance for the RPV level control currently exists in the emergency operating procedures, see paragraph 2.5.

## 2.7 Operator Training and Qualifications for the Required Actions

A Licensed Reactor Operator will perform the two required operator actions in the control room. RPV level control is an existing part of the emergency procedures and as such the operators receives training on this action. The qualification required for this action is to be a Licensed Reactor Operator.

The training required for the manual suction transfer is to train the operators on the action and then validate that the action is correctly implemented in the simulator. A Licensed Reactor Operator would perform this action.

2.8 Additional Support Personnel/Equipment and Instrumentation Required

There is no additional support personnel or additional equipment required for these actions. The cue for operators to control RPV level is any entry condition for the RPV Control or Level/Power Control emergency operating procedures, step RC/L-4 (reference 8). The cue for initiating the HPCI suction transfer is suppression pool high level. The suppression pool high level condition is alarmed in the control room 23'9"±2", reference 18. The level switches that actuate the alarm are safety related and powered by the 1E 125VDC power (reference 7). A 1E battery powers the control room annunciator however it is via a non-1E electrical panel (reference 9). There are also two safety related suppression pool level indicators in the control room on 1/2C601 (reference 10). The operator will receive indication of a successful suction transfer by the valve position indicating lights in the control room (reference 5 & 6).

There is also a level indicator LI-1(2)5775B on the HPCI panel in the control room. The tolerance of this level indicator is ±4", (reference 19). A credible error in implementing the manual suction transfer is an error closing the suction source from the CST without the suppression pool suction valve being open. If this error occurs, HPCI will trip on low suction pressure. If the CST suction valve is inadvertently closed and there still is a valid HPCI initiation signal, the valve will automatically reopen and HPCI will automatically restart when the low suction pressure condition clears (reference 5 & 7). The hand switch, which initiated the close signal, is a spring return to "auto" so the close signal is not continuous. Hence, the potential operator error of closing the CST suction source is of no consequence to Reactor inventory assuming HPCI restarts. The duration of the less than full suction flow condition is expected to be approximately 27 seconds (reference 11). Note, no credit is taken in the risk model for the suction transfer from the CST to the Suppression Pool.

It should be noted that if the HPCI suction valve from the suppression pool is opened, the 100% open limit switch on this valve will initiate a close signal to the HPCI suction valve from the CST.

2.9 The risk significance of operator action

The operator action to control reactor water level with HPCI is not risk significant as defined in paragraph 3.7. There is no change in CDF for the operator action always being successful and there is only a 1% increase in CDF if the operator action always fails compared to the mean operator error rate. The LERF is insensitive to HPCI reactor level control errors.

2.10 The LOCA event trees and LOCA and ATWS sequence quantification are provided in Attachment 3.

## 3.0 ASSUMPTIONS AND INPUTS

3.1 Operator error rate will be obtained from reference 4 Group 1, Probability of Failure to Manually Operate Critical Components. It is appropriate to use Group 1 data since the actions taken are being performed from the control room on major pieces on equipment using critical parameters and alarms as cues for the actions.

3.2 Given a small break LOCA has occurred the reactor will SCRAM on high drywell pressure. For this scenario a ten-minute time delay period will be assumed before the operator controls reactor water level.

3.3 CAFTA will be used for CDF and LERF calculations. Fault Tree SUSQUEHANNA15 was used for both the automatic transfer and manual suction transfer.

3.4 Water intrusion into the 20" HPCI exhaust line following a system trip (pipe center line elevation at 26.5') is assumed to cause HPCI to fail due to high exhaust pressure upon attempting to restart.

3.5 The modification that is removing the HPCI suction swap will not delete the alarm that comes in when the auto transfer was initiated. Calculation EC-052-1025 assumed the suppression pool was at the auto transfer level at the start of the transient. Therefore, the operator has a minimum of 21 minutes from the alarm on suppression pool level to transfer the HPCI suction to the suppression pool.

3.6 Manual Rod Insertion (MRI) is assumed failed for the current Susquehanna design, HPCI automatic suction swap. MRI is a relatively slow process to shutdown the reactor and is only successful if HPCI is available for makeup. If an ATWS occurs, the suppression pool temperature will rise above the 190°F, the short-term HPCI limit, before the reactor can be brought to Hot Shutdown by MRI. When the automatic suction transfer does occur, there are not enough rods driven into the core to shut down the reactor. HPCI is assumed failed shortly after the automatic transfer occurs due to the high suction temperature. Without HPCI adding water, the RPV level will drop and the RPV must be depressurized while critical, which will cause core damage.

3.7 Risk significant operator actions will be determined with the methodology described in NUMARC 93-01.

3.8 If the operator is successful at initially taking control of the RPV level with HPCI, RPV level control is deemed successful. It is felt that the dominant error would be to initially fail to take level control. Once the operator takes control he will maintain control.

3.9 The LOCA and ATWS sequence quantifications will have a truncation limit four orders of magnitude lower than the sum of the sequence cutsets except if the cutset total is less than 1E-11. For these sequences the truncation will be low enough to demonstrate that the sequence will quantify. A sequence total of 1E-11 will not make a significant change to the total CDF or LERF.

## 4.0 METHOD

### 4.1 Modification Method

The CAFTA fault tree replicating PPL's Individual Plant Evaluation (IPE) was modified to suit the two scenarios, HPCI automatic suction swap and HPCI manual suction swap.

#### 4.1.1 HPCI Automatic Suction Swap

In the HPCI automatic suction swap case, a basic event was added, HPCI-SWAP, to the MRI "or" gate, 156-N-N-MRI. The sole input to this "equal gate" is HPCIMOD. Assigning a probability of one to HPCIMOD assures that MRI will always fail for a high-powered ATWS event with SLCS failure. MRI needs HPCI to be successful in order to give the operators enough time to drive the control rods in manually. During the ATWS the suppression pool level rises due to the HPCI exhaust and Safety Relief Valves (SRV) lifting. As a consequence of the steam condensing in the suppression pool, the suppression pool temperature increases. Hence as the suppression pool level rises, the automatic transfer occurs and the suppression pool water temperature exceeds the HPCI limit of 190°F thus failing HPCI. Therefore it is appropriate to fail MRI.

For ATWS scenarios in which SBLC is operable, HPCI will have completed its mission before suppression pool temperature reaches 190°F (Ref. 2). If one SBLC pump is operable, RCIC, CRD, and SBLC can maintain RPV water level above top of active fuel at the time when suppression pool temperature reaches 190°F given that HPCI initially ran and failed after the suction transfer.

#### 4.1.2 HPCI Manual Suction Swap

In the manual suction swap case, MRI is not defeated. HPCI can remain on the CST until the manual rod insertion is complete.

The manual suction swap case then imposes an operator action for the small liquid LOCAs. With a small liquid LOCA the suppression pool level rises due the liquid from the break and the HPCI exhaust steam and the level will exceed the manual transfer point (25 feet suppression pool level) in a minimum of 21 minutes (reference 13 page 63). However, the temperature of the pool does not exceed 140°F (reference 13 page 63). A suppression pool level above the 25 feet does not automatically fail HPCI. HPCI will continue to exhaust steam into the suppression pool and the suppression pool level will not exceed the suppression pool load limit (Ref. 13, p. 63a). However, if the operator does not control RPV level with HPCI, level 8 will be reached and HPCI will trip. HPCI will automatically restart if the RPV water level reaches level 2. This restart of HPCI with high suppression pool level may cause HPCI to trip on high exhaust pressure.

The HPCI level concerns outlined above, are addressed in the fault tree as follows:

An *AND Gate*, 152-II-N-CTRLLVL was added to the HPCI *OR* Gate, 152. The *AND Gate* has three inputs: a small liquid LOCA initiator, a basic event of an operator error to control RPV water level, and a switch to defeat this logic for the case using the automatic transfer. If the operator controls RPV level below level 8 HPCI will not trip off and try to restart. Continued operation of HPCI when the suppression pool level is above the automatic suction swap level is not a problem, as the exhaust steam will maintain the turbine exhaust piping free of water.

A small liquid break is the only scenario when it is desirable to align the HPCI suction to the suppression pool. For other initiators, the suppression pool level does not reach the HPCI suction manual transfer point of 25 feet or the pool level is above the manual transfer point but the suppression pool temperature exceeds 190°F (the ATWS sequence using MRI) (Ref. 1, pp.165-166).

### 4.1.3 Operator Actions

Failure to take RPV level control is one operator action that needs to fail to fail HPCI for small break LOCAs as previously discussed. The operator error rate for this action was determined as follows:

Given a small break LOCA has occurred, the reactor will SCRAM on high drywell pressure. If feedwater is available it will continue to control level. Regardless of feedwater availability HPCI will start and inject. Per assumption 3.2 the operator does not initiate any level control for the first ten minutes. For HPCI to be successful, RPV level must be controlled so that the level does not reach level 8 during the time that the horizontal portion of the HPCI exhaust line is subject to water intrusion, suppression pool level ≥ 25 feet. Level 8 will cause HPCI to trip. Hence, the operator is allowed 11 minutes (21 minutes-10 minutes) to control level to avoid a level 8 trip. Using reference 4 Group1 error rates (Table 5-46) the probability of failure is 0.023 for a time of 11 minutes and the upper 95% confidence limit is 0.061. According to input 3.8, if the operator is successful in initially taking control of level, then the operator will succeed in RPV level control with HPCI.

RPV level control is performed from the control room. Controlling RPV level is part of the existing Emergency Operating procedures, EO-100/200-102 step RC/L-4.

Note that a HPCI failure for this event, in of itself, does not result in core damage. If HPCI does fail, the RPV will be depressurized and low pressure ECCS will be used for makeup.

4.2 Method for HPCI Failure at 140°F

If HPCI is postulated to fail at 140°F it will fail for all ATWS sequences which involve closure of the MSIVs (Ref. 1). To evaluate the significance of failing at 140°F and failing at 190°F (current assumption) a basic event, 152-II-N-ATWSSWITCH was added to the HPCI *OR Gate*, 152. There are 4 ATWS sequences in which HPCI randomly fails (reference 3) that result in core damage (PDS-1 is no core damage), ATWS_11, 12, 13 & 14. These four sequence were run with HPCI failing randomly and failure being certain, probability = 1.0. The CDF for the four sequences was then totaled. The other ATWS sequences either do not rely on HPCI or use HPCI success. If HPCI is successful for a sequence it is not in the fault tree. Hence, failing HPCI is of no consequence for these sequences. If the event tree was rewritten for certain HPCI failure, there would be no "up" leg and the logic would use the four sequences with HPCI failed. Therefore it is proper to only evaluate these four sequences to determine the effect certain HPCI failure has on the ATWS contribution to CDF.

4.3 The risk significance of the operator action will be determined by comparing the CDF and LERF for successful actions and for actions which always fail, to the mean operator error rate. If the successful operator action reduces CDF or LERF by 0.005 or more, then the operator action is risk significant or if the operator action always fails, and CDF or LERF increase by a factor of 2 or greater, the operator action is risk significant, reference section 3.7.

4.4 The small break LOCA event trees are in Attachment 4. The Small Liquid Breaks and Small Steam Break LOCAs are provided. Each of these event trees ends with a "class" label. The label refers to the LOCA Transfer number. LT-#. Each LOCA transfer number is the entrance point on a LOCA transfer event tree. The only transfers that result in core damage or in a large early release are LOCA transfers LT-2 and LT-3. Hence, LOCA transfer trees LT-2 and LT-3 are also included. From these transfers the only sequences that result in core damage or large early release are:

| Sequence | Plant Damage State | Sequence | Plant Damage State |
|---|---|---|---|
| LT-2-1 | PDS-4 | LT-3-1 | PDS-4 |
| LT-2-2 | PDS-10 | LT-3-2 | PDS-10 |
| LT-2-4 | PDS-13 | LT-3-4 | PDS-13 |
| LT-2-5 | PDS-4 | LT-3-5 | PDS-4 |
| LT-2-6 | PDS-10 | LT-3-6 | PDS-10 |
| LT-2-8 | PDS-4 | LT-3-7 | PDS-4 |
| LT-2-9 | PDS-10 | LT-3-8 | PDS-10 |
| LT-2-27 | PDS-11 | LT-3-10 | PDS-4 |
| LT-2-29 | PDS-13 | LT-3-11 | PDS-10 |
| LT-2-30 | PDS-7 | LT-3-14 | PDS-11 |
| | | LT-3-16 | PDS-13 |
| | | LT-3-17 | PDS-7 |

The Plant Damage States are defined and dispositioned as follows:

| Plant Damage State | Definition | Disposition |
|---|---|---|
| PDS-1 | No Core Damage | None |
| PDS-2 | Limit Cycle Operation During an ATWS | CDF |
| PDS-3 | No Core Damage, Containment Vented at 30 PSIA | None |
| PDS-4 | Core Damage, Vessel OK, Containment OK | CDF |
| PDS-5 | Core Damage, Vessel OK, Containment Over Pressure Failure | None, not an early release* |
| PDS-5L | Core Damage, Vessel OK, Containment Over Pressure Failure | LERF |
| PDS-6 | Core Damage, Vessel Fails, Containment OK | None* |
| PDS-7 | Core Damage, Vessel Fails, Containment Over Temperature Failure | LERF |
| PDS-8 | Core Damage, Vessel Fails, Containment Over Pressure Failure | None, not an early release* |
| PDS-9 | No Core Damage, Containment Over Pressure Failure | None, not an early release |
| PDS-10 | Core Damage, Vessel OK, Containment Isolation Failure | LERF |
| PDS-11 | Core Damage, Vessel Fails, Containment Isolation Failure | LERF |
| PDS-12 | ATWS flag | None |
| PDS-13 | Core Damage and Liner Failure | LERF |
| * These states are not included in CDF since the cutsets would be non-minimal to the CDF cutsets. Core Damage would have occurred with less failures than listed in these damage states. | | |

## 5.0 RESULTS

### 5.1 Modification results

The results of running CAFTA for the removal of the automatic transfer is given in the table below.

| | Pre-Modification | Post Modification | | | |
|---|---|---|---|---|---|
| | n/a | Mean Op Er | 95% Op Er | Certain Op Er | No Op Er |
| CDF | 5.29E-07 | 4.86E-07 | 4.86E-07 | 4.92E-07 | 4.86E-07 |
| LERF | 1.45E-08 | 1.39-08 | 1.39E-08 | 1.39E-08 | 1.39E-08 |

5.2 Results for HPCI Failure at 140°F

The results of running CAFTA for HPCI failing at 140°F and failing at 190°F (random failures) are given in the table below.

| | |
|---|---|
| HPCI Random Failure | |
| | |
| Sum of ATWS11, 12, 13, and 14 CDF | 6.14E-8 |
| | |
| HPCI Failure Certain | |
| | |
| Sum of ATWS11, 12, 13, and 14 CDF | 6.56E-7 |
| | |
| Factor difference between random and certain failure | 10.68 |

5.3 Taxonomy of Initiating Events

See table on next page.

## Taxonomy of CDF

| | Plant Configuration | Pre Modification | | Post Modification | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Operator Error Rate for controlling Reactor Level during a Small Liquid Break | N/A | | No Error (0) | | Mean (0.023) | | 95$^{th}$ Percentile (0.061) | | Certain (1) | |
| | Cumulative & % CDF per Initiator | Cul. CDF | %CDF | Cul. CDF | %CDF | Cul. CDF | %CDF | Cul. CDF | %CDF | Cul. CDF | %CDF |
| Initiators | %LOOPMAN* | 1.63E-07 | 30.78% | 1.63E-07 | 33.50% | 1.63E-07 | 33.50% | 1.63E-07 | 34.77% | 1.63E-07 | 33.08% |
| | %ISOMAN | 7.42E-08 | 14.03% | 3.17E-08 | 6.53% | 3.17E-08 | 6.53% | 3.17E-08 | 6.53% | 3.17E-08 | 6.45% |
| | %LOACBUS | 8.27E-10 | 0.16% | 4.40E-10 | 0.09% | 4.40E-10 | 0.09% | 4.40E-10 | 0.09% | 4.40E-10 | 0.09% |
| | %LOCA-LG-LQD | 2.05E-12 | 0.00% | 2.05E-12 | 0.00% | 2.05E-12 | 0.00% | 2.05E-12 | 0.00% | 2.05E-12 | 0.00% |
| | %LOCA-LG-STM | 3.53E-10 | 0.07% | 3.53E-10 | 0.07% | 3.53E-10 | 0.07% | 3.53E-10 | 0.07% | 3.53E-10 | 0.07% |
| | %LOCA-MD-LQD | 7.55E-11 | 0.01% | 7.55E-11 | 0.02% | 7.55E-11 | 0.02% | 7.55E-11 | 0.02% | 7.55E-11 | 0.02% |
| | %LOCA-SM-LQD | 1.89E-10 | 0.04% | 1.89E-10 | 0.04% | 2.78E-10 | 0.06% | 5.01E-10 | 0.10% | 6.44E-09 | 1.31% |
| | %LOCA-SM-STM | 3.82E-10 | 0.07% | 3.82E-10 | 0.08% | 3.82E-10 | 0.08% | 3.82E-10 | 0.08% | 3.82E-10 | 0.08% |
| | %LODCBUS_624 | 2.14E-07 | 40.43% | 2.14E-07 | 44.01% | 2.14E-07 | 44.00% | 2.14E-07 | 43.98% | 2.14E-07 | 43.45% |
| | %NONISO | 1.78E-08 | 3.36% | 1.78E-08 | 3.65% | 1.78E-08 | 3.65% | 1.78E-08 | 3.65% | 1.78E-08 | 3.61% |
| | %RBCCW | 2.43E-10 | 0.05% | 9.33E-11 | 0.02% | 9.33E-11 | 0.02% | 9.33E-11 | 0.02% | 9.33E-11 | 0.02% |
| | %TBCCW | 6.12E-09 | 1.16% | 6.12E-09 | 1.26% | 6.12E-09 | 1.26% | 6.12E-09 | 1.26% | 6.12E-09 | 1.24% |
| | SBO | 5.21E-08 | 9.86% | 5.21E-08 | 10.73% | 5.21E-08 | 10.73% | 5.21E-08 | 10.72% | 5.21E-08 | 10.59% |
| | Total | 5.29E-07 | 100.00% | 4.86E-07 | 100.00% | 4.86E-07 | 100.00% | 4.86E-07 | 100.00% | 4.92E-07 | 100.00% |
| | | | | | | | | | | | |
| | CDF due to ATWS | 1.27E-07 | 24.07% | 8.43E-08 | 17.35% | 8.43E-08 | 17.35% | 8.43E-08 | 17.34% | 8. 43E-08 | 17.13% |
| | | | | | | | | | | | |
| | *The number reported for the LOOP initiator is a LOOP without SBO. | | | | | | | | | | |

## Taxonomy of LERF

| | Plant Configuration | Pre Modification | | Post Modification | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Operator Error Rate for controlling Reactor Level during a Small Liquid Break | N/A | | No Error (0) | | Mean (0.023) | | 95th Percentile (0.061) | | Certain (1) | |
| | Cumulative & % LERF per Initiator | Cul. LERF | %LERF | Cul. LERF | %LERF | Cul. LERF | %LERF | Cul. LERF | %LERF | Cul. LERF | %LERF |
| Initiators | %LOOPMAN* | 6.45E-09 | 44.55% | 6.45E-09 | 44.55% | 6.45E-09 | 44.55% | 6.45E-09 | 44.55% | 6.45E-09 | 44.55% |
| | %ISOMAN | 7.16E-10 | 4.95% | 9.70E-11 | 0.70% | 9.70E-11 | 0.70% | 9.70E-11 | 0.70% | 9.70E-11 | 0.70% |
| | %LOACBUS | 6.76E-11 | 0.47% | 6.40E-11 | 0.46% | 6.40E-11 | 0.46% | 6.40E-11 | 0.46% | 6.40E-11 | 0.46% |
| | %LOCA-LG-LQD | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %LOCA-LG-STM | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %LOCA-MD-LQD | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %LOCA-SM-LQD | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %LOCA-SM-STM | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %LODCBUS_624 | 8.43E-10 | 5.82% | 8.43E-10 | 6.09% | 8.43E-10 | 6.09% | 8.43E-10 | 6.09% | 8.43E-10 | 6.09% |
| | %NONISO | 2.10E-10 | 1.45% | 2.10E-10 | 1.52% | 2.10E-10 | 1.52% | 2.10E-10 | 1.52% | 2.10E-10 | 1.52% |
| | %RBCCW | 1.16E-12 | 0.01% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% | 0.00E+00 | 0.00% |
| | %TBCCW | 3.71E-09 | 25.61% | 3.71E-09 | 26.76% | 3.71E-09 | 26.76% | 3.71E-09 | 26.76% | 3.71E-09 | 26.76% |
| | SBO | 2.48E-09 | 17.14% | 2.48E-09 | 17.14% | 2.48E-09 | 17.14% | 2.48E-09 | 17.14% | 2.48E-09 | 17.14% |
| | Total | 1.45E-08 | 100.00% | 1.39E-08 | 100.00% | 1.39E-08 | 100.00% | 1.39E-08 | 100.00% | 1.39E-08 | 100.00% |
| | | | | | | | | | | | |
| | LERF due to ATWS | 7.06-09 | 48.75% | 6.43E-09 | 46.44% | 6.43E-09 | 46.44% | 6.43E-09 | 46.44% | 6.43E-09 | 46.44% |
| | | | | | | | | | | | |
| | *The number reported for the LOOP initiator is a LOOP with out SBO. | | | | | | | | | | |

Each of the initiating events is described below:

| Initiating Event | Description |
|---|---|
| %LOOPMAN | Loss of Off Site Power with on-site available |
| %ISOMAN | Isolation Transient |
| %LOACBUS | Loss of ESS 4.16KV Bus "A" - 1A201 |
| %LOCA-LG-LQD | Loss of Coolant Accident Large Liquid Break |
| %LOCA-LG-STM | Loss of Coolant Accident Large Steam Break |
| %LOCA-MD-LQD | Loss of Coolant Accident Medium Liquid Break |
| %LOCA-SM-LQD | Loss of Coolant Accident Small Liquid Break |
| %LOCA-SM-STM | Loss of Coolant Accident Small Steam Break |
| %LODCBUS_624 | Loss of 125VDC Panel 624 - "B" 125VDC |
| %NONISO | Non-Isolation Transient |
| %RBCCW | Loss of Reactor Building Closed Cooling Water |
| %TBCCW | Loss of Turbine Building Closed Cooling Water |
| SBO | Station Black Out – LOOP and loss of on-site power |

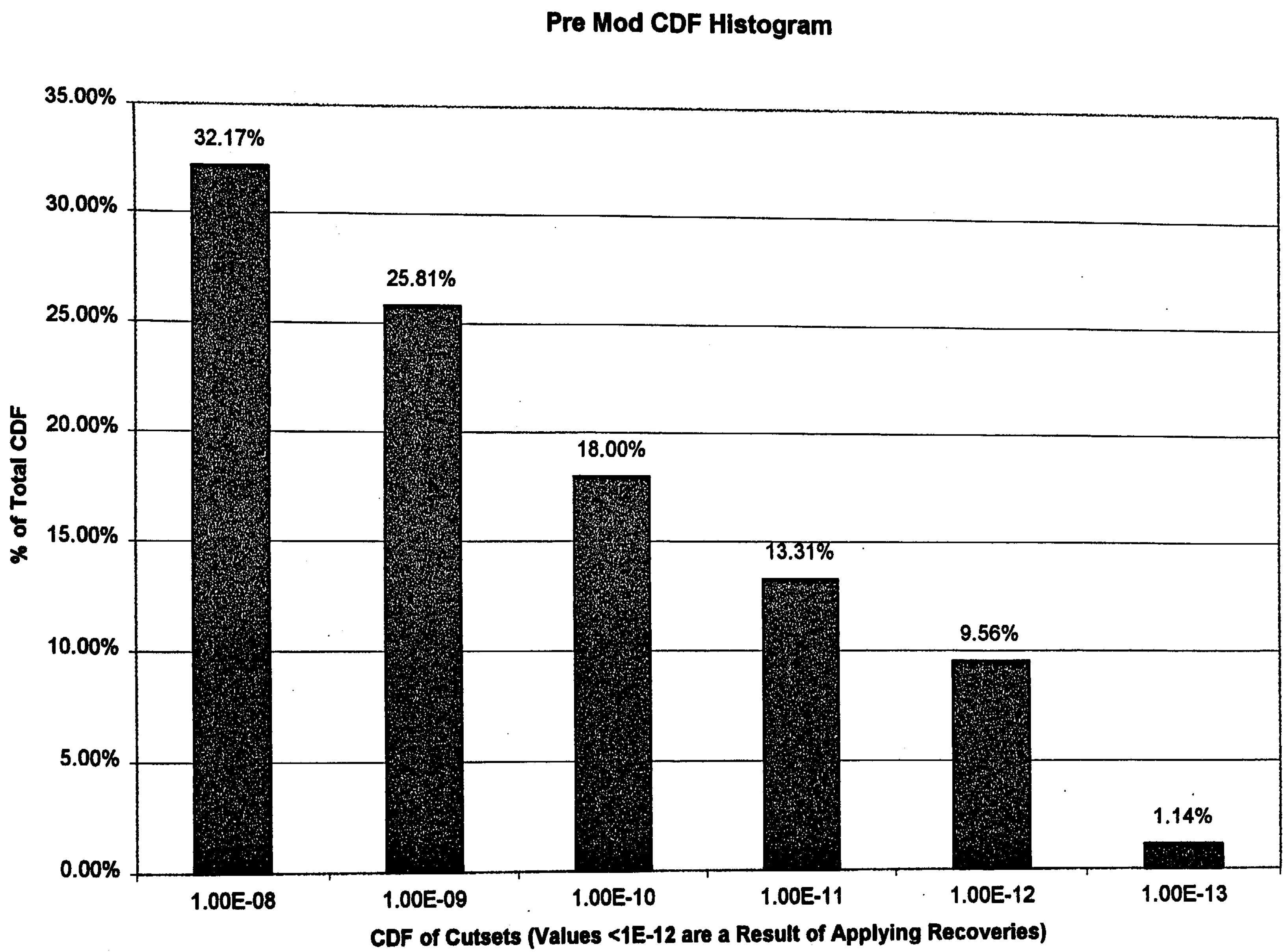
The LERF results indicate insensitivity to the operator errors for controlling RPV water level during a small break LOCA. This insensitivity can be explained by separately (separately form the entire LERF fault tree) quantifying the LOCA contributors to LERF. This quantification (LPDS-7), run with a truncation limit of 1E-15, shows that the most probable cutset (LOCAs contributing to LERF) is 6.9E-14. Therefore the apparent insensitively of LERF to LOCAs is due to the fact that the LOCA contributors to LERF are below the truncation limit (1E-12) at which the LERF case was run.

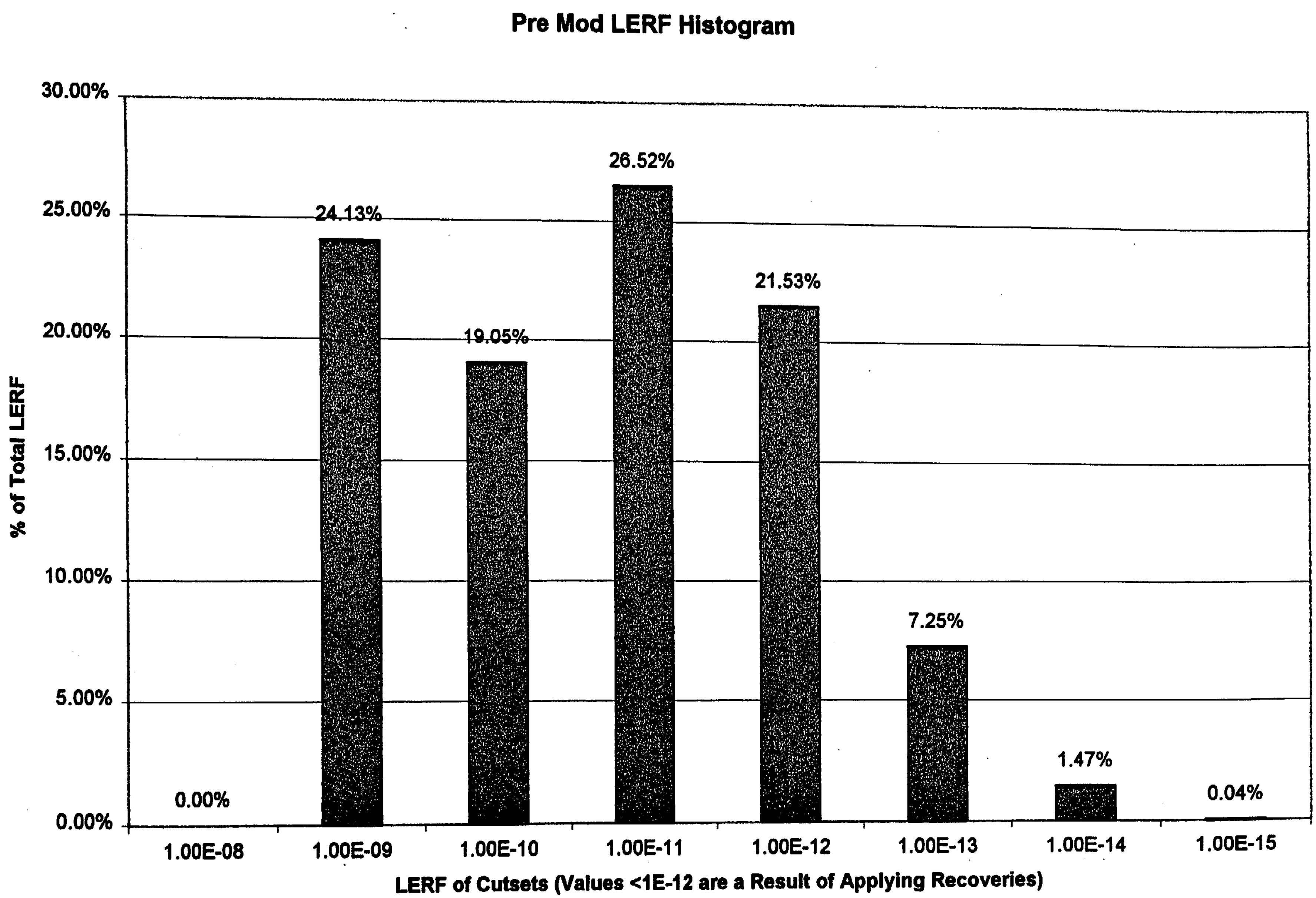
5.4 Station Black-Out Contribution

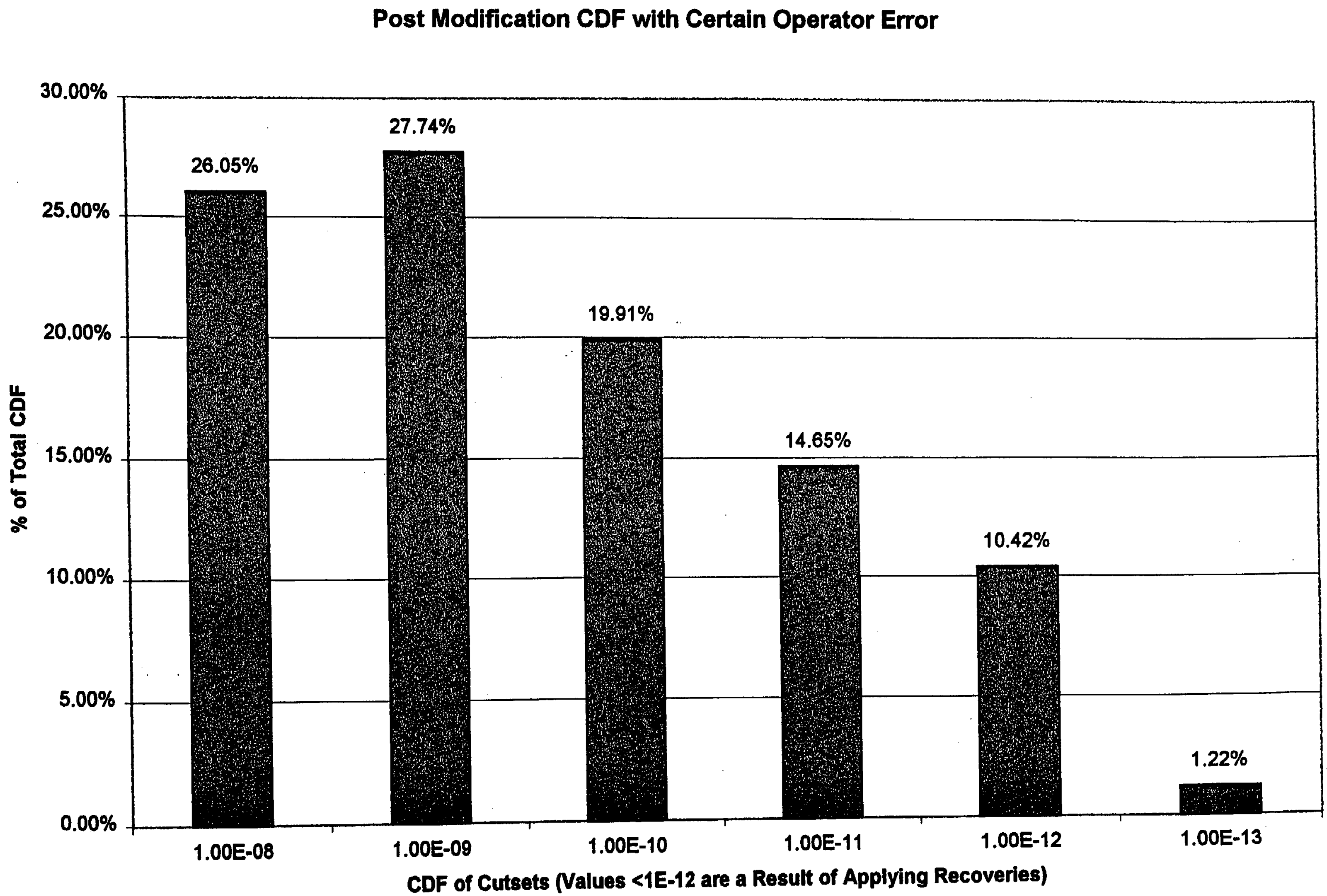
A station black out (SBO) is not an initiating event. However a loss of off site power (LOOP) and the failure of four diesel generators is an SBO. Since it is of interest to know the contribution of an SBO to CDF and LERF all the sequences involving a LOOP and failure(s) that would cause the failure of all the diesel generators were identified. The SBO contribution to CDF and LERF is shown in the matrix in section 5.3.

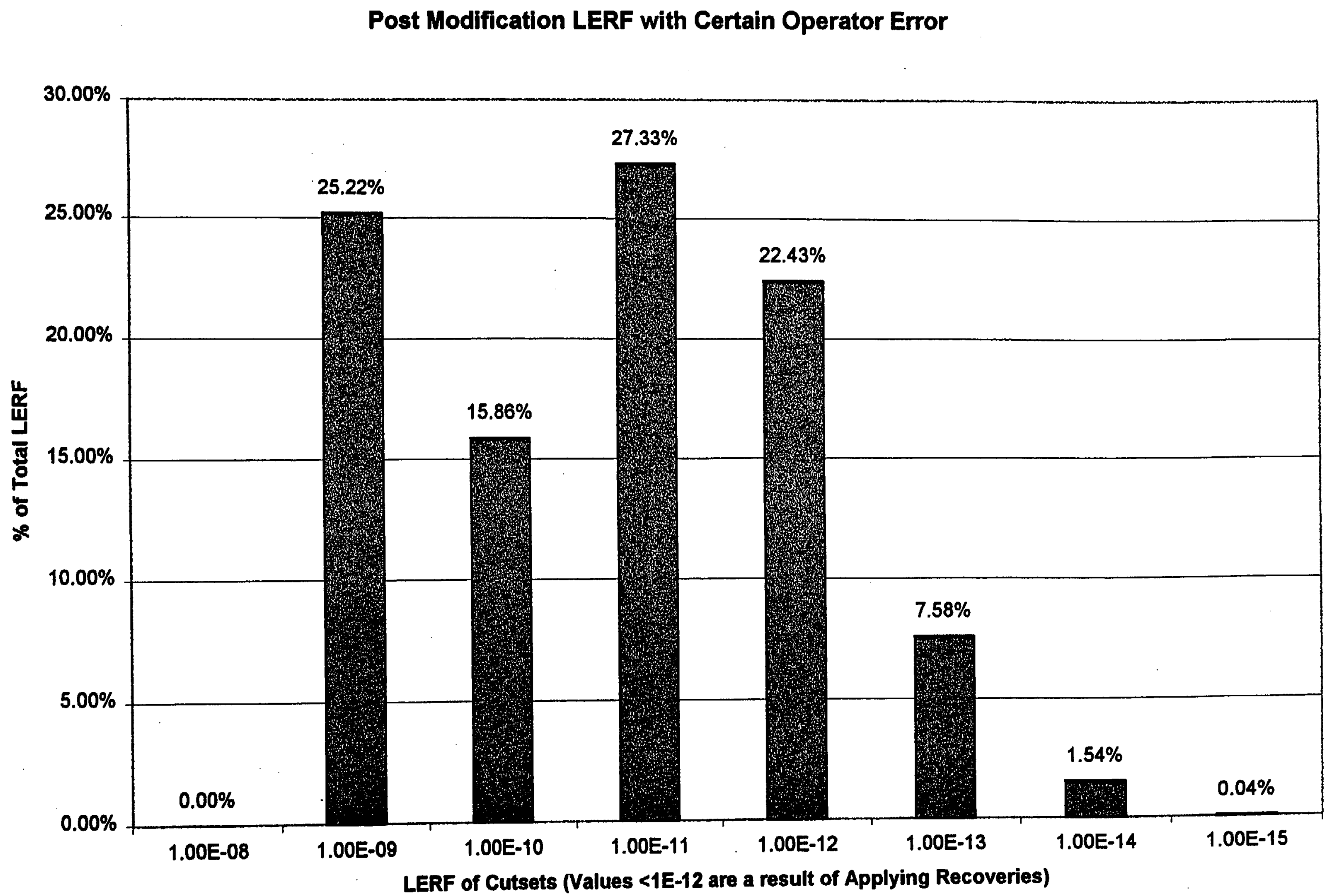
5.5 Histogram

The truncation of the CDF and LERF calculations was performed at 1E-12. A histogram plot of each for the Pre and Post Modification cases are shown below.


Pre Mod CDF Histogram
35.00%
30.00%
25.00%
20.00%
15.00%
10.00%
5.00%
0.00%
% of Total CDF
32.17%
25.81%
18.00%
13.31%
9.56%
1.14%
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
CDF of Cutsets (Values <1E-12 are a Result of Applying Recoveries)


Pre Mod LERF Histogram
30.00%
25.00%
20.00%
15.00%
10.00%
5.00%
0.00%
% of Total LERF
0.00%
24.13%
19.05%
26.52%
21.53%
7.25%
1.47%
0.04%
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
1.00E-14
1.00E-15
LERF of Cutsets (Values <1E-12 are a Result of Applying Recoveries)


Post Modification CDF with Certain Operator Error
30.00%
25.00%
20.00%
15.00%
10.00%
5.00%
0.00%
% of Total CDF
26.05%
27.74%
19.91%
14.65%
10.42%
1.22%
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
CDF of Cutsets (Values <1E-12 are a Result of Applying Recoveries)


Post Modification LERF with Certain Operator Error
30.00%
25.00%
20.00%
15.00%
10.00%
5.00%
0.00%
% of Total LERF
0.00%
25.22%
15.86%
27.33%
22.43%
7.58%
1.54%
0.04%
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
1.00E-14
1.00E-15
LERF of Cutsets (Values <1E-12 are a result of Applying Recoveries)

## 5.6 LOCA Sequences

The CDF and LERF, Pre and Post Modification (Mean operator error) for the Small Liquid and Small Steam LOCA sequences are shown below:

| | Pre Modification | | | | Post Modification | | | |
|---|---|---|---|---|---|---|---|---|
| | LOCA-SM-LQD | | LOCA-SM-STM | | LOCA-SM-LQD | | LOCA-SM-STM | |
| Sequence | CDF | LERF | CDF | LERF | CDF | LERF | CDF | LERF |
| LT-2-1 | 1.87E-11 | | 1.87E-11 | | 2.45E-11 | | 1.87E-11 | |
| LT-2-2 | | 1.87E-15 | | 1.87E-15 | | 2.45E-15 | | 1.87E-15 |
| LT-2-5 | 0.00E+00 | | 1.87E-11 | | 0.00E+00 | | 1.87E-11 | |
| LT-2-6 | | 0.00E+00 | | 1.87E-15 | | 0.00E+00 | | 1.87E-15 |
| LT-2-8 | 0.00E+00 | | 1.27E-14 | | 0.00E+00 | | 1.27E-14 | |
| LT-2-9 | | 0.00E+00 | | 1.03E-18 | | 0.00E+00 | | 1.03E-18 |
| LT-2-27 | | 2.57E-16 | | 2.57E-16 | | 2.57E-16 | | 2.57E-16 |
| LT-2-30 | | 7.19E-15 | | 7.19E-15 | | 7.19E-15 | | 7.19E-15 |
| LT-3-1 | 3.68E-10 | | 3.68E-10 | | 5.09E-10 | | 3.68E-10 | |
| LT-3-2 | | 1.74E-14 | | 1.74E-14 | | 2.59E-14 | | 1.74E-14 |
| LT-3-5 | 0.00E+00 | | 3.68E-10 | | 0.00E+00 | | 3.68E-10 | |
| LT-3-6 | | 0.00E+00 | | 3.22E-14 | | 0.00E+00 | | 3.22E-14 |
| LT-3-7 | 0.00E+00 | | 3.44E-11 | | 0.00E+00 | | 3.44E-11 | |
| LT-3-8 | | 0.00E+00 | | 3.44E-15 | | 0.00E+00 | | 3.44E-15 |
| LT-3-10 | 0.00E+00 | | 1.25E-14 | | 0.00E+00 | | 1.25E-14 | |
| LT-3-11 | | 0.00E+00 | | 1.25E-18 | | 0.00E+00 | | 1.25E-18 |
| LT-3-14 | | 6.30E-16 | | 6.30E-16 | | 9.31E-16 | | 6.30E-16 |
| LT-3-17 | | 2.69E-14 | | 2.69E-14 | | 3.81E-14 | | 2.69E-14 |

The cutsets for these sequences are in Attachment 3.

Sequences LT-2-4, LT-2-29, LT-3-4, and LT-3-16 were omitted from this table. These sequences are for liner plate failure and did not quantify at a truncation of 1E-20.

The CDF and LERF values listed in this section are for the truncation limits shown in section 5.8. If the truncation limit is 1E-12, same as that used for the one top model, the sum of the small LOCA events compare well with the one top model results in section 5.3. The following table lists the sequence results with the 1E-12 cutoff.

| | Pre Modification | | Post Modification Mean Op. Error | |
|---|---|---|---|---|
| | CDF | LERF | CDF | LERF |
| LOCA-SM-LQD | 1.87E-10 | 0 | 2.76E-10 | 0 |
| LOCA-SM-STM | 3.79E-10 | 0 | 3.83E-10 | 0 |

## 5.7 ATWS Sequences

The CDF and LERF, Pre and Post Modification (Mean operator error) for the ATWS sequences are shown below:

| | Pre Modification | | Post Modification | |
|---|---|---|---|---|
| Sequence | CDF | LERF | CDF | LERF |
| ATWS_4 | 8.58E-09 | | 8.58E-09 | |
| ATWS_4-LERF | | 2.13E-10 | | 2.13E-10 |
| ATWS_9 | 5.75E-08 | | 1.45E-08 | |
| ATWS_9-LERF | | 6.06E-09 | | 5.43E-09 |
| ATWS_12 | 1.92E-11 | | 1.92E-11 | |
| ATWS_13 | 5.75E-08 | | 5.75E-08 | |
| ATWS_14 | 3.90E-09 | | 3.90E-09 | |
| ATWS_14-LERF | | 9.15E-10 | | 9.15E-10 |

The cutsets for these sequences are in Attachment 3.

The CDF and LERF values listed in this section are for the truncation limits shown in section 5.8. If the truncation limit is 1E-12, same as that used for the one top model, the sum of the ATWS sequences are the same as one top model results in section 5.3. The following table lists the sequence results with the 1E-12 cutoff.

| | Pre Modification | | Post Modification Mean Op. Error | |
|---|---|---|---|---|
| | CDF | LERF | CDF | LERF |
| ATWS | 1.27E-7 | 7.06E-9 | 8.43E-8 | 6.43E-9 |

## 5.8 Truncation Limits for the LOCA and ATWS Sequence Cases (Sections 5.6 and 5.7)

| Sequence | Truncation |
|---|---|
| LT-2-1 | 1E-14 |
| LT-2-2 | 1E-18 |
| LT-2-5 | 1E-14 |
| LT-2-6 | 1E-18 |
| LT-2-8 | 1E-18 |
| LT-2-9 | 1E-20 |
| LT-2-27 | 1E-18 |
| LT-2-30 | 1E-16 |
| LT-3-1 | 1E-14 |
| LT-3-2 | 1E-16 |
| LT-3-5 | 1E-14 |
| LT-3-6 | 1E-17 |
| LT-3-7 | 1E-14 |
| LT-3-8 | 1E-18 |

| Sequence | Truncation |
|---|---|
| LT-3-10 | 1E-16 |
| LT-3-11 | 1E-20 |
| LT-3-14 | 1E-16 |
| LT-3-17 | 1E-16 |
| ATWS_4 | 1E-14 |
| ATWS_4-LERF | 1E-16 |
| ATWS_9 | 1E-12 |
| ATWS_9-LERF | 1E-12 |
| ATWS_12 | 1E-15 |
| ATWS_13 | 1E-13 |
| ATWS_14 | 1E-13 |
| ATWS_14-LERF | 1E-14 |

## 6.0 REFERENCES

1. Calculation EC-EOPC-0519, Rev. 4, 6-6-01
2. Calculation EC-052-1051, Rev. 0, 2-1-01.
3. ATWS Event tree, Attachment 1
4. Human Reliability & Safety Analysis Data Handbook, by David I. Gertman and Harold S. Blackman
5. E-152 sheet 12
6. E-152 sheet 14
7. M1-E41-69
8. EO-000-102 revision 1
9. FF-104140 sheets 5701 and 5702
10. J-802 sheets 3 and 8
11. EC-052-0547 page 72 (HV155F004 is similar to HV155F042)
12. M-157 sheet 8
13. Calculation EC-052-1025, Revision 2 page 63, 7/13/01
14. PLA-5322
15. EO-000-103 revision 1, page 16 is Attachment 2

16. Design Basis Document for High Pressure Coolant Injection System DBD004 page 33

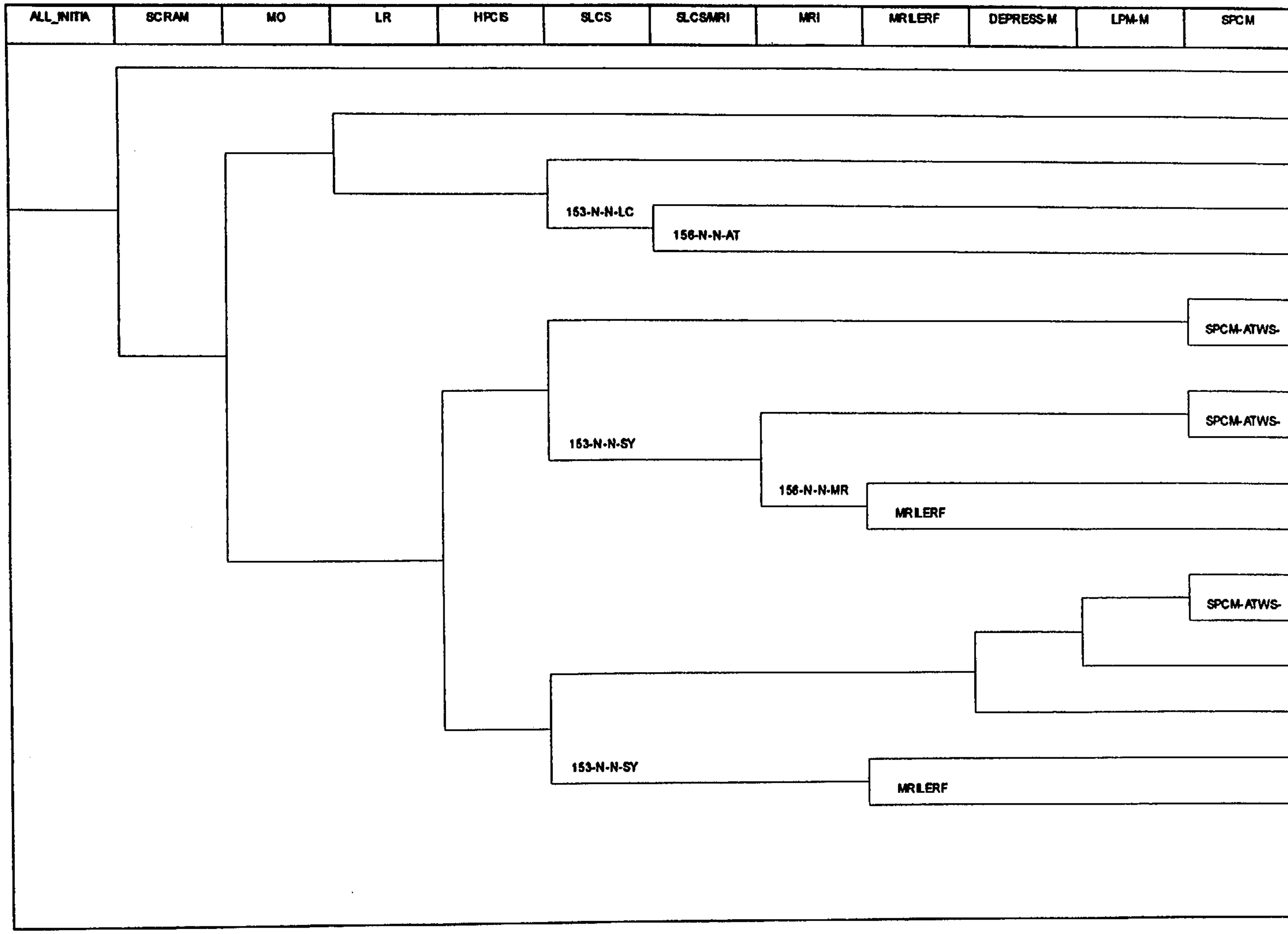
17. CAFTA model and cutsets for all cases (CD) Attachment 3

18. SI-1(2)52-310, Quarterly Calibration of Suppression Pool High Water Level Channels LSH-E41(2)-1N015A&B.

19. SI-1(2)59-306, 24 MONTH CALIBRATION OF SUPPRESSION POOL WATER LEVEL CHANNELS LT-15775A&B (NARROW RANGE)

ATWS Event Tree

Attachment 1




| Class | Frequency | Name |
|---|---|---|
| PDS-1 | 1.00E+00 | ATWS_1 |
| PDS-1 | 0.00E+00 | ATWS_2 |
| PDS-1 | 0.00E+00 | ATWS_3 |
| PDS-2 | 0.00E+00 | ATWS_4 |
| PDS-5L | 0.00E+00 | ATWS_4L |
| PDS-1 | 0.00E+00 | ATWS_5 |
| PDS-9 | 0.00E+00 | ATWS_6 |
| PDS-1 | 0.00E+00 | ATWS_7 |
| PDS-9 | 0.00E+00 | ATWS_8 |
| PDS-2 | 0.00E+00 | ATWS_9 |
| PDS-5L | 0.00E+00 | ATWS_9L |
| PDS-1 | 0.00E+00 | ATWS_10 |
| PDS-9 | 0.00E+00 | ATWS_11 |
| PDS-2 | 0.00E+00 | ATWS_12 |
| PDS-2 | 0.00E+00 | ATWS_13 |
| PDS-2 | 0.00E+00 | ATWS_14 |
| PDS-5L | 0.00E+00 | ATWS_14L |

SP/L-10 MAINTAIN SUPP POOL LVL < 26'
USING:

SUPP POOL CLEANUP SYSTEM
BYPASSING ISO AS NECESSARY
IAW ES-159-002(ES-259-002)
OR
RHR SUPP POOL COOLING LETDOWN
BYPASSING ISO AS NECESSARY
IAW ES-159-002(ES-259-002)

Water level is maintained below the elevation of the bottom of the HPCI turbine exhaust line which begins to flood at a suppression pool water level of 25' 7". The line slopes toward the suppression pool and does not actually fill until approximately 27', therefore a limit of 26' is imposed.

Since removal of water from the suppression pool may be prevented by isolation signals, permission is given in ES-159-002(ES-259-002), Primary Containment Letdown Isolation Bypass, to bypass these isolations.

(Reference: SSES-EPG SP/L-3.2)

SP/L-11 **WHEN** SUPP POOL LVL REACHES **26'**

ENSURE HPCI AND RCIC RUNNING

Intent of this step is to ensure neither HPCI nor RCIC auto-start with a flooded turbine exhaust line.

The HPCI and RCIC turbine exhaust lines begin to flood at a suppression pool water level above 26'. If either were to auto start with a flooded exhaust line, there is no guarantee that the systems would remain functional. Therefore, both HPCI and RCIC are ensured to be running when pool level reaches 26'. If the turbines are running, continued operation with levels above 26' will not result in adverse consequences. Adding heat to the suppression pool from HPCI and RCIC steam turbines is acceptable. Adding water to the suppression pool if HPCI and RCIC are operating with minimum flow valves open is acceptable. If HPCI or RCIC subsequently trip, restart is acceptable if the system is needed for adequate core cooling or pressure control.

(Reference: SSES-EPG SP/L-3.2)

SP/L-12 **WHEN** SUPP POOL LVL CANNOT BE MAINTAINED < **38'**

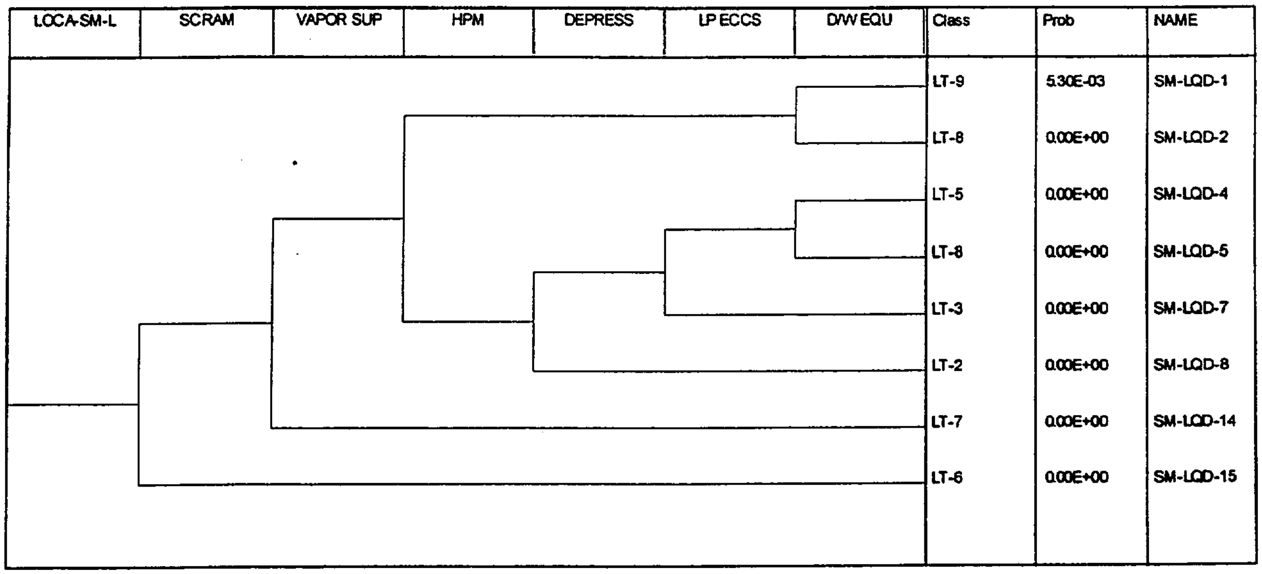
1 GO TO RPV CONTROL

Attachment 3

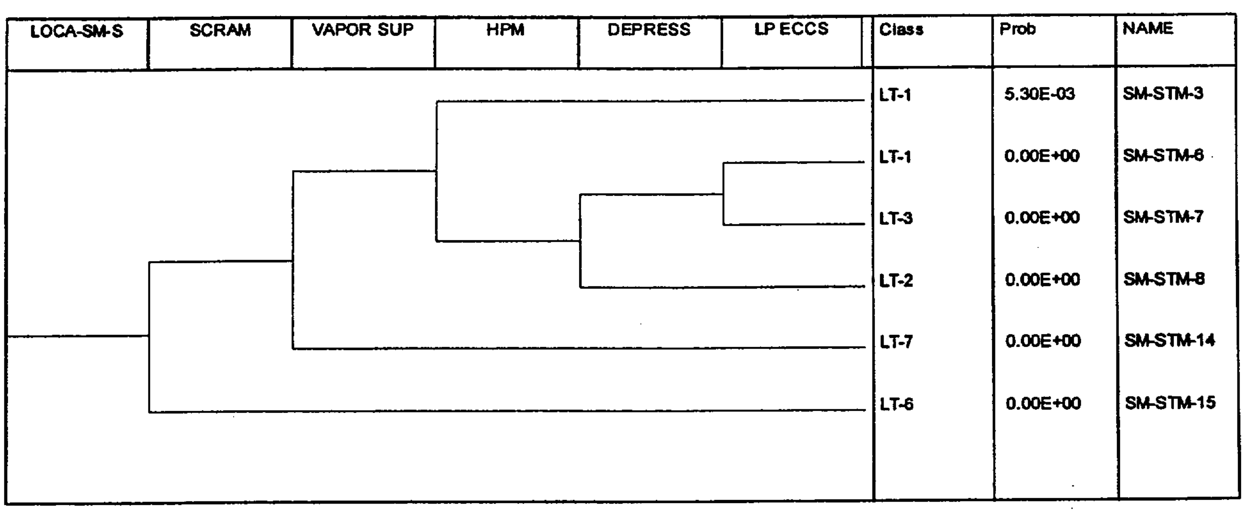
CAFTA Model and Cutsets for all Cases
(see attached CD)

Attachment 4

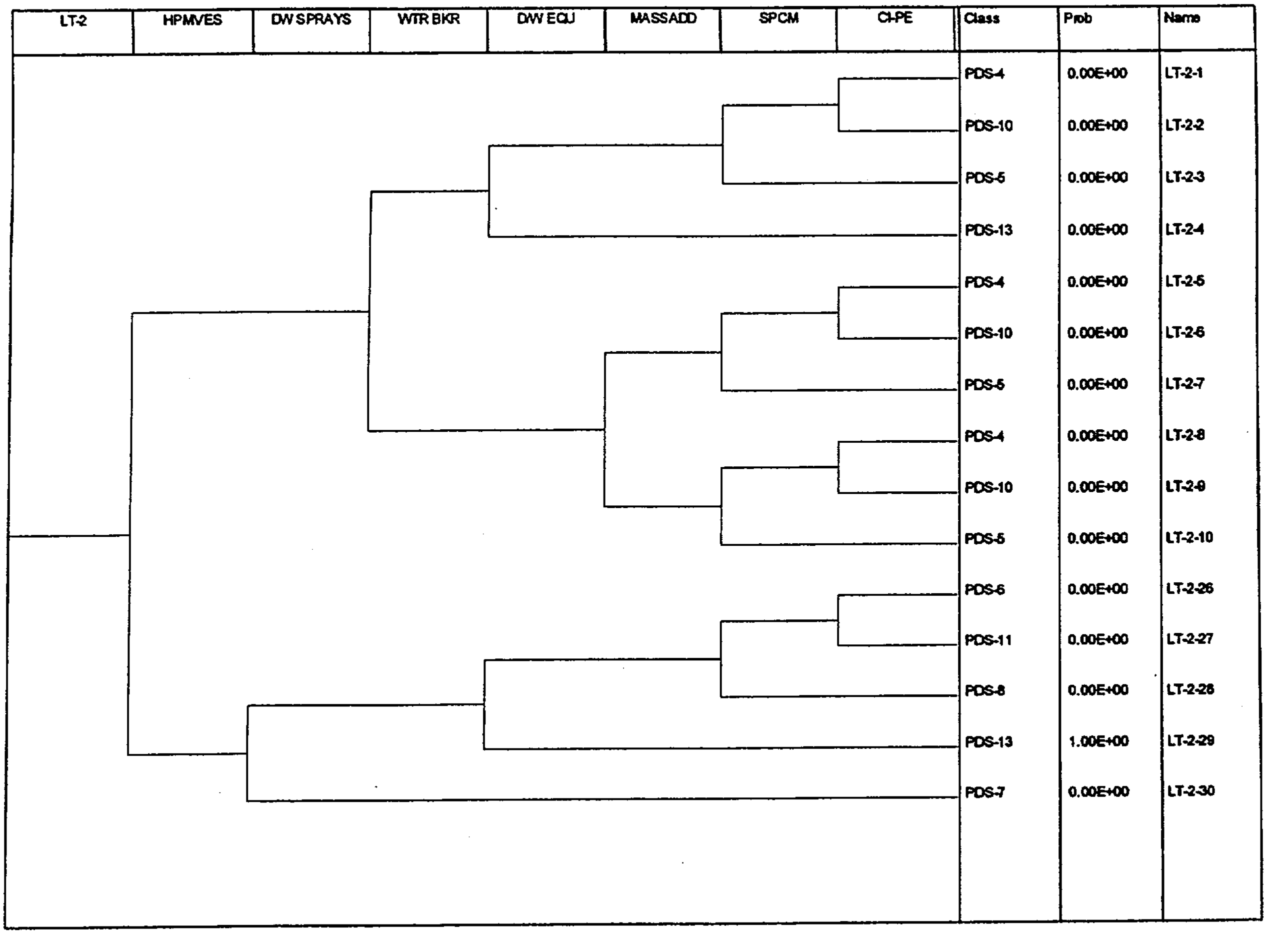
## Event Tree for Small Liquid LOCAs




## Event Tree for Small Steam LOCAs



LOCA transfer 2 (LT-2)




Some sequence number intentionally skipped.

## LOCA Transfer 3 (LT-3)