Britt T. McKinney
Vice President-Nuclear Site Operations

PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3149 Fax 570.542.1504
btmckinney@pplweb.com

ppl™

MAY 0 6 2005

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington DC 20555-0001

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FROM NRC ON PROPOSED RELIEF REQUEST NO. 3RR-01 TO THE THIRD 10-YEAR INSERVICE INSPECTION PROGRAM FOR SUSQUEHANNA SES UNITS 1 AND 2
PLA-5768

Docket Nos. 50-387 and 50-388

Reference: Letter from R. V. Guzman (NRC) to B. L. Shriver (PPL), "Request for Additional Information (RAI) – Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2) – Third 10-Year Inservice Inspection Interval Program Plan RE: Alternate Risk-Informed Selection and Examination Criteria for Pressure Retaining Piping Welds, (TAC Nos. MC1181 and MC1182)," dated April 28, 2004.

This letter is in response to the above referenced letter. Attachment 1 to this letter contains PPL Susquehanna, LLC's (PPL) response to the Request for Additional Information. On June 15, 2004, a teleconference between PPL and the NRC was held to discuss four additional concerns the NRC had. Attachment 2 contains the responses to these concerns. Attachment 3 contains the tables referenced in response to the NRC's verbal questions. Attachment 4 contains revised Relief Request No. 3 RR-01 as a result of the update of the PRA model.

The additional information requested in the April 13, 2005 teleconference has been incorporated in our responses to the previous questions.

PPL has made the following new commitments:

- PPL will review changes to the plant, operating parameters, and PRA model changes for the effects on the RI-ISI Program once per inspection period.
- PPL will resolve the remainder of the Peer Review Level B Facts and Observations prior to the next scheduled model periodic update.

If you have any questions, please contact Mr. C. T. Coddington at (610) 774-4019.

Sincerely,

B. T. McKinney

A047

Attachment 1: Response to NRC Request for Additional Information Relating to Relief Request No. 3RR-01
Attachment 2: Response to NRC Verbal Request for Additional Information Relating to Relief Request No. 3RR-01
Attachment 3: Section 6 Tables
Attachment 4: Relief Request No. 3 RR-01, Revision 1

Copy: NRC Region I
Mr. A. J. Blamey, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

Attachment 1 to PLA-5768

Response to NRC Request for Additional Information Relating to Relief Request 3RR-01

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATING TO RELIEF REQUEST 3RR-01

NRC Question No. 1:

Regulatory Guide (RG) 1.178, "An Approach for Plant-Specific Risk-Informed Decision Making for Inservice Inspection of Piping," Revision 1, dated September 2003, replaced the original "For Trial Use" RG 1.178, dated September 1996. RG 1.178, Revision 1, includes guidance on what should be included in Risk-Informed Inservice Inspection (RI-ISI) submittals. Particularly, in RG 1.178, Section 4.1, the following information is requested:

> "A description of the staff and industry reviews performed on the [Probabilistic Risk Assessment] PRA. Limitations, weakness, or improvements identified by the reviewers that could change the results of the PRA should be discussed. The resolution of the review comments, or an explanation of the insensitivity of the analysis, used to support the submittal, should be provided."

a) Please briefly describe all weaknesses and limitations identified by the Nuclear Regulatory Commission (NRC) staff during the review of the individual plant examination (IPE) and how these issues have been resolved or an explanation of the insensitivity of the analysis used to support the submittal to the comment. Your submittal also described an expert review on May 29, 1997. Please provide any weakness or limitation identified by the expert and how these issues have been resolved or an explanation of the insensitivity of the analysis used to support the submittal to the comment.

b) Has your PRA been peer-reviewed by one of the industry-based groups using a format similar to Nuclear Energy Institute (NEI) 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance (Revision A-3)? If so, please provide the facts and observations that the peer review team identified as important and necessary to address (Significance Level A and B in NEI 00-02) and describe how these issues have been resolved or provide an explanation of the insensitivity of the analysis used to support the submittal to the comment.

c) If your PRA has not been peer-reviewed by one of the industry groups, please explain how the reviews that have been performed provide confidence that the quality of the PRA is sufficient to support your RI-ISI analysis.

PPL Response:

NRC Question Number 1 concerns the quality of Susquehanna's PRA. The responses include resolution of the IPE SER identified weaknesses, results of the Peer Review, and detailed discussions of the Peer Review B Level Fact & Observations (there were no A Level F&Os).

The peer-reviewed model w/ improvements, which was used for this application was not a simple extension of the original IPE. The current model is based on success criteria developed from extensive thermal hydraulic calculations for post power uprate conditions (i.e., current fuel type and current rated power). Using the new success criteria, new event trees were developed based on the calculated accident progression and current emergency operating procedures.

Resolution of IPE SER Weaknesses

PPL's Individual Plant Evaluation (IPE) was submitted to the NRC and received an SER on August 11, 1998. There were three weaknesses identified in the SER. The model used for this submittal, as well as the Peer Review model, contain significant modifications from the IPE model in response to these SER comments. Excerpts from the 1998 SER describing these weaknesses and the model changes made as a result are given below.

Identified Weakness #1

"In the licensee's analysis, the accident sequence progression was terminated if the containment failed prior to core damage; all sequences were then assumed to go to core damage in the reported CDF. Radionuclide releases were not calculated for these containment failures nor was a detailed understanding of the plant response obtained."

Response

Subsequent to the SER on PPL's IPE, substantial changes to the event trees were made that addressed the issue of accident sequence progression. In PPL's Peer Review model and the model used for this submittal, events progress beyond containment failure given no prior core damage. In the case of containment failure and no prior core damage, available sources of injection into the core are evaluated. If injection is successful, the end-state is no core damage and containment failure. If injection is not successful, core damage occurs and the sequence can go to a LERF end-state depending on the sequence timing.

The event trees used for the Peer Review and for this submittal include injection from sources outside the reactor building given containment failure and no prior core damage. The success criteria are based on detailed thermal hydraulic analyses. The event trees are also annotated with the timing for a General Emergency declaration and timing for containment failure and core damage, if it occurs. Thus, the sequence can be readily identified as a LERF sequence if appropriate. The event tree logic is reflected in the fault tree model.

It should be noted that for the Peer Review model and the model used for this submittal, a Large Early Release is conservatively considered to be any release that meets the "early" requirements. In the new level 2 model being developed to support License Renewal and Extended Power Uprate, LERF is expected to be approximately three times lower than the current value. Thus, the new level 2 model is expected to reduce the calculated ICLERP values used to determine the needed weld inspections.

Identified Weakness #2

"The impact on conditional containment failure probability of some severe accident phenomena and resulting containment failure modes appear to have been understated. As a result, all early and late containment failures, other than the containment failures resulting from loss of DHR discussed in item 1 above, are reported by the licensee to occur in less than one percent of core damage events, including ATWS and station blackout.

Appendix 1 to GL 88-20 recommended that licensees consider a maximum coolable debris bed to be 25 cm. For depths in excess of that (as proposed by the SSES IPE) both coolable and noncoolable outcomes should be considered and documented, even in the presence of a water layer provided by the drywell sprays, because of the possibility of the formation of a noncoolable debris crust. Noncoolable outcomes may lead to the occurrence of phenomena such as containment overpressure failure from noncondensible gas generation due to core-concrete interaction or containment failure from corium attack on the drywell liner/concrete containment boundary.

The licensee assumed, however, that core debris released from the vessel post-accident will always be quenched on the drywell floor and, consequently, core-concrete interactions with the drywell floor, steel liner, or concrete containment will be prevented, as long as the drywell sprays provide a water pool on the drywell floor. Similarly, core debris attack on other structures, such as the downcomer vents, resulting in suppression pool bypass or loss of pool scrubbing, would not be possible, according to the licensee, given spray operation. Additionally, the licensee did not consider the possible negative effects of water on the drywell floor, such as containment pressurization due to ex-vessel steaming resulting from fuel-coolant interactions."

Response:

Subsequent to the SER on PPL's IPE, substantial changes to the event trees were made that address the issues of containment failure modes. The current SSES PRA model considers the following containment failure modes:

a. Containment Overpressure
b. Containment isolation failure
c. In-vessel steam explosion (Alpha Mode failure)
d. Ex-vessel steam explosion (Shock loading)
e. Direct containment heating (DCH)
f. Failure Induced by Corium Attack on the Containment Structures, including:

    1. Drywell head flange failure
    2. Loss of vapor suppression due to downcomer melt through
    3. Drywell liner melt through
    4. Overpressure failure due to non-condensable gas generation

The Susquehanna containment design is not susceptible to in-vessel and ex-vessel steam explosions. In addition, evidence from NUREG/CR-5623 exists to show that any core debris generated is not expected to cover a uniform area greater than that extending to the innermost ring of downcomers. Therefore, the drywell liner is not susceptible to failure in the event of vessel melt-through. Each of the other containment failure mechanisms is considered in the current PRA model.

NUREG/CR-5623 calculates containment conditions for core melt core-concrete interaction and the production of non-condensable gases. These calculations conclude that containment pressure will remain less than the ultimate pressure capacity, as long as sufficient drywell spray is available to establish a water pool on the drywell floor up to the downcomer overflow. The drywell spray flow must also continue in sufficient quantity to remove decay heat from the corium. This drywell spray requirement is transferred to the event tree model by requiring that the containment spray function be available in sufficient time to generate the required pool on the drywell floor prior to reactor vessel failure.

Under LOCA sequences, a further requirement for containment integrity is that the vacuum breakers between drywell and suppression chamber are required to operate following the initiation of the containment spray function. It is assumed in the LOCA evaluations that, at the time when drywell spray is initiated, the drywell will be devoid of non-condensable gases and filled with steam from the break. Therefore, the drywell spray will cause a rapid drywell depressurization and at least one vacuum breaker must operate in order to prevent containment failure resulting from implosion.

Based on this discussion, it is concluded that the current Peer Review PRA model does include both the coolable geometry issue and the potential negative effects of water vapor and noncondensable gas generation following core melt extrusion from the reactor vessel. It should also be noted that for the current model, a Large Early Release is conservatively considered to be any release that meets the "early" requirements. In the new level 2 model being developed to support License Renewal and Extended Power Uprate, LERF is expected to decrease by a factor of three from the current value. Thus, the new level 2 model is expected to reduce the calculated ICLERP values used to determine the needed weld inspections.

Identified Weakness #3

"The treatment of ISLOCA was characterized as limited in the staff's October 27, 1997, SER. The licensee has not revisited its ISLOCA analysis and, consequently, it remains a weakness."

Response

PPL has fully addressed ISLOCA in the model used for this application. PPL has performed a formal calculation to evaluate the initiation frequency of an ISLOCA for the following systems:

- RCIC
- HPCI
- Core Spray
- Reactor Water Cleanup
- RHR

PPL has included in the model ISLOCA initiators, which are greater than the ISLOCA cutoff frequency outlined in NUREG-CR-5928, "ISLOCA Research Program," published July 1993.

The contribution of ISLOCAs to the CDF is about 4% and the contribution to the LERF is about 10%. The location of the ISLOCA in both cases is from the RHR system.

Peer Review Results and F&O Dispositions

The risk model version used for the September 16, 2003 ISI submittal is a prior version of the peer-reviewed version. Subsequent to the peer review, additional changes were made to the peer-reviewed model to address the more significant peer review Facts and Observations (F&Os). The risk-informed ISI program as described in RR-01 has been revised to incorporate the post peer review model results. All affected phases of the RI-ISI methodology have been revisited including the Consequence Evaluation, Risk Ranking, Element Selection, and Risk Impact Assessment. The final selection results are presented in a series of tables included as Attachment 3 to this response.

Subsequent model revisions will be evaluated for their impact on the program. This process will occur as part of the requirement to maintain the RI-ISI Program as a living program which reviews impacts of new PRA models, industry experience, plant events, new degradation mechanisms, etc. Should new model revisions affect the consequence evaluation in the SSES RI-ISI Program, the program will be updated to reflect those changes including the impacts on risk ranking and element selections.

The October 2003 BWROG peer review provided PPL with Level B, C, D and S F&Os; PPL did not receive any Level A F&Os. PPL has resolved the Level B F&Os that were determined to be the most significant in their effect on PRA results (more than half of the Level B F&Os) before the model was released. The remainder of the Level B F&Os will be resolved prior to the first scheduled model periodic update.

The peer review team used Revision A-3 NEI draft "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," NEI 00-02 dated June 2, 2000 as the basis for the review.

The Peer Review process uses grades to assess the relative technical merits and capabilities of each technical element and sub-element reviewed. The grades and criteria were developed, in the BWROG program, considering attributes of a PRA necessary to ensure quality, elements of a PRA that are critical to its technical adequacy, and elements needed to support PRA applications. The grades and criteria, which have been adopted for this program, provide guidance on appropriate use of the information covered by the sub-element for risk-informed applications, and convey the ability of the PRA sub-element to support particular types of applications. Four grade levels are used to indicate the relative quality level of each technical element and sub-element based on the criteria at hand. The grading and criteria are:

Grade 1 – Supports Assessments of Plant Vulnerabilities
Grade 2 – Supports Risk-Ranking Applications
Grade 3 – Supports Risk Significance Evaluations w/Deterministic Input
(Risk-Informed Decisions)
Grade 4 – Provides Primary Basis For Application (Risk-Based Decisions)

It is important to note that the PRA does not receive one overall grade. Each element is graded based on the criteria for the element. Then, based on the criteria grades, a summary grade is provided for each of the eleven technical elements.

The minimum grade, the average grade, and the summary grade for each of the eleven elements are listed in the following table along with the overall assessment (extracted from the 2003 Peer Review Report):

PRA PEER REVIEW REPORT			
OVERALL ASSESSMENT			
PRA ELEMENT	GRADE BASED ON SUB-ELEMENTS		
	Minimum	Average	Summary
Initiating Events	2	2.86	3
Accident Sequence Evaluation	2	2.92	3
Thermal Hydraulic Analysis	2	3.00	3
System Analysis	3	3.04	3
Data Analysis	2	2.94	3
Human Reliability Analysis	2	2.89	3
Dependencies	2	3.00	3
Structural Response	3	3.40	3
Quantification	2	2.97	3
Containment Performance	2	2.57	2
Maintenance & Update	2	2.27	2

Overall Assessment: Based on the PRA Peer Review Team review, the PRA can be effectively used to support applications involving absolute risk determination. The Level 1 PRA is fully supportive of Grade 3 applications when the footnotes identified on sub-elements are dispositioned. Level 2 is a useful screening tool to assess applications.

Areas Requiring Enhancement:

Re-examine the following specific issues.

<u>Conservatisms</u>:

- The HRA Peer review identified the quantitative assessment of dependencies among HEPs as an area potential of improvement that could reduce excess conservatisms for absolute risk determination.
- Reassess the DCH conditional probability.
- Reassess the over-temperature failure assumption used in Level 2.
- LERF and CDF definitions.

PRA PEER REVIEW REPORT
OVERALL ASSESSMENT
<u>Non-Conservatisms</u>: SBO events may have sequence dependencies not fully accounted for. This may adversely impact the SBO sequence frequency. <u>Other Issues</u>: The accident sequence evaluation should be reviewed to ensure that the key safety functions are included [e.g., consider including reactivity control, SRV reset (i.e., no SORV) for ATWS, and CRD as a long-term "required" injection method] in those sequences that would challenge the safety functions. A search for plant-unique uncertainties and the associated sensitivity studies to support the uncertainty ranges should be performed. The Level 2 analysis has a number of items that would appear useful to re-examine. These include: • Inclusion of containment isolation in selected sequences. • Inclusion of energetic failure modes including hydrodynamic loads. • Removal of excess conservatisms in the LERF definition.
Areas Recommended For Enhancement: See Facts and Observations sheets for specific recommendations.

The Level B Facts and Observations were divided into two groups according to their proposed implementation schedule. The group labeled "Closed/Resolved in current PRA (FEB05)" contains the F&Os that were resolved prior to releasing the current model version for use. The group labeled "Do for first periodic update" contains the F&Os that will be resolved prior to the model release following the first scheduled periodic update.

The Level B F&Os by group and their current dispositions are provided in Appendix A.

As a result of the Peer review (per NEI 00-02) results and the resolution of the significant Level B F&Os as described in Attachment A, the current version of the Susquehanna PRA model is adequate to support this application.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: AS Sub-element: 5 Observation 1A INDEX: 2

250 VDC Load Shed

One of the assumptions used in the model is that procedure EO-100-030 is implemented to shed 250 VDC loads. There is currently not an explicit HEP (human error probability) in the model to represent the failure of this action and the consequential inability to achieve at least 4 hours of HPCI/RCIC operation. The procedure directs this to be accomplished after 30 minutes and before 45 minutes.

Disposition:

Created new HEP where the operator fails to shed 250 VDC loads. This 250 VDC load shed only impacts Unit 1. Unit 2 does not require 250 VDC load shed because Unit 2 has a separate non-1E battery bank. Incorporated the new basic event into the PRA model and updated the HRA Notebook with all information relevant to this HEP. This F&O and resolution is a duplicate of F&O Index 59.

Element: AS Sub-element: 5 Observation 1B INDEX: 3

Control of HPCI/RCIC

After 4 hours into an SBO (station blackout), 250 VDC may be unavailable. This creates the need to control HPCI and RCIC flow such that they do not trip and require restart. The ability to perform such control actions does not appear to be included as a HEP.

Disposition:

An HEP for operator failure to control level was developed, analyzed, documented, and included in the PRA model. Nothing further required for this F&O.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
<u>Closed/Resolved in Current PRA (FEB05)</u>

Element: AS **Sub-element:** 5 **Observation** 4 **INDEX:** 5

ATWS – Sequence TR-6, TR-7

These sequences assume HPCI operated initially but SLC has failed and Manual Rod Insertion (MRI) is underway. If such a scenario could be successful, it would likely make pool temperature above HCTL (Heat Capacity Temperature Limit).

The SSES EOPs deviate from the BWROG recommended guidelines by allowing operation under ATWS conditions above PSL (Pressure Suppression Limit) and HCTL. The consequences of subsequent RPV emergency depressurization due to low RPV water level does not currently account for the plant conditions above PSP and above HCTL on the accident sequence impacts.

Disposition:

The ATWS event tree has been revised to require success of high-pressure injection and suppression pool cooling in order to have a successful outcome for sequences where SLCS is failed and MRI is available for reactor shutdown.

Simulation of reactor shutdown with MRI shows that pool temperature is well above the HCTL (Heat Capacity Temperature Limit) and suppression chamber pressure is well above the Pressure Suppression Limit. If high-pressure makeup were to fail in an accident sequence where MRI alone accomplishes shutdown, it is likely that RPV depressurization would cause containment pressure to exceed 82 psig, the pressure at which SRVs close on insufficient pneumatic supply. This would lead directly to core melt, vessel failure, and containment failure.

Venting of the containment at 65 psig would also be a concern in this situation if sufficient time were available to carry out the venting. Venting would disable all low-pressure ECCS due to the harsh environment in the Reactor Building. Consequently, there are no ATWS success paths that involve failure of high-pressure makeup and SLCS, which is reflected in the event trees.

**Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)**

Element: AS Sub-element: 5 Observation 10 INDEX: 11

The RPT (recirculation pump trip) is credited in ATWS to prevent early core damage. There is logic to generate an RPT on Level 2, end of cycle (EOC) turbine trip[1], and high RPV pressure. The Level 2 trip occurs too late to be effective in preventing very high RPV pressure under certain accident sequences. Therefore, it should not be credited in the model[2]. The risk model credits the high RPV pressure trip and the EOC RPT. The present structure of the model has these two trips as redundant methods for the RPT. The fault tree should be revised for the RPT to remove the EOC RPT for non-turbine-trip events. The PRA group identified this would be incorporated into the model.

Disposition:

PPL agrees with the comment that the Rx Level 2 trip will come in too late to be effective for mitigating an ATWS and that the EOC RPT (End Of Cycle - Recirculation Pump Trip) is ineffective for non-turbine-trip events.

The resolution of the Rx Level 2 issue requires no changes to the RPT logic. The fault tree does not credit Rx Level 2 for RPT. However, Rx Level 2 was credited in the PRA for automatic ARI (alternate rod insertion) initiation. PPL's review of Level 2 for ARI automatic initiation indicated that this input should be removed, since the reactor may not reach Level 2 for some ATWS transients (e.g., those with feedwater available). Hence, the Rx Level 2 gates were removed as inputs from the ARI logic gates.

The resolution of the EOC RPT issue required adding input to fail the EOC RPT "OR" gate for non-turbine-trip events. Gate %1MSIVATWS was added as input to EOC RPT. %1MSIVATWS is an "OR" gate including all initiators that would close the MSIVs (i.e., non-turbine-trip events).

The described changes have been incorporated into the current PRA.

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[1] Turbine stop valve position.
[2] This has been confirmed by the PRA group.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: AS Sub-element: 6 Observation 1 INDEX: 13

SRVs

The successful prevention of overpressure failure under ATWS conditions requires RPT and SRVs opening. The ATWS event tree should include both.

Disposition:

The ATWS event tree has been revised to require successful RPT and SRV operation in order to have a successful outcome. The ATWS event tree in the revised Event Tree Notebook contains a branch which goes to core damage, vessel failure, and containment over-pressure failure if the ATWS RPT and a sufficient number of SRVs are not both successful.

Element: AS Sub-element: 6 Observation 3 INDEX: 15

ATWS (E.T. Notebook, p. H.2 and p. H.21) Sequence TR-6-1

End State sequence TR-6-1 appears to be optimistic given the fact that no reactivity control method has been successful.

It is judged important to incorporate an evaluation of a successful reactivity control method before assigning success.

Disposition:

A requirement for reactivity control, either SLCS or MRI, was added to the ATWS event trees replacing TR-6-1 with three new sequences. The three new sequences are: level reduction with SLCS success (no core damage), level reduction with SLCS failure and MRI success (no core damage), and level reduction with both SLCS and MRI failure (core damage).

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: AS **Sub-element:** 6 **Observation** 4 **INDEX:** 16

ATWS

There are a number of functional failures that are not addressed in the ATWS event tree. These include the following:

- Reactivity Control for Main Condenser Available
- Failure of all High Pressure and Low Pressure Injection

Initiation of Containment Vent and consequential failure of ECCS is not asked on Branches 27, 29, 37, and 39 of the ATWS tree where pressure is above 82 psig. The procedural direction to open the containment vent does not appear to be accounted for in the ATWS scenarios for Branches 27, 29, 37, and 39. This could lead directly to core damage due to the loss of ECCS makeup.

Disposition:

1. A requirement for reactivity control, either SLCS or MRI, was added to the ATWS event trees replacing TR-6-1 with three new sequences. The three new sequences are: level reduction with SLCS success (no core damage), level reduction with SLCS failure and MRI success (no core damage), and level reduction with both SLCS and MRI failure (core damage).

2. A branch corresponding to failure of all high-pressure and low-pressure injection has been added to the ATWS event tree. The additional branch is Branch 34 described in the Event Tree Notebook.

3. On ATWS Event Tree Branches 27 and 29, the containment vent would not be opened because core damage from power/flow instabilities exists on these branches. As discussed in the Event Tree Notebook, plant procedures recommend against containment venting with core damage. Similarly, core damage exists on Branches 37 and 39. On Branch 37, core damage exists from the operator failing to throttle low-

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

pressure injection after reactor depressurization. While on Branch 39 of the ATWS event tree calculation, core damage exists from operation of a critical reactor in a depressurized state without reactivity control (SLCS is failed and MRI is too slow to stabilize core when depressurization is required).

4. Branch 27 of the peer-reviewed ATWS event tree is equivalent to Branches 39 and 41 in the revised event tree. On Branches 39 and 41, core damage exists and, as discussed above, the containment would not be vented at 65 psig. The equivalent of Branch 29 in the peer-reviewed event tree does not exist in the revised ATWS event tree because credit is no longer taken for MRI when core damage exists. Branches 37 and 39 in the peer-reviewed event tree correspond to Branches 36 and 22, respectively, in the revised event tree. Branch 36 goes to LERF because failure of the operator to control low-pressure injection is now assumed to lead to loss of the RPV and containment integrity. Branch 22 also goes directly to LERF because credit is no longer taken for MRI in scenarios involving RPV depressurization and failure of SLCS. In scenarios where SLCS is failed and RPV depressurization is required, containment pressure will likely exceed 82 psig, the pressure at which SRVs go closed. Closure of the SRVs will cause loss of low-pressure injection, which will lead to vessel failure and containment over-pressurization.

Element: AS Sub-element: 7 Observation 1 INDEX: 18

MRI as an option for successful control of reactivity requires control rods to be individually inserted into the core.

There may be mechanical common cause failure modes that defeat both the scram function and MRI. The combination of all of these mechanical modes of failure (e.g., channel obstruction possibly due to high fuel burn-up effects or interference due to movement of vessel internals) should be factored into the assessment regarding whether MRI offers a truly independent method of reactivity insertion.

Disposition:

Previously at Susquehanna, control rod friction due to channel bow was identified as a potentially significant issue. Although there is no expectation that channel bow would prevent control rods from inserting to at least notch position 02 during a scram, it could cause significant degradation in the insertion speed when rods are driven manually using the CRD system. Calculations show that there is little margin available to the

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

containment venting pressure (65 psig) in an isolation ATWS where SLCS is failed and shutdown is achieved by MRI (manual rod insertion). If control cell friction causes rods to insert significantly slower than the 60 second/rod, then the containment will reach the venting pressure before hot shutdown is achieved. Venting of the containment would lead to failure of RCIC, HPCI, and all low-pressure ECCS. In the ATWS event tree, these sequences proceed to LERF. In order to account for failure of MRI to achieve shutdown before the containment vent pressure is reached, a failure probability of 0.5 associated with control cell friction is included in the MRI fault tree. The probability of MRI failure due to movement of vessel internals is expected to be in order of magnitude smaller than for channel bow and, therefore, this effect is already included in the specified failure probability of 0.5.

Discussion addressing MRI failure due to high control cell friction has been included in of the Event Tree Notebook.

Element: AS **Sub-element:** 7 **Observation** 4 **INDEX:** 21

RPV Rupture

The excessive LOCA evaluation has been included as an initiating event in the quantification. Core damage and LERF is assumed. This is conservative because core spray would be a potential success for prevention of core damage by design of the core spray system. Containment should remain intact and capable of mitigating the event, i.e., vapor suppression is adequate for mitigation of the initial pressurization for the spectrum of excessive LOCAs, except possibly the largest of postulated instantaneous ruptures of the RPV.

Disposition:

The evaluation for peak containment pressure following a complete reactor vessel rupture has been written into the Event Tree Notebook (Appendix O). The conclusion is that peak containment pressure exceeds 250 psig following complete reactor vessel rupture; therefore, reactor vessel rupture leads directly to LERF. The frequency for reactor vessel rupture has been documented in the initiating event notebook.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: AS Sub-element: 8 Observation 3 INDEX: 24

ATWS Event Tree (Appendix H) Section H.23

The discussion of the low pressure makeup use in ATWS response is subject to the following comments:

- The success of LP (low pressure) injection conflicts with discussions in Section 2.8 of the main report.
- The sequences with controlling RPV level too low are neglected as probabilistically insignificant.

The assertion that containment failure can be prevented even though there is a loss of control of low-pressure injection would appear optimistic without significantly more analysis regarding boron washout, RPV integrity during the reactivity excursion, and the power level following loss of low-pressure injection control.

Disposition:

1. The conflict between the discussion in Section 2.8 of the Event Tree Notebook and the success criteria for low-pressure injection during ATWS appears to be caused by unclear wording in Section 2.8. Based on wording in the EOP calculation that formed the basis of the event tree success criterion, it could have been concluded that use of LP ECCS always leads to early containment failure and core damage regardless of RPT success, but this was not the intent. The wording in Section 2.8 has been clarified to indicate that low-pressure makeup cannot prevent core damage if the RPT is failed.

2. Accident sequences that lead to core damage from insufficient low-pressure injection have been added to the ATWS event tree. The ATWS event tree also includes sequences that lead to core damage if the operator fails to take control of LP RPV injection.

3. The ATWS event tree has been revised to specify core damage, vessel failure, and COPF (containment overpressure failure) if the operator fails to take control of low-pressure makeup.

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Element: AS **Sub-element:** 9 **Observation** 1 **INDEX:** 26

Injection Without Heat Removal

Sequences involving no available heat removal result in SRVs reclosing as containment pressure exceeds 82 psig. For such sequences, a high pressure injection source is required for core damage prevention. CRD is such a viable long-term injection source.

CRD should be credited consistently in the ability to prevent core damage when no heat removal is available and adequate core cooling has been maintained for an extended time by other means.

For TR-3 Branch 35 – Only CRD is a success?
For TR-8, Following Branch 14 – Should CRD be credited as a success?

On Branch 35 of TR-3, any of the following are currently credited as success: 1 CRD pump, condensate pump, fire pump, or RHRSW pump. This appears incorrect since the SRVs would reclose on high containment pressure causing SRVs to close and the RPV to repressurize. For TR-3 Branch 35, only CRD is capable of injection prior to containment overpressure failure because the RPV repressurizes. This node should be re-evaluated because it apparently credits a low-pressure system as a success (i.e., RHRSW).
On Branch 14 of TR-8, availability of 1 CRD pump would provide success, but at this time it is only credited on Branch 1 along with the other low-pressure injection systems. The fact that CRD will continue to inject after SRVs close on High DW (drywell) pressure is not included in the event tree logic. It is recommended to include CRD as a separate top, after the containment vent top. If CRD is available and the vent fails, then core damage could be avoided on the COPF branch. For TR-8, CRD would be a success following Branch 14. This should be credited. This will reduce conservatism in this sequence. A branch for late injection should be added to credit CRD here.

PPL indicated that this is currently under investigation to be added to the model.

Note: CRD pumps are located in the Turbine Building and are therefore not subjected to the adverse environment in the Reactor Building following vent or containment overpressure failure (COPF).

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Disposition:

Event Tree TR-2, High-pressure boil off, has been modified to include a top event which checks for availability of 2 CRD pumps to save the vessel in scenarios where HPCI, RCIC, FW, and ADS are failed.

Success criteria for extended high-pressure makeup (HP makeup after 4 hours) in the Event Tree Notebook have been revised to include 2 CRD pumps. Two CRD pumps can maintain the core covered at high reactor pressure for times greater than 4 hours.

Therefore, the extended high-pressure makeup top event (LATE_INJ2) has been revised to include functional success (i.e., no core damage) if 2 CRD pumps are available in sequences where the vent fails and SP Cooling is unavailable. Failure of the containment vent leads to DW pressure >82 psig which causes SRVs to close on insufficient gas supply pressure. The reactor repressurizes until SRVs open in safety mode via springs. Injection from 2 CRD pumps prevents core damage in sequences of this type. LATE_INJ2 has been revised to fail injection from Condensate, RHRSW, and fire pumps if the vent fails because SRVs will close and reactor will repressurize. This revision has been incorporated into the event trees (TR-3, TR-5, and TR-8).

It is not necessary to check for availability of CRD injection after Branch 14 on TR-8 because extended high-pressure makeup (high-pressure makeup after 4 hours) has been revised to be successful if 2 CRD pumps are available (see Revision 5 to §A.9). Since it has already been determined that extended high-pressure makeup is failed before TR-8 is entered (determination is made on Branch 10 of main transient tree in Appendix A), it is not necessary to check for availability of 2 CRD pumps again after Branch 14 on TR-8.

Element: AS Sub-element: 22 Observation 1 INDEX: 31

Core Damage

The definition of core damage is critical to the quantification process and the understanding of the resulting risk measures. (As background, see Attachment AS-22A).

The PSA Applications Guide offers a core damage definition of the following:

> A state of "Uncovery and heatup of the reactor core to the point where prolonged clad oxidation and severe fuel damage is anticipated."

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The ASME PRA Standard provides an example definition:

> Collapsed liquid level less than 1/3 core height or code-predicted peak core temperature >2,500°F (BWR).

Finally, an alternative definition of severe core damage used in many BWR PRAs is:

> RPV water level below 1/3 core height;
> AND
> Core nodal temperature (using a nodalization like MAAP) to be greater than 1800°F for more than 1 minute.

To this could be added the criteria regarding excessive reactivity insertion to require it to be less than 280 cal/second.

The Susquehanna PRA uses a core damage definition for ATWS events that: NEDE-24222 demonstrates significant margin to 10 CFR 50.46 fuel limits for non-oscillation ATWS event and these are not considered core damage events in the Susquehanna PRA. However, due to the potential for fuel cladding dryout and clad melt, any ATWS which exhibits unstable core power oscillations is assumed to lead to gross clad failure in multiple fuel pins and is defined as a core damage event.

The above definitions are quite close and all are generally consistent. The Susquehanna definition is the most restrictive and results in the possibility of assigning "core damage" to states where there is large flow/power oscillations ("instabilities") due to ATWS conditions.

Disposition:

A formal calculation was performed to document a revised ATWS core damage criterion for use in the PRA. This criterion is related to the amount of time before feedwater flow is reduced to suppress large power oscillations that can result in excessive cladding temperatures. PPL also defines core damage as core nodal temperature greater than 1800°F.

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Element: DA Sub-element: 15 Observation 2 INDEX: 130

Conditional LOOP

LOOP given a scram and LOOP given a LOCA event have not been included in the model.

Disposition:

The fault tree was revised to incorporate the conditional LOOP given LOCA and LOOP given a trip. The conditional probability for LOOP given LOCA is 2.4E-2 and for LOOP given plant trip is 2.4E-3. The referenced letters from the Office of Nuclear Regulatory Research provide the bases for these numbers. The Kuritzky letter (June 14, 2002) provides a basis for the LOOP given plant trip. The Thadani letter (July 31, 2002) establishes a factor of 10 difference between the two conditional probabilities with the LOOP given LOCA being 10 times higher than LOOP given plant trip. Therefore, the conditional probability of a LOOP given a LOCA is 2.4E-2. Erin Engineering is also using these values in the risk models for the Exelon plants.

Element: DE Sub-element: 8 Observation 1 INDEX: 45

2nd DC Bus Failure

The CCF (common cause failure) of a 2nd DC Bus failing given failure of the first is considered underestimated. Consider use of NUREG-0666 or alternative to assess.

1) Common Hardware Issues
2) Common Environment
3) Crew error is post initiator repair actions.

Disposition:

The final analysis of the F&O concludes that the CCF value used in the model is adequate.

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The CCF number used in the model is not directly comparable to the value listed in NUREG–0666. The NUREG–0666 value, 6E-5, for the CCF of two buses failing is a probability for two buses failing per reactor year and is considered the total probability of two buses failing. The total probability is the sum of the probability of A bus failing and the CCF probability for B bus failing plus the probability of B bus failing and the CCF for the A bus. The two bus CCF value used in the SSES model is 9.88E-9. This number is based on CCF multiplier from NUREG/CR-5485 adjusted for run time common cause failure by dividing the Table 5-11 value by 2 and the independent failure rate of 1.166E-7, reference EC-RLIB-0504 p. 18.

To make a valid comparison, the model number will be adjusted for total probability and expressed in terms of a yearly frequency. Also, NUREG–0666 only addresses a CCF of two buses while the model has CCF for 2, 3, and 4 buses. The CCF for the 3 and 4 buses failing must be added to the CCF for the two buses failing since any failure mode that can fail 3 or 4 buses will also fail two buses.

Model Data

CCF for 2 of 4 buses 9.88E-9
CCF for 3 of 4 buses 4.67E-9
CCF for 4 of 4 buses 2.59E-8

CCF for 24 hours	Total CCF probability for 24 hours	Total CCF probability for one year
CCF probability for 2 of 4 buses * # of combinations of 2	9.88E-9*6 = 5.93E-8	
CCF probability for 3 of 4 buses* # of combinations of 3	4.67e-9*4 = 1.87E-8	
CCF probability for 4 of 4 buses* # of combinations of 4	2.59E-8*1 = 2.59E-8	
Total	1.04E-7	1.04E-7*365= 3.79E-5

Hence, the equivalent "model" CCF is 3.79E-5 and is somewhat lower than the NUREG–0666 value of 6E-5. However, the NUREG number includes common cause due to closing a tiebreaker between DC buses, which is cited as causing most of the two bus failures. Since SSES does not have any tiebreaker between DC channels, this failure mode does not need to be considered. Recognizing that "most" dual failures were

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attributable to closing the bus tiebreaker, it can be reasonably assumed that the other dual bus failures would have amounted to less than 3E-5 per year. Therefore, it is concluded that the SSES dual DC bus failure CCF value of 3.79E-5 compares well with the value from NUREG–0666, 3E-5, and does not need to change as a result of this F&O. Thus, no change to the model is required.

Element: HR Sub-element: 10 Observation 1 INDEX: 53

MRI

The model takes significant credit for Manual Rod Insertion (MRI). SSES has made a plant modification to make this action more efficient and easier to perform. This is a very positive reflection of the active risk management program at PPL.

The MRI action has been reassessed with revised timing by PPL reflecting the power uprate condition and the latest T&H calculations. The HEP was readjusted using the IPE HRA methods to reflect the latest timings (time available). However, the following items are considered not to have been assessed as part of the analysis:

- Confirmation of the feasibility of the assumed manipulation and diagnosis time by simulator observation.

- Confirmation that sufficient manpower is available within the time frame.

- Confirmation that the T&H case performed adequately models that situation.

Specifically, for the events involving MSIV closure, does it include FW coastdown, enhanced CRD injection, maximum HPCI and RCIC flows, failure of CST refill?

Finally, the success of MRI in overcoming the mechanical common cause failure is difficult to assess and has not been attempted by other BWR utilities. It involves an assessment of the conditional failure probability of MRI to insert control rods given a mechanical common cause failure to scram has occurred due to the following:

- Core barrel tilted or loose and was the cause of the control rod and fuel movement that caused binding of the control rods.

- Other mechanical failures that interfere with control rod movement.

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Disposition:

1. Based on simulator data (seventeen data points) for MRI during an ATWS, MRI initiation times range from 5 minutes to 12.5 minutes with only one data point exceeding 12 minutes. In the PRA, Manual Rod Insertion must begin by 12 minutes or it is considered to be failed. The operator failure rate for initiation of MRI within 12 minutes is specified as 0.061 in the PRA (Susquehanna Human Reliability Analysis Notebook). This error rate shows excellent agreement with the available simulator data. Using a log normal distribution, the error rate based on simulator data is 0.066.

2. Simulator exercises demonstrate that sufficient manpower would be available in the control room to initiate MRI during an ATWS event.

3. Thermal-hydraulic calculations for reactor shutdown via MRI account for continued feedwater injection after the MSIVs are closed. In a SABRE code analysis, feedwater continues to inject to the RPV for 100 seconds after the MSIVs are closed. At 100 seconds into the event, the SABRE model indicates that steam line pressure decays to the point where it can no longer power the feedwater turbines. As discussed in calculations supporting the Emergency Procedures, successful shutdown via MRI requires operator action to throttle HPCI injection by 20 minutes. Prior to 20 minutes, full HPCI and RCIC flow (5600 gpm) is assumed. CRD flow is not included in the SABRE Run; however, the CRD injection rate is very small (63 gpm) compared to full HPCI and RCIC flow (5600 gpm). Success of MRI also requires makeup to the CST within 18 minutes using demineralized water transfer pumps and a condensate pump.

4. The PRA has been revised to include failure of MRI due to control cell friction caused by channel bow. MRI failure due to channel bow induced friction is deemed possible and its probability is specified as 0.5. The probability of MRI failure due to core barrel tilt is expected to be orders of magnitude smaller than that assigned for channel bow, and therefore, this effect is already included in the specified failure probability of 0.5.

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Element:	**HR**	**Sub-element:**	**Observation**	**3**	**INDEX:**	**56**

Manual Local Recoveries

There is extensive use of local manual recoveries in the assessment of RHR for suppression pool cooling and RHRSW for very late RPV injection. The following 6 items are of note:

1. The HEPs apply at long times.
2. The HEPs are quite low (6E-4).

   Disposition:

   Both observations are correct. All non-ATWS sequences in the PRA model require local valve manipulation at a time period of greater than 5 hours. Applying Table 5-54 of the Human Reliability & Safety Analysis Handbook; Gertman & Blackman; 1994, the data only goes out to 300 minutes. For HEP evaluation >300 minutes, as is the case in our PRA model, the reference instructs use of the 300 minutes human error probability of 6E-04. For ATWS sequences that would require valve manipulation <300 minutes, values are used from the same referenced table for the appropriate time. The ATWS valve recovery times are logically differentiated in the model as required per the sequence into HEP values corresponding to 2, 3.4, and 5 hours. No model changes required.

3. The HEPs need to be dependent on the HEP for suppression pool cooling initiation (i.e., applies to the use of HEPs for RHRSW injection initiation).

   Disposition:

   An extensive HEP dependency analysis was performed on the PRA model. All significant dependent HEP combinations (HEP combinations recurring in the top 1500 cutsets) have been analyzed and incorporated into the PRA model.

4. The access, cue, timing, training, manipulation time need to be addressed for each valve or group of valves under the assumed conditions.

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Disposition:

The HEPs given to these groups of valves rely not only on the research conducted by Gertman & Blackman (Human Reliability & Safety Analysis Handbook; Gertman & Blackman; 1994), but also on the large time available to complete the local recovery based on thermal-hydraulic accident analyses performed. Manipulation time is assumed negligible when compared to available time. Operator qualification is assumed to be sufficient training (also based on available time).

Valve use and access is described in parts 5 and 6 of this response.

5. Specifically, has the valve been physically manipulated locally to demonstrate that it is feasible to accomplish the assumed action.

Disposition:

The valves have been physically manipulated locally at least once during start-up testing.

6. A specific access related issue that should be addressed on an accident sequence specific basis is the following related to high radiation:

6a. For ATWS scenarios, it should be assumed that noble gases are present in the containment causing both shine and leakage related radiation in the Reactor Building. Under such conditions, access to the SPC return valves and RHR HX valves may be compromised. (See HR-12-4).

6b. For core damage events, the HEPs for local action are even more in question because of the high radiation environment likely to exist.

Disposition:

High radiation considerations in questions 6, 6a, and 6b have been handled as follows: Manual valve recoveries have been logically updated in the model as guaranteed failed in sequences where core damage occurs prior to valve recovery via local manipulation. The assumption in the PRA model is that operators will not operate the valves locally if core damage has occurred.

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Element: HR Sub-element: Observation 4 INDEX: 57

ATWS – RHR Recoveries-Local Manual Actions

For ATWS, we cannot preclude core fuel perforations and radiation in the containment. This plant state may preclude crew actions to effectively complete the local action because of health physics concerns; i.e., high radiation to personnel.

Disposition:

The PRA model was changed such that no credit is given for manual recovery of Rx Building valves if core damage has occurred. Impact on U1 & U2 PRA models determined to be minimal upon implementation and sensitivity analysis. Manual recovery of Rx Building valves is credited if core damage has not occurred.

Element: HR Sub-element: 16 Observation 1 INDEX: 58

Control of HPCI/RCIC

After 4 hours into an SBO with successful load shed, 250 VDC may be unavailable. This creates the need to control HPCI and RCIC flow such that they do not trip and require restart. The ability to perform such control actions does not appear to be included as an HEP.

This same issue may also be present prior to 4 hours in an SBO without successful 250 VDC load shed.

Disposition:

An HEP for operator failure to control level was developed, analyzed, documented, and included in the PRA model. Nothing further required for this F&O.

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Element: HR Sub-element: 16 Observation 2 INDEX: 59

250 VDC Load Shed

One of the assumptions used in the model is that procedure EO-100-030 is implemented to shed 250 VDC loads. There is currently not an explicit HEP in the model to represent the failure of this action and the consequential inability to achieve at least 4 hours of HPCI/RCIC operation.

The procedure directs this to be accomplished after 30 minutes and before 45 minutes.

Disposition:

Created new HEP - Operator fails to shed 250 VDC loads. This 250 VDC load shed only impacts Unit 1. Unit 2 does not require 250 VDC load shed because Unit 2 has a separate non-1E battery bank. Incorporated new basic event into the PRA model and updated the HRA Notebook with all information relevant to this HEP. This F&O and resolution is a copy of F&O Index 2.

Element: HR Sub-element: 17 Observation 1 INDEX: 60

Recovery

Manual manipulation of valves has been included in the model with very high reliability.

The valves and their manipulation need to be examined for:

1. Accessibility
2. Time Available
3. Cue
4. Time Required
5. Environment

These performances shape factors are not currently documented in the HRA.

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Disposition:

This F&O is covered by F&O Index 56. The analysis and actions performed to resolve the F&O Index 56 correspondingly resolve this F&O as well. See resolution to F&O Index 56.

Element: HR Sub-element: 26 Observation 1 INDEX: 65

ATWS HEP Dependency (See HR-26-2)

The HEPs that model response to ATWS may have dependencies that are not yet explicitly addressed. These dependencies can be incorporated into the model by:

> Making the actions dependent (conditional); "hardwire" the conditional probabilities
> OR
> Performing a second HEP dependent sensitivity case with RPS mechanical failure set higher than 2.1 E-6. This will allow the ATWS HEPs to be included in the top cutsets examined.

Disposition:

A full SSES PRA HEP dependency analysis was completed for all HEPs in the model (which includes ATWS HEP dependencies addressed/questioned in this F&O.) All updated HEP dependency analyses are documented in the HRA Notebook and are included in the model.

Element: HR Sub-element: 26 Observation 2 INDEX: 66

ATWS (See HR-26-1)

- ADS inhibit and SLC Failure may need to be treated explicitly.
- They can show up together.
- Their combination may not have been captured in the dependent HEP assessment.

(There may be a need for a diagnosis error that applies to all ATWS HEP combinations.)

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Disposition:

A full SSES PRA HEP dependency analysis was completed for all HEPs in model (which includes the specific HEP dependencies addressed/questioned in this F&O.) All updated HEP dependency analyses are documented in the HRA Notebook and included in the model.

Element: IE **Sub-element:** 5 **Observation** 3 **INDEX:** 73

Initiating event Loss of Drywell Cooling (LDWC) is not modeled because "The drywell chillers provide cooling to the drywell during normal operation. If they are lost, a manual SCRAM or, ultimately an automatic SCRAM, on high drywell pressure will occur. No safety systems are affected. Loss of the drywell chillers is considered to be bounded by the turbine trip initiating event."

But, HPCI initiation occurs from High Drywell Pressure Relays 95E211K5A/B and 95E211K6A/B. This initiating of HPCI would more likely cause a Level 8 trip, which causes a feedwater trip.

Disposition:

Loss of drywell cooling leads directly to high drywell pressure resulting from the increased drywell temperature. The high drywell pressure condition causes the HPCI system to initiate and begin reactor vessel injection, an event that would be very similar to an inadvertent HPCI startup initiator. The inadvertent HPCI startup initiator is evaluated in Section 15.5 of the SSES FSAR. The reload licensing analysis evaluation of the inadvertent HPCI start event from rated conditions concludes that the level control system is expected to reduce feedwater flow in time to prevent reactor vessel level from reaching the Level 8 trip setting.

The reload licensing analysis also concludes that the inadvertent HPCI start at normal power level is similar to the loss of feedwater heating (LFWH) event. This conclusion is based on the fact that the feedwater flow reduction resulting from decreased feedwater demand combined with the low injection enthalpy of the water injected by the HPCI system will cause an increase in core inlet subcooling. In the event that the Level 8 trip and subsequent scram does not occur (as the full power analysis shows), the reactor stays at power and the event is not considered an initiating event for the PRA. If the lower power HPCI initiator causes a Level 8 trip, the result is a turbine trip. The inadvertent

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HPCI initiation event initiator is already classified as a turbine trip with bypass initiator in the PRA. (See Appendix A, Section 2.6 of the Initiating Event Notebook, EC-RISK-1121, Revision 0.) Thus, no further action is warranted.

Element: IE Sub-element: 6 Observation 2 INDEX: 75

Dual Unit Effects

Dual unit effects and insights with a single diesel operating should be included in the summary notebook discussion (as sensitivities if desired) to address:

Effects of switching RHR high AMP loads.
On RHR Motors.
On D/G.
RWST (refueling water storage tank) adequacy to support.
Loss of SW (service water) on Unit 1.
Loss of Instrument Air on Unit 1 should also be discussed.

Disposition:

A dual unit shutdown with less than 4 diesels would require cycling the RHR pumps on and off in each unit for suppression pool cooling due to the present electrical restrictions on the bus and DG. Only one RHR pump is presently allowed to be on any one channel for both units. This process is required per Susquehanna Operating Procedure. Thus, the dual unit shutdown will not change the generated cutsets (results). This discussion covers the dual unit effects on:

- Effects of switching RHR high AMP loads.
- On RHR Motors.
- On D/G.

The dual unit effects of requiring makeup from the RWST have been addressed in the event tree notebook in the development of the success criteria. Therefore, the dual unit effects on the RWST have been addressed.

The loss of service water and the loss of instrument air on Unit 1 really have no dual unit effects. There is outage capability to cross-tie certain service water loads between units

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but this cross-tie is normally closed and is not credited in the PRA. The instrument air system can be cross-tied between units but this is not normally done and again is not credited in the PRA. Therefore, there are no dual unit effects on service water and instrument air.

Element: IE Sub-element: 15 Observation 1 INDEX: 92

A grid reliability of GR1 was selected for Susquehanna. Do calculations and procedures exist that verify black start capability from off-site power within 30 minutes?

Disposition:

The Initiating Event Notebook only cites GR1 as one of two comparisons to the grid loss frequency used in the model. The main comparison was against actual PJM experience. As discussed in the response to F&O Index Number 88, the Susquehanna LOOP initiation frequencies and recovery times provide reasonable results compared to the results which would be obtained if INEEL/EXT-04-02326, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003" were used as a basis for LOOP initiation frequency.

Element: MU Sub-element: 4 Observation 1 INDEX: 115

The monitoring and collection of new information for an update is not presently a fully implemented and controlled process. The update guidance procedure provides for sending a model update information package to designated site personnel but does not establish a process for interface with the operator training program to ensure that insights are reviewed with the plant operators and EOOS support personnel. This may provide additional feedback pertaining to the fidelity of the PRA model.

Disposition:

Subsequent to the peer review, the above mentioned PRA maintenance and update procedure (NDAP-QA-1002) was formally issued. The purpose of this procedure is to define the basic process used by PPL to develop, control, and update the Susquehanna Probabilistic Risk Assessment (PRA). The procedure provides criteria to determine when updates are needed plus requirements for the PRA group to review changes in plant procedures and plant modifications to ensure the PRA continues to be consistent with the

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as-built / as-operated plant. The procedure also provides requirements for communicating PRA results to the organization, including Training, Work Management, Operations, Nuclear Regulatory Affairs, the Maintenance Rule expert panel, and Station Management. A revision to the maintenance and update procedure made after the peer review requires that the Training Group be informed of significant PRA changes (risk significant systems, risk significant operator actions, risk significant scenarios, etc.).

Training modules on risk concepts have been developed and presented to Engineering, Operators, and the STAs. Other training has been provided to Work Management and the STAs (users of EOOS). Significant changes to the model would be reflected in the training modules.

Element: MU Sub-element: 6 Observation 1 INDEX: 117

No detailed process has been established for the configuration control of the PSA model files including backup, storage and retrieval from a secure controlled location. Also, no formal benchmark process has been established to validate that retrieved model files are satisfactory for use in performing an application.

Disposition:

Three procedures are currently in place for management of controlled model files. These controls will apply to the CAFTA developed model and it's associated files. PPL's SQA procedure is the primary procedure that addresses control of software and data products. This procedure requires that all controlled data sets (i.e., PRA models) be placed in a "QA" data directory. PPL has established a directory that will be used to store the controlled PRA models. The SQA procedure also requires that controlled files must be documented, reviewed, and approved prior to being released for use.

The documentation for the PRA models will be done in accordance with PPL's recorded calculation procedure. Once the controlled files are moved into the QA directory, access permissions are set to "Read Only", preventing unintentional changes and assuring that the files as documented will be the same files that will be used in application calculations.

Finally, the PPL SQA procedure requires that a data custodian be established for all controlled data files. The data custodian will be notified of system or environment changes that may impact the correct operation of the data file. The data custodian will be notified prior to changes to the PPLNet environment so that he or she can evaluate the impact of the change. This protocol will assure that the controlled files will continue to yield the expected results.

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Element: MU Sub-element: 11 Observation 1 INDEX: 119

A process for review of prior PRA applications has not been fully implemented.

Disposition:

A process for review of prior PRA applications has been implemented through procedure NDAP-QA-1002. NDAP-QA-1002 states that: following a PRA Model Update, an information package describing the changes, the new PRA Taxonomy (risk significant operator actions and systems, and most risk significant MOVs and AOVs, ISI inputs, etc.), and the review of previous applications shall be prepared. This information package shall be transmitted to the following individuals via a calculation package so that review by each is formally documented:

1. Manager – Nuclear Fuels & Analysis
2. Manager – Station Engineering
3. Manager – Work Management
4. Manager – Nuclear Operations
5. Manager – Nuclear Regulatory Affairs
6. Manager – Nuclear Design Engineering
7. Manager – Nuclear Training
8. Manager – Quality Assurance
9. General Manager – Nuclear Assurance
10. General Manager – Nuclear Engineering
11. VP-Nuclear Operations
12. Nuclear Records
13. PORC Secretary
14. Supervisor NDE - SSES

Element: MU Sub-element: Observation 1 INDEX: 120

Sufficient documentation reflecting the process used for configuration control of the current PRA model update and maintenance does not exist. This detailed documentation of the update process is important to the configuration control and traceability of the model changes and review process provided for on PRA model update.

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Disposition:

Subsequent to the peer review, NDAP-QA-1002, was formally issued. The purpose of this procedure is to define the process used by Plant Analysis to develop, control and update the Susquehanna Probabilistic Risk Assessment (PRA). Details on the process used to develop the current SSES PRA are provided below.

The current PRA model is documented and controlled under PPL QA procedures. All documentation packages include an independent technical review and final approval by qualified PPL engineers. Extensive model documentation includes:

1) Individual System Notebooks for all key systems important to risk (e.g., HPCI, RCIC, ADS and MSIVs, RHR, Electrical Distribution system, etc.);
2) Event Tree Notebook which documents the accident or transient progression from an initiating event to a plant damage state;
3) Initiating Events Notebook which documents the initiating events which are considered in the Susquehanna PRA and their associated frequencies;
4) Human Reliability Notebook which identifies human actions and their associated failure probabilities;
5) Dependency Matrix Notebook which provides an overall summary of the inter-relationships of plant systems;
6) Internal Flooding Notebook which identifies the impact of internal floods on key equipment and equipment or train availability; and,
7) Summary Notebook which documents the final PSA model including all software files developed as part of the model and the sensitivities on key input parameters.

Changes to any of the above documentation packages is also done under PPL QA procedures. As with the initial preparation, all changes are prepared, independently reviewed and approved prior to releasing the revised model for general use by plant personnel.

Plant procedures are in-place which assure that the Plant Analysis group will be informed of any plant or procedure changes which may affect the current risk model. If changes are warranted, all affected documentation will be revised to assure the PRA reflects the current as-built, as-operated plant.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

The fault tree model and associated databases, which are developed and documented in the packages discussed above, are controlled via applicable PPL QA procedures. These procedures provide requirements and guidance for configuration control. After these files have been developed and approved for use, the model files are stored in special directories to prevent inadvertent changes by users.

The software used for risk analysis is controlled and documented in accordance with PPL QA procedures. These procedures provide requirements that must be met for all quality-related software, including configuration control of the software and future updates. Documentation packages have been developed for all risk analysis software to document the procurement, installation, V&V and configuration control of this software. Changes to the software must be documented in revisions to these software packages and are thus subject to independent technical review and approval prior to their use in risk related analyses.

Element: QU Sub-element: 9 Observation 1 INDEX: 33

Common Cause of 4 EDGs and D/G "E"

The CCF of the 5th D/G may be too conservative. This dependency should be assessed considering diverse features of the D/G "E."

Location
Environment
Manufacturer
Design
Maintenance Practices

Disposition:

This F&O states that the Common Cause Failure (CCF) probability of the fifth diesel may be too conservative. The CCF probability for the diesel generators was developed from NUREG/CR-5497. The CCF probability for the E diesel generator (D/G E) is a conditional probability of it failing given one or more of the A – D diesel generators fails. The CCF probability for the A – D diesel generators is based on a group of four while the CCF probability for the E diesel generator is based on a group of 5. There is not much of an argument to be made for maintaining that the D/G E should have a different CCF probability because:

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

- The D/G E is manufactured by Cooper Bessmer as are the A – D.
- The D/G E model type is KSV as is the model type of the other four, except that D/G E has 20 cylinders and the A – D have 16.
- The same maintenance practices are used on all five DGs.

Therefore, the CCF currently being used is considered appropriate.

Element: QU Sub-element: 14 Observation 1 INDEX: 34

The Summary Document does not identify how circular logic is identified and resolved in the PRA model. A consistent means of highlighting circular logic paths in the model, such as a gate naming convention, is not being employed.

Disposition:

Circular logic breaks are discussed in the Summary Notebook, which has been prepared, reviewed and approved per PPL documentation procedures. However, the model is not completely consistent with regard to a gate naming convention for circular logic. It should be noted that if circular logic exists in the model, the fault tree will not quantify. The naming convention does not affect the model results.

Element: QU Sub-element: 19 Observation 1 INDEX: 35

Designed Documentation

Using the fault tree recovery method allows for sequence based recoveries. This portion of the quantification is the least documented. The tree is large enough to require a documentation section.

Disposition:

The Summary Notebook section discussing recoveries was expanded following the Peer Review to include more detailed discussions of the approaches used for using sequence based recoveries in the model.

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Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: QU Sub-element: 31 Observation 1 INDEX: 39

The PSA results summary should identify dominant contributors.

A detailed description of the Top 10 accident cutsets should be provided because they are important in ensuring that the model results are well understood and that modeling assumption impacts are likewise well known.

Similarly, the dominant accident sequence groups or functional failure groups should also be discussed. These functional failure groups should be based on a scheme similar to that identified by NEI 91-04, Appendix B.

Disposition:

A discussion of the top 10 cutsets is included in the Summary Notebook. The "SCHEME SIMILAR TO THAT IDENTIFIED BY NEI IN NEI 91-04" would require revising the event trees for different plant damage states. The plant damage states as defined for SSES are technically adequate and do not require revision to resolve this F&O.

Element: QU Sub-element: 34 Observation 2 INDEX: 41

The PRA model update is still in progress and will require a comprehensive review once the model is finalized to ensure consistency between the model content and all supporting documentation, including the results presented in the Summary Document.

Disposition:

The Summary Notebook was in draft form when the Peer Review Team evaluated it. It was since issued as a formal calculation (prepared, reviewed and approved per PPL documentation procedures) in April 2004. The model content, supporting documentation, and detailed model results were provided in the calculation.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: SY Sub-element: 5 Observation 1 INDEX: 141

HPCI

For transient events with the flow rate for injection relatively low, HPCI minimum flow valve could remain open and increase the drain rate from the CST.

Disposition:

The relevant technical evaluation in the Event Tree Notebook has been revised to address the effect of HPCI minimum-flow valve operation on CST inventory. The model now includes logic representing the timing evaluations related to the CST drain rates.

Element: SY Sub-element: 13 Observation 1 INDEX: 149

Adequate Inventories

The following "inventories" do not appear to address the demands that may be imposed under accident conditions:

> 250 VDC adequacy (i.e., required DC load shed which is not currently included in the model).
>
> CST/RWST inventory is not explicitly addressed.

Disposition:

The 250 VDC load shed is applicable to Unit 1 only. Unit 2 does not require 250 VDC load shed because Unit 2 has a separate non-1E battery bank. The system fault tree model was revised to include dependency on the 250 VDC load shed by creating the new HEP where the operator fails to shed 250 VDC loads. Incorporated new basic event into PRA model and updated the HRA Notebook with all information relevant to this HEP.

Event Tree Notebook has been revised to address CST/RWST inventory. Effect of HPCI minimum-flow valve operation on CST inventory is also addressed in the Event Tree Notebook. The model now includes logic related to CST/RWST inventory demands during accident conditions.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: TH Sub-element: 4 Observation 1 INDEX: 162

Technical Support (See AS-5-3)

The technical support for some of the success criteria should be re-examined to consider the following issues:

DW/T when recirc pump seal leakage is induced during an SBO.
Effect of minimum flow valve being opened.
Effect of HCL on timing of sequence.

In addition, the description of the procedure directions in an SBO appear to give directions different than those assumed in the T&H (thermal hydraulic) calculations used in support of the PRA sequence for SBO.

Disposition:

Technical evaluation in Event Tree Notebook has been revised to address the effect of recirculation pump seal leakage on Drywell temperature response during a SBO.

Effects of RCIC and HPCI minimum-flow valves failed open on CST inventory are addressed in the revised Event Tree Notebook.

The effect of the HCTL on operation of HPCI and RCIC is also addressed in the revised Event Tree Notebook.

Additional discussion has been provided in the Event Tree Notebook to show that the TH calculations are consistent with the expected response of the plant in a SBO event. The discussion pertains specifically to the situation where the HCTL is reached and RCIC is the only injection system available in the plant. Based on discussion in the SBO procedure, it is not expected that the operator would deliberately depressurize the RPV in this case because the action would lead directly to core melt and vessel failure.

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Disposition of B Level F&Os
Closed/Resolved in Current PRA (FEB05)

Element: TH Sub-element: 8 Observation 3 INDEX: 166

Charger Room Cooling

No evidence of an evaluation of charger room cooling has been performed.

It is noted that the team walkdown of the plant on Wednesday of the visit identified that the chargers were likely not subject to thermal conditions that would induce failure within the PRA mission time despite loss of ventilation based on the size of the room and its normal temperature.

Disposition:

This F&O states that no evidence of an evaluation of charger room cooling was performed. However, a formal calculation had been prepared that addresses the charger room cooling requirements. This calculation concludes that no cooling is required to the charger rooms. The calculation does require that the battery charger room doors be open prior to 6 hours from the time of loss of Control Structure HVAC. A plant off-normal procedure addresses this requirement. Therefore, the charger rooms do not require cooling which is how they are modeled.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Do for First Periodic Update

Element: DA **Sub-element:** 4 **Observation** 2 **INDEX:** 124

A limited set of failure data was updated with plant specific data prior to 1999. The majority of the failure data is based on generic values.

Generic data tends to be more conservative than plant data. Using plant data would also help identify any potential plant outliers.

Develop program to periodically update failure data using accumulated plant data.

Disposition:

PPL intends to develop and implement, prior to a future PRA model update, a program to periodically update component failure data with plant specific data. The program will consider utilizing plant specific data to define failure rates for the most risk significant components. (HPCI, for example, will be considered as a potential candidate for update with plant specific data.)

The 'generic' values currently used in the plant PRA model are accepted industry values. Although utilizing overly conservative component failure data in the plant PRA model can theoretically distort quantification results, industry accepted component failure rates generally have the tendency (as stated in the F&O) of being somewhat conservative relative to plant data. The industry accepted data used in the plant PRA is not considered to be overly conservative (i.e., use of the generic data does not skew the results or the risk insights obtained from the PRA), and is thus deemed sufficient for risk-informed applications.

The use of more plant specific data is not expected to significantly influence the calculated inputs used for the ISI program. Therefore, the ISI consequences are expected to be unaffected.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Do for First Periodic Update

Element: DA Sub-element: 4 Observation 4 INDEX: 126

The plant specific components receiving a data update do not include the HPCI pump which has a relatively high Fussell-Vesely importance.

Include the HPCI pump in the component population for periodic plant specific data update. Consider whether any other components merit plant specific data update.

Disposition:

PPL intends to develop and implement, prior to a future PRA model update, a program to periodically update component failure data with plant specific data. The program will consider utilizing plant specific data to define failure rates for the most risk significant components. (HPCI, for example, will be considered as a potential candidate for update with plant specific data.)

The 'generic' values currently used in the plant PRA model are accepted industry values. Although utilizing overly conservative component failure data in the plant PRA model can theoretically distort quantification results, industry accepted component failure rates generally have the tendency (as stated in the F&O) of being somewhat conservative relative to plant data. The industry accepted data used in the plant PRA is not considered to be overly conservative (i.e., use of the generic data does not skew the results or the risk insights obtained from the PRA), and is thus deemed sufficient for risk informed applications.

The use of more plant specific data is not expected to significantly influence the calculated inputs used for the ISI program. Therefore, the ISI consequences are expected to be unaffected.

Element: DE Sub-element: 7 Observation 1 INDEX: 42

Missing Human Interactions (See also DE-7-3)

The human interactions that can cut across system trains and can cause failure of multiple trains due to pre-initiator should be identified and documented. (See Element HR-26.)

Identify and document pre-initiator unavailabilities and ensure that it is consistently treated for all relevant systems.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Do for First Periodic Update

Disposition:

Twenty-one pre-initiator human errors are currently documented in the HRA Notebook and are included in the plant PRA model. Pre-initiators have been evaluated for the diesel generators, LPCI, RCIC, HPCI, Core Spray, SLC, and CRD. In the model quantification, the pre-initiators contribute 3.66% of the Unit 1 CDF and 3.67% of the Unit 2 CDF with more than half of the contribution coming from the A and B diesel generators. In general, the pre-initiators are comparable to the 16 pre-initiators included in the Limerick PRA model.

With regard to this F&O, SSES HRA pre-initiators are currently deemed sufficient for risk-informed applications. The pre-initiators will, however, be comprehensively reevaluated in a future model update. Adding more pre-initiators is not expected to affect the insights presently realized.

As discussed above, additional pre-initiators are not expected to significantly influence the calculated inputs used for the ISI program. Therefore, the ISI consequences are expected to be unaffected.

Element: HR Sub-element: 4 Observation 1 INDEX: 50

Missing Pre-initiator Human Error Probabilities.

Only a limited number of pre- initiator Human Errors are included in the fault trees.

The pre-initiators included in the model are considered to be adequate except for possible common cause events. However, further consideration of plant specific procedures could identify other pre-initiators for inclusion.

Disposition:

Twenty-one pre-initiator Human Errors are currently documented in the HRA Notebook and are included in the plant PRA model. Pre-initiators have been evaluated for the diesel generators, LPCI, RCIC, HPCI, Core Spray, SLC, and CRD. In the model quantification, the pre-initiators contribute 3.66% of the Unit 1 CDF and 3.67% of the Unit 2 CDF with more than half of the contribution coming from the A and B diesel generators. In general, the pre-initiators are comparable to the 16 pre-initiators included in the Limerick PRA model.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
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With regard to this F&O, SSES HRA pre-initiators are currently deemed sufficient for risk-informed applications. The pre-initiators will, however, be comprehensively reevaluated in a future model update. Adding more pre-initiators is not expected to affect the insights presently realized.

Additional pre-initiators are not expected to significantly influence the calculated inputs used for the ISI program. Therefore, the ISI consequences are expected to be unaffected.

Element: IE Sub-element: Observation 1 INDEX: 88

LOOP Frequencies Developed for Susquehanna are not based on NUREG 1032.

However, EPRI database was used as a source of LOOP data for the Susquehanna area. An attempt was made to sub-divide the LOOP events into grid related, severe weather and extremely severe weather related events. This approach differs from using NUREG-1032 to develop the LOOP frequency and recovery terms. Using NUREG 1032, a value for the plant-centered frequency would be obtained and then using the correlations provided estimates for the grid-related, severe weather and extremely severe weather contributions to LOOP would be computed. The LOOP contributions due to non-plant centered events would be added to the plant centered LOOP frequency to obtain the total LOOP frequency.

Susquehanna started with a total frequency, however, rather than using NUREG 1032 to obtain additional contributions due to rare weather events. NUREG 1032 and Regulatory Guide 1.155 were used to sub-divide the total frequency into plant centered, grid related, severe weather and extremely severe weather related contribution. Comparisons to NUREG 1032 were made to valid results.

Using the Susquehanna approach, if the plant-centered LOOP frequency is 3.0E-02 per year and the plant is susceptible to severe weather events (say once every 50 years) it is likely that a severe weather event would not be included in the prior data distribution, which typically would cover a time span of 10 to 20 years. A 1 in 50 years severe weather event, according to the Susquehanna approach would reduce the plant center LOOP frequency to about 1.0E-02 per year.

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Disposition of B Level F&Os
Do for First Periodic Update

The result of the Susquehanna approach is that the plant-centered LOOP frequency is less for Susquehanna than the national average (1.58E-02/year versus 1.86E-02/year). The Susquehanna plant-centered LOOP frequency is also less than what would be obtained using the 4 of 5 PJM events and the 2.98E-02/year updated mean LOOP frequency (i.e., a plant-centered LOOP of 2.38E-02/year).

Since the rare events (grid related, severe weather and extremely severe weather events) may not be included in the database used for the prior distribution, these terms should be added to the mean LOOP frequency. Since LOOP is a significant contributor to CDF, LOOP frequency/recovery will have a significant impact on results.

Disposition:

The main issue in this F&O is the inconsistency in references for the development of the LOOP initiator frequency and LOOP recoveries. A future update of the model will consider using INEEL/EXT-04-02326, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003" which includes the August 14, 2003 power outage. This source of data has a LOOP initiator frequency specific to SSES and recovery curves for five different causes of a LOOP. The use of this document will provide a consistent data source for the LOOP initiator frequency and recoveries.

An assessment of the impact of the proposed change was performed by running a sensitivity case with the Grid, Extreme Weather and Severe Weather frequencies set to the INEEL values for SSES. To account for the less optimistic recoveries, the least optimistic recovery curve, extreme weather, values were manually inserted for the highest worth LOOP cutsets caused by extreme weather. The result of this effort was an increase of 10% for CDF. It is concluded, from this sensitivity case, that changing to the INEEL data would not result in a substantial change to the model results.

As the CDF and LERF increase as a result of lower LOOP recovery probabilities, the mitigation value of the systems dependent on AC decreases. This decrease is expected to be relatively uniform for all affected systems. This decrease in a system's accident mitigation capability will result in a decrease in ICCDP and ILERP when the system is assumed failed. The fact that the ICCDP and ILERP would decrease makes the current calculated inputs and the ISI consequences conservative.

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Do for First Periodic Update

Element:	**IE**	**Sub-element:**	**5**	**Observation**	**2**	**INDEX:**	**72**

Missing or incomplete documentation for exclusion.

1. Loss of GSW (general service water) is not included in the fault tree. It is assumed to be no worse than the loss of RBCCW or TBCCW. This does not account for the impact on both RBCCW and TBCCW being lost at the same time. If this has been taken into account, then the basis should be documented.
2. Medium Steam LOCA, or SORV3 (3 or more SORVs).
3. Feedwater ramp-up initiator.
4. Reference Leg break initiator should be added to the model.

Disposition:

1. The loss of general service water (GSW), referred to as normal service water (NSW or SW) at SSES, is discussed in the Initiating Events Notebook. The Initiating Events Notebook discussion states that the 'loss of normal service water is subsumed by and conservatively reflected in the loss of offsite power initiator category.'

   The conclusion that the loss of normal service water is subsumed by the LOOP event is based on the fact that the loss of normal service water event has impacts similar to those of the LOOP event (MSIV closure). Loss of normal service water is; however, less severe because the emergency on-site AC power sources are not the only AC power sources required for mitigation. In addition, the event frequency for loss of normal service water is evaluated to be much smaller than the LOOP frequency. Therefore, the loss of normal service water event is assumed to be subsumed by the LOOP event and a separate initiating event is not included in the current model. Since this approach may be slightly conservative, consideration will be given to including the loss of service water event as a specific initiating event as part of a future PRA update.

2. The Event Tree Notebook discusses the LOCA sequences in detail. A determining factor for a steam break is whether or not the high pressure makeup systems (HPCI or RCIC) are sufficient to mitigate the event and prevent core damage. Small steam breaks are defined as those breaks for which the high pressure makeup systems are required for mitigation. Large steam breaks are defined as those for which the break depressurizes the reactor vessel in sufficient time so that the low

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Do for First Periodic Update

pressure injection systems (LPCI and core spray) prevent core damage. Small break events will result in success by having 3 ADS valves (to effect depressurization) and injection via low pressure injection systems. Therefore, the break consisting of three or more open SRVs will depressurize the reactor vessel and is already analyzed and considered to be a large steam break event.

3. The feedwater ramp-up initiator is discussed in Section 15.1.2 of the SSES FSAR as feedwater controller failure – maximum demand. An increase in feedwater flow at power would lead directly to feedwater pump trips on high reactor level. Therefore, the feedwater ramp-up event is already included as part of the loss of feedwater initiator. The loss of feedwater initiator frequency includes loss of feedwater events caused by the feedwater ramp-up.

4. The Initiating Events Notebook discusses the methodology for evaluating the LOCA event frequencies (instrument line breaks are considered small steam or liquid breaks, depending on location). The frequency of breaks in the reference leg piping is part of the total frequency calculation for small liquid breaks. Breaks in the reference leg would also cause false high level signals to be generated from the affected instruments. However, the resulting high pressure in the drywell will cause the reactor to scram and the high pressure systems required for level control following LOCA (HPCI and RCIC) have redundant level instrumentation. Therefore, the false high level signal generated by the affected instrumentation would have no effect on the resulting small break LOCA event mitigation, and including a reference leg break as a specific initiating event is not required.

If the subject initiators are added to the model, the calculated inputs used for the ISI program are not expected to change. Therefore, the ISI consequences are expected to be unaffected.

Element: **IE** **Sub-element:** ***6*** **Observation** **1** **INDEX:** **74**

LOIA

Loss of Instrument Air can result in the shutdown of both plants and have relatively significant impacts:

MSIV closure.
Loss of TBCCW.

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Disposition of B Level F&Os
Do for First Periodic Update

Disposition:

The loss of instrument air (LOIA) event can cause a shutdown of both units (resulting from the loss of TBCCW and the subsequent MSIV closure) only if the instrument air systems are cross-tied between the units. Operating with the instrument air system cross-tied is not a normal mode of operation at SSES, therefore, should the instrument air systems need to be cross-tied for any reason, a specific risk assessment would be required prior to such operation.

The loss of instrument air (LOIA) event is considered to be subsumed by the loss of TBCCW initiating event, as discussed in the Initiating Event Notebook. However, consideration will be given to adding the LOIA event to the SSES PRA model as an initiating event as part of a future PRA update.

Since the effect of including LOIA as an initiator is expected to be small, and the effect is already subsumed by the loss of TBCCW initiator, it is expected that there will be no change to the number of welds requiring inspection if LOIA were to be included as an initiator.

Element: **IE** **Sub-element:** **7** **Observation** **1** **INDEX:** **76**

BOC

The BOC should be retained in the quantitative model and not prematurely screened.

The BOC could be a significant LERF contributor.

Disposition:

Breaks outside containment (BOC) have not been prematurely screened. BOCs have been evaluated in the Initiating Events Notebook. The frequency of BOC events has been evaluated to be a factor of at least 15 less than the frequency of interfacing system LOCA (ISLOCA) events. ISLOCA events are included as initiating events in the current PRA model and the highest frequency ISLOCA event contributes approximately 3.5% to the overall CDF and approximately 8.4% to LERF. Based on the evaluated initiating event frequency for BOC events, BOC would contribute approximately 0.2% to CDF and

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Disposition of B Level F&Os
Do for First Periodic Update

0.5% to LERF. These frequencies were evaluated as insignificant for the current PRA model for SSES. However, since BOC events have been included in PRAs performed by other utilities, PPL will evaluate adding BOC events to the PRA model.

Including explicit BOC's in the PRA is not expected to influence the calculated inputs used for the ISI program, since the BOC's initiating event frequency is low.

Element: IE Sub-element: 7 Observation 2 INDEX: 77

Loss of Instrument Air and BOCs are not modeled because of their core damage frequency contribution. Although this may be true, they should be modeled for use in Maintenance Rule A4 calculations and SDP.

Disposition:

The loss of instrument air (LOIA) event is considered to be subsumed by the loss of TBCCW initiating event, as discussed in the Initiating Event Notebook. However, consideration will be given to adding the LOIA event to the SSES PRA model as an initiating event as part of a future PRA update.

Breaks outside Containment (BOCs) were evaluated for their frequency in the Initiating Events Notebook. It was demonstrated, following the frequency evaluation, that BOCs have 'an insignificant impact on both CDF (<1%) and LERF (<1%). As such, the Break outside of Containment Initiating Events are not explicitly included in the SSES model.'

Thus, results of the current model would not be significantly impacted by including the LOIA and BOC as initiating events, and the calculated inputs used for the ISI program would be expected to be unaffected.

Element: IE Sub-element: 13 Observation 2 INDEX: 89

Missing from the analysis.

The results of the initiating event analysis should be compared with generic data sources to provide a reasonableness check of the quantitative and qualitative results.

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Disposition of B Level F&Os
<u>Do for First Periodic Update</u>**

Disposition:

The data sources for the event frequencies generated in the Initiating Events Notebook incorporated both SSES specific and external industry sources. The results of the SSES PRA model for CDF and LERF appear to be consistent with other industry analyses; therefore, the frequency of initiating events used in the model should not be significantly different from other analyses in the industry.

However, an examination of the SSES initiating event frequencies versus industry sources will be undertaken and documented as part of the first periodic update.

Given that the initiating event frequencies should not be significantly different from other source in the industry, the calculated inputs used for the ISI program would not significantly change. Therefore, the ISI consequences should be unaffected.

Element: L2 Sub-element: 5 Observation 1 INDEX: 99

Success Criteria

If needed use RMIEP (LaSalle) NUREG/CR-5305 analysis to support success criteria decisions regarding phenomena for which no plant specific thermal hydraulic analysis is available. This includes:

- Containment overtemperature failure.

Disposition:

The SSES success criteria for preventing Containment over-temperature failure are discussed in the Performance Criteria Notebook. These success criteria are considered acceptable for the current SSES PRA that evaluates CDF and LERF. Further definition of these success criteria will be considered as part of the SSES Level 2 PRA, currently under development as part of the License Renewal and Extended Power Uprate Projects.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

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Disposition of B Level F&Os
Do for First Periodic Update

Element: L2 Sub-element: 8 Observation 2 INDEX: 101

Containment Isolation

The placement of the CI node at the end of the event tree is workable. However, in certain cases (see LT2, BRANCH LT-2-3, LT-2-7, LT-2-10;TR-3 BRANCHES TR-3-1 to TR-3-9) the event tree does not branch at CI. The end state is currently identified as core damage and a release, but it is not LERF. However, if the CI node was asked, the contribution due to LERF would be calculated.

Disposition:

The event tree package will be reexamined as part of the full SSES Level 2 PRA model development currently being undertaken as part of the License Renewal and Extended Power Uprate projects.

The LT2 branch, referenced above should reflect the LERF potential resulting from CI failure, because core damage exists on entry into LT2. Therefore, the failure of the containment isolation function would lead directly to radioactive material release and LERF.

Revising the LT2 branch to include the CI failure event as LERF will result in a minimal increase for the SSES LERF value. However, the LERF value, as evaluated by the present PRA model, is conservative. Therefore, the LERF increase resulting from the CI failure events is judged to be inconsequential to the overall result.

Adding the CI node to TR3 would have no effect on the LERF calculation. In the TR3 event, core damage does not occur until at least 6 hours following the General Emergency declaration.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld repairs.

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Disposition of B Level F&Os
Do for First Periodic Update

Element: L2 Sub-element: 10 Observation 1 INDEX: 105

Containment Over Temperature Failure (COTF)

The assumption that COTF occurs for RPV breach events without drywell sprays is considered to be too pessimistic. MAAP and MELCOR calculations for Mark II plants demonstrate substantial containment temperature and pressure capability for extended times (many hours) after RPV breach. This can occur both with LPCI/CS injection to the failed RPV or with no RPV injection. (See related comment on the definition of "early").

Disposition:

A Susquehanna specific calculation for RPV breach was added to the Event Tree Notebook (EC-RISK-1092, Appendix O, added in Revision 5) concluded that the pressure generated by the water from the reactor vessel flashing to steam would result in immediate containment failure on overpressure (COPF). A MAAP input file for Susquehanna is being prepared as part of a full Level 2 PRA (being developed to support the License Extension and Extended Power Uprate (EPU) projects). The RPV breach event will be reconsidered during the development of the Level 2 PRA.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

Element: L2 Sub-element: 15 Observation 1 INDEX: 108

Class 4 Containment Failure

The definition of containment failure during an ATWS and its size and location should be identified. The attached discussion of ATWS-induced dynamic loads is included for your use in considering the plant specific evaluation. Attachment L2-15 provides some considerations regarding containment failure modes that may require consideration under ATWS conditions.

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Do for First Periodic Update

Disposition:

The current Susquehanna PRA evaluates Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). As such, no specifics on containment failure modes or quantification of release amounts or paths are documented in the current PRA. A full Level 2 PRA, with quantification of containment failure releases and locations, is under development in support of the License Renewal project. The impact of ATWS induced dynamic loads on containment failure size and location is being included as part of the full Level 2 PRA model development.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

Element: L2 Sub-element: 22 Observation 3 INDEX: 112

LERF

The magnitude of the release is not included as a determining factor in the LERF definition in the SSES simplified LERF model. Only the fact that a release occurs (greater than leakage) is included as the basis for the LERF determination. This would appear to be extremely conservative.

The timing definition for LERF used for the SSES PRA is within 12 hours after a General Emergency. This is atypical in the industry (usually 4-6 hours). The bad weather evacuation for SSES may indicate as much as 9 hours. This time estimate should be made to be more consistent (i.e., not overly conservative) relative the definition in Regulatory Guide 1.174.

Disposition:

For the current SSES PRA (which evaluates CDF and LERF), no quantification of magnitude of the radioactivity release rate is performed. A full Level 2 PRA, with quantification of containment failure releases and locations, is under development in support of the License Renewal project.

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Disposition of B Level F&Os
Do for First Periodic Update

The 12-hour break point for LERF following the declaration of General Emergency was judged to be overly conservative. The current version of the Event Tree Notebook re-evaluated the LERF timing definition as within 6 hours of a General Emergency declaration. Thus, the current Susquehanna PRA defines LERF as a release within 6 hours of declaration of a General Emergency.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

Element: MU Sub-element: Observation 1 INDEX: 113

The update process is currently defined by only a high level Maintenance and Update guidance procedure. The procedure does not go into effect until December 31, 2003. As such, the Peer Review team was unable to review the implementation of the Maintenance and Update process. The intent of the program as specified in the procedure was evaluated. Grades recorded that reflect the lack of an active program. The overall process is deemed inadequate for configuration control of the details of the change process and does not allow review by affected plant programs consistent with current industry practice. A detailed procedure driven process should be implemented for PRA model updates to ensure consistency in work practices and to capture detailed information such as specific model modifications performed, the revised model assembly, the quantification plan, results evaluation, required reviews and approvals, and review of prior applications.

Disposition:

Subsequent to the peer review, the above mentioned PRA maintenance and update procedure (NDAP-QA-1002) was formally issued. The purpose of this procedure is to define the basic process used by PPL to develop, control, and update the Susquehanna Probabilistic Risk Assessment (PRA). The procedure provides: criteria to determine when updates are needed, requirements for the PRA group to review changes in plant procedures and plant modifications, and requirements for documentation. The procedure also provides requirements for communicating PRA results to the organization, including

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
Do for First Periodic Update

Training, Work Management, Operations, Nuclear Regulatory Affairs, the Maintenance Rule expert panel, and station management. Details on the process used to develop and control the current SSES PRA are provided below:

The current PRA model is documented and controlled under PPL documentation procedures. All documentation packages include an independent technical review and final approval by qualified PPL engineers. Extensive model documentation includes:

1) System Notebooks for all key systems important to risk (e.g., HPCI, RCIC, ADS and MSIVs, RHR, Electrical Distribution system, etc.),
2) An Event Tree Notebook which documents the accident or transient progression from an initiating event to a plant damage state,
3) An Initiating Events Notebook which documents the initiating events considered in the Susquehanna PRA and their associated frequencies,
4) A Human Reliability Notebook which identifies human actions and their associated failure probabilities,
5) A Dependency Matrix Notebook which provides an overall summary of the inter-relationships of plant systems,
6) An Internal Flooding Notebook which identifies the frequencies and the impact of internal floods on key equipment and equipment or train availability, and
7) A Summary Notebook, which documents the final PSA model including all software files, developed as part of the model and the sensitivities on key input parameters.

Changes to any of the above documentation packages is also done under PPL documentation procedures. As with the initial preparation, all changes are prepared, independently reviewed and approved prior to releasing the revised model for general use by plant personnel.

The fault tree model and associated databases, which are also developed and documented in the packages discussed above, are controlled via applicable PPL procedures. These procedures provide requirements and guidance for configuration control. After these files have been developed and approved for use, the model files are stored in special directories to prevent inadvertent changes by users.

The software used for risk analysis is controlled and documented in accordance with PPL Software QA procedures. These procedures provide requirements that must be met for all quality-related software, including configuration control of the software and future updates. Documentation packages have been developed for all risk analysis software to

**Appendix A to NRC Question No. 1
Disposition of B Level F&Os
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document the procurement, installation, V&V and configuration control of this software. Changes to the software must be documented in revisions to these software packages and are thus subject to independent technical review and approval prior to their use in risk related analyses.

A more detailed procedure for documenting the PRA model assembly process could help ensure consistent model development in the future. The absence of this procedure does not have any impact on the current model results. Any changes to the current model will still need to go through the calculation process, which provides for a review and approval of the revision.

This F&O only applies to the update process. As such it has no effect on the model results.

Element: QU Sub-element: Observation 1 INDEX: 32

A process for documenting PRA model assembly does not exist that describes how the different elements (functional top logic, event tree and fault tree development, system model integration, circular logic resolution, recovery fault tree development, mutually exclusive file development, and flag file development and model file use) of the PRA model are developed. Such documentation ensures consistency in model assembly and awareness of the process employed for future model and file updates.

Disposition:

The current PRA model and associated PRA elements are documented, reviewed and approved in calculation packages per PPL calculation procedure (with the exception of a few system notebooks for the less important systems).

A detailed written procedure for documenting the PRA model assembly would help provide consistent model development in the future. Lack of this procedure does not have any impact on the current model results. Any changes to the model will need to go through the calculation process, which provides for a review and approval of the revision. Therefore, it is not necessary to have this documentation in place to have a model that represents the "as-built/as-operated" plant.

This F&O only applies to model documentation. As such it has no effect on the model results.

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Disposition of B Level F&Os
<u>Do for First Periodic Update</u>

Element: ST Sub-element: 5 Observation 2 INDEX: 136

Containment Over Temperature Failure (COTF)

The mechanistic treatment of containment failure due to the combination of high temperatures and pressures is not included in the structural analysis. A default conservative assumption is used.

Disposition:

A PPL recorded calculation addresses the success criteria for maintaining an intact containment on the basis of both temperature and pressure. Containment over-pressure failure (COPF) is defined to occur at 140 psig, as discussed in the calculation. Containment over-temperature failure is defined to occur when RPV melt-through occurs with the drywell floor dry and with insufficient drywell spray available. Containment failure due to COPF or COTF is evaluated on these bases in the Event Tree Notebook. However, because the current PRA model is a modified Level 1 PRA (CDF and LERF are evaluated), no quantification of containment break location or radioactivity release rate has been performed. The evaluation of containment break location and radioactivity release rates will be undertaken as part of the full Level 2 PRA model for SSES, currently under development as part of the License Renewal and Extended Power Uprate projects.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

Element: ST Sub-element: 5 Observation 3 INDEX: 137

Hydrodynamic Loads

The structural analysis does not examine the possible effects associated with containment barrier unavailability due to ATWS events that include:

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Disposition of B Level F&Os
Do for First Periodic Update

Hydrodynamic loads.
Pool bypass above temperatures above 240F (Sonin experiments).
Containment vent.
Stuck open tailpipe vacuum breakers.
High pool water level (and hydrodynamic loading).

See discussion associated with L2-15.

Disposition:

In the current SSES PRA (which evaluates CDF and LERF), no quantification of containment breach location or radioactivity release rate is performed. A full Level 2 PRA, with quantification of containment failure releases and locations, is under development in support of the License Renewal project. The impact of ATWS induced dynamic loads on containment failure size and location is being included as part of the full Level 2 PRA model development.

In the new level 2 model being developed, LERF is expected to be approximately three times lower than the current value and the CDF is expected to be the same or go down due to additional LOOP recoveries being applied to some sequences. Thus, the new level 2 model will tend to reduce the calculated ICLERP and ICCDP values used to determine the needed weld inspections. Therefore, the use of the current model is conservative for determining weld inspections.

Element: SY Sub-element: 4 Observation 1 INDEX: 140

The quality and content of system notebooks are good. Several other system notebooks are in various stages of development. All modeled systems should have these books completed and reviewed.

Disposition:

It was planned to develop system notebooks for the 27 systems credited in the PRA model. Of the 27, notebooks were issued for 17 of the most risk significant systems. Of the 10 remaining, five notebooks have been drafted and five have not yet been prepared. PPL intends to complete and formally document the remaining 10 system notebooks.

Appendix A to NRC Question No. 1
Disposition of B Level F&Os
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However, given that the most important systems have been addressed by specific system notebooks and that the remaining systems are relatively straightforward to model, no significant model impacts are foreseen once the 10 remaining system notebooks are issued.

This F&O only applies to model documentation. As such it has no affect on the model results.

Element: SY Sub-element: 8 Observation 2 INDEX: 144

Missing Pre-initiator Human Errors Probabilities (HEPs)

Selected Pre-initiator Human Errors are included in the system model. PPL should ensure that the pre-initiators are examined relative to plant design and procedures and are incorporated and quantified.

Disposition:

Twenty-one pre-initiator human errors are currently documented in the HRA Notebook and are included in the plant PRA model. Pre-initiators have been evaluated for the diesel generators, LPCI, RCIC, HPCI, Core Spray, SLC, and CRD. In the model quantification, the pre-initiators contribute 3.66% of the Unit 1 CDF and 3.67% of the Unit 2 CDF with more than half of the contribution coming from the A and B diesel generators. In general, the pre-initiators are comparable to the 16 pre-initiators included in the Limerick PRA model.

With regard to this F&O, SSES HRA pre-initiators are currently deemed sufficient for risk informed applications. The pre-initiators will, however, be comprehensively reevaluated in a future model update. Adding more pre-initiators is not expected to affect the insights presently realized.

As discussed above, additional pre-initiators are not expected to significantly influence the calculated inputs used for the ISI program. Therefore, the ISI consequences are expected to be unaffected.

NRC Question 2:

Your submittal discusses the use of the Markov piping analysis method to estimate the change in risk due to adding and removing inspection locations from the inspection program. The submittal refers to Section 3.7.2 of Electric Power Research Institute (EPRI), Topical Report (TR) 112657. The safety evaluation (SE) approving the EPRI methodology (Adams accession No. ML013470102) approved the use of the Markov model as a basis for the direct estimation of pipe failure frequencies instead of the bounding pipe failure frequencies. The SE did not approve the use of the Markov model to estimate the inspection efficiency factor (IEF) that is used in Equation 3-9 in TR-112657 because there is insufficient information in EPRI TR-112657 to fully define the method. The methodology description for estimating the IEF is located in EPRI TR-110161, "Piping System Reliability and Failure Rate Estimation Models for Use in Risk-Informed Inservice Inspection Applications," and EPRI TR-111880, "Piping System Applications," (both proprietary). The use of the Markov method to estimate the IEF in Equation 3-9 in EPRI TR-112657 has been approved in some relief requests after sufficient information was provided to fully define the method and the staff found the specific method acceptable (i.e., the SE for Dresden (Adams accession No. ML012050103).

Please clarify how the Markov method was used to estimate the changes in risk due to adding and removing inspection locations from the inspection program. If it is possible to identify a previously approved risk-informed inservice inspection relief request that used the same methodology that was used in your submittal, a more limited discussion of the method and identification of the relevant relief request should be sufficient.

PPL Response:

This question concerns the "inspection efficiency factor." Section 7 of the Susquehanna Tier 2 RI-ISI document entitled, "Risk Informed Inservice Inspection Evaluation Final Report July 2003, Susquehanna Steam Electric Station, Unit 1 and 2" documents the methodology and input for calculating the inspection effectiveness factor. This factor is calculated from the ratio of hazard rates which are intermediate solutions in the Markov methodology. This methodology is consistent with that used in the RISI analysis for Exelon's nine (9) RISI submittals, including Dresden Station as one example.

NRC Question 3:

In Tables 7 and 8 of your submittal dated September 16, 2003, there are columns provided for ASME Code, Section XI, core damage frequency (CDF), RI-ISI CDF, and delta CDF (that can be obtained as the difference of the first two columns). Although there are no columns for Section XI large early release frequency (LERF) and

RI-ISI LERF, there is a column for delta LERF. Please confirm that delta LERF reported in the tables was calculated as it was for DCF and that the two LERF values were intentionally not provided in the tables.

PPL Response:

The Delta LERF values in Tables 7 and 8 were calculated using the same approach as that used for determining the Delta CDF values. While the system values for LERF were not provided in these tables, these values are available within the Markov database.

Attachment 2 to PLA-5768

Response to NRC Verbal Request for Additional Information Relating to Relief Request 3RR-01

The following RAI questions were provided to PPL in a teleconference between PPL and the NRC that was held on June 15, 2004. Answers to each question as discussed in the teleconference are provided.

NRC Question A1:

The CDF/LERF of the PRA model used for this application was not provided in PPL's submittal.

PPL Response:

The current PRA model including the F&O dispositions discussed in the response to Question 1 is documented in the Summary Notebook and is referred to as the FEB05 version. Results for the current PRA model (including random maintenance) are:

	CDF	LERF
Unit 1	2.96E-6	1.18E-6
Unit 2	2.78E-6	1.18E-6

As noted previously, future revisions to the PRA model will be evaluated for effects on the ISI program.

NRC Question A2:

The Calc Model EC-RISK-1083 Rev. 1 is in the list of references, but in the narrative PPL states that the PRA model is "continually updated". PPL's write-up suggests that they are continually updating calc model EC-RISK-1083, Rev 1. If this is the case, then the specific update of this model should be uniquely identified.

PPL Response:

The current PRA model including the F&O dispositions discussed in the response to Question 1 is documented in the Summary Notebook and is referred to as the FEB05 version. As noted previously, future revisions to the PRA model will be evaluated for effects on the ISI program.

NRC Question B1:

Weld count versus number of examinations. PPL has only identified how many were considered, how many welds were actually eliminated from the program? We do not know that the 25%/10% criteria were actually met. (Also, the 10% criterion for examining Class 1 welds needs "numerical proof"). There is further confusion in that the weld count includes hundreds of welds in risk Category 1 (high consequence and high failure potential). This appears to be an outlier as Susquehanna has many more high consequence FAC related welds compared to other utilities. Most utilities eliminated such welds to their FAC augmented inspection program, and only those few welds with multiple degradation mechanisms remained in the RISI program.

PPL Response:

Based upon discussions during the June 15, 2004 teleconference, PPL agreed to provide two tables from Section 6 of the RISI Program. These tables are provided in Attachment 3. Two additional tables are also provided in Attachment 3 to answer the portion of this question regarding Class 1 selection percentages. Further answering the question, PPL does treat augmented inspection programs (e.g., FAC and IGSCC) as independent programs.

Initially, all welds within the RISI scope are evaluated for consequence and degradation as required by the methodology. When the only mechanism assigned to a given location is treated under a separate augmented program such as FAC or IGSCC, that location is removed from the RISI population used to determine the number and locations of required element selections and the augmented program itself drives the examination requirements for those locations.

These augmented-only locations which default to those associated augmented programs for examination requirements are however maintained in the Risk Impact Assessment which follows the element selection. Any such locations removed from the RISI Program which previously received, or were scheduled to receive, an examination under ASME Section XI are treated in the Risk Impact Assessment as deleted exams and thus accounted for in the analysis against the risk acceptance criteria.

NRC Question B2:

Regarding the commitment to review the risk ranking of piping element on an ASME interval, rather than an ASME periodic frequency. Other utilities committed to doing this on a periodic frequency. Susquehanna proposes doing this only every interval. Guidance is not that black and white, but suggests that it should be done on a periodic frequency.

PPL Response:

PPL's RI-ISI Program is a "living program" in that changes to the plant, operating parameters, PRA model, industry experience, or known degradation mechanisms would require a review for impact on the RISI Program. Since these changes can not be predicted, PPL will, as a minimum, require that a review of the RI-ISI Program be undertaken at least once per inspection period.

Attachment 3 to PLA-5768

Section 6 Tables

Table 6-1

SSES Unit 1 Summary of Element Selection

HIGH AND MEDIUM RISK CATEGORY TOTALS [1]							
Risk Category	Total RI Element Population	Minimum Elements to Select for I3	Elements Selected for I3 [2]	Extra Class 1 Butt Welds Selected for I3 [3]	Total RI Elements Selected for I3	SCXI Welds Selected Previously for I2	% RI Elements Selected
1	69	17.3	23	0	23	26	33.3%
2	51	12.8	13	0	13	34	25.5%
3	3	0.8	1	0	1	3	33.3%
4	325	32.5	35	0	35	58	10.8%
5	63	6.3	8	0	8	6	12.7%
TOTALS:	511	69.7	80	0	80	127	15.7%

Notes: [1] This table does not include elements whose sole degradation mechanism is inspected for under a separate program (i.e., FAC and IGSCC.)

[2] These Summations across Risk Categories are obtained from the table below. Since the minimum percentage requirements are applied at the "system" level, the risk category roll-ups will tend to show selections that exceed the Minimum to Select column due to the summation of round up.

[3] Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Table 6-1 (Continued)

SSES Unit 1 Summary of Element Selection

SYSTEM BREAKDOWN[1]								
Risk Cat	System	Total RI Element Population	Minimum Elements to Select for I3	Elements Selected for I3 [2]	Extra Class 1 Butt Welds Selected for I3 [3]	Total RI Elements Selected for I3	SCXI Welds Selected Previously for I2	% RI Elements Selected
1	FW	63	15.8	17	0	17	20	27.0%
	RPV-E	6	1.5	6	0	6	6	100.0%
2	CS	8	2.0	2	0	2	6	25.0%
	RHR	27	6.8	7	0	7	12	25.9%
	RPV-E	16	4.0	4	0	4	16	25.0%
3	FW	3	.75	1	0	1	3	33.3%
4	CS	4	0.4	1	0	1	0	25.0%
	HPCI	22	2.2	3	0	3	7	13.6%
	RHR	139	13.9	14	0	14	8	10.1%
	RPV-E	10	1.0	1	0	1	8	10.0%
	RR	79	7.9	8	0	8	25	10.1%
	RWCU	71	7.1	8	0	8	10	11.3%
5	HPCI	24	2.4	3	0	3	1	12.5%
	RCIC	28	2.8	3	0	3	3	10.7%
	RHR	11	1.1	2	0	2	2	18.2%

Notes: (1) This table does not include elements whose sole degradation mechanism is inspected for under a separate program (i.e., FAC and IGSCC).

(2) The required minimum number of inspection elements in each system is the percentage of the total number of elements rounded up to the next higher whole number.

(3) Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Table 6-2

SSES Unit 2 Summary of Element Selection

HIGH AND MEDIUM RISK CATEGORY TOTALS[1]							
Risk Category	Total RI Element Population	Minimum Elements to Select for I3	Elements Selected for I3 [2]	Extra Class 1 Butt Welds Selected for I3 [3]	Total RI Elements Selected for I3	SCXI Welds Selected Previously for I2	% RI Elements Selected
1	81	20.3	26	0	26	21	32.1%
2	44	11.0	12	0	12	29	27.3%
3	3	0.8	1	0	1	2	33.3%
4	307	30.7	33	0	33	44	10.7%
5	62	6.2	8	0	8	9	12.9%
TOTALS:	497	69.0	80	0	80	105	16.1%

Notes: [1] This table does not include elements whose sole degradation mechanism is inspected for under a separate program (i.e., FAC and IGSCC.)

[2] These Summations across Risk Categories are obtained from the table below. Since the minimum percentage requirements are applied at the "system" level, the risk category roll-ups will tend to show selections that exceed the Minimum to Select column due to the summation of round up.

(3) Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Table 6-2 (Continued)

SSES Unit 2 Summary of Element Selection

SYSTEM BREAKDOWN[1]								
Risk Cat	System	Total RI Element Population	Minimum Elements to Select for I3	Elements Selected for I3 [2]	Extra Class 1 Butt Welds Selected for I3 [3]	Total RI Elements Selected for I3	SCXI Welds Selected Previously for I2	% RI Elements Selected
1	FW	75	18.8	20	0	20	15	26.7%
	RPV-E	6	1.5	6	0	6	6	100.0%
2	CS	7	1.8	2	0	2	5	28.6%
	RHR	22	5.5	6	0	6	9	27.3%
	RPV-E	15	3.8	4	0	4	15	26.7%
3	FW	3	0.8	1	0	1	2	33.3%
4	CS	5	0.5	1	0	1	2	20.0%
	HPCI	19	1.9	2	0	2	10	10.5%
	RHR	131	13.1	14	0	14	9	10.7%
	RPV-E	10	1.0	1	0	1	4	10.0%
	RR	77	7.7	8	0	8	13	10.4%
	RWCU	65	6.5	7	0	7	6	10.8%
5	HPCI	22	2.2	3	0	3	1	13.6%
	RCIC	24	2.4	3	0	3	3	12.5%
	RHR	16	1.6	2	0	2	5	12.5%

Notes: (1) This table does not include elements whose sole degradation mechanism is inspected for under a separate program (i.e., FAC and IGSCC).
(2) The required minimum number of inspection elements in each system is the percentage of the total number of elements rounded up to the next higher whole number.
(3) Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Table 6-3

RISI Program Weld Selection of Class 1 Butt Welds

Unit(s)	Class 1 Butt Welds RI Population	Class 1 Butt Welds RI Selection	Class 1 Butt Welds Additional Selection[1]	Class 1 Butt Welds Total Selection	Class 1 Butt Welds % Selected
1	338	51	0	51	15.1%
2	315	53	0	53	16.8%

Notes: [1] Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Table 6-4

RISI Program Weld Selection of All Class 1 Welds

Unit(s)	Class 1 Welds RI Population	Class 1 Welds RI Selection[1]	Class 1 Welds Additional Selection[2]	Class 1 Welds Total Selection	Class 1 Welds % Selected
1	394	51	0	51	12.9%
2	379	53	0	53	14.0%

Notes: [1] Some socket welds are selected, but are not selected for volumetric examination, so the socket weld exams are not included in these totals.

[2] Additional welds selected to achieve 10% of the Class 1 Butt Weld population as desired by the USNRC.

Attachment 4 to PLA-5768

Relief Request No. 3RR-01 Revision 1

RELIEF REQUEST NUMBER 3RR-01
(Page 1 of 5)

COMPONENT IDENTIFICATION

Code Class:	1 and 2
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.140, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.51, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Pressure Retaining Piping
Reference:	1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure" 2) W. H. Bateman (NRC) to G. L. Vine (EPRI) letter dated October 28, 1999 transmitting "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" 3) American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B" 4) Risk-Informed Inservice Inspection Evaluation, Final Report - Susquehanna Steam Electric Stations Units 1 and 2

CODE REQUIREMENT

Table IWB-2500-1, Examination Category B-F, requires volumetric and/or surface examinations on all welds for Items B5.10 and B5.140.

RELIEF REQUEST NUMBER 3RR-01 (Continued)
(Page 2 of 5)

Table IWB-2500-1, Examination Category B-J, requires volumetric and/or surface examinations on a sample of welds for Items B9.11, B9.21, B9.31, B9.32 and B9.40. The weld population selected for inspection includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.

2. All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

    a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.
    b. cumulative usage factor U of 0.4.

3. All dissimilar metal welds not covered under Category B-F.

4. Additional piping welds so that the total number of circumferential butt welds, branchconnections, or socket welds selected for examination equals 25% of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds excluded by IWB-1220.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and/or surface examinations on a sample of welds for Items C5.11, C5.51, and C5.81. The weld population selected for inspection includes the following:

1. Welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Category C-F-1) or of all carbon and low alloy steel welds (Category C-F-2) not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

RELIEF REQUEST NUMBER 3RR-01 (Continued)
(Page 3 of 5)

a. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and high allow welds (Category C-F-1) or carbon and low alloy welds (Category C-F-2) in each system;
b. within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and
c. within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

BASIS FOR RELIEF

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative utilizing Reference 1 along with two enhancements from Reference 3 will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (Reference 2):

> "The staff concludes that the proposed RI-ISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10CFR50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."

The Risk Impact Assessment completed as part of the baseline RISI Program evaluation is an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the Third Interval ISI update, there is no traditional ASME Section XI selection to compare with

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under the new code of assessed record since this is a new inspection interval. As such, the transition impact was between the previous second interval selection and the new RISI selection.

The actual evaluation and ranking procedure including the Consequence Evaluation, Degradation Mechanism Assessment, and Risk Ranking processes are summarized in the attached "Risk-Informed Inservice Inspection Evaluation, Final Report, Executive Summary". These processes are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These portions of the RISI Program are reevaluated as major revisions of the site PRA occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, and Element Selection steps define the *living program* process applicable to the RISI Program.

PROPOSED ALTERNATE PROVISIONS

The proposed alternative described in Attachment A, "Risk-Informed Inservice Inspection Evaluation, Final Report, Executive Summary", along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10CFR50.55a(a)(3)(i).

The Third Interval RISI Program will be an EPRI TR-112657, Revision B-A, application and will be maintained as a living program as described in the Basis For Relief above. The following two enhancements will be implemented.

> In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, SSES will utilize the requirements of Subarticle-2430", Additional Examinations" contained in Code Case N-578-1 (Reference 3). The alternative criteria for additional examinations contained in Code Case N-578-1 provides a more refined methodology for implementing necessary additional examinations.
>
> To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination methods" of EPRI TR-112657, SSES will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" contained in Code Case N-578-1 (Reference 3). To implement Note 10 of this table, paragraphs and figures from

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the 1998 Edition through the 2000 Addenda of ASME Section XI (SSES's Code of record for the Third Interval) will be utilized which parallel those referenced in the Code Case for the 1989 Edition. Table 1 of Code Case N-578-1 will be used as it provides risk informed Category/Item Numbers, a detailed breakdown for examination method, and a categorization of parts to be examined where the TR is either silent or ambiguous.

The SSES RISI Program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, and 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by Code Case N-578-1.

In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to receive Code required pressure testing as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the SSES pressure testing program, which remains unaffected by the RISI Program.

APPLICABLE TIME PERIOD

Relief is requested for the third ten-year inspection interval of the Inservice Inspection Program for SSES Units 1 and 2.

EXECUTIVE SUMMARY

Risk Informed Inservice Inspection Program

Susquehanna Steam Electric Station
Units 1 and 2

EXECUTIVE SUMMARY

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EXECUTIVE SUMMARY

1. INTRODUCTION

The objective of this submittal is to request the use of a risk-informed inservice inspection (RISI) program for Class 1 and Class 2 piping that is currently inspected as part of the ASME Section XI based ISI program. The RISI program is proposed as an alternative to the 1998 Edition, 2000 Addenda of the ASME Section XI requirements for the third inspection interval. The risk-informed process used in this submittal is described in EPRI RISI Topical Report (Reference 1) and the accompanying NRC staff SER on the EPRI method.

The RISI Program will be incorporated during the first period of the Third Ten-Year Inservice Inspection Interval for PPL's Susquehanna Steam Electric Station (SSES), Units 1 and 2. The Third Ten-Year Inspection Interval will start June 1, 2004 and is projected to end May 31, 2014 for both units. At that time, the SSES Units 1 and 2 ISI Programs will require updating for the Fourth Inservice Inspection Interval.

As a risk-informed application, this submittal meets the principles of Regulatory Guides 1.174 and 1.178 as well as those set forth in the EPRI RISI Topical Report and the NRC staff SER on the EPRI RISI method.

<u>PRA Quality</u>

The SSES PRA model (Reference 2) used for the risk determinations within the current revision of this regulatory application is a peer reviewed model using a new code, CAFTA, which was implemented after the Individual Plant Examination (IPE) submitted to the NRC by letter dated December 13, 1991. The IPE had been accepted by the NRC by Staff Evaluation Report (SER) letter dated August 11, 1998.

Since this time, PPL has converted to the CAFTA code for fault tree analysis. Three models have been produced with the new software, the latest of which has been peer reviewed. The peer-reviewed model is not based on, nor is it an extension of, the original IPE. The peer review model started with the success criteria being developed from revised thermal hydraulic calculations for the post power up-rate conditions (i.e., current fuel type and current rated power). Using the success criteria, new event trees which were developed based on the accident progression and current emergency operating procedures.

The October 2003 BWROG peer review provided PPL with Level B, C, D, and S Facts and Observations (F&Os). PPL did not receive any Level A F&Os. PPL resolved the Level B F&Os that were determined to be the most significant in their effect on PRA results before the model was released. This accounted for more than half of the Level B F&Os. The remaining Level B F&Os will be resolved prior to the first scheduled model update.

The overall assessment of the model based on the peer review team findings is that the PRA can be used effectively to support applications involving absolute risk determination. The Level 1 PRA is fully supportive of Grade 3 applications and the Level 2 is a useful tool to assess applications.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM REQUIREMENTS

2.1 ASME Section XI

ASME Section XI Categories B-F, B-J, C-F-1, and C-F-2 currently contain the requirements for examining these Class 1 and Class 2 piping components via Non Destructive Examination (NDE) methods.

2.2 Alternate RISI Program

The alternative RISI program for piping is described in EPRI RISI Topical Report. The RISI program will be substituted for the 1998, 2000 Addenda ASME Section XI Code Edition examination program for Class 1 Category B-J and B-F welds and Class 2 Category C-F-1 and C-F-2 welds in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other portions of the ASME Section XI Code imposed inservice inspection program outside of this RISI scope will be unaffected. The EPRI Topical Report provides the requirements for defining the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI.

2.3 Augmented Programs

As discussed in Section 6 of the EPRI Topical Report, certain augmented inspection programs may be integrated into the RISI program. At this time, no augmented programs are integrated in the RISI Program, with the exception of the IGSCC Category A welds. The following augmented programs were not integrated into the RISI program and remain unaffected:

- IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter 88-01 and NUREG-0313). Only IGSCC Category A welds will be subsumed into the RISI program.
- Flow Accelerated Corrosion (FAC) (Generic Letter 89-08)
- High Energy Line Breaks (USNRC Branch Technical Position MEB 3-1.)

Elements in the scope of this evaluation that were also covered by these augmented programs were included in the consequence assessment, degradation assessment, and risk categorization evaluations, to determine the damage mechanisms at those elements and whether the affected piping was subject to damage mechanisms other than those addressed by the augmented program. If no other damage mechanism was identified, the element was removed from the RISI element selection population and retained in the appropriate augmented inspection program. If another damage mechanism was identified, the element was retained within the scope of consideration for element selection as part of the RISI program. In the Feedwater System, many of the elements covered by the FAC program were also assessed for the potential for other damage mechanisms that are evaluated as part of the EPRI RISI methodology. The entire scope of the RISI evaluation including those elements covered by augmented programs and not included in the RISI selection population were included in the risk impact assessment phase of the evaluation described below.

2.4 Multiple Damage Mechanisms

The vast majority of pipe elements that were evaluated in the RISI evaluation were found to be susceptible to none of the damage mechanisms addressed in the EPRI RISI methodology. A number of elements were found to be susceptible to one specific damage mechanism, and a relatively small number were identified to be subject to the potential for two or more damage mechanisms. Specific examples are welds in the Feedwater and Reactor Pressure Vessel systems that are subject to both FAC and thermal fatigue, as well as welds in the Residual Heat Removal and Reactor Pressure Vessel systems that have the potential for both IGSCC and thermal fatigue. If one of the damage mechanisms was FAC, the element was assigned to the High failure potential category to be consistent with the EPRI Topical Report. If that assignment led to the decision to select that element for inspection in accordance with the 25% sampling requirement, it was retained in the FAC program for inspection for FAC as well as inspected for the remaining damage mechanism as part of the RISI program. The potential for synergy between two or more damage mechanisms working on the same location was considered in the estimation of pipe failure rates and rupture frequencies which was reflected in the risk impact assessment.

3. RISK-INFORMED ISI PROCESS

The process used to develop the RISI program is consistent with the methodology described in the EPRI Topical Report for ASME Code Case N-578-1 (Reference 5) applications. The process involves the following steps:

- Definition of RISI Program Scope
- Consequence Analysis
- Degradation Mechanism Assessment
- Risk Categorization
- Inspection Location Selection and NDE Selection
- Program Relief Requests

- Risk Impact Assessment
- Implementation and Monitoring Program

3.1 Definition of RISI Program Scope

The systems to be included in the RISI program are provided in Table 1. This scope covers ASME Class 1 and 2 piping systems within the scope of the existing ASME Section XI inspection program. The as-built and as-operated isometric and piping and instrumentation diagrams and additional plant information were used to define the system boundaries. The RISI evaluation system boundaries were defined using the system boundaries established in the existing plant ISI program.

3.2 Consequence Analysis

The consequences of pressure boundary failures were evaluated and ranked based on their impact on conditional core damage probability (CCDP) and conditional large early release probability (CLERP). The impact on these measures due to both direct and indirect effects was determined using the PRA model described in Section 1. Consequence categories (High, Medium or Low) were assigned according to Table 3-1 of the EPRI RISI Topical Report. One of the enhancements that was incorporated into this application of the EPRI RISI methodology was the direct use of the PRA models to support the estimation of CCDP and CLERP values for each pipe element in the scope of the RISI evaluation, in lieu of the consequence tables in the EPRI Topical Report. This step was taken to reduce some of the conservatisms inherent in the consequence tables and to support a more complete and realistic quantification of the risk impacts of the RISI program in comparison with previous applications of this methodology. Another motivation was to increase consistency with other risk informed applications at SSES that directly utilize the plant-specific PRA models.

3.3 Degradation Mechanism Assessment

Failure potential was assessed using the deterministic criteria in the EPRI Topical Report to evaluate the potential for each damage mechanism that an ISI exam could identify, and be supported by industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in the EPRI Topical Report.

Table 2 summarizes the degradation mechanism assessment by system for each damage mechanism that was identified as a potential failure cause. In addition, failure rates and rupture frequencies were assessed for each piping element within the scope of the RISI evaluation using information in Reference 4 and described in the Tier 2 documentation (Reference 3).

3.4 Risk Categorization

In the preceding steps, each element within the scope of the RISI program was evaluated to determine the consequences of its failure, as measured by CCDP and CLERP. Each element was also evaluated to determine its potential for pipe rupture based on the potential for degradation mechanisms that were identified. The results of the consequence assessment were then combined with the results of the degradation assessment, using the risk matrix shown in Figure 1. This provides a risk ranking and risk category for each element.

The results of this evaluation in terms of the number of elements in each of the EPRI RISI risk categories per system are summarized in Table 3 and Table 4 for SSES Unit 1 and Unit 2, respectively.

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	CONSEQUENCES OF PIPE RUPTURE IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	HIGH Category 3	HIGH Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	HIGH Category 2
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

Figure 1

EPRI RISI Matrix for Risk Ranking of Pipe Elements (Reference 1)

3.5 Inspection Location Selection and NDE Selection

In general, an ASME Code Case N-578-1 application of RISI, per the EPRI RISI Topical Report, requires that 25% of the elements that are categorized as "High" risk (Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (Risk Categories 4 and 5) be selected for inspection and appropriate non-destructive examination (NDE). Inspection locations are generally selected on a system-by-system basis, so that each system with "High" risk category elements will have approximately 25% of the system's "High" risk elements selected for inspection and similarly 10% of the elements in systems having "Medium" risk category welds will be selected. During the selection process, an attempt is made to ensure that all damage mechanisms and all combinations of damage mechanisms are represented in the elements selected for inspection. An element ranking process was used to incorporate several factors into the selection of specific elements to satisfy the above sampling percentages. These factors include whether the element has been previously selected for ISI exams, whether previous exams had indications of possible damage, presence of radiation fields in the vicinity of the elements, accessibility of the element for inspection, and numerical estimates of the pipe rupture frequencies at these locations. The results of the selection are presented in Tables 5 and 6 for Units 1 and 2, respectively. Section 4 of the EPRI Topical Report and ASME Code Case N-578-1 (Reference 7) were used as guidance in determining the examination methods and requirements for these locations. From the Class 1 butt welded elements that are considered within the scope of the RISI evaluation at Unit 1, a total of 15.1% are selected for volumetric examination as part of the risk informed inspection program. The corresponding percentage for Unit 2 is 16.8%. The total Class 1 welds selected for RISI evaluation is 12.9% for Unit 1 and 14.0% for Unit 2. As noted above, elements found to be susceptible to two or more damage mechanisms are given enhanced treatment by retaining them within the scope of the augmented programs and in the risk informed program for the applicable damage mechanisms.

In addition, all in-scope piping components, regardless of risk classification, will continue to receive Code-required pressure and leak testing, as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the station's pressure and leak test program, which remains unaffected by the RISI program.

<u>Additional Examinations</u>

Examinations performed that reveal flaws or relevant conditions exceeding the applicable acceptance standards shall be extended to include additional examinations. The additional examinations shall include piping structural elements with the same postulated failure mode and the same or higher failure potential.

(1) The number of additional elements shall be the number of piping structural elements with the same postulated failure mode originally scheduled for that fuel cycle.

(2) The scope of the additional examinations may be limited to those high safety significant piping structural elements (i.e., Risk Categories 1 through 5) within systems, whose material and service conditions are determined by an evaluation to have the same

postulated failure mode as the piping structural element that contained the original flaw or relevant condition.

If the additional required examinations reveal flaws or relevant conditions exceeding the referenced acceptance standards, the examination shall be further extended to include additional examinations.

(1) These examinations shall include all remaining piping elements whose postulated failure modes are the same as the piping structural elements originally examined.

(2) An evaluation shall be performed to establish when those examinations are to be conducted. The evaluation must consider failure mode and potential.

No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

For the inspection period following the period in which the original examination discovering the flaw or relevant condition was completed, the examinations shall be performed as originally scheduled.

3.6 Program Relief Requests

In instances where a location may be found at the time of the examination that does not meet the >90% coverage requirement, the process outlined in the EPRI Topical Report will be followed.

As required by Section 6.4 of EPRI TR-112657, PPL has completed an evaluation of existing relief requests to determine if any should be withdrawn or modified due to changes that occur from implementing the RISI Program. There are no existing relief requests required to be withdrawn. There are no existing relief requests that are required to be modified due to RISI expansion of the examination volume.

3.7 Risk Impact Assessment

The RISI program has been conducted in accordance with Regulatory Guides 1.174 and 1.178, and the EPRI Topical Report, which require an evaluation to show that implementation of a risk informed inspection program would result in acceptably small changes, if any, in core damage frequency (CDF) and Large Early Release Frequency (LERF).

The risk impact assessment performed in this RISI application included a qualitative evaluation as well as a comprehensive quantitative evaluation of the changes in CDF and LERF due to changes in the ISI program for each piping element in the scope of the RISI evaluation. This is another enhancement that was made that goes well beyond the limited quantitative analyses that are needed to implement the methods described in the EPRI Topical Report.

Individual elements were evaluated for consequence and degradation mechanism and then assigned to a risk category and risk ranking as part of the risk characterization step. The elements were then grouped by system and the changes in risk for each element was summed to provide the change in risk for the system due to increases and decreases in the number of exams and for the potential for increases in the NDE probability of detection where the "inspection for cause" principle was applied.

Per Section 3.7.2 of EPRI TR-112657, the Markov piping reliability analysis method was used to estimate the change in risk due to adding and removing locations from the inspection program. The actual CCDP and CLERP values calculated for each element in the consequence assessment was used in the risk impact calculation. Realistic quantitative estimates of failure frequencies, rupture frequencies, and risk impacts were performed for all elements within the scope of the RISI evaluation, in lieu of the qualitative analysis and bounding risk estimates that are permitted under most circumstances in the EPRI RISI Topical Report.

The changes to the ASME Section XI ISI program include changing the number and location of inspections within the risk segment, and in many cases improving the effectiveness of the inspection to account for the results of the RISI degradation mechanism assessment. For example, for locations subject to thermal fatigue, examinations are to be conducted on an expanded volume and are to be focused to enhance the probability of detection (POD) during the inspection process. For other damage mechanisms, this "inspection for cause" principle is also expected to favorably impact the POD.

Limits are imposed by the EPRI methodology (TR-112657) to ensure that the change in risk of implementing the RISI program meets the requirements of Regulatory Guides 1.174 and 1.178. The criteria established require that the cumulative increase in CDF and LERF be less than 1×10^{-7} and 1×10^{-8} per year per system, respectively. Meeting these limits is consistent with meeting Regulatory Guide 1.174 risk significant thresholds of 1×10^{-6} per year and 1×10^{-7} per year for changes in CDF and LERF, respectively, for a full plant scope RISI application.

The technical basis for the Markov model input parameters that were used in this evaluation are documented in the Tier 2 documentation (Reference 3). These parameters include a set of failure rates and rupture frequencies for piping systems in General Electric BWR plants subject to several degradation mechanisms that were identified for these systems as part of the degradation mechanism assessment. The failure rates and rupture frequencies that were used in this evaluation are those developed in Table A-11 in EPRI TR-111880 (Reference 4).

Separate Markov calculations were performed for the change in CDF and the change in LERF. This calculation was performed so that pipe elements whose failure could create a potential containment failure or bypass concern were factored into the LERF evaluation. Unlike previous applications of the EPRI methodology, realistic estimates of CDF and LERF contributions and changes in CDF and LERF due to all changes in the RISI program were quantified for all pipe elements, in addition to a qualitative evaluation that is part of the EPRI procedure.

The results of the risk impact assessment for each system at SSES Unit 1 are summarized in Table 7 and key aspects are plotted in Figures 2 and 3 for comparison against the risk significant criteria established in the EPRI RISI Topical Report. A similar set of results is presented in Table 8 and Figures 4 and 5 for Unit 2. As seen in these figures and tables, the RHR system group at Unit 1 and Unit 2 and the FW system group at Unit 2 exhibited a small decrease in CDF due to the changes from the RISI program. The remaining systems evaluated across the two reactor units exhibited small increases in CDF and LERF. In each case in which a risk increase was identified, the estimated increases in CDF and LERF are much smaller than the risk acceptance criteria by a large margin. With the exception of the RPV-E system, each system was found to have a change in LERF that is less than or equal to 15% of the EPRI RISI risk significance threshold of $1x10^{-8}$/system-year, and a change in CDF that is less than 5% of the associated threshold of $1x10^{-7}$/system-year.

The total change in CDF and LERF due to the combined changes in the RISI program for the entire scope of Class 1 and 2 systems is well below the risk acceptance criteria for both SSES units, approximately an order of magnitude for the change in LERF and more than an order of magnitude for the change in CDF.

As a sensitivity case, an evaluation was performed assuming that all NDE exams were removed from the ISI program, indicating that the EPRI RISI combined risk significance threshold still would not be exceeded.

As indicated above, the risk impact evaluation has demonstrated that no significant risk impacts will occur from implementation of the RISI program for the entire scope of Class 1 and 2 piping that was included in this evaluation. This satisfies the risk significance criteria of Regulatory Guide 1.174 and the EPRI RISI Topical Report.

Defense-In-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds," this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and ASME Code Case N-578-1 provide a more robust selection process founded on actual service experience taken from nuclear plant piping failure data.

This process has two key independent ingredients: (1) a determination of each location's susceptibility to degradation and (2) an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, the consequence assessment effort has a single failure criterion. As such, no matter how unlikely a failure scenario is, it is ranked High in the consequence assessment, and no lower than Medium in the risk assessment (i.e., Risk Category 4), if, as a result of the failure, there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability, with less credit given to less reliable equipment.

All locations within the reactor coolant pressure boundary will continue to receive a system pressure test and visual VT-2 examination as currently required by the Code regardless of its risk classification.

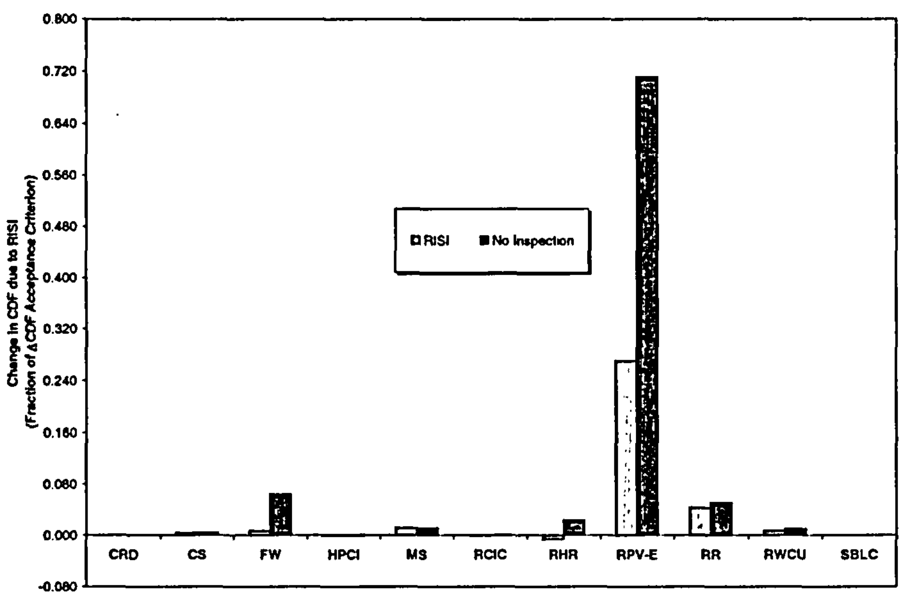

Figure 2

Change in Pipe Rupture CDF for SSES Unit 1 Systems

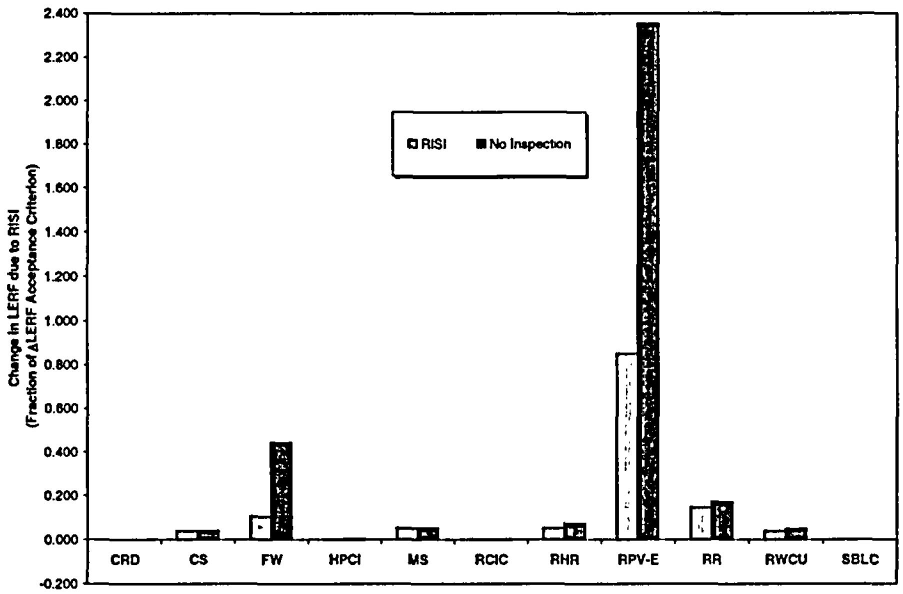

Figure 3

Change in Pipe Rupture LERF for SSES Unit 1 Systems

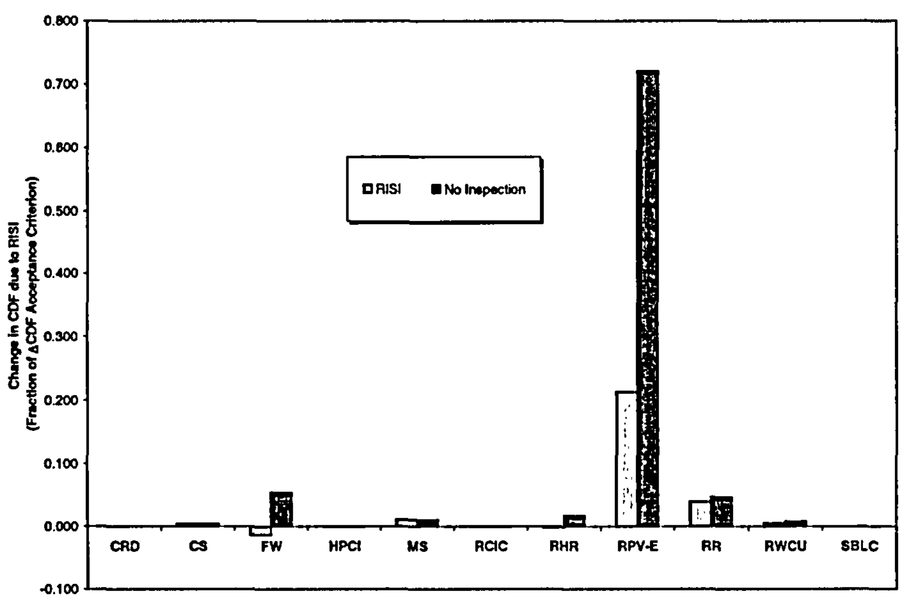

Figure 4
Change in Pipe Rupture CDF for SSES Unit 2 Systems

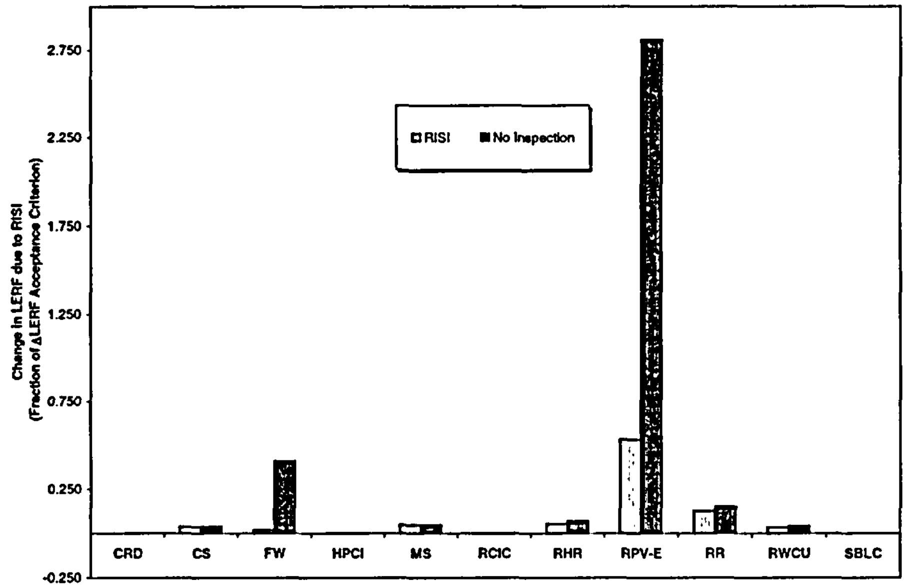

Figure 5
Change in Pipe Rupture LERF for SSES Unit 2 Systems

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RISI program, procedures that comply with the guidelines described in EPRI RISI Topical Report will be prepared to implement and monitor the program. The new program will be integrated into the third ten-year interval for SSES Unit 1 and into the third ten-year interval for SSES Unit 2. No changes to the Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change are to be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures are to be retained and modified to address the RISI process, as appropriate.

The RISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. Such relevant information would include major updates to the SSES Units 1 and 2 PRA models which could impact both the risk characterization and risk impact assessments, any new trends in service experience with piping systems at SSES and across the industry, and new information on element accessibility that will be obtained as the risk informed inspections are implemented. As a minimum, risk ranking of piping element selections will be reviewed and adjusted on an

ASME ISI period basis. In addition, changes may occur more frequently as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific service experience feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the new RISI program and the applicable portions of the previous Second Interval ISI Program (Categories B-F, B-J, C-F-1, and C-F-2 of ASME Section XI, 1989 Edition with no Addenda) is provided in Table 5 and Table 6 for Unit 1 and Unit 2, respectively. The number of exams for Unit 1 is reduced from 205 Section XI program exams to 80 RISI program exams, a net reduction of 125 exams. An additional 127 Section XI exams default to the FAC and IGSCC augmented program welds for a total reduction of 252 exams compared to the 332 Section XI total (80% reduction). Unit 2 is reduced from 171 exams to 80 exams, a net reduction of 91 exams. An additional 118 Section XI exams default to the FAC and IGSCC augmented program welds for a total reduction of 209 exams compared to the 289 Section XI total (72% reduction). Inspections scheduled as part of the FAC or IGSCC augmented programs remain in effect, as these augmented programs are unchanged by the RISI program. As shown in Tables 7 and 8, the total change in CDF and LERF due to the net changes in number and location of inspections in all systems that were evaluated in this risk informed evaluation was found to be well below the risk acceptance criteria. These risk impacts are acceptable in relation to the risk significance thresholds of the EPRI Topical Report and those in Regulatory Guide 1.174.

6. REFERENCES

1. EPRI, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," TR-112657 Rev. B-A, December 1999.
2. Susquehanna Steam Electric Station PRA Calculation EC-RISK-1127.
3. Risk Informed Inservice Inspection Evaluation, Susquehanna Steam Electric Station Units 1 and 2 – Final Report.
4. T.J. Mikschl and K.N. Fleming, "Piping System Failure Rates and Rupture Frequencies for Use in Risk informed Inservice Inspection Applications," EPRI TR-111880, 1999, September 1999. *EPRI Licensed Material*
5. ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1.

Table 1
System Selection for SSES Unit 1 and Unit 2

System Description
Control Rod Drive and Scram Discharge Volume (CRD)
Core Spray (CS)
Feedwater (FW)
High Pressure Coolant Injection (HPCI)
Main Steam (MS)
Reactor Core Isolation Cooling (RCIC)
Residual Heat Removal (RHR)
Reactor Pressure Vessel (RPV-E)
Reactor Recirculation (RR)
Reactor Water Cleanup System (RWCU)
Standby Liquid Control (SBLC)

NOTES: This table shows the systems that contain welds that are Class 1 or Class 2 category B-J, B-F, C-F-1, or C-F-2.

Table 2
Failure Potential Assessment Summary for SSES Unit 1 and Unit 2[2]

System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CRD[1]											
CS	X		X								
FW	X	X									X
HPCI		X									
MS											X
RCIC		X									X
RHR	X	X	X							X	
RPV-E	X	X	X						X		X
RR			X								
RWCU			X								X
SBLC											

NOTES:

1. Includes scram discharge volume.
2. This table shows the assessed failure mechanisms for each system. The RISI Program addresses the cumulative impact of all mechanisms that were identified in each system.

   TASCS – thermal stratification, cycling and stripping, TT – thermal transients, IGSCC – intergranular stress corrosion cracking, TGSCC – transgranular stress corrosion cracking, ECSCC – external chloride stress corrosion cracking, PWSCC – primary water stress corrosion cracking, MIC – microbiologically influenced corrosion, PIT – pitting, CC – crevice corrosion, E-C – erosion-cavitation, FAC – flow accelerated corrosion

Table 3
Number of Elements (Welds) by Risk Category for SSES Unit 1[3]

	High Risk[2]			Medium Risk[2]		Low Risk[2]	TOTAL
System	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]						37	37
CS		8 + (2)[4]		4	0 + (2)[4]	183	199
FW	63 + (33)[5]		3				99
HPCI				22	24	117	163
MS	0 + (256)[5]		(20)[5]				276
RCIC	0 + (18)[5]				28	53	99
RHR		27 + (4)[4]		139	11 + (23)[4]	339 + (7)[4]	550
RPV-E	6 + (4)[5]	16 + (6)[4]		10			42
RR		0 + (60)[4]		79			139
RWCU	0 + (55)[5]	0 + (6)[4]		71			132
SBLC						53	53
TOTAL	435	129	23	325	88	789	1789

NOTES:
1. Includes scram discharge volume.
2. See Figure 1 for definition of EPRI Risk Categories.
3. This table shows the results of the Risk Categorization for Unit 1. The risk categories are defined in Figure 3-4 of EPRI TR-112657 (Reference 1).
4. Includes the number (in parentheses) of the 110 welds in the augmented program for IGSCC.
5. Includes the number (in parentheses) of the 386 welds in the augmented program for FAC.

Table 4
Number of Elements (Welds) by Risk Category for SSES Unit 2[3]

System	High Risk[2]			Medium Risk[2]		Low Risk[2]	TOTAL
	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]						39	39
CS		7 + (2)[4]		5	0 + (2)[4]	179	195
FW	75 + (22)[5]		3				100
HPCI				19	22	111	152
MS	0 + (276)[5]		(21)[5]				297
RCIC	0 + (18)[5]				24	67	109
RHR		22 + (4)[4]		131	16 + (27)[4]	327	527
RPV-E	6 + (4)[5]	15 + (7)[4]		10			42
RR		0 + (62)[4]		77			139
RWCU	0 + (66)[5]	0 + (6)[4]		65			137
SBLC						41	41
TOTAL	467	125	24	307	91	764	1778

NOTES:
1. Includes scram discharge volume.
2. See Figure 1 for definition of EPRI Risk Categories.
3. This table shows the results of the Risk Categorization for Unit 2. The risk categories are defined in Figure 3-4 of EPRI TR-112657 (Reference 1). The minor differences are due to slight differences in the number of welds in these systems.
4. Includes the number (in parentheses) of the 110 welds in the augmented program for IGSCC.
5. Includes the number (in parentheses) of the 407 welds in the augmented program for FAC.

Table 5
Number of Inspections by Risk Category for SSES Unit 1[3]

	High Risk[2]						Medium Risk[2]				Low Risk[2]		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7			
System	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI
CRD[1]											3		3	0
CS			6	2				1			17		23	3
FW	20	17			3	1							23	18
HPCI							7	3	1	3	10		18	6
RCIC									3	3	5		8	3
RHR			12	7			8	14	2	2	31		53	23
RPV-E	6	6	16	4			8	1					30	11
RR							25	8					25	8
RWCU							10	8					10	8
SBLC											12		12	0
TOTAL	26	23	34	13	3	1	58	35	6	8	78	0	205	80

NOTES:
1. Includes scram discharge volume.
2. See Figure 1 for definition of EPRI RISI risk categories.
3. This table provides a comparison of the RISI element selection to the original ASME Section XI program. The total number of inspections is significantly lower for the RISI program. Some RISI inspection locations are new when compared to the Section XI program (i.e., they were previously not addressed).

Table 6
Number of Inspections by Risk Category for SSES Unit 2[3]

System	High Risk[2]						Medium Risk[2]				Low Risk[2]		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7			
	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI	Sec XI	RISI
CRD[1]											3		3	0
CS			5	2			2	1			16		23	3
FW	15	20			2	1							17	21
HPCI							10	2	1	3	5		16	5
RCIC									3	3	4		7	3
RHR			9	6			9	14	5	2	29		52	22
RPV-E	6	6	15	4			4	1					25	11
RR							13	8					13	8
RWCU							6	7					6	7
SBLC											9		9	0
TOTAL	21	26	29	12	2	1	44	33	9	8	66	0	171	80

NOTES:
1. Includes scram discharge volume.
2. See Figure 1 for definition of EPRI RISI Risk Categories.
3. This table provides a comparison of the RISI element selection to the original ASME Section XI program. The total number of inspections is significantly lower for the RISI program. Some RISI inspection locations are new when compared to the Section XI program (i.e., they were previously not addressed).

Table 7
Impact of RISI and No Inspections on CDF and LERF Due to Pipe Ruptures for SSES Unit 1 Systems

System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Section XI	RISI	No Inspection	RISI	No Inspection	Acceptance Criterion	RISI	No Inspection	Acceptance Criterion
CRD[1]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	<1.00E-07	0.00E+00	0.00E+00	<1.00E-08
CS	7.70E-10	1.20E-09	1.24E-09	4.29E-10	4.75E-10	<1.00E-07	3.91E-10	4.05E-10	<1.00E-08
FW	2.74E-08	2.81E-08	3.39E-08	7.04E-10	6.45E-09	<1.00E-07	1.06E-09	4.44E-09	<1.00E-08
HPCI	6.77E-10	7.17E-10	7.22E-10	3.91E-11	4.49E-11	<1.00E-07	2.35E-11	2.92E-11	<1.00E-08
MS	6.63E-09	7.80E-09	7.80E-09	1.17E-09	1.17E-09	<1.00E-07	5.27E-10	5.27E-10	<1.00E-08
RCIC	1.21E-10	1.51E-10	1.52E-10	3.05E-11	3.08E-11	<1.00E-07	1.36E-11	1.38E-11	<1.00E-08
RHR	2.38E-08	2.32E-08	2.62E-08	-6.14E-10	2.37E-09	<1.00E-07	5.23E-10	7.34E-10	<1.00E-08
RPV-E	6.05E-08	8.76E-08	1.32E-07	2.71E-08	7.11E-08	<1.00E-07	8.53E-09	2.35E-08	<1.00E-08
RR	1.91E-08	2.34E-08	2.42E-08	4.33E-09	5.12E-09	<1.00E-07	1.46E-09	1.71E-09	<1.00E-08
RWCU	7.78E-09	8.50E-09	8.81E-09	7.17E-10	1.02E-09	<1.00E-07	3.99E-10	4.98E-10	<1.00E-08
SBLC	7.02E-10	7.06E-10	7.06E-10	3.49E-12	3.49E-12	<1.00E-07	1.16E-12	1.16E-12	<1.00E-08
TOTAL	1.47E-07	1.81E-07	2.35E-07	3.39E-08	8.78E-08	<1.00E-06	1.29E-08	3.19E-08	<1.00E-07

NOTES:

1. Includes scram discharge volume.

Table 8

Impact of RISI and No Inspections on CDF and LERF due to Pipe Ruptures for SSES Unit 2 Systems

System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Section XI	RISI	No Inspection	RISI	No Inspection	Acceptance Criterion	RISI	No Inspection	Acceptance Criterion
CRD[1]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	<1.00E-07	0.00E+00	0.00E+00	<1.00E-08
CS	7.12E-10	1.19E-09	1.23E-09	4.74E-10	5.21E-10	<1.00E-07	3.93E-10	4.07E-10	<1.00E-08
FW	3.25E-08	3.12E-08	3.79E-08	-1.36E-09	5.40E-09	<1.00E-07	5.40E-10	4.43E-09	<1.00E-08
HPCI	6.90E-10	7.13E-10	7.19E-10	2.28E-11	2.87E-11	<1.00E-07	7.56E-12	9.51E-12	<1.00E-08
MS	6.87E-09	7.97E-09	7.97E-09	1.10E-09	1.10E-09	<1.00E-07	5.91E-10	6.11E-10	<1.00E-08
RCIC	1.16E-10	1.51E-10	1.52E-10	3.52E-11	3.55E-11	<1.00E-07	3.24E-11	3.27E-11	<1.00E-08
RHR	2.12E-08	2.11E-08	2.30E-08	-1.25E-10	1.79E-09	<1.00E-07	5.65E-10	6.98E-10	<1.00E-08
RPV-E	6.25E-08	8.39E-08	1.35E-07	2.13E-08	7.21E-08	<1.00E-07	5.33E-09	2.81E-08	<1.00E-08
RR	1.87E-08	2.26E-08	2.34E-08	3.94E-09	4.73E-09	<1.00E-07	4.39E-15	4.62E-15	<1.00E-08
RWCU	8.44E-09	9.03E-09	9.29E-09	5.81E-10	8.50E-10	<1.00E-07	1.94E-09	1.94E-09	<1.00E-08
SBLC	4.25E-10	4.28E-10	4.28E-10	2.62E-12	2.62E-12	<1.00E-07	8.74E-13	8.74E-13	<1.00E-08
TOTAL	1.52E-07	1.78E-07	2.39E-07	2.60E-08	8.66E-08	<1.00E-06	1.09E-08	3.82E-08	<1.00E-07

NOTES:
1. Includes scram discharge volume