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JUN 18 2004

U. S. Nuclear Regulatory Commission
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Washington DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO NRC REQUEST FOR ADDITIONAL
INFORMATION (RAI) RELATIVE TO SUSQUEHANNA
STEAM ELECTRIC STATION, UNITS 1 AND 2
REVISED RESPONSE TO GENERIC LETTER 94-02:
LONG-TERM STABILITY SOLUTION
(TAC NOS. MC 1659 AND MC 1660)**
PLA-5771

**Docket Nos. 50-387
and 50-388**

References: 1) *Letter from R. V. Guzman (NRC) to B. L. Shriver (PPL), "Request for Additional Information (RAI) - Susquehanna Steam Electric Station, Units 1 and 2 - Revised Response to Generic Letter 94-02: Long-Term Stability Solution (TAC NOS. MC 1659 and MC 1660)," dated May 19, 2004.*

2) *PLA-4925, R. G. Byram (PPL) to USNRC, "Proposed Amendment to License NPF-14: Final Response to GL 94-02: Long-Term Stability Solution," dated June 19, 1998.*

3) *PLA-4956, R. G. Byram (PPL) to USNRC, "Proposed Amendment to License NPF-22: Final Response to GL 94-02: Long-Term Stability Solution," dated August 5, 1998.*

4) *PLA-5686, B. L. Shriver (PPL) to USNRC, "Proposed Amendment No. 259 to License NFP-14 and Proposed Amendment No. 224 to License NFP-22: Revised Response to GL 94-02: Long-Term Stability Solution," dated December 22, 2003.*

Enclosed with this letter is the PPL Susquehanna, LLC response to the NRC Request for Additional Information in Reference 1 above.

There are no new commitments made in this letter. If you have any questions, please contact Mr. Duane L. Filchner at (610) 774-7819.

Sincerely,

B. L. Shriver

Enclosure: Response to NRC Request for Additional Information Relating to Revised Response to Generic Letter 94-02: Long Term Stability Solution

Attachment 1: Conformance of the SSES OPRM System design and implementation with the NRC approved Generic Topical Report, CENPD-400-P-A, Rev. 1 and associated Safety Evaluation Report

A001

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

Enclosure to PLA-5771

Response to NRC Request for Additional Information Relative to the Revised Response to Generic Letter 94-02:

LONG-TERM STABILITY SOLUTION

Response to NRC Request for Additional Information Relating to Revised Response to Generic Letter 94-02: Long Term Stability Solution

NRC Question 1:

Confirm the current validity of the plant-specific actions stated in CENPD-400-P-A, Rev. 1, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor."

PPL Response:

The plant specific actions stated in CENPD-400-P-A, Rev. 1, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor," are currently valid for the proposed OPRM implementation. Attachment 4 of PPL letters PLA-4925 and PLA-4956 (References 2 and 3), describes the degree to which the Susquehanna Unit 1 and Unit 2 design and implementation conforms to the applicable NRC approved generic topical reports and NRC Safety Evaluation Report. This previously submitted information is included as Attachment 1 to this letter and has been marked as a revision to show the proposed removal of the channel Period Based Algorithm (Sp) Allowable Value from the LCO statement.

NRC Question 2:

Describe the current implementation and monitoring status (i.e., system tests and operability basis) of the OPRM system, and the analytical approach stated in NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications."

PPL Response:

Hardware implementation and monitoring status: The OPRM hardware and software (designed and implemented in accordance with CENPD-400-P-A, Rev. 1, Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM)) is currently installed in a monitoring and test mode only. There is no current requirement for the OPRM system to be operable, and there are no operator interfaces or active outputs to the plant trip system or to control room alarms.

When made operable in accordance with the proposed Technical Specifications, the equipment and its software are designed and installed to support its safety-related protective function. Implementing setpoints and settings are currently consistent with the expected operating values; the OPRM function is continuously monitored by the plant process computer and it is in the process of being tested for efficacy and reliability by controlled plant test procedures. All Surveillances referenced in the proposed Technical Specifications will be current when the Technical Specifications take effect.

When implemented, all OPRM setpoints and settings will be within the specified bounds of the Licensing Topical Report NEDO-32465-A (Table 3-1).

Analytical Approach: The licensing approach to determine analytical setpoints for the OPRM System is described in Section 4.0 of NEDO-32465-A. The three portions of the Option III licensing methodology are:

1. Calculation of Pre-Oscillation MCPR: This portion of the methodology captures the cycle specific MCPR margin to the safety limit. In demonstrating that the MCPR SL is not violated for anticipated oscillations, the MCPR margin prior to an event is clearly important.

2. Statistical Calculation of Peak Oscillation Magnitude: This portion of the methodology captures the effect of plant characteristics, trip system definition, and setpoint values on the magnitude of the hot bundle power oscillation (before it is terminated by the scram). The output to the analysis is a 95% probability/95% confidence upper bound on the normalized oscillation peak, (Peak – Minimum) / Average, for any fuel bundle in the core as a function of amplitude setpoint. See Section 4.3 of NEDO-32465-A.

3. MCPR Performance of the Hot Bundle: This portion of the methodology captures the effect of fuel design and other factors on the core minimum MCPR due to an oscillation. However, the generic DIVOM curve described in Section 4.4 of NEDO-32465-A was previously determined to be potentially non-conservative as documented by GE as a 10CFR21 report.

   As a result, the fuel vendor for PPL Susquehanna, LLC (Areva/Framatome-ANP), will generate a plant specific DIVOM relationship for the Susquehanna units. This plant specific DIVOM relationship will be used to generate the OPRM setpoints as described in Section 4.5 of NEDO-32465-A.

The status of these three analytical elements is discussed below:

1. Calculation of Pre-Oscillation MCPR

   This analysis will be performed using the 3-D nodal simulator code CASMO-4/MICROBURN-B2 (EMF-2158 (P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors, Evaluation and Validation of CASMO-4/MICROBURN-B2.") using the approach described in Section 4.2 of NEDO-32465-A. The analysis will simulate a flow reduction to natural circulation and calculate the increase in MCPR due to the flow and power decrease for a number of exposures during the cycle. CASMO-4/MICROBURN-B2 is NRC approved for use in licensing analyses of steady state physics events.

2. Statistical Calculation of Peak Oscillation Magnitude

   This analysis is performed by GE based on the Susquehanna OPRM trip system. These calculations have been performed as a function of assumed amplitude setpoint using the statistical methodology described in Section 4.3 of NEDO-32465-A.

3. MCPR Performance of the Hot Bundle

   This analysis will be performed by FRA-ANP using the 3-D transient RAMONA5 code. RAMONA5 models the core MCPR response for a growing oscillation. The neutronics input for RAMONA5 is generated with CASMO-4/MICROBURN-B2. FRA-ANP is developing the RAMONA5 methodology performing validation analyses, and documenting the results per their 10 CFR 50 Appendix B program.

NRC Question 3:

Provide a description of the actions taken by PPL according to the Boiling Water Reactor Owner's Group (BWROG) letter, BWROG-03049 dated September 30, 2003, "Utility Commitment to NRC for OPRM Operability at Option III Plants." Please identify any plant-specific differences from the generic values specified in NEDO-32465-A such as Period Based Detection Algorithm (PBDA) period confirmation setpoints in Table 3-1, PBDA trip setpoints in Table 3-2, and generic DIVOM (Delta critical power ratio (CPR) to Initial CPR vs. Oscillation Magnitude) curve slope.

PPL Response:

The BWROG letter essentially recommends that each utility inform the NRC of their plans to arm the OPRM. PPL's Technical Specification submittal, (Reference 4) fulfills the intent of the BWROG letter. The generic DIVOM curve in NEDO-32465-A is not used. A plant specific curve is being generated as discussed in the response to Questions 2 and 5.

The values of the period tolerance and the conditioning filter cutoff frequency setpoints used in the OPRM for the Susquehanna units will be within the range specified in Table 3-1 of NEDO-32465-A. Also, the values of the period confirmation count setpoint will be within the ranges specified in Table 3-2 of NEDO-32465-A.

NRC Question 4:

Provide the rationale for locating the OPRM Period Based Algorithm Allowable Value and Confirmation Counts in the COLR.

PPL Response:

The Technical Specification 5.6.5, CORE OPERATING LIMITS REPORT, (COLR) requires the core operating limits to be prepared prior to each reload cycle and documented in a unit specific COLR for each Susquehanna unit (1 and 2). It is proposed to include the setpoints for the new OPRM Technical Specification 3.3.1.3 in the COLR because these setpoints are developed for each reload cycle, utilizing NRC-approved methods, and established such that all applicable limits of the plant safety analysis are met. Including this information in the COLR is consistent with the stated criteria for the contents of the COLR. By reference from the COLR, the Period Based Algorithm setpoints become integral to the Operability basis for the OPRM protective function.

NRC Question 5:

Identify the methodologies and key inputs used to determine the OPRM setpoints for TS 3.3.1.3 and indicate whether they are NRC-approved methods.

PPL Response:

The methodology used to determine the OPRM setpoints for TS 3.3.1.3 is contained in NEDO-32465-A which is NRC approved. NEDO-32465-A makes allowance for the fuel vendors to generate plant specific DIVOM curves in the event of the introduction of new fuel types. As stated in "BWROG-0368," the BWROG has concluded and the NRC has agreed that use of plant specific DIVOM curves is an acceptable way to resolve the 10 CFR 21 Reportable Condition. Use of a plant specific DIVOM curve will capture the impact of plant specific differences (operating domains, energy loading, core loading strategies, etc.). The following is a list of codes and key inputs used to develop the SSES plant specific OPRM setpoints.

Input	Code	Status
Initial MCPR (IMCPR)	CASMO4/MICROBURN-B2	NRC approved for other similar licensing applications (EMF-2158(P)(A)) Consistent with Section 4.2 of NEDO-32465-A.
Hot Channel Oscillation Magnitude (function of assumed amplitude setpoint)	Statistical Model described in Section 4.3 of NEDO-32465-A.	NRC Approved
DIVOM Curve	RAMONA5 Neutronic input taken from CASMO4/MICROBURN-B2	Code is being documented and validated for DIVOM application by FRA-ANP NRC approved for other licensing applications (EMF-2158(P)(A))

NRC Question 6: Describe how the alternate method will be employed to detect and suppress thermal hydraulic instability oscillations stated in TS 3.3.1.3 Actions A.3 and B.1.

PPL Response: The actions relating to the alternate method referenced in TS 3.3.1.3 Actions A.3 and B.1 to detect and suppress instability oscillations are currently located in TS 3.4.1 and will be removed by this proposed change. These actions are being relocated to the Technical Requirements Manual. The instability regions will be re-evaluated each cycle and the regions specified in the COLR portion of the Technical Requirements Manual.

Attachment 1 to PLA-5771

Conformance of the SSES OPRM System design and implementation with the NRC approved Generic Topical Report, CENPD-400-P-A, Rev. 1 and associated Safety Evaluation Report

The following information is provided from References 2 and 3. It has been revised as indicated for these proposed TS changes to remove the setpoint from the LCO statement.

Conformance of the SSES OPRM System design and implementation with the NRC approved Generic Topical Report, CENPD-400-P-A, Rev. 1 and associated Safety Evaluation Report.

The NRC, in their Safety Evaluation Report accepting the Asea Brown Boveri Combustion Engineering (ABB-CE) Option III - OPRM system as a permanent long-term solution for the long standing thermal-hydraulic instability issue, required that each plant referencing the OPRM system licensing topical report, CENPD-400-P-A, Rev. 1, properly address each area of the design and implementation of the system in their Technical Specification submittal.

The following are the applicable areas identified in the NRC SER and the conformance of the SSES OPRM system design and implementation to these requirements:

1) System Description

Implementation of the OPRM system at SSES is consistent with the 'system description' as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1.

The only deviation from the system as described in the NRC approved licensing topical report is the implementation of the OPRM 'Trouble' alarm annunciation in the control room. This alarm function in the control room was not implemented at SSES due to the fact that the alarm only indicates several process conditions that do not impact the OPRM's performance of its safety-related function (e.g., the deviation between the Master OPRM module's APRM power signal from that of the Slave OPRM module's APRM power signal). However, the alarm is indicated locally at the OPRM installed location and SSES has implemented the option of providing all the OPRM annunciation outputs to the operators via the plant integrated computer system (PICSY).

2) Single Failures

Implementation of the OPRM system at SSES is in conformance with the 'single failure criterion' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1. For the case of the OPRM modules residing in the unassigned LPRM pages, their APRM power signals and Total Flow signals are derived through analog isolation devices from APRMs selected to preserve the system function in the event of a single failure.

3) Equipment Qualification/Channel Integrity

Implementation of the OPRM system at SSES is in conformance with the 'equipment qualification and channel integrity' requirements as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1. The installed location of the OPRM is a controlled, mild environment. The OPRM equipment environmental qualification testing envelopes the following Lower Relay Room environmental conditions as listed in FSAR Table 3.11-1, with the exception of the Relative Humidity:

Temperature:	Normal 60°F to 80°F, Abnormal 90°F
Relative Humidity:	Normal 10 to 60%
Radiation:	Less than 1.8×10^2 Rads TID (normal + abnormal)

The OPRM equipment was tested at a Relative Humidity range of 40% (±5) to 95% (+4%) which did not envelope the lower end of the SSES Lower Relay Room Relative Humidity value of 10%. However, this deviation and its possible consequences has been analyzed by the system vendor, ABB-CE, and determined to be acceptable.

The electromagnetic interference (EMI) environment of the installed location of the OPRM (i.e., Lower Relay Room) was tested and the OPRM was found to be electromagnetically compatible with the existing equipment.

The OPRM is seismically qualified by type testing per IEEE-344. The addition of these modules will not affect the overall seismic qualification of the PRNMS Panels (1C608) or the APRM system components within them.

4) Channel Independence

Implementation of the OPRM system at SSES is in conformance with the 'channel independence' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1.

The OPRM is designed with signal isolation to ensure that there are no safety impacts to existing plant systems. Signal level isolation and cable separation are provided where system interfaces cross channel boundaries. The OPRM modules' outputs to the RPS trip system are through safety-related trip relays and are compatible with the existing neutron monitoring system trip channels.

The OPRM modules are individually located in APRM/LPRM pages within the PRNMS panels (1C608/2C608) and inter-module communication is through fiber optic data links. The OPRM only interfaces with LPRMs in a compatible separation group. Qualified isolation is provided where the OPRM interfaces with an APRM associated with another separation group. The OPRM communicates with the PICSY via fiber optic cable to maintain the electrical isolation between the 1E and non-1E equipment. This feature also provides the isolated output of each APRM and LPRM to PICSY, allowing the existing hard-wired interface to be removed.

The interface between the OPRM and the non-1E annunciator equipment is through qualified relay isolation. The cables for the non-1E portions of the equipment maintain appropriate separation within the panels.

5) Channel Bypass or Removal from Operation/Access to Setpoint and Test Points

Implementation of the OPRM system at SSES is in conformance with the 'channel bypass or removal from operation' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1. Each OPRM module is provided with a keylock switch for module bypass at the installed location of the OPRM. Each module manual bypass or manual enable function is independent of all other channels. Access to the OPRM functional capabilities (e.g., OPRM module configuration or setpoint changes) is password protected and requires the use of the administratively controlled OPRM keys and Maintenance Terminal.

6) Indication of Bypasses

Implementation of the OPRM system at SSES is in conformance with the 'indication of bypass' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1. When the OPRM is bypassed, this condition is alarmed in the control room panels (1C651/2C651).

7) Information Readout

The SSES implementation of the OPRM system is in conformance with the 'information readout' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1. The OPRM equipment status is available to the operator in the control room on the indication panel (1C651/2C651) and locally at the PRNMS panel (1C608/2C608). Additionally, the OPRM communicates with PICSY providing operating data and status and all LPRM and APRM data needed for the plant computer interface with the PRNMS.

The addition of the OPRM system annunciation windows to the indication panel (1C651/2C651) did not require changes to this panel beyond the activation of existing annunciation windows and etching of the activated windows with the specific OPRM alarmed condition (e.g., 'Bypass/INOP', 'Trip Enable', 'Alarm'). The changes to the indication panel (1C651/2C651) are in accordance with SSES' human factors manual.

8) Technical Specification

Implementation of the OPRM system at SSES is in conformance with the 'technical specification' requirement as stated in the NRC SER approving the generic licensing topical report for the OPRM system, CENPD-400-P-A, Rev. 1.

The new OPRM system technical specification section and bases are consistent with the NRC approved generic technical specification.

The following differences with the NRC approved Generic Technical Specifications are proposed:

~~LCO 3.3.1.3 OPRM Period Based Algorithm (PBA) and Confirmation Count Permissive (N_p)~~

~~The SSES OPRM Period Based Algorithm (PBA) Allowable Value (S_p) and associated Confirmation Count Permissive (N_p) are delineated in the LCO statement for the OPRM; the LCO Basis reflects the setpoint requirements. The NRC approved generic technical specification does not specifically list these parameters, however their SER requires that the plant specific setpoint values be provided on a plant specific basis.~~

OPRM Instrumentation Bases 3.3.1.3 Applicability Basis

The Applicability Basis statement has been rewritten to more clearly and accurately reflect the basis for requiring OPRM operability at or above 25% RTP. No change in purpose or intent is reflected in the proposed change.

Surveillance Requirement 3.3.1.3.5 Reactor Core Flow Vs. Drive Flow

OPRM functional requirements stipulate that the trip function be enabled based on Reactor Recirculation Core Flow, rather than on Reactor Recirculation Drive Flow, as specified in the Generic Technical Specification. The analytical basis for enabling the OPRM protective function is based on the core operating conditions under which thermal-hydraulic instabilities have been observed to occur. This implies and requires system functionality below the specified core flow limit. Reactor Recirculation Drive Flow will be used to determine the core flow, based on the known drive flow/core flow relationship, for the system operating bypass.

Bryce L. Shriver
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Chief Nuclear Officer

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JUN 18 2004

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**SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL
INFORMATION FROM NRC ON PROPOSED
RELIEF REQUEST NOS. 3RR-01, 3RR-02, AND 3RR-04
TO THE THIRD 10-YEAR INSERVICE INSPECTION
PROGRAM FOR SUSQUEHANNA SES UNITS 1 AND 2**
PLA-5767

**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: Pressure Retaining Welds," dated April 28, 2004.*

This letter is in response to the above referenced letter. The Enclosure 1 to this letter contains PPL Susquehanna, LLC's response to the Request for Additional Information questions on Relief Request 3RR-01. The Enclosure 2 to this letter contains PPL Susquehanna, LLC's response to the Request for Additional Information questions on Relief Request 3RR-02. The Enclosure 3 to this letter contains PPL Susquehanna, LLC's response to the Request for Additional Information questions on Relief Request 3RR-04.

There are no new commitments made in this letter. If you have any questions, please contact Mr. C. T. Coddington at (610) 774-4019.

Sincerely,

B. L. Shriver

Enclosures:
Enclosure 1: Response to NRC Request for Additional Information Relating to Relief Request No. 3RR-01
Enclosure 2: Response to NRC Request for Additional Information Relating to Relief Request No. 3RR-02
Enclosure 3: Response to NRC Request for Additional Information Relating to Relief Request No. 3RR-04

Attachments:
Attachment A to Enclosure 1 - Revised Relief Request 3RR-01
Attachment A to Enclosure 2 - Second Interval Inspections for RHR and Core Spray Pumps
Attachment B to Enclosure 2 - Revised Drawing for 3RR-02
Attachment A to Enclosure 3 - Additional Drawings for Relief Request 3RR-04

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

Enclosure 1 to PLA-5767

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

Response to NRC RAI on Relief Request 3RR-01

NRC Question No. 1:

PPL references an Item Number B5.140 from Table IWB-2500-1, Examination Category B-F. Table IWB-2500-1, Examination Category B-F, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1998 Edition through the 2000 Addenda does not list an Item Number B5.140. Please delete or correct the item number.

PPL Response:

The reference to Item Number B5.140 is incorrect. This item number does not exist in ASME Section XI, 1998 Edition through the 2000 Addenda. It will be deleted from the Relief Request Text. See Attachment A to Enclosure 1 for corrected version (Revision 1).

NRC Question No. 2:

In PPL's Risk Informed Inservice Inspection (RI-ISI) program Report, Section 3.5, "Inspection Location Selection and NDE Selection under Additional Examinations," PPL states that "examinations performed that reveal flaws or relevant conditions exceeding the applicable acceptance standards shall be extended to include additional examination." Please provide a specific time line for all scenarios that would require an inspection sample increase. Also, be more specific as to the time frame of additional inspections that may become necessary if flaws are found in the first sample expansion. The Nuclear Regulatory Commission (NRC) staff expects that sample expansion examinations will be performed in the same time frame that is outlined in the ASME, Section XI IWB-2430.

PPL Response:

In accordance with the Proposed Alternative in Relief Request 3RR-01, the RI-ISI program will perform any additional examinations required as a result of flaws or relevant conditions in accordance with Code Case N-578-1 Subarticle 2430. In addition to the provisions of this Subarticle, PPL will perform any examinations of either a first or second sample expansion during the same outage timeframe as when the original flaws or relevant conditions are found.

NRC Question No. 3:

PPL states that it will use Table 1, Examination Category R-A, "Risk Informed Piping Examinations," contained in Code Case N-578-1. Does PPL have any elements that fall under Item No. R1.19 that are socket welds? If so, does PPL intend to take exception to Note 12 of the table and perform a surface exam?

PPL Response:

PPL does not have any components identified under Item No. R1.19, Elements Subject to External Chloride Stress Corrosion Cracking (ECSCC), and therefore will not be taking exception to Note 12.

NRC Question No. 4:

In PPL's RI-ISI Program Report, the introduction section states, "To strengthen the technical basis for the RI-ISI program beyond the minimum requirements implied by the Electric Power Research Institute, RI-ISI Topical Report (TR), a number of enhancements were made to the process that are described in the paragraphs below." Please provide a list of all enhancements.

PPL Response:

The EPRI Topical Report and NRC acceptance guidelines for RI-ISI provide a degree of flexibility in how the approaches are implemented on specific plants. In addition, there are new insights available from the NRC and other independent reviews that give rise to enhancements that will be implemented. These enhancements will adhere to the prevailing guidance documents but will further strengthen the technical basis of the risk-informed program. These enhancements are summarized below and compared against the EPRI Topical Report in Table 4-1.

Use of the Existing PRA Models

The original EPRI method attempted to minimize the use of the existing plant-specific PRA models. A process of using look-up tables to determine the conditional core damage probabilities from assumed pipe breaks was developed that obscures the PRA process. This was initially confusing to the NRC in the early stages of RI-ISI and is also foreign to utility PRA groups that are trying to use plant PRA models in a consistent way to support a variety of applications. PPL will use the PRA models for all calculations involving CDF or CCDP. In this way, the resulting RI-ISI program can be updated in the future by the PPL PRA team without having to learn an independent PRA procedure.

Multi-attribute Decision Metric for Element Prioritization

The enhanced approach includes a multi-attribute decision metric in which each pipe element is assigned an index that captures the following factors potentially influencing the selection of locations for examination: scaffolding requirements, radiation levels, expected exposures, evidence of damage in previous exams, existence of baseline inspections, risk impact of increases and decreases in the number of exams, etc. The valuation of these indices and weighting factors are controlled by the user so that the optimum element selection is achieved. This process ensures that risk acceptance guidelines are optimized while factoring in other critical aspects of a solid element selection process.

Table 4-1
Comparison of EPRI and PPL Enhanced RI-ISI Methods

Element of RI-ISI Methodology	EPRI Method per TR-112657	Enhanced RI-ISI Method per PPL RI-ISI Program
Damage Mechanism Evaluation	Use of Screening Evaluation to evaluate potential for specific damage mechanisms; no consideration of what to do for multiple damage mechanisms in same location.	Use of EPRI method is augmented in two respects: failure rates for leaks and ruptures quantified for each pipe element to provide an assessment with and without credit for inspections; special treatment of elements subject to multiple damage mechanisms to account for synergy effects.
Consequence Evaluation	Use of consequence tables supplement by some PRA runs to calibrate worth of system "trains," focus on CDF with limited consideration of LERF.	Direct use of existing PRA models to calculate both CCDPs and CLERPs for all pipe elements in lieu of EPRI consequence tables; results in more realistic assessment of consequences for a range of pipe breaks.

Element of RI-ISI Methodology	EPRI Method per TR-112657	Enhanced RI-ISI Method per PPL RI-ISI Program
Risk Characterization of Pipe Segments	Use of EPRI Risk Matrix	Use of EPRI Risk Matrix supplemented by calculation of incremental delta risk for each pipe element; yields better risk optimization of initial element selection and enables the delta risk evaluation to be done at the same time that elements are selected.
Element Selection Process	25% of High risk elements, 10% of Medium risk elements, and 0% of Low risk segments selected by multidiscipline panel (Element Selection Meeting); elements selected by engineering judgement.	Uses same sampling percentages as they are the ones approved by the NRC. Uses a multi-attribute decision metric to assign priorities for each element to increase quality of selected elements.
Risk Impact Assessment	Uses combination of qualitative and bounding quantitative evaluations; does not emphasize risk impact quantification. Markov method one option available to support this task.	Realistic risk impact assessment of all pipe elements in terms of change in pipe failure and rupture frequency, CDF and LERF. Enhanced Markov Model used to calculate inspection effectiveness factors.
Documentation	Documentation Template for NRC Submittal plus additional reports for each aspect of evaluation.	Documentation Template for all project documentation; uses similar submittal revised to account for additional quantitative risk information.

Element of RI-ISI Methodology	EPRI Method per TR-112657	Enhanced RI-ISI Method per PPL RI-ISI Program
Completed Full Scope Plant Evaluations (N-578-1)	Based on pilot studies at ANO-2 and Fitzpatrick.	Additional insights gained from applying same enhanced process at Byron 1 and 2, Braidwood 1 and 2, LaSalle 1 and 2, Quad Cities 1 and 2, Dresden 2 and 3, Clinton 1, Limerick 1 and 2, Peach Bottom 2 and 3, and Three Mile Island 1.

Calculation of Pipe Rupture Risk For All Elements

Another aspect of the EPRI method is that numerical estimates of pipe rupture frequencies are only introduced in the last steps when the delta risk evaluation is performed. The PPL RI-ISI Team has already developed the methods and databases to very easily and quickly make these estimates. Prior to the element selection process, the RI-ISI Team calculated not only the conditional core damage probabilities from the most recent plant-specific PRA models, but also estimated the rupture frequencies for each piping element.

This additional quantification of risk impacts, that goes beyond the standard EPRI method, can then be used to refine the risk characterization of elements to support the selection process.

NRC Question No. 5:

Identify any portion of Code Case N-578-1 that PPL has used in the development of its RI-ISI program that are not specifically incorporated into or referenced by EPRI TR-11265, Revision B.

PPL Response:

The two requests specified in the Proposed Alternative section of Relief Request 3RR-01 are the only portions of Code Case N-578-1 that will add to and/or supercede requirements in the EPRI Topical Report. PPL views these two modifications to the EPRI process as enhancements rather than alternatives.

Attachment A to Enclosure 1 to PLA-5767

Revised Relief Request 3RR-01

RELIEF REQUEST NUMBER: 3RR-01, Rev. 1

COMPONENT IDENTIFICATION

Code Class:	1 and 2
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, 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 Station Units 1 and 2 (Dated July 2003)

CODE REQUIREMENT

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

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, branch connections, 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:

   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 alloy 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.

RELIEF REQUEST NUMBER: 3RR-01, Rev. 1

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(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 10 CFR 50.55a 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 RI-ISI Program evaluation is an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RI-ISI methodology. For the Third Interval ISI update, there is no traditional ASME Section XI selection to compare with 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 RI-ISI 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 RI-ISI 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 RI-ISI 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 10 CFR 50.55a(a)(3)(i).

The Third Interval RI-ISI 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 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 RI-ISI 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 RI-ISI 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.

Enclosure 2 to PLA-5767

Response to NRC Request for Additional Information Relating to Relief Request 3RR-02

Response to NRC RAI on Relief Request No. 3RR-02

NRC Question No. 1:

PPL did not clearly state what the impracticality is in performing the required inspections. Based on the understanding that the NRC staff currently has regarding the licensee's request, the staff would not consider the licensee's situation to be impractical and would need supporting information to support a review under Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.55a(a)(3)(i) or (ii). Should PPL decide to apply for relief under (3)(i), it would need to show that its alternative provides an acceptable level of quality and safety. Should PPL choose to apply for relief under (3)(ii), it would need to make a clear case that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

PPL Response:

This relief request has been approved as written for the second 10-year inspection interval in Relief Request RR-1. Having obtained NRC approval for this relief during the second ten-year inspection interval, PPL resubmitted this relief as a proactive measure knowing that the referenced welds are not accessible for inspection due to plant configuration. In the "Basis For Relief" section in Relief Request 3RR-02, PPL states that conformance with the specified ASME Section XI requirement has been determined to be impractical, pursuant to 10 CFR 50.55a(g)(5)(iii). ASME Section XI requires a surface examination of Class 2 pump casing welds. The specific welds on the Residual Heat Removal (RHR) and Core Spray (CS) pump casings for which relief is requested are located either inside the pump or embedded in concrete, and are therefore impractical to examine. Access for examination would only be possible through destructive methods or possibly some limited coverage could be achieved from the internal surface depending upon how much disassembly of the pumps could be performed. The disassembly of the pumps for the sole purpose of examination is a major effort that could result in damage to the pumps with only very minimal exam area coverage being accessible based on past experience. Thus, even with disassembly, the majority of the welds in question are still inaccessible and could not be inspected without destructive measures on the pump and/or surrounding structure. Therefore, based on this configuration as shown on the drawings attached to Relief Request 3RR-02 and the other supporting information provided, PPL believes that the ASME Section XI required examinations are impractical for these welds. For the remaining eight (8) RHR Pump casing welds and seven (7) CS Pump casing welds, the Code required exams will be completed in accordance with Section XI.

Previously relief was granted under 10 CFR 50.55a(g)(6)(i) for the second 10-year inservice inspection interval in NRC Safety Evaluation Report (SER) dated June 7, 1996. The conclusion in this SER stated, "Based on the impracticality of meeting the Code requirements and the burden associated with the disassembly of the pumps, it is recommended that the relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) provided that the pumps are examined if disassembled. The examination of other RHR pump casing welds and the performance of the system pressure tests will provide reasonable assurance of the pumps operational readiness."

Situations where a component's design or the design of the component surroundings and the plant structure do not allow for examination are typical examples for the use of 10 CFR 50.55a(g)(5)(iii) and approval under 10 CFR 50.55a(g)(6)(i). If the NRC believes that the subject of Relief Request 3RR-02 fits better under Section 50.55a(a)(3)(ii), PPL would not object to the Staff authorizing relief based on the justifications already provided.

NRC Question No. 2:

Please provide information regarding previous pump weld inspections including scope of inspections and results.

PPL Response:

During the second inspection interval the Unit 1 "B" RHR pump was pulled for normal maintenance. Due to the size and configuration of the pump casing, a surface examination of the casing welds was deemed impractical. Based upon an approved Relief Request for Columbia Generating Station (Relief Request 2ISI-20), Susquehanna performed a Visual (VT-1) examination of the welds. This approach was documented in the PPL system and was approved by the ANII. These examinations found no indications in the pump casing welds.

A listing of the ASME Section XI examinations that were able to be performed during the second 10-year interval is provided in Attachment A to Enclosure 2. No indications have been found on the pump casing welds that are accessible for examination. Limerick Generating Station (LGS) has a similar relief request (RR-07) for Pressure Retaining Welds in Pumps (pump casing welds) associated with the RHR and CS system pumps.

NOTE: A typographical error was discovered on Figure 3RR-02.1 for the CS Pump number. This error has been corrected in Revision 1 of the subject relief request (see Attachment B to Enclosure 2). Also, it should be noted that the weld nomenclature is slightly different between the pumps, but the number and location of the welds is consistent.

NRC Question No. 3:

If the pumps are disassembled for maintenance or repair, does PPL intend to perform the required examination in accordance with the Code requirements?

PPL Response:

If welds or portions of welds are made accessible for inspection due to maintenance or repair activities, PPL will perform the required ASME Section XI examinations of those portions of welds made accessible through the disassembly activities. Additional disassembly will not be performed for the sole purpose of inspection.

NRC Question No. 4:

PPL does not provide an adequate explanation/description of its alternative in the "Proposed Alternative Examinations" section of its request. Please explain in clear and concise manner, the proposed alternative.

PPL Response:

In the event the subject welds, or portions of the welds, become accessible upon disassembly of any one (1) of the pumps in each of these systems (pump groups), the accessible portions of the welds will be surface examined from the inside surface to the maximum extent practicable and in accordance with the applicable Section XI requirements for surface examinations. In addition, all pumps will be subject to the visual examination requirements of Examination Category C-H, thus, providing assurance of pump structural integrity.

NRC Question No. 5:

PPL states that other required examinations/test of pumps will be performed to provide reasonable assurance of structural integrity, but the examinations/tests and the frequency that PPL refers to is not provided. Please provide more information and basis for relief regarding the performance and frequency of the other required examination and or tests of pumps.

PPL Response:

Quarterly flow surveillance tests are performed on both the RHR and Core Spray pumps. These surveillances are conducted per PPL Procedures and are designed to test both loops of the systems.

Attachment A to Enclosure 2 to PLA-5767

Second Interval Inspections for RHR and Core Spray Pumps

Core Spray Pumps								
Unit	Component	Description	System	Interv.	Period	Outage	Exam	Results
1	1P206A-361-1-5	SH-FL	CS	2	1	08-09	SUR	NRI
1	1P206A-361-3-13	FL-P	CS	2	1	08-09	SUR	NRI
1	1P206A-361-4-6	E-FL	CS	2	3	13	SUR	NRI
1	1P206A-361-5-13	P-SH	CS	2	3	13	SUR	NRI
1	1P206A-361-5-6	E-SH	CS	2	3	13	SUR	NRI
1	1P206A-361-5-8	PLT-SH	CS	2	2	10	SUR	NRI
1	1P206A-361-5-L	LS(ADJ 361-5-8)	CS	2	2	10	SUR	NRI
2	2P206D-361-1-5	SH-FL	CS	2	3	11	SUR	NRI
2	2P206D-361-3-13	FL-P	CS	2	1	07	SUR	NRI
2	2P206D-361-4-6	E-FL	CS	2	1	07	SUR	NRI
2	2P206D-361-5-13	P-SH	CS	2	1	07	SUR	NRI
2	2P206D-361-5-6	E-SH	CS	2	1	07	SUR	NRI
2	2P206D-361-5-8	PLT-SH	CS	2	3	11	SUR	NRI
2	2P206D-361-5-L	LS(ADJ 361-5-8)	CS	2	3	11	SUR	NRI

RHR Pumps								
Unit	Component	Description	System	Interv.	Period	Outage	Exam	Results
1	1P202A-361-1-5	SH-FL	RHR	2	1	09	SUR	NRI
1	1P202A-361-13-L	LS(UP 361-5-13)	RHR	2	3	13	SUR	NA
1	1P202A-361-3-13	FL-P	RHR	2	2	11	SUR	NRI
1	1P202A-361-4-6	P-FL	RHR	2	2	11	SUR	NRI
1	1P202A-361-5-13	P-SH	RHR	2	2	11	SUR	NRI
1	1P202A-361-5-6	E-SH	RHR	2	1	09	SUR	NRI
1	1P202A-361-5-8	PLT-SH	RHR	2	3	13	SUR	NRI
1	1P202A-361-5-L	LS(ADJ 361-5-8)	RHR	2	3	13	SUR	NRI
1	1P202B-359-1-C	RHR Pump Welds	RHR	2	2	11	VT-1	NRI
1	1P202B-359-2-C	RHR Pump Welds	RHR	2	2	11	VT-1	NRI
1	1P202B-359-2-L	RHR Pump Welds	RHR	2	2	11	VT-1	NRI
1	1P202B-361-2-6	FL-E	RHR	2	2	11	VT-1	NRI
1	1P202B-361-6-7	E-P	RHR	2	2	11	VT-1	NRI
1	1P202B-361-7-8	P-PLT	RHR	2	2	11	VT-1	NRI
2	2P202D-361-1-5	SH-FL	RHR	2	2	09	SUR	NRI
2	2P202D-361-13-L	LS(UP 361-5-13)	RHR	2	2	09	SUR	NRI
2	2P202D-361-3-13	FL-P	RHR	2	2	09	SUR	NRI
2	2P202D-361-4-6	E-FL	RHR	2	3	11	SUR	NRI
2	2P202D-361-5-13	P-SH	RHR	2	2	09	SUR	NRI
2	2P202D-361-5-6	E-SH	RHR	2	2	09	SUR	NRI
2	2P202D-361-5-8	PLT-SH	RHR	2	3	11	SUR	NRI
2	2P202D-361-5-L	LS(ADJ 361-5-8)	RHR	2	2	09	SUR	NRI

Attachment B to Enclosure 2 to PLA-5767

Revised Drawing for 3RR-02

RELIEF REQUEST NUMBER: 3RR-02

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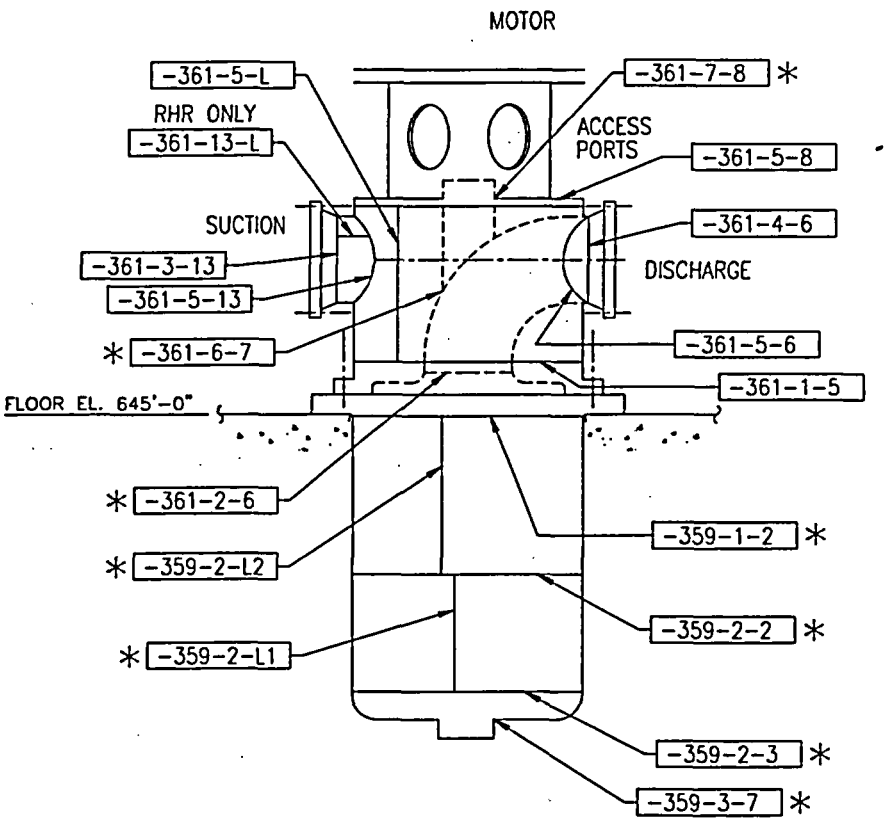
FIGURE 3RR-02.1, Rev. 1

PUMP CASING WELDS
CORE SPRAY PUMPS (1P206A, B, C, D AND 2P206A, B, C, D)
AND
RESIDUAL HEAT REMOVAL PUMPS (1P202A, B, C, D AND 2P202A, B, C, D)



ELEVATION VIEW THRU FLOOR

* Inaccessible welds for which relief is requested.

(Note that the weld nomenclature is slightly different between the pumps, but the number and location of the welds is consistent.)

Enclosure 3 to PLA-5767

Response to NRC Request for Additional Information Relating to Relief Request 3RR-04

Response to NRC RAI on Relief Request No. 3RR-04

NRC Question No. 1:

Please provide a drawing that clearly shows interference that prevents a Code-required inspection from being performed (including dimensions).

PPL Response:

PPL Drawings C-198623, FF113010 sheet 5501, and FF113011 sheet 8101 are provided to illustrate the obstructions which preclude complete examination of the CRD housing welds discussed in Relief Request 3RR-04. (See Attachment A to Enclosure 3).

NRC Question No. 2:

Discuss any present or past degradation issues concerning the control rod drive housing welds at SSES 1 and 2 or other similar units.

PPL Response:

There are no degradation issues related to the Susquehanna Unit 1 or Unit 2 CRD housing welds for which relief is requested.

NRC Question No. 3:

Discuss the inspection scope of the affected welds for the second 10-year inservice inspection interval.

PPL Response:

Since relief from inspection of the peripheral CRD Housing-to-Flange welds was granted prior to the start of the second interval in Relief Request 2RR-8, these welds were not inspected through the required ASME Section XI examinations (surface or volumetric). Rather, they were inspected as part of the vessel VT-2 examination prior to startup at the end of each refueling outage. The CRD Housing pipe-to-pipe welds (not included as part of 3RR-04) are accessible and 10% of the peripheral units will be examined as required by Section XI.

NRC Question No. 4:

PPL does not clearly state what the impracticality is in performing the required inspections. Based on the understanding that the NRC staff currently has regarding the licensee's request, the staff would not consider the licensee's situation to be impractical and would need supporting information to support a review under Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.55a(a)(3)(i) or (ii). Should PPL decide to apply for relief under (3)(i), it would need to show that its alternative provides an acceptable level of quality and safety. Should PPL choose to apply for relief under (3)(ii), it would need to make a clear case that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Provide an explanation of the hardship or unusual difficulty in performing the ASME Section XI required examination with as much supporting information as possible.

PPL Response:

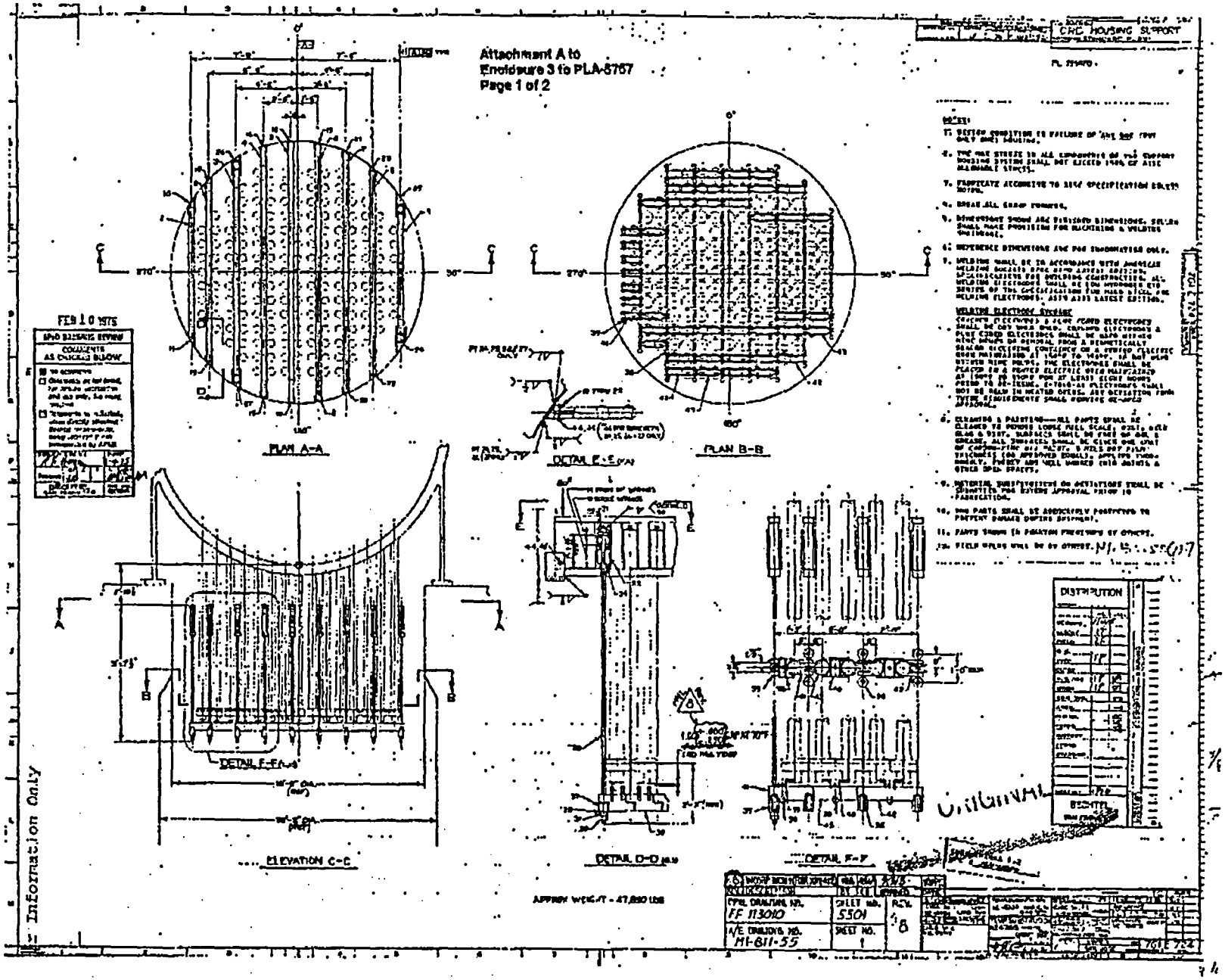
This relief request has been approved as written for the second 10-year inspection interval. Having obtained NRC approval for this relief during the second ten-year inspection interval, PPL resubmitted this relief as a proactive measure knowing that the referenced welds are not accessible for inspection due to plant configuration. In the "Basis For Relief" section in relief request 3RR-04, PPL states that conformance with the specified ASME Section XI requirement has been determined to be impractical, pursuant to 10 CFR 50.55a(g)(5)(iii). The CRD Housings at SSES each have two pressure-retaining welds, the (housing-to-flange) weld and the (housing tube A-to-housing tube B) weld. ASME Section XI requires a surface or volumetric examination of the pressure retaining welds in 10% of the peripheral CRD housings. However, the examination of the (housing-to-flange) weld is impractical due to numerous physical obstructions that prohibit access to this area. The lower (housing-to-flange) welds are located well below the tube-to-tube weld in an area with little access. To gain access for examination, the CRD housings and surrounding obstruction would require design modifications and destructive activities. Therefore, based on the restrictions noted as shown on the drawings contained in Attachment E and the other supporting information provided, PPL believes that the ASME Section XI required examinations are impractical for the housing-to-flange welds. For 10% of the tube-to-tube welds, the Code required exams will be completed in accordance with Section XI.

Previously relief was granted for the second 10-year inservice inspection interval via NRC SER dated June 7, 1996 per relief request 2RR-08. The conclusion in this SER stated, "Based on the impracticality of performing the Code-required examinations on the subject CRD welds, it is recommended that the relief be granted pursuant to 10 CFR 50.55a(g)(6)(i). The examinations that will be performed provide reasonable assurance of the operational readiness of the CRD housing welds."

Situations where a component's design or the design of the component surroundings and the plant structure do not allow for examination are typical examples for the use of 10 CFR 50.55a(g)(5)(iii) and approval under 10 CFR 50.55a(g)(6)(i).

Attachment A to Enclosure 3 to PLA-5767

Additional Drawings for Relief Request 3RR-04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CRC HOUSING SUPPORT
PLAN A-A
PLAN B-B
DETAIL E-E
ELEVATION C-C
DETAIL D-D
DETAIL F-F
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