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OCT 0 2 2006

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SUSQUEHANNA STEAM ELECTRIC STATION
10 CFR 50.59 SUMMARY REPORT AND
CHANGES TO REGULATORY COMMITMENTS **Docket Nos. 50-387**
PLA-6111 **and 50-388**

Reference: 1) PLA-5823, B.L. Shriver (PPL) to USNRC "10 CFR 50.59 Summary Report," dated October 8, 2004

Attachment 1 is the summary report of PPL Susquehanna, LLC 50.59 Evaluations. This report is required by 10 CFR 50.59(d)(2) and is to be submitted at intervals not to exceed 24 months. The previous report (Reference 1) included the period from April 01, 2003 to September 30, 2004. This report provides summaries of those 50.59 Evaluations of Changes, Tests, and Experiments approved between October 01, 2004 and August 31, 2006.

The format of the report is as follows:

50.59 Evaluation No:	Unique number for each evaluation.
Cross-Reference:	Reference to the document for which the 50.59 Evaluation was prepared.
Description of Change:	A brief description of the changes, tests, and experiments.
Summary:	A summary of PPL Susquehanna, LLC's basis for concluding that a license amendment was not required pursuant to 10 CFR 50.59(c)(2).

Attachment 2 is a summary of Changes to Regulatory Commitments for the commitments that were changed in accordance with the guidance of NEI 99-04 "Guidelines for Managing NRC Commitment Changes" and SECY-00-045.

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Per NEI 99-04, commitment changes are required to be reported to the NRC either annually or with an FSAR update per 10 CFR 50.71(e). PPL is providing commitment changes along with the Summary of 50.59 Evaluations report rather than including them in an FSAR update. The next FSAR update is required to be issued to the NRC in the fall of 2006.

For each PPL commitment change, a description of the change and the justification for the commitment change is provided.

If you have any questions regarding this information, please contact Mr. Rich Tombasco at (610) 774-7720.

Sincerely,

B. T. McKinney

B. T. McKinney

Attachments:

Attachment 1 – 10 CFR 50.59 Summary of Changes, Tests, and Experiments

Attachment 2 – Changes to Regulatory Commitments

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

Attachment 1 to PLA-6111

10 CFR 50.59

Summary of Changes, Tests, and Experiments

50.59 Evaluation No.: E-01-29

Cross-Reference: LDCN No. 3912

Description of Change:

This is a change to the FSAR description for processing liquid radwaste through the Liquid Radwaste (LRW) filters and LRW demineralizer to the liquid radwaste sample tanks.

The LRW Demineralizer effluent conductivity instrumentation was obsolete, unreliable and inaccurate. Therefore, it was considered Class C instrumentation, not functional, maintained, or operated. Alternate more conservative means of controlling LRW demineralizer effluent quality are utilized as described by this FSAR change. These alternate controls ensure the LRW Sample Tank quality is not affected adversely as a result of this change in LRW processing. The quality of water recycled to the CST or discharged to the river from the LRW sample tanks is not adversely affected by this change.

Summary:

The change only affects the quality of water processed to the LRW sample tanks and returned to the CST or discharged to the Susquehanna River. This change does not adversely affect the quality of water in the LRW sample tanks or water returned to the CST or water discharged to the Susquehanna River from the sample tanks due to the alternate processing control.

The analyses in FSAR Sections 2.4.12, 2.4.13, and 15.7.3 bound all conceivable small releases outside containment in the LRW systems. The accident analysis in FSAR Sections 2.4.13 and 15.7.3 are based on the same postulated failure causing the complete release of the radioactive inventory in the liquid containing waste tank with the largest quantity of volatile radionuclides in the Radwaste Waste Management Systems. This change in LRW processing does not affect the radioactive inventory in the RWCU phase separator or operation of the phase separator in any way that could result in failure of the tank. Therefore, there is no increase in the frequency of occurrence or consequences of an accident previously evaluated in the FSAR.

Since this change involves LRW processing and the affected SSC's are all within the LRW system, there are no nuclear safety-related functions affected. This change does not directly or indirectly affect any SCC's outside the LRW system. Any failure in the LRW system that releases activity is no different than any analyzed in FSAR

Section 15.7.3. Therefore, this change does not result in any increase in the likelihood of occurrence or the consequences of a malfunction of an SSC important to safety as previously evaluated in the FSAR. The change does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR, and the change does not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

Finally, the change is strictly limited to LRW Demineralizer equipment operation and is not related to a method of evaluation described in the FSAR. No analytical methods are altered as a result of this change.

50.59 Evaluation No.: E-01-33

Cross-References: LDCN No. 3820, DCP No. 538557

Description of Change:

Components of the Unit 2 Reactor Building Crane were at the end of their useful life and needed to be upgraded to improve overall crane reliability. This change upgraded the entire crane controls, including the static stepless drives, motors and brakes. The limit switch technology was replaced with a laser positioning system (analog to digital upgrade). This new positioning system allows the addition of crane travel restrictions to prevent the main hook from impacting the refueling floor equipment. A radio control system was also installed.

Additional items addressed by this change consist of: 1) Replacement of the hoist geared limit switch with a GEMCO geared limit switch; 2) Provide rigging points for the four inboard bogie wheels for wheel and bearing maintenance; 3) Provide overspeed protection system to detect main hoist overspeed in excess of 115% of critical (load on the hook) and 115% of non-critical (unloaded hook) speed limits; 4) Provide a micro-speed selector switch in the cab and on the three radio controllers that, when selected, reduce the maximum speeds on all crane motions.

Summary:

A review of the FSAR, the SSES SER, the ODCM, and the FPRR, identified that the only accident that is applicable is the "SPENT FUEL CASK DROP ACCIDENT." This modification does not create any new interfaces with equipment important to safety, and existing interfaces are not adversely affected. Therefore, dynamic qualification is retained. The transfer of new fuel assemblies from the railroad/truck bay is not adversely affected since no design parameters are changed as a result of this modification.

The consequences of an accident are not increased because there are no adverse impacts to the safety-related function of the cranes, i.e., the safe approach distance for these cranes has not been changed to the reactor building or control structure. Therefore, this change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, as previously evaluated in the FSAR.

This modification does not create adverse impacts to existing interfaces with plant equipment. The programmable logic controller, laser positioning system, flux vector and frequency drives installed by this modification represent an analog to digital upgrade.

Although the equipment may fail in a different manner than the previous relay logic, the failures do not result in a different type of crane failure.

The new radio controls installed have a FCC licensed frequency and transmit at a power level of less than 0.02 Watts (20 mW). The use of this frequency is properly licensed by the FCC. This frequency does not interfere with any existing approved wireless communications equipment on-site including the plant radio system, wireless phones or HP/Security wireless video.

The installation of upgraded controls provides increased reliability of the crane and an increased ability to troubleshoot operational errors. Therefore, this modification does not create the possibility for an accident of a different type than evaluated previously in the FSAR.

50.59 Evaluation No.: E-01-34

Cross-References: LDCN No. 3825, LDCN No. 3735

Description of Change:

This FSAR change revised the Pump Seizure Analysis (FSAR 15.3.3) and Misplaced Bundle (FSAR 15.4.7) Accident Analysis Results to support Unit 2, Cycle 13 Core Loading.

Summary:

As part of the Unit 2, Cycle 13 core loading change, the design basis accidents described in Chapter 15 of the FSAR were evaluated. The results of that evaluation showed that the consequences for the Pump Seizure and Misplaced Bundle Accidents increased slightly from the values documented in the FSAR. The change is less than a minimal increase as defined by 10 CFR 50.59; therefore, the Unit 2 Cycle 13 Core Loading change is considered to not be adverse.

Since a License Amendment is required prior to implementation of the U2C13 Core Loading, changes to TS 2.1.1.2 (MCPR Safety Limit) and 5.6.5.b (COLR references). were submitted to the NRC for approval as a license amendment.

50.59 Evaluation No.: E-01-36

Cross-References: LDCN No. 3807, LDCN No. 3808

Description of Change:

This Technical Requirements Manual (TRM) table change revised the Control Rod Block Instrumentation applicability requirements during refueling outages when control rod movement in cells containing no fuel assemblies is performed coincident with SRM and IRM maintenance.

In addition, the TRM Bases Section B 3.1.3 was revised to require a second licensed operator or other qualified member of the technical staff to verify control rod movement in accordance with an approved control rod withdrawal sequence.

The net effect of these changes was to replace the automatic control rod block IRM and SRM functions with an administrative control.

Summary:

The addition of footnote (j) to all Mode 5 TRM Table 3.1.3-1 entries for SRM's and IRM's (Functions 2.a., 2.b., 2.c., 2.d, 3.a., 3.b., 3.c., and 3.d.) did not affect the Rod Withdrawal Error analysis provided in the FSAR. No SSC's relied upon in Safety Analyses were affected. The use of a second licensed operator or other qualified member of the technical staff to verify the correct control rod is selected prior to control rod withdrawal from a defueled cell further provides added assurance that the protection normally provided by control rod block instrumentation (SRM's and IRM's) is maintained. Using the second person compensates for disabling the automatic rod block function performed by the SRM's and IRM's in Mode 5. These changes did not introduce any new failure modes or cause changes in any accident frequency or radiological consequences.

50.59 Evaluation No.: E-01-38

Cross-Reference: LDCN 3827, DCP 482087

Description of Change:

This FSAR change clarified the SSES design basis for internal flood protection. It reflects the results of the investigation and analyses performed under the Corrective Action Program. As a result of the investigation, several sections of the FSAR related to internal flood protection were found to be unclear and required clarification. These changes represent new FSAR described design functions, which are necessary to demonstrate compliance with existing internal flooding design requirements.

Summary:

The FSAR changes are summarized below:

- Clarified that ECCS/RCIC rooms were designed to be "Watertight" from the ground up only, since the equipment hatches above these rooms were not designed to be leaktight.
- Provided detailed discussion regarding the "worst case" pipe crack in the reactor building and clarified that credit is taken for operator action to terminate the event. Also added discussions regarding internal flooding in the reactor building to demonstrate that the current internal flood design basis requirements remain satisfied.
- Clarified that adequate separation of essential systems and components is maintained, as defined in Branch Technical Position (BTP) ASB 3-1.
- Clarified the scope of piping evaluated as high energy piping, which is consistent with BTP ASB 3-1.
- Provided additional details regarding the role of the plant drainage system in mitigating the consequences of internal flooding in the reactor building.

A review of the SSES Internal Flood Design basis requirements revealed that through wall leakage cracks are required to be postulated in moderate energy piping located in areas containing systems important to safety. The SSES internal flood design requires that adequate equipment remains available following an internal flooding event to safely shutdown and maintain safe shutdown of the units, assuming an additional single failure, as defined in the FSAR Section 3.6.1.1. In addition, sufficient physical separation between systems must be maintained to prevent loss of more than one ECCS/RCIC system as a result of an internal flooding event.

Internal flooding design basis calculations were revised to account for leakage into ECCS and RCIC rooms through equipment hatches. The revised design basis flooding calculations demonstrate compliance and are consistent with the guidelines provided in BTP ASB 3-1. The original internal flooding design basis requirements are maintained, even with the additional leakage pathways considered via the equipment hatches. The FSAR changes represent new FSAR described design functions, which are necessary to demonstrate compliance with existing internal flooding design requirements. The evaluations performed in support of these new design functions demonstrated that the changes do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR, since the existing internal flooding design basis requirements are maintained.

50.59 Evaluation No.: E-01-40

Cross-References: LDCN 3904, DCP 656068

Description of Change:

This change provided the conversion from a temporary plant modification to a permanent plant modification in the common recombiner system logic. The modification allows the Common Motive Steam Jet Condenser Outlet Valve to open in the STANDBY mode to prevent a pressure build-up in the recombiner system, from through-seat leakage at the valve, by venting it to the aligned U1 or U2 charcoal adsorber trains.

Summary:

The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. FSAR Section 15.7.1.1 describes the Ambient Charcoal Offgas Treatment System Failure. The section identifies the following events, which could lead to a gross failure of the offgas treatment system:

(1) A seismic occurrence greater than design basis;
(2) A hydrogen explosion in housing unit;
(3) A fire in the filter assembly;
(4) Failure of spatially related equipment.

Providing a vent path for the Common Recombiner when in STANDBY MODE has no effect on the probability of occurrence or the consequences of these events. The offgas system continues to be monitored for hydrogen concentration and process temperatures, in accordance with plant procedures. The system is capable of processing noncondensible radioactive offgas from the main condenser. The flow from the system is monitored by the Turbine Building SPING exhaust vent system.

Technical Specification Basis 3.7.5 Main Condenser Offgas identifies the need to restrict the gross radioactivity rate of the noble gases from the main condenser, to assure that the total body exposure to an individual at the exclusion area boundary does not exceed a small fraction of the limits of 10 CFR Part 100. Technical Specification Basis 3.7.5 is not adversely affected by this change because the capability of the offgas system is not reduced.

This modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The additional flow from the Common recombiner to the in-service offgas train is insignificant. Failure of the valve to remain open in the Standby mode results in isolation of the Common Recombiner, which is the pre-modification configuration.

The modification does not reduce the margin of safety as defined in the basis for any technical specification.

50.59 Evaluation No.: E-01-41

Cross-References: LDCN 3850, LDCN No. 3848, DCP 618882, PLA-5880

Description of Change:

The Susquehanna Steam Electric Station (SSES), Unit 1 Power Range Neutron Monitoring (PRNM) system, including the Oscillation Power Range Monitor (OPRM), was replaced by GE's Nuclear Measurement Analysis and Control (NUMAC) PRNM system. This system performs the same plant-level functions as the previously installed systems, including the OPRM. (The Unit 2 PRNM and OPRM will be replaced during a future outage.)

The modification replaced the existing APRM, RBM, LPRM, OPRM, and recirculation flow units, all part of the existing PRNM system. The modification excluded the LPRM detectors and signal cables, which were retained with the NUMAC PRNM replacement. The reactor recirculation flow transmitters were also replaced. The complexity of the modification required plant procedure changes for support and operation of the equipment.

Summary:

The Power Range Neutron Monitoring (PRNM) System replacement consisted of replacing the PRNM System (including the APRM, the Rod Block Monitor, the LPRM System excluding the detectors and signal cables, and OPRM) with GE's Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring (PRNM) System. The Option III stability solution was integrated into the PRNM System electronics.

License Amendment Request (PLA-5880) dated June 27, 2005, addressed Susquehanna's Technical Specification submittals for this system upgrade. PLA-5880 provided responses to the plant-specific utility actions required by GE License Topical Report (LTR) NEDC-32410P-A, and included descriptions and justifications for deviations from the LTR as well as changes that are not addressed in the LTR. Changes addressed in PLA-5880 were submitted to NRC for prior review and approval.

The new equipment was specifically designed to assure that it fully meets the necessary requirements and has been specifically designed to have the same or more conservative failure modes as the previous system. The replacement equipment is fully qualified to operate in its installed location. Postulated failures can cause a certain loss of system function and these conditions are bounded by existing analysis.

50.59 Evaluation No.: E-01-42

Cross-References: LDCN 3940, LDCN 3950

Description of Change:

This change to the Technical Specification Bases (TSB) Table B 3.6.1.3-1 provides a clarification to allow maintenance on the HPCI & RCIC Minimum Flow checkvalves while at power.

Summary:

Per FSAR 6.2.4.3.3.2 and 6.2.4.3.3.3, penetrations listed in TSB Table B 3.6.1.3-1 are provided with a check valve for automatic isolation in the short-term. Long-term isolation is provided by a motor-operated valve that can be closed from the Control Room. This change allows the motor-operated valve to be closed at the start of a transient, thus precluding the possibility of an isolation malfunction, and maintaining the TS and TRM requirements for the penetration while undergoing maintenance.

This change does not affect the normal operational conditions of the HPCI or RCIC Minimum Flow line. This change only applies when HPCI or RCIC are out of service for maintenance. This condition is governed by the respective LCO (3.5.1 or 3.5.3), and has no affect on the safety function for the HPCI or RCIC system.

50.59 Evaluation No.: E-01-44

Cross-References: LDCN 3945, DCP 688900

Description of Change:

This change installed a continuous air purge to dilute and move accumulating hydrogen through the Offgas System when the common offgas recombiner is in standby. The air purges hydrogen through the offgas recombiner skid to the Unit 1 or Unit 2 Offgas charcoal beds when the common offgas recombiner is in standby mode. Operating procedures were changed to reflect alignment, set-up, and isolation of the air purge.

Summary:

The installation of a purge air system to the common offgas recombiner system did not introduce the possibility of a change in the frequency of an accident, the possibility of a change in the likelihood of a malfunction, the possibility of a change in the consequences of an accident, the possibility of a change in the consequences of a malfunction, the possibility of a new accident, or the possibility for a malfunction of an SSC with a different result.

The installation of a purge air system to the common offgas recombiner system does not have the potential to affect fission product barriers and does not result in a departure from a method of evaluation, described in the FSAR, used in establishing the design bases or in the safety analyses.

FSAR Section 15.7.1.1 addresses the Ambient Charcoal Offgas Treatment System failure. In this section of the FSAR, the offgas treatment system is assumed to fail, resulting in releases from the offgas system charcoal adsorption beds, delay line, and the Steam Jet Air Ejector (SJAE). This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment. Any breach in the new purge air piping with a postulated release of offgas is enveloped by failure of the offgas system described in FSAR Section 15.7.1.1.

The service air/offgas piping and valves are designed to the same requirements as the existing service air/offgas system piping and valves. The change is an enhancement to the offgas system. A new means of adding, diluting, and moving hydrogen in the common offgas recombiner system when in standby without placing an unacceptable demand on the Service Air System.

50.59 Evaluation No.: E-01-46

Cross-Reference: LDCN 4313

Description of Change:

Changes were made to FSAR Sections 15.3.3 and 15.3.4 as a result of a deficiency in the Recirculation Pump Seizure Analysis. These changes were: 1) Change from FANP analytical method to GENE analytical method, and 2) Change to Recirculation Pump Seizure consequences. Minor changes are made to FSAR Section 15.3.4, Recirculation Pump Shaft Break, to reflect the revised Recirculation Pump Seizure Accident Analysis.

Summary:

The Recirculation Pump Seizure Analysis documented in FSAR Section 15.3.3 was previously performed by Framatome-ANP (FANP) using NRC approved methods. Due to an error in the FANP analysis, GE Nuclear Energy (GENE) performed the Recirculation Pump Seizure Analysis using GENE's NRC approved methods. The analysis demonstrates that the Recirculation Pump Seizure event does not result in fuel failure and that the assumed initial MCPR for the event is non-limiting compared to the MCPR Operating Limits contained in the unit / cycle specific Core Operating Limit Reports. Therefore, since the Recirculation Pump Seizure Analysis results are non-limiting, the GENE methods do not need to be added to TS 5.6.5.b.

GENE confirmed that the analysis of the Recirculation Pump Seizure was performed using GENE's NRC approved standard methodology for reload licensing and that application is within the code application capabilities. PPL also performed a review of the GENE NRC approved methods used for the Recirculation Pump Seizure Analysis and concluded that the methods are applicable to Susquehanna and have been reviewed and approved by the NRC for this application. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

50.59 Evaluation No.: E-01-47

Cross-Reference: DCP 677648, 677650 and PLA-6002 (Proposed Licensing Amendment Request)

Description of Change:

As part of the Extended Power Uprate (EPU) project, the main turbine High Pressure (HP) turbine is being replaced.

As a result of the increase in the thermal conditions and the HP turbine design process, an analysis and revision to the existing SSES turbine missile report, previously prepared by Siemens Power Generation, Inc. was necessary. The revised Siemens Missile Report documents the missile analysis to the uprated condition of 3952.8 MWt.

This 50.59 Evaluation was to determine if the new Siemens Missile Report results in a departure from a method of evaluation described in the FSAR and to determine if the plant-specific actions required in the NRC Safety Evaluation of the Siemens generic missile analysis conclusions are applicable.

Summary:

The turbine missile probability analysis evaluates the probability of damage from postulated turbine missiles to safety-related components at Susquehanna SES using NRC-approved methodology in the Siemens Report, which is applied and documented to Susquehanna Units 1 & 2. The probability of unacceptable damage to safety-related components due to turbine missiles at SSES has been calculated to be within the required probability as given by the NRC in Reg. Guide 1.115. Based upon these low probability results, the updated turbine missile hazard is not considered as a design basis event for Susquehanna SES and is consistent with the design basis conclusions outlined in FSAR Section 3.5.1.3 for the existing turbine.

Attachment 2 to PLA-6111

Changes to Regulatory Commitments

Commitment Change No.: LDCN 3831

Description of the Change:

On October 09, 2003, PPL revised the commitment date from the Spring 2003 RIO to the Spring 2005 RIO for inspection of the Unit 2 Core Shroud Vertical Weld H6B/H7. This commitment was documented in PPL's letter to the NRC (PLA-5234), dated October 4, 2000).

The schedule to inspect this vertical weld was changed again from the Spring 2005 RIO to the Spring 2007 RIO. This latest schedule change is considered a change in commitment.

Summary:

PLA-5234 stated that the H6B/H7 vertical welds were scheduled for inspection in the Unit 2 Spring 2003 RIO. The basis for this conclusion is contained in an engineering calculation that provides an acceptable range of years for weld inspections.

The 2003 inspection date was chosen because it was four years from the date of the last inspection (April 1999). This date was more conservative than a 2005 inspection date, which was still within the acceptable range of dates.

Subsequently, BWRVIP-76 was approved and issued for use in November 1999. PPL is committed to implement the BWRVIP Program. Therefore, the provisions of BWRVIP-76 supersede previous NRC commitments on Core Shroud inspections. A new calculation was performed using approved BWRVIP-76 methodology which demonstrates that the inspections of the Unit 2 Core Shroud Vertical Welds H6B/H7 can be postponed until the Spring 2007 RIO and still provide the assurance that the shroud is structurally sound.