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U. S. Nuclear Regulatory Commission
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SUSQUEHANNA STEAM ELECTRIC STATION
10 CFR 50.59 SUMMARY REPORT AND
CHANGES TO REGULATORY COMMITMENTS **Docket Nos. 50-387**
PLA-6437 **and 50-388**

Reference: 1) PLA-6111, B. T. McKinney (PPL) to USNRC "10 CFR 50.59 Summary Report," dated October 2, 2006.

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 October 01, 2004 to August 31, 2006. This report provides summaries of those 50.59 Evaluations of Changes, Tests, and Experiments approved between September 01, 2006 and August 31, 2008.

The summary for each 50.59 Evaluation is formatted 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).

No Regulatory Commitments were changed in accordance with the guidance of NEI 99-04 "Guidelines for Managing NRC Commitment Changes" and SECY-00-045 for the period from September 01, 2006 and August 31, 2008.

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NRR

If you have any questions regarding this information, please contact Mr. Jason Welch at (570) 542-3251.

Sincerely,

B. T. McKinney

Attachment - 10 CFR 50.59 Summary of Changes, Tests, and Experiments

cc: NRC Region I
Mr. R. Janati, DEP/BRP
Mr. F. Jaxheimer, NRC Sr. Resident Inspector
Mr. B. Vaidya, NRC Project Manager

Attachment to PLA-6437

10 CFR 50.59

Summary of Changes, Tests, and Experiments

50.59 Evaluation No.: E-01-43

Cross-Reference: LDCN No. 3958

Description of Change:

FSAR sections 9.2.5.2 and 9.2.6.2 were revised to state that the outside surfaces of certain buried pipes were intentionally left without any coating or protective wrapping at two locations. At these locations, copper-copper sulphate reference electrodes belonging to the cathodic protection system were installed and were used to measure pipe to electrolyte potential. The pipe to electrolyte potentials at various locations of buried pipes were measured at regular intervals selected for the cathodic protection system.

These FSAR sections previously stated that all piping outside of the pumphouse, main plant, and spray pond is buried and it is coated and wrapped for corrosion protection.

Summary:

The Systems, Structures, or Components (SSCs) that were affected by the activity were the Emergency Service Water/Residual Heat Removal Service Water (ESW/RHRSW) return loop piping due to damaged pipe coating and/or wrapping at two locations. The potential effect is ESW/RHRSW return loop piping water leakage due to soil corrosion produced over a long period if not protected by the cathodic protection. Pipe leakage of the piping is addressed in the FSAR section 9.2.5.6 and 9.2.6.6.

It was decided that the surfaces of the affected ESW/RHRSW return loop piping may be left as-found because repairing these surfaces would create additional damage to good wrapping on the other surfaces. It was determined that the cathodic protection system would provide the necessary corrosion protection provided pipe to electrolyte potential is measured at regular intervals.

Safety evaluation of the ESW/RHRSW system is addressed in the FSAR sections 9.2.5.3 and 9.2.6.3. Accident analysis due to failure of RHR Shutdown cooling is addressed in FSAR section 15.2.9. The activity did not introduce any new malfunctions or increase the frequency of the malfunction; therefore, there is no increase in the radiological consequence of an accident. As per FSAR sections 9.2.5.2 and 9.2.6.2, it is not considered credible to have a common cause loss of the common ESW/RHRSW loops that affects the system capability to bring either or both units to a safe shutdown condition under emergency conditions. No credit is taken for the cathodic protection system for the operability of any SSC in the FSAR.

50.59 Evaluation No.: E-01-45

Cross-Reference: LDCN No. 4306

Description of Change:

The normal U1 and U2 Refueling Platform power supplies contain 480V AC power and control circuitry for the RMCS refueling interlocks. The U1 and U2 platform can be connected to either unit for refueling. When using the opposite unit's refueling platform on the refuel unit for fuel handling activities (e.g. U1 platform to refuel U2 reactor), the refuel unit's platform is unable to be used for any purpose, since the normal power supply and interlock interface is already in use by the fuel handling platform. The refuel unit's platform must sit idle over the dryer-separator storage pool.

This change allowed the idle platform to be powered from an alternate source, which does not have the RMCS refuel interlock interface. When powered from the alternate source the refuel unit's platform becomes an auxiliary work platform over the dryer-separator storage pool or reactor vessel. The RMCS refuel interlocks for this platform are defeated when powered from the alternate source. In this configuration, the Main Hoist on the work platform is in a stowed position and therefore physically disabled from handling fuel. The Auxiliary Hoists (i.e. Frame and Monorail Hoists) on the work platform are now administratively controlled from operation in the vessel if the Steam Separator is removed. Operation of the Rigid Pole Handling System was not affected by this change. In addition to the RMCS refueling interlocks, any boundary zone or travel interlocks may also be defeated for the platform functioning as an auxiliary work platform.

Summary:

This action allows either refueling platform to be used as an auxiliary work platform when the opposite unit's platform is used for fuel handling activities on the refuel unit. The Main Hoist on the work platform will be in a stowed position and therefore physically disabled from handling fuel. The Auxiliary Hoists on the work platform are administratively controlled from operation in the vessel if the Steam Separator is removed. In this configuration, the RMCS refueling interlocks are not functional for the platform functioning as a work platform. However, the interlocks are not safety related and are only required when handling fuel or control rods within the reactor pressure vessel. This action did not impact any safety related function. This action did not interfere with the refueling interlocks of the fuel handling platform, which will be used for the refuel unit fuel handling operations.

This change reduced outage time required for refueling activities by allowing some of the non-fuel handling work that would have been done by the Refueling Platform to be performed concurrently by the other platform, which would normally have sat idle.

This action did not have an adverse effect on any accident or malfunction previously evaluated in the FSAR, nor does it create the possibility for an accident or malfunction of a different type than previously evaluated in the FSAR, nor does it adversely effect the fission product barriers as described in the FSAR, nor does it adversely effect the evaluation methodologies described in the FSAR.

50.59 Evaluation No.: 50.59 SE 00002

Cross-Reference: LDCN Nos. 3838, 3840, 3849, 3917, 3919, 3982, 3985

Description of Change:

The Susquehanna Steam Electric Station (SSES), Unit 2 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.

This also included an ARTS/MELLLA change that was associated with the APRM RBM Technical Specifications (ARTS) improvements program and the extension of the SSES operating domain to permit original rated core thermal power operation at core flows as low as 75% of rated (known as Maximum Extended Load Line Limit Analysis (MELLLA)).

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.

Upon installation of PRNMS, the Average Power Range Monitor (APRM) flow biased flux scram setpoint and the APRM flow-biased rod block trip setpoints were revised to permit operation in the MELLLA domain.

Summary:

The Power Range Neutron Monitoring (PRNM) System and ARTS/MELLLA change 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. The ARTS/MELLLA change was associated with the APRM RBM Technical Specifications (ARTS) improvements program and the extension of the SSES operating domain to permit original rated core thermal power operation at core flows as low as 75% of rated (known as Maximum Extended Load Line Limit Analysis (MELLLA).

This 50.59 evaluation addressed plant functional impacts as a result of installing NUMAC PRNMS and was required by LTR NEDC-32410P-A. It also reviewed the plant functional impacts as a result of the hardware changes necessary to incorporate the settings and setpoints outlined in the Technical Specification proposed amendment submittal. It did not provide an additional evaluation of the settings, setpoints, and analysis already documented in the Technical Specification proposed amendment submittal.

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.

This action did not have an adverse effect on any accident or malfunction previously evaluated in the FSAR, nor does it create the possibility for an accident or malfunction of a different type than previously evaluated in the FSAR, nor does it adversely effect the fission product barriers as described in the FSAR, nor does it adversely effect the evaluation methodologies described in the FSAR.

50.59 Evaluation No.: 50.59 SE 00005

Cross-Reference: LDCN Nos. 4472, 4473

Description of Change:

The purpose of this change was to eliminate the following post maintenance Main Steam Isolation Valve (MSIV) leakage rate acceptance criteria statement from the Unit 1 and 2 TS Bases Surveillance Requirement 3.6.1.3.12:

"If the leakage from the MSIVs requires internal work on any MSIV, the leakage will be reduced for the affected MSIV to ≤11.5 scfh."

This action eliminated unnecessary internal work on the MSIVs. Unnecessary internal work has been shown to introduce defects that could result in exceeding leakage limits. As stated in NUREG 1169, performing an MSIV repair to a low leakage rate can potentially cause actual defects to be introduced. In addition, unnecessary worker dose is eliminated. The dose estimated to be saved by not performing internal MSIV work to improve the leak rate of an MSIV is 0.75 – 1.0 REM each attempt.

Summary:

Removal of the post maintenance MSIV leakage rate acceptance criterion in TS Bases section SR 3.6.1.3.12 did not represent an activity that requires prior NRC approval. This change was not an initiator of any accident and no new failure modes were introduced. Consequently, there was no possibility of affecting the frequency of an accident or malfunction previously evaluated in the FSAR, or creating a new accident or malfunction. This activity did not result in exceeding a design basis limit for a fission product barrier as described in the FSAR, since the change did not affect a design basis limit for a fission product barrier. This change was not a change to the methodology described in the FSAR for analyzing the radiological consequences due to MSIV leakage post-LOCA.

50.59 Evaluation No.: 50.59 SE 00009

Cross-Reference: LDCN No. 4542

Description of Change:

The purpose of the change was to update Note 1 of FSAR Table 3.6-1. Note 1 of Table 3.6-1 previously stated that the Turbine Building does not contain any safety related equipment. This change modified Note 1 to indicate that the Turbine Building does not contain any essential systems and components. Essential systems and components are defined in FSAR Section 3.6.3 as "systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power."

Summary:

The change to Note 1 of FSAR Table 3.6-1 did not represent an activity that requires prior NRC approval. This change was not an initiator of any accident and no new failure modes were introduced. Consequently, there was no possibility to affect the frequency of an accident or malfunction previously evaluated in the FSAR, or to create a new accident or malfunction. This change did not result in exceeding a design basis limit for a fission product barrier. This change was not a change to the methodology described in the FSAR for analyzing a High Energy Line Break and the associated consequences.

The change met the requirements for evaluating the effects of high energy line breaks and moderate energy pipe cracks presented in FSAR Section 3.6, NRC Branch Technical Position (BTP) ASB 3-1 and BTP MEB 3-1. The evaluation concluded that safe shutdown could be maintained with the available equipment following a postulated high energy line break in the turbine building.

50.59 Evaluation No.: 50.59 SE 00010

Cross-Reference: LDCN No. 4584

Description of Change:

This Engineering Change (EC) provided a justification for reducing the Residual Heat Removal (RHR) flow requirements through the Unit 1 RHR heat exchanger under Design Basis Accident – Loss Of Coolant Accident (DBA-LOCA) conditions, when Emergency Operating Procedures (EOPs) require all four Low Pressure Coolant Injection (LPCI) pumps to be in service for Reactor Pressure Vessel (RPV) flooding. Previously, an RHR heat exchanger could not be aligned to satisfy the 9750 gpm minimum shell side flow requirement with two LPCI pumps in service and emergency operating procedures did not allow a LPCI pump to be removed from service under these conditions. This EC demonstrated that with two LPCI pumps in service on a loop, the flow diverted through the RHR heat exchanger with the normally open heat exchanger inlet/outlet isolation valves fully open and the normally open heat exchanger bypass valve fully open was enough to support operation of the RHR heat exchanger, as credited in the SSES Containment Analysis for long term containment cooling.

This temporary change supported operation with the EOPs as they were written and was designed to provide sufficient time for permanent resolution of the issue regarding the lack of consistency between the SSES EOPs and the SSES Containment Analysis described in the FSAR.

Summary:

This EC reduced the RHR heat exchanger shell side design flow and heat exchanger fouling factor in order to credit the alignment of an RHR heat exchanger in LPCI mode of operation, for the specific DBA-LOCA scenarios described above. The Emergency Core Cooling System (ECCS) function of flooding the reactor core in the LPCI mode of operation was not impacted since this change did not change the RHR system alignment under these DBA-LOCA conditions and did not reduce the mass flow of water available to the reactor pressure vessel. No other RHR system operating modes were impacted by this change.

The RHR heat exchangers continued to perform their design basis safety function. The RHR heat exchangers are designed with a tube side design fouling factor of 0.002 hr-ft^2°F/BTU and shell side design fouling factor of 0.0005 hr-ft^2-°F/BTU. This equates to an overall fouling factor of 0.0028 hr-ft^2-°F/BTU, in terms of outside area. The reduction in Unit 1 RHR heat exchanger shell side flow is realized by reducing the RHR heat exchanger overall design fouling factor from 0.0028 to 0.00142 hr-ft^2-°F/BTU. This reduction in overall heat exchanger fouling factor is based on performance testing conducted on RHR Heat Exchanger 1E205B in 2004. The maximum overall fouling

resistance for the RHR heat exchanger performance test was found to be 0.0005 hr-ft^2-°F/BTU, including test uncertainty, which is approximately 1/3 the value assumed in support of this EC. These testing results are consistent with inspection records for these heat exchangers, which show these heat exchangers contain only minor amounts of mud and silt after 4 years of service. At these reduced fouling/RHR flow conditions, the heat transfer analysis demonstrates that the RHR heat exchanger capacity will meet or exceed the capacity assumed in the FSAR and the SSES Containment Analysis. The remaining fouling margin, combined with other conservative design assumptions used in this analysis, ensure the RHR heat exchangers will continue to perform their FSAR described design functions.

No additional testing was required to support this LPCI mode of operation, since a conservative evaluation was performed to support the RHR flow used in the RHR heat exchanger heat transfer analysis. In addition, since there were no additional system components required to function to support the alignment of the RHR heat exchanger in the LPCI operating mode, no new testing requirements were imposed.

Based on the discussion above, this change did not adversely impact RHR system operation. The RHR heat exchangers continued to perform their design basis safety function and the SSES containment analysis was not adversely impacted by this change. Therefore, there was no increase in the radiological consequences of an accident previously evaluated in the FSAR. This change did 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.