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OCT 08 2004

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-5823 **and 50-388**

Reference: 1) PLA-5678, B.L. Shriver (PPL) to USNRC "10 CFR 50.59 Summary Report," dated September 25, 2003.

Enclosed 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 02, 2001 to March 31, 2003. This report provides summaries of those 50.59 Evaluations of Changes, Tests, and Experiments approved between April 01, 2003 and September 30, 2004.

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).

Also enclosed 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 for the first time rather than including them in an FSAR update. The next FSAR update is required to be issued to the NRC in the fall of 2005.

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. Duane L. Filchner at (610) 774-7819.

Sincerely,

B. T. McKinney

Enclosures:

10 CFR 50.59 Summary of Changes, Tests, and Experiments
Changes to Regulatory Commitments

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

10 CFR 50.59

Summary of
Changes, Tests, and Experiments

50.59 Evaluation No.: E-01-22

Cross-Reference: DCP Number: 467199 (Unit Common)
LDCN No. 3566, LDCN No. 3567

Description of Change:

The change removes the Primary Coolant Degasifier Filter Exhaust (PCDFE) System from the Technical Requirements Manual (TRM), the Final Safety Analysis Report (FSAR) and procedures since it has been evaluated that its design function is no longer required. As such, the system can be administratively controlled out of service until it can be abandoned through the modification process. The administrative controls will allow the PCDFE to be removed from service without entering the actions in the associated TRM Section.

Summary:

The function of the PCDFE is to filter radioactive particulate and remove radioactive iodine from non-condensable gases when either Unit 1 or Unit 2 is shut down. This function ensures the radioactive release limits in the licensing documents are not exceeded. This function is no longer required based upon current analysis.

There is no effect on the design function of the degasifier system when the PCDFE is removed from service. The system will still be able to maintain oxygen concentrations in reactor water below 250 ppb during plant shutdown periods when it is required. Also, by removing the PCDFE System from service, particulate and iodine activity that could be stripped from condensate, along with non-condensables in the vacuum tower, would be discharged directly to the turbine building exhaust vent and discharged to the environment. Calculations have determined that offsite dose limits will not be exceeded as long as reactor coolant activity is below that presented in the licensing basis effluent releases analysis. In addition, as part of the existing design, the discharge will be monitored by the turbine building vent activity sampling system (SPING). Administrative controls have been put in place through revision to the operating procedures to secure the vacuum degasifier, if operating, in the event that the offgas noble gas release rate increases by more than 50%.

Therefore, since the design function of the degasifier system is not affected by this change, a license amendment is not required.

50.59 Evaluation No.: E-01-23

Cross-Reference: DCP 479590 (U1); DCP 479591 (U2)
LDCN No. 3602; LDCN No. 3603

Description of Change:

The Reactor Recirculation Pump #2 Runback Limiter Setpoint was changed in the design change process from 45% to 48%. Also, FSAR Sections 7.7.1.3 and 10.4.7 were revised to account for this change. This new setpoint was determined based on analysis of historical data and evaluation of the current Power/Flow-Map relationships.

Summary:

The Reactor Recirculation Pump #2 Runback Limiter Setpoint is currently set at 45% recirculation pump speed. Power level increases, fuel design changes, and changes in methodologies for determining prohibited regions of operation have contributed to dynamic changes in core operation. The result of these changes is that a runback to 45% recirculation pump speed may require a manual SCRAM due to entry into Region 1 on the Power/Flow map, particularly when operating at high rod lines and low flow rates.

For the #2 Runback Limiter Setpoint, the speed controller signal is automatically limited to 45% if:

- 1 of 4 operating circulating water pumps trips, or

- Reactor Low Water Level 4 is present (as sensed by the feedwater system) and other BOP conditions exist. These conditions include feed pump discharge low flow, condensate pump discharge low pressure and feedwater heater high-high water level.

  The speed controller signal limits the Reactor Recirculation Pump to 45% rated speed to avoid a reactor SCRAM that could occur due to low reactor water level. The #2 Runback Limiter functions to assure that a feedwater transient or loss of vacuum allows the plant to remain on line but at a lower power level.

In order for the plant to stay on line for the transients discussed above, the #2 Runback Limiter Setpoint must be set high enough to prevent the requirement for a manual SCRAM, but low enough in order to maintain operation without the potential to lose equipment.

The loss of one of the circulating water pumps affects the plant's ability to reject waste heat, which in turn affects the condenser vacuum. The loss of a single feedwater pump affects the amount of feedwater flow and, consequently, the power level that the feedwater system can support. The loss of a condensate pump affects the suction pressure to the feedwater pumps and the amount of flow and power level the feedwater system can support. A feedwater heater high-high water level signal results in the isolation of that feedwater heater string. This isolation reduces the number of feedwater heater flow paths from three to two and, consequently, affects the amount of flow that the feedwater system can generate and the reactor power level that the feedwater system can support.

There are two issues related to changing the Reactor Recirculation Pump #2 Runback Limiter Setpoint. First, the setpoint must be low enough so that the remaining plant equipment can accommodate the power level, and the transient does not cause a trip on low feedwater pump suction pressure. Secondly, the setting must be high enough to avoid entry into stability Region I on the power to flow map which requires an immediate manual SCRAM. The setpoint change associated with this 50.59 Evaluation provides for the Reactor Recirculation Pump #2 Runback Limiter Setpoint to be changed from 45% to 48%. This new setpoint was determined based on analysis of historical data and evaluation of the current Power/Flow-map relationships.

This change in Reactor Recirculation Runback Limiter Setpoints provides a reliable and stable flow control operation. All safety design requirements and functions of the Reactor Recirculation System are maintained. Therefore, a license amendment is not required.

50.59 Evaluation No.: E-01-25

Cross-Reference: LDCN No. 3676
Unit 1 Cycle 14 Core Loading Report

Description of Change:

Revise the Pump Seizure (FSAR 15.3.3) Accident Analysis to support the Unit 1 Cycle 14 Core Loading. During the Unit 1 13th refueling and inspection outage, PPL will replace 280 irradiated Framatome-ANP (FANP) ATRIUM-10 fuel assemblies with unirradiated ATRIUM-10 fuel assemblies. The remaining assemblies in the reactor core (316 once burned and 168 twice burned ATRIUM-10 fuel assemblies) will be shuffled to obtain a core configuration that will provide the required energy for Cycle 14 operation.

Summary:

As part of the Unit 1 Cycle 14 Core Loading change, the design basis accidents described in Chapter 15 of the FSAR were evaluated. The results of that evaluation showed that the radiological consequences of the Pump Seizure Accident increased slightly from the values currently in the FSAR. Therefore, this 50.59 Evaluation was performed for that change.

The increase in the Pump Seizure Accident consequences is due to the following:

- normal cycle-to-cycle variation in the results;
- change in fuel assembly burn-up to accommodate an increase in the allowed burnup for the ATRIUM-1 mechanical design; and
- the fuel assembly radial peaking factor
  (i.e., the amount of power produced by the assembly).

The Pump Seizure Accident results are documented in the Susquehanna Unit 1 Cycle 14 Reload Licensing Analysis Report and were performed in accordance with NRC-approved methods and regulatory guidelines. The Unit 1 Cycle 14 Core Loading did not change any structures, systems, or components that would affect the consequences of the Pump Seizure Accident. These results show that the increase in thyroid and whole body dose is less than a minimal increase.

Since the change is less than a minimal increase defined by 10 CFR 50.59 and below the applicable fraction of the 10 CFR 100 limits, a license amendment is not required.

50.59 Evaluation No.: E-01-30

Cross-Reference: LDCN 3726

Description of Change:

Revise the Battery Duty load profiles for Class 1E 125 VDC batteries 1D610 and 1D620 in FSAR Table 8.3-6J. The first minute load is changed from 325 amps to 300 amps and the 1-239 minute load is changed from 95 amps to 115 amps.

Summary:

This revision to FSAR Table 8.3-6J load profiles for the Class 1E 125 VDC batteries 1D610 and 1D620 was initiated after verifying that all the design requirements specified in the FSAR were met. The total load profile ampere-hours energy requirement remains within the battery capacity at end-of-life conditions. The revised FSAR load profiles envelop the actual calculated load profile and provide margin for future load growth. Battery testing performed in accordance with Technical Specification Surveillance requirements assures that the batteries are sized correctly and can supply power to the calculated LOCA/LOOP load profile.

As the actual DC system load is increased due to new load additions, the available margin to the FSAR load profile is reduced. It is SSES practice to maintain margin between the actual load and the load specified in the FSAR table. Therefore, whenever actual load is increased within the available margin, the battery and battery charger re-sizing is not required.

These changes to the battery load profiles do not affect the ability of the batteries to perform their safety function, nor the frequency of an accident, nor the likelihood of occurrence of a malfunction of an SSC important to safety, nor the possibility of creating a new or different accident. Accordingly, a license amendment is not required to implement these changes.

50.59 Evaluation No.: E-01-31

Cross-Reference: Plant Operating Procedures

Description of Change:

This 50.59 Evaluation was performed to allow operation of either unit at 100% power while the feedwater (FW) heater drain system operates with one level control valve in one heater string in a fixed (gagged, jacked, or otherwise manually controlled) position. The activity is a compensatory action for a failure of the instrumentation and controls of the valve or for a mechanical failure of the valve itself. This action will be implemented when it is necessary to provide an alternate method for maintaining proper level in a feedwater heater.

Summary:

The action is designed to have a minimal impact on the operation of the FW heater vent and drain system while providing optimum level control of the affected heater using the emergency dump valve while the normal level control valve is degraded. This action represents a controlled compensatory action until an opportunity to perform corrective maintenance restores the function of all affected components.

Operation of the affected components is described in FSAR sections 10.4.7 and 10.4.10 and results in a departure from the FSAR described design function for the normal level control valve and the emergency dump valve. Failure of the activity to control FW heater level will result in a Feedwater heater string isolation which is described in FSAR section 15.1.1. The frequency of a loss of FW heating event will not be increased as a result of this activity per a risk evaluation performed by engineering.

The activity will not cause a result that represents an increase in the likelihood of occurrence of a malfunction of an SSC and has no radiological consequences associated with it per FSAR 15.1.1.5. The activity does not create the possibility of a different accident than previously evaluated or different results from a malfunction of an SSC important to safety. Since the frequency or severity of a loss of FW heating is not increased, there is no impact on the fission product barrier. The evaluation methodologies described in the FSAR are not impacted by this activity because no FW temperature reduction in excess of 100 degrees F were anticipated as a result of the implementation. Therefore, this activity may be implemented without prior NRC approval as a license amendment.

Changes to Regulatory Commitments

Commitment Change No: LDCN 3358

Description of the Change:

This change eliminated the commitment to conduct an inspection in accordance with plant procedures, of all safety-related panels for open sliding links (states links) following every outage of greater than six weeks duration. This commitment was made to the NRC at the May 31, 1985 Enforcement Conference and documented as Item 1.8 in NRC Combined Inspection Reports 50-387/86-09 and 50-388/86-09 dated June 12, 1986.

Justification for the Change:

This change eliminates the need to perform states links inspections during each refueling outage to identify links that are not in their normally closed position or are administratively controlled.

This commitment was established in 1985 in response to a Notice of Violation. The violation occurred because a contract electrician inappropriately opened links in the control circuit of an ESW bypass valve, thereby rendering it inoperable. Corrective actions included establishing a procedure which has been performed during the final states of each refueling outage. The purpose for the commitment was to assure plant states links are maintained in their normally closed position or are administratively controlled.

A review of the safety-related states link inspections was performed of data from the Unit 1 Refuel outages in 1998 and 2000 and the Unit 2 Refuel outages in 1999 and 2001. This review identified that the states links that were found open during the inspections were being controlled by other work management mechanisms such as bypasses, surveillances, DCPs or owner tags.

The current programs and procedures which have been implemented since the 1985 ESW incident have proven to be sufficient and adequate to maintain proper control of states links at SSES. This issue is closed in an NRC inspection report; and since there have been no recurrences within the last two years, it is acceptable to no longer require inspection of all states links following every outage of greater than six weeks duration.

Commitment Change No: LDCN 3376

Description of the Change:

Letter PLA-4893 (from PPL to NRC) dated May 4, 1998 contains a statement that the Process Control Program (PCP) and its implementing procedures are to be reviewed by the Plant Operations Review Committee (PORC). The revised commitment is to have only the PCP reviewed by PORC as required by FSAR Section 13.4.1.3.

Justification for the Change:

The procedure for the PCP provides administrative control, guidance, and records for processing, packaging, transportation, and disposal of radioactive waste. The PCP is required by the OQA program and is applicable to low-level radioactive waste generated as a result of the operation of SSES. It is not applicable to spent fuel or greater than Class C waste. Some of the program requirements for processing solid radioactive waste contained in this procedure are: contracted vendor services, testing/treatment of solidified radwaste for combustible gases, radioactive waste dewatering, container inspections, storage of packaged radioactive waste, and changes to the solid radioactive waste process control program. The implementing procedures are not required to be reviewed by PORC since they do not contain new requirements, and FSAR Section 13.4.1.3 only requires PORC to review the PCP. The implementing procedures only implement the requirements that are stated in the procedure for the PCP.

Eliminating review by PORC of the PCP implementing procedures has no effect on the capability of any SSC or the ability of personnel to keep the plant safe. During the review of the implementing procedures, the technical reviewer verifies that the implementing procedures implement the requirements of the PCP. Therefore, it is acceptable to no longer require PORC review of PCP implementing procedures as a regulatory commitment.

Commitment Change No: LDCN 3478

Description of the Change:

Delete the use of a watchman key code station system.

Justification for the Change:

The reason for the existing regulatory commitment was to compensate for a failure by contractor fire-watch supervision to provide adequate assurance that contractor fire-watch personnel were performing fire-watch rounds, as required. All contractor fire-watch personnel have since been replaced by PPL employee personnel, and assurance of proper performance of fire-watch rounds is provided by direct PPL supervision, use of hand-logs to document rounds, and periodic QA surveillances/audits. Since 11/99, fire-watch personnel are full time PPL employees.

This change could not decrease the effectiveness of fire-watches because the original problem was attributed, in part, to lack of contractor supervision of contractor personnel, whereas current practice is to use PPL personnel for fire watches, and those personnel are adequately supervised. In addition, manual logs provide assurance of proper fire-watch performance. Therefore, it is acceptable to no longer require use of a watchman key code station system.

Commitment Change No: LDCN 3487

Description of the Change:

Letter PLA-751 (PPL to NRC) responded to IE Bulletin 80-14. It contained a commitment to a 24-hour report for the failure of Scram Discharge Volume (SDV) Vent or Drain Valves. The commitment is revised to report SDV valve failures in accordance with 10 CFR 50.72 and 10 CFR 50.73 reporting requirements in lieu of the 24-hour report commitment in PLA-751.

Justification for the Change:

IE Bulletin 80-14 was issued to address a generic industry problem with SDV components. The 24-hour reporting requirement was appropriate to help the NRC assess the extent of the problem. Subsequently, TS requirements for SDV components were established to ensure reliability. Also, 10 CFR 50.72 and 10 CFR 50.73 were issued to establish comprehensive reporting requirements. The 24-hour reporting requirement of SDV valve failures from IE Bulletin 80-14 is not consistent with the significance of 24-hour reports as defined in 10 CFR 50.72 and 10 CFR 50.73, and is no longer appropriate.

10 CFR 50.72 and 10 CFR 50.73 as clarified by NUREG 1022 delineate the current event report guideline and requirements. These documents do not require any special reporting for SDV vent and drain valves. Thus, current regulations do not require a 24-hour report for inoperable SDV vent and drain valves. The reporting of equipment failures has no effect on the ability of the system to perform its safety function, or the ability of plant personnel to perform their duties. Therefore, it is acceptable to no longer require a 24-hour report for inoperable SDV vent and drain valves.

Commitment Change No: LDCN 3594

Description of the Change:

Letter PLA-5239 (PPL to NRC) was issued in response to Licensee Event Report (LER) 387/00-009-00. The letter contained a commitment "to revise the Technical Specification Bases (TSB) for Technical Specification (TS) 3.6.1.3 to clarify the appropriate action statement in the event that the subject valves fail to meet the surveillance requirement." This change revises the commitment as follows: "to revise applicable surveillance procedures that implement Surveillance Requirement 3.6.1.3.2 to clarify the appropriate action statement to be entered in the event that the subject valves fail to meet the surveillance requirement."

Justification for the Change:

The transmittal of LER 387/00-009-00 via PLA-5239 commits to the revision of the TSB document to communicate the required information. The information can be more conventionally communicated through a revision to procedures. This modified approach constitutes a change to the commitment.

The TSB document has not previously been used to direct licensed operators in the application of TS action statements upon failure to meet TS surveillance requirements. This "assistance" is, however, provided in station procedures developed for ensuring satisfactory fulfillment of surveillance requirements. As such, the intention and purpose of the original regulatory commitment can be accomplished through procedural revision instead of TSB alteration. This approach conforms to established conventions in use at SSES.

This commitment change maintains no impact to safety function capabilities of plant systems, structures or components. The proposed commitment change only alters the document used to communicate information to licensed operators. The purpose and intent of the original commitment is not adversely affected by this change. As such, the commitment change does not affect the ability of licensee personnel to keep the plant safe relative to the existing commitment, and it is acceptable to revise the applicable surveillance procedures instead of the TS Bases.

Commitment Change No: LDCN 3613

Description of the Change:

Letter PLA-5234 (PPL to NRC) dated October 4, 2000 committed PPL to inspect the Unit 2 Core Shroud Vertical Weld H6B/H7 during the Spring 2003 Refueling and Inspection Outage (RIO). This commitment is being changed from the Spring 2003 RIO to the Spring 2005 RIO.

Justification for the Change:

The commitment is to inspect the Core Shroud welds on a frequency that detects serious degradation in the welds prior to failure. Based on the calculation which uses a BWRVIP-approved methodology, the inspection date can be extended and the weld will still maintain adequate safety factors.

PLA-5324 states that the H6B/H7 vertical welds were scheduled for inspection in the Unit 2 Spring 2003 RIO. A section of an engineering calculation provides a range of acceptable years before the weld needs to be inspected. This range of years is from 3.76 to 6.23 years. The time chosen for inspection was four years from the date of the last inspection (April 1999). This was more conservative than extending the inspection six years to the 2005 outage which also was within the range of the calculation.

The reason a range of years exists is due to an input to the calculation. There are two variables for that input:

1) The first input was the "maximum crack" depth in the H6B horizontal weld, which had been used by GE in their evaluation of vertical welds. This evaluation was included in the calculation.

2) The other input was the average crack depth that is now included in the requirements in BWRVIP-76 (approved and issued for use in November 1999). At the time the calculation was issued (April), BWRVIP-76 was still in draft and had not been issued to the utilities.

Even though one value is less conservative than the other, both values are sufficient to provide a valid calculation and resultant conclusion. The inspection of the H6B/H7 vertical welds can occur between the 3.76 and 6.23 years and still provide the assurance that the shroud is structurally sound. Therefore, delaying the inspection of the H6B/H7 vertical welds to the Unit 2 2005 RIO has no impact, and it is acceptable to change this commitment.