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SEP 2 5 2003

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SUSQUEHANNA STEAM ELECTRIC STATION
10 CFR 50.59 SUMMARY REPORT **Docket Nos. 50-387**
PLA-5678 **and 50-388**

Reference: 1) PLA-5370, R. G. Byram (PPL) to USNRC, "10 CFR 50.59 Summary Report – 1998/2001," dated September 25, 2001.

Enclosed is the summary report of PPL Susquehanna, LLC 50.59 Evaluations that is required to be submitted at intervals not to exceed 24 months by 10 CFR 50.59(d)(2). The previous report (Reference 1) issued September 25, 2001 covered the period from October 30, 1998 to April 1, 2001. This report provides summaries of those 50.59 Evaluations approved during the period of April 2, 2001 to March 31, 2003.

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).

If you have any questions regarding this report, please contact Mr. Duane L. Filchner at (610) 774-7819.

Sincerely,

B. L. Shriver

IE47

Enclosure

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

10 CFR 50.59

Summary of Changes, Tests, Experiments

50.59 Evaluation No. E-01-1

Cross-Reference DCP Numbers: 294805 (U1) 294809 (U2); MSL RM-MSIV Closure and Scram Deletion; LDCN No. 3332, LDCN No. 3529, TRAR # 3530 (U1); TRAR # 3531 (U2)

Description of Change

The action eliminated the reactor protection system (RPS) trip, main steam line isolation valve (MSIV) trip closure, and the main steam drain line valve trip closure. These trips received their signal from the main steam line (MSL) radiation monitor system. The MSL radiation monitor alarm and isolation function for the main condenser mechanical vacuum pump (MVP) and the reactor recirculation sample line isolation were not affected by the change.

Summary

The change to eliminate the RPS trip and vessel isolation closures is recognized as a generic improvement in BWR design and operation. The SSES changes are consistent with the BWR Owners' Group position and recommendations documented in the GE Topical Report NEDO-31400A, Class 1, dated October 1992, "Safety Evaluation for Eliminating the Boiling Water Reactor, Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor."

The topical report was reviewed by the NRC staff and found conditionally acceptable in a letter and accompanying safety evaluation sent to the BWR Owners' Group dated May 15, 1991. The SSES modification complies with the NRC specified conditions for acceptance of the topical report applicability with the exception that, due to the low background radiation level at the monitor location, the alarm and trip setpoint for the pre-treatment radiation monitor continues to be established at a level that is consistent with Technical Specification 3.7.5 and its bases in lieu of the NRC recommended value of 1.5 times the nominal N-16 background dose rate.

Eliminating these trip functions provides for the following benefits:

1) Reduction in inadvertent scram frequency and unnecessary reactor vessel isolations
2) Improved availability of the main condenser for removal of decay heat; and
3) Increased operator control over the pathway for radioactive releases by permitting continued use of the augmented offgas system to process radioactivity.

50.59 Evaluation No. E-01-6

Cross-Reference Unit 2 Technical Specification Bases 3.3.5.1 "ECCS Instrumentation" – LDCN 3395

Description of Change

A condition was identified where an inoperable and un-tripped Reactor Vessel Water Level Low Low Low – Level 1 Switch resulted in declaring both off-site circuits and two diesel generators (DG) inoperable placing both Unit 1 and Unit 2 into LCO 3.0.3. Revised guidance was incorporated into the Unit 2 Technical Specification Bases Section 3.3.5.1 – "ECCS Instrumentation." This 50.59 Evaluation and a PPL Calculation are the regulatory and technical basis documenting the acceptability for the changes.

The equipment impacted included the ECCS initiation logic, LOCA Load Shed logic, and all equipment that is initiated from these logics including but not limited to Core Spray (CS), Residual Heal Removal (RHR), Reactor Building Chillers, Turbine Building Chillers, Drywell Cooling equipment, Emergency Service Water, Emergency Diesel Generators and both Offsite Circuits.

Summary

The revised Unit 2 Technical Specification bases for Section 3.3.5.1 does not require entering LCO 3.0.3 for an inoperable switch in any channel of Reactor Vessel Water Level – Low, Low, Low, Level 1, Drywell pressure – High, and Reactor Steam Dome Pressure – Low associated with initiating Core Spray or Low Pressure Coolant Injection (LPCI) Systems provided both offsite circuits are OPERABLE.

The Technical Specification Bases change revised the guidance on the loss of redundant automatic initiation capability as it related to REQUIRED ACTION B.1 based upon a more detailed analysis. The change also clarified when the Allowed Performance Time (APT) can be applied for the above ECCS Instruments. The change enhanced plant safety in that it prescribed the plant configuration necessary (i.e. two offsite circuits OPERABLE when an ECCS instrument is inoperable and un-tripped) to safely operate both units and to avoid the operational challenges of an unnecessary two unit shutdown. The change had no effect on accidents and malfunctions previously evaluated in the FSAR. The change did not alter the physical design of the plant, nor did it change any procedure that would introduce an accident initiator. Since the change did not introduce any accident initiators, it did not create the potential for a new type of event not previously evaluated in the FSAR. The revised guidance presented in the Technical Specification Bases change does not alter the existing accident analysis; therefore, it does not impact fission product barriers as described in the FSAR. Methodologies described in the FSAR are not impacted by the change.

50.59 Evaluation No. E-01-8

Cross-Reference Unit 1 Cycle 13 Core Loading and Core Operating Limits Report for 3489 MWt Rated Power

Description of Change

The action included the change to Unit 1 reactor core loading to support Cycle 13 operation at a rated power of 3489 MWt.

The evaluation also addressed the implementation of GE Marathon control blades. The GE Marathon control blade design was first used at SSES for SSES Unit 2 in Cycle 11. This control blade design was previously reviewed and approved by the NRC to serve as direct replacements for the GE Duralife 160C control blades currently in use at SSES in both units.

The COLR, FSAR and Power/Flow Map were updated as a result of the changes.

These activities supported SSES Unit 1 for 24 month cycle operation at a rated power of 3489 MWt for Cycle 13.

Summary

The applicable sections of the FSAR related to the core loading and the licensing events that were evaluated for U1C13, including the use of Marathon Control Blades, are Chapters 4, 5, 6, 9, and 15.

All fuel in the U1C13 core was determined to meet the required mechanical, thermal-hydraulic, and nuclear design criteria, and therefore the fuel is fully capable of performing its intended design functions. Furthermore, the U1C13 core loading did not directly or indirectly affect the functioning, performance, reliability, response time, power supplies, cooling, or lubrication of any plant SSC. The GE Marathon control blades were determined to be directly interchangeable with the existing GE Duralife 160C control blades currently in use at SSES (i.e., the form, fit, and function of the Marathon control blades are identical to that of the Duralife 160C control blades). The Power/Flow map was revised to maintain the appropriate stability regions.

FSAR Chapter 15 potentially limiting anticipated operational occurrences were evaluated using methodology which is NRC approved and included in Section 5.6.5 of the Technical Specifications. Results from these analyses were used to determine Core Operating Limits contained in U1C13COLR. These limits shall be used by the core monitoring system.

Accidents that have potential dose consequences were evaluated. The changes to the core loading and the use of the GE Marathon control blades in their designated locations does not cause the consequences of these accidents to exceed the criteria previously evaluated and approved by the NRC. For analyses that are not performed as bounding analyses, the change in the applicable doses is less than a minimal increase.

All analyses that have design basis limits for fission product barriers were evaluated and it was determined that the design basis limits were not exceeded and that they need not be altered.

The following reactivity related evaluations were performed:

1) Core Shutdown Margin (SDM),
2) Standby Liquid Control System (SLCS) and
3) Fuel Storage Criticality

Results of these evaluations (including the use of GE Marathon control blades) determined that U1C13 has adequate shutdown margin, and the SLCS can provide sufficient boron to keep the core subcritical (cold, xenon-free). The new fuel vault and spent fuel pool met their acceptance criteria and remain subcritical.

Additional analyses and evaluations addressed the impact of ATRIUM™ 10 fuel and the 24 month cycle operation on decay heat, the radioactive source terms, heavy loads (movement of heavy loads over irradiated fuel), Post-LOCA hydrogen generation (hydrogen recombiners), Equipment Qualification (In-Containment Emergency Equipment), LOCA electrical time lines (electrical supply), Suppression Pool Heat Load, Spray Pond Analysis, Spent Fuel Pool Boiloff Analysis, Public and Occupational Dose, ATWS, Recirculation Pump Performance, LOCA offsite dose, the Emergency Plan, and the EOPs. The results of these analyses demonstrated that the applicable acceptance criteria for these evaluations were met for U1C13.

All methodology used to evaluate the U1C13 core loading, including the use of GE Marathon control blades, was applied as approved by the NRC.

This was the first time that the U1C13 safety analyses for the Rotated Bundle Analysis (RBA) and the Recirculation Flow Controller Failure (RFCF) event used the new NRC approved methodology. The FSAR was updated to reflect the change in methodology for these two events. Since this new methodology was NRC approved and was contained in Section 5.6.5 of the Unit 1 Technical Specifications, this did not represent a departure from a method of evaluation described in the FSAR. This was only a change from one NRC approved methodology to another.

50.59 Evaluation No. E-01-9

Cross-Reference Unit 1 Technical Specification Bases Section 3.3.5.1 "ECCS Instrumentation" - LDCN 3398

Description of Change

A condition was identified where an inoperable and un-tripped Reactor Vessel Water Level Low Low Low – Level 1 Switch resulted in declaring both off-site circuits and two diesel generators (DG) inoperable placing both Unit 1 and Unit 2 into LCO 3.0.3. Revised guidance was incorporated into the Unit 1 Technical Specification Bases Section 3.3.5.1 – "ECCS Instrumentation." This 50.59 Evaluation and a PPL Calculation are the regulatory and technical basis documenting the acceptability for the changes.

The equipment impacted includes the ECCS initiation logic, LOCA Load Shed logic, and all equipment that is initiated from these logics including but not limited to Core Spray (CS), Residual Heat Removal (RHR), Reactor Building Chillers, Turbine Building Chillers, Drywell Cooling Equipment, Emergency Service Water, Emergency Diesel Generators and both Offsite Circuits.

Summary

The revised Unit 1 Technical Specification Bases for Section 3.3.5.1 does not require entering LCO 3.0.3 for an inoperable switch in any channel of Reactor Vessel Water Level – Low Low Low, Level 1, Drywell pressure – High, and Reactor Steam Dome Pressure – Low associated with initiating Core Spray or Low Pressure Coolant Injection (LPCI) Systems provided both offsite circuits are OPERABLE.

The Technical Specification Bases change revised the guidance on the loss of redundant automatic initiation capability as it relates to REQUIRED ACTION B.1 based upon a more detailed analysis. The change also clarified when the Allowed Performance Time (APT) can be applied for the above ECCS Instruments. The proposed change enhanced plant safety in that it prescribed the plant configuration necessary (i.e. two off site circuits OPERABLE when an ECCS instrument is inoperable and un-tripped) to safely operate both units and to avoid the operational challenges of an unnecessary two unit shutdown. The change had no effect on accidents and malfunctions previously evaluated in the FSAR. The change did not alter the physical design of the plant, nor did it change any procedures that would introduce an accident initiator. Since the change did not introduce any accident initiators, it did not create the potential for a new type of event not previously evaluated in the FSAR. The revised guidance presented in the Technical Specification Bases change does not alter the existing accident analysis; therefore, it does not impact fission product barriers as described in the FSAR. Methodologies described in the FSAR are not impacted by the change.

50.59 Evaluation No. E-01-10

Cross-Reference DCP 337343, 337347 – Control Room Panels 1/2C601 Recorder Replacement

Description of Change

Replace obsolete analog recorders on 1/2C601, Reactor Core Cooling Bench Board, with digital recorders.

Summary

The modifications did not involve revising or replacing any FSAR described methodology, although analog instruments were replaced with digital instruments. The modifications did not involve any test or experiment that was not described in the FSAR for any of the SSC's that are impacted. All required post modification testing was implemented in accordance with established work control practices.

Although use of a microprocessor-based digital instrument introduced the potential of a software failure or EMI/RFI failure, the resulting failure of the instrument is functionally equivalent to failure of the existing recorders. The functionality of the data displayed was not changed. Failure of the hardware and software was minimized by testing. The recorder software is vendor tested, validated and verified (V&V) in accordance with IEEE 7-4.3.2-1993. The replacement recorders comply with EMI/RFI testing based on EPRI TR-102323 Guidelines. The new recorders are seismically and dynamically qualified in accordance with IEEE-344-1987. The modifications did not adversely impact the dynamic qualification of panel 1/2C601and did not adversely affect the power sources. They did not adversely impact the fire protection program. The new recorders are highly dependable and meet the functional requirements for the data to be displayed. They do not degrade any design requirements for the recorders nor did they degrade supporting SSCs for the affected instruments.

Operators utilize data from the new recorders the same way they used data from the old recorders; therefore, the design function remains unchanged. These modifications did not require a new method of performing existing design functions and did not adversely affect any design function described in the FSAR. A failure of a new recorder produces the same functional results as failure of the replaced recorders.

Analog isolators provided electrical isolation between the Non-Class 1E-conductivity inputs reassigned from the old conductivity recorder to the existing Class 1E recorder.

50.59 Evaluation No. E-01-11

Cross-Reference DCPs 99-3058, 99-3059 "Control Room Panels 1/2C614 Recorder Replacement"

Description of Change

Replace analog temperature monitoring recorders on Panels 1/2C614, Nuclear Steam Supply Shutoff, System Steam Leak Detection System Vertical Panels, with digital recorders.

Summary

The modifications did not involve revising or replacing any FSAR described methodology, although it replaced an analog instrument with a digital instrument. Also, the modifications did not involve any test or equipment not described in the FSAR for any SSC impacted. All testing required by the change was normal post maintenance testing and was implemented in accordance with established procedures and work control practices.

Although use of a microprocessor-based digital instrument introduced the potential of software failures and potential failures due to EMI/RFI, the resulting failure of the digital instrument is functionally equivalent to a failure of the replaced recorders. The functionality of the recorder and its data was not changed. Failure of the hardware and software was minimized by vendor testing. The recorder software was vendor tested, validated and verified (V&V) in accordance with IEEE 7-4.3.2-1993. The replacement recorders comply with EMI/RFI testing based on EPRI TR-102323 Guidelines. The new recorders are seismically and dynamically qualified in accordance with IEEE-344, 1975. The modifications did not adversely impact the dynamic qualification of Panel 1/2C614 and did not adversely affect the power sources or adjacent components due to electromagnetic impact. They did not adversely impact the fire protection program. The new recorders are highly dependable and meet the functional requirements for the data to be displayed. They do not degrade any design requirements for the recorders, nor do they degrade supporting SSCs of the affected instruments.

Operators utilize the new recorders the same way they used the replaced recorders; therefore, their design function remained unchanged. These modifications did not require a new method of performing existing design functions and did not adversely affect any design function described in the FSAR. A failure of the new recorders produces the same functional results as failure of the replaced recorders.

50.59 Evaluation No. E-01-13

Cross-Reference Revision to the Turbine Building Crane Operating Procedure

Description of Change

This Turbine Building Crane operating procedure was changed to reflect upgraded controls and to allow lifting of loads over an operating turbine/generator in support of the Unit 1 & 2 Turbine Retrofit Project (TRP).

Summary

The activity was in a non-safety-related area and did not involve a change to a SSC described in the FSAR. Lifting of any loads by the Turbine Building Cranes (of a possible load drop onto the turbine/generator) has no direct effect on any SSC design functions described in the FSAR. Neither the main turbine nor the generator are SSCs important to safety since they are not relied upon to mitigate accidents or transients, and their failure would not prevent safety-related SSCs from fulfilling their intended design function.

FSAR Section 3.5.1.3 contains turbine missile probability analysis which results in probabilities low enough that a turbine missile, and its effect on SSCs important to safety, is not considered as a design basis event for SSES. The initiating event for the probability analysis is a turbine overspeed condition. A load drop onto the turbine was not considered as an initiating event which generates turbine missiles and no conditions have changed to cause a re-evaluation. The FSAR or design basis for SSES does not exclude lifting loads over operating turbine/generators. A plant procedure administratively restricted this lifting. Therefore, this analysis and the results still conclude that a turbine missile is not considered as a design basis event for SSES.

Lifting loads over an operating turbine/generator has no effects on accidents and malfunctions previously evaluated in the FSAR, and will not cause accidents or malfunctions whose effects are not bounded by previous analyses.

From a plant and personnel safety basis, a mitigation strategy was developed to reduce the risks associated with lifting over an operating turbine/generator. This included limiting the number of these engineered lifts, crane and rigging inspections, crane and rigging load factors of safety, safety briefings, and dose control activities. An additional precaution was taken on the turbine deck with the evacuation of the immediate area of the operating turbine/generator during the lifts. Taking into account all considerations presented above, it was concluded that the proposed lifting over an operating turbine/generator could be accomplished safely and prior NRC approval was not required.

50.59 Evaluation No. E-01-14

Cross-Reference SSES Updated Turbine Missile Analysis
PLA-5565, B. L. Shriver to USNRC, "Replacement of Main Turbine Rotors," dated January 22, 2003

Description of Change

The turbine missile analysis has been updated to reflect the material properties and design characteristics of the new turbine rotors.

This 50.59 Evaluation reflected that:

- The final probability of unacceptable damage per unit per year, P_4, is within the NRC recommended value of <1 x 10^{-7}.
- The method used to calculate P4, in accordance with NUREG 1048 Appendix U, is an acceptable method and does not require prior NRC approval,
- The final value for the probability of a turbine failure resulting in the ejection of a missile per unit per year P_1, is <1 x 10^{-5} in accordance with NUREG 1048 Appendix U.

The data used to populate the updated turbine missile damage probability as a result of the Siemens turbine retrofit was calculated using the methods described in NUREG-1048, Supplement No. 6, Safety Evaluation Report Related to the Operation of Hope Creek Generating Station, Appendix U, Probability of Missile Generation in General Electric Nuclear Turbines.

Summary

The turbine missile probability analysis evaluated the probability of damage from postulated turbine missiles to safety-related components at Susquehanna Steam Electric Station using the methodology defined and accepted by the NRC in NUREG-1048, Supplement 6, Appendix U. The probability of unacceptable damage, in accordance with NUREG 1048 Appendix U methodology, to safety-related components due to turbine missiles at Susquehanna Steam Electric Station was calculated as $3x10^{-8}$ per unit per year. This calculated value is within the required probability, less than $1x10^{-7}$ per unit per year, as given by the NRC in Reg. Guide 1.115 and NUREG-1048 Supplement 6, Appendix U. Based upon this low probability, the updated turbine missile hazard is not considered as a design basis event for Susquehanna Steam Electric Station and is consistent with the design basis conclusions outlined in FSAR section 3.5.1.3 for the existing turbine. Therefore, prior NRC approval is not required.

The updated missile probability analysis was prepared by Siemens and is based on NUREG 1048 which has been reviewed by the NRC and accepted for use at the Limerick Generating Station, Grand Gulf, Comanache Peak, and Connecticut Yankee. Accordingly, PPL determined that it is also acceptable to use NUREG 1048, Supplement 6, Appendix U for the Susquehanna.

Using the recommended values for $P_2 \times P_3$, as outlined in NUREG 1048 Appendix U, the calculated value for P_1 is $< 1 \times 10^{-5}$ per unit per year meets the criteria for using the method outlined in NUREG-1048.

50.59 Evaluation No. E-01-16

Cross-Reference Surveillance Requirement (SR) 3.5.1.12 – Technical Specification Bases Change to Remove Requirements to Verify Flow Through SRV Tailpipes

Description of Change

Technical Specification Surveillance Requirement 3.5.1.12 tests the operation of the ADS valves by requiring the valves to be manually actuated. The TS bases for SR 3.5.1.12 provides a testing method to verify that the valve and solenoid are functioning properly and that no blockage exists in the SRV discharge lines. The requirement to verify that no blockage exists in the SRV discharge lines has been deleted. The intent of passing flow through the valve to verify that no blockages exist in the SRV discharge lines is met with the FME program. An alternate and preferred testing method for ADS SRVs has been added to allow for testing of the SRV at atmospheric temperature and pressure without passing steam flow through the valve.

Summary

The change to the TS Bases 3.5.1.12 did not change operation of the ADS valves. This change provided an alternate and preferred test method for the ADS valves. This preferred testing ensures that TS 3.5.1.12 is met without contributing to valve seat damage and further leakage of SRVs. The testing will be performed during cold shutdown when the valves are not required for operation. Therefore, this change did not affect safe operation and it did not increase the frequency or consequences of an accident. It did not affect any of the eight criteria for NRC approval; therefore, NRC approval was not required prior to implementation.

50.59 Evaluation No. E-01-17

Cross-Reference "Change Methodology for HELB Outside Primary Containment"

Description of Change

This FSAR revision changed the computer codes used for the analysis of postulated high energy line breaks outside primary containment, from the FLUD, COPDA and COTTAP2 computer codes to the COTTAP4 computer code. The methodology used in the FLUD and COPDA is discussed in details in FSAR Appendices 3.6A and 6B, therefore a change to elements of this methodology required a 50.59 Evaluation.

Summary

This change did not represent an activity that required prior NRC approval in order to be implemented. The 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. The activity did not result in exceeding a design basis limit for a fission product barrier. The change was a conservative change to elements of the methodology described in the FSAR.

50.59 Evaluation No. E-01-18

Cross-Reference Installation of Handwheels and Use of Manual Operators on the Unit 2 SDV Vent and Drain Valves

Description of Change

Install handwheels on the Unit 2 SVD Vent and Drain Valves. Use of manual handwheels is administratively controlled.

Summary

Administrative means are being used to ensure manual operation of the Unit 2 SDV Vent and Drain Valves is controlled and safe for the operation of Susquehanna SES. Administratively controlling manual operation of the valves ensures that the FSAR design function of the valves (i.e. to close) is maintained. Use of these administrative controls ensures no more than a minimal increase in the frequency and consequences of accidents or malfunctions of equipment important to safety. Use of the Unit 2 SDV Vent and Drain Valve manual operators does not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. Use of the Unit 2 SDV Vent and Drain Valve manual operators does not involve departure from a method of evaluation described in the FSAR.

50.59 Evaluation No. E-01-19

Cross-Reference Removal of the BDID Closure Requirement

Description of Change

This FSAR change removed the requirement to isolate reactor building HVAC ductwork after a high energy line break outside primary containment by not taking credit for the closure function of the Back Draft Isolation Dampers (BDID).

Summary

Removal of the BDID closure requirement did not represent an activity that required prior NRC approval. The 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, nor did it create a new accident or malfunction. The change did not result in exceeding a design basis limit for a fission product barrier as described in the FSAR, since it did not affect a design basis limit for a fission product barrier. There was no change to the methodology described in the FSAR for analyzing a High Energy Line Break and the associated consequences.

50.59 Evaluation No. E-01-20

Cross-Reference Revised Design Basis Accident Results for Unit 2, Cycle 12 Core Loading

Description of Change

The Unit 2 design basis accidents described in Chapter 15 of the FSAR were re-evaluated based upon Unit 2, Cycle 12 Core Loading.

Summary

The design basis accidents described in Chapter 15 of the FSAR were evaluated as part of the Unit 2 Cycle 12 Core Loading change. The results of that evaluation showed that the consequences for the Pump Seizure and Fuel and Equipment Handling Accidents increased slightly from the values currently in the FSAR; however, the change is less than a minimal increase as defined by 10 CFR 50.59, therefore prior NRC approval is not required.

50.59 Evaluation No. E-01-21

Cross-Reference Revised Ultimate Heat Sink (UHS) Analysis to Account for Potential Boundary Valve Leakage in ESW System

Description of Change

Inadequate IST program testing of boundary valves between the ESW and RBCCW/TBCCW heat exchanger isolation valves was considered a violation of Tech Spec 5.5.6 "In-Service Testing." These valves form the "Q" boundary for the ESW system and serve to isolate the ESW system from the non-Q service water system in a design basis accident (DBA). In a DBA, the non-Q service water system would not be available, therefore, there is a potential leak path between the ESW system and the plant service water system via these TBCCW/RBCCW heat exchanger isolation valves. Any leakage past these valves would result in loss of Spray Pond Inventory that was not accounted for in the original Spray Pond design analyses. The UHS Spray Pond analysis was revised to account for potential isolation valve leakage between the ESW and service water system boundary valves.

Summary

The Ultimate Heat Sink (UHS) Maximum Water Loss analysis and Minimum Heat Transfer Analysis were revised to account for potential boundary valve leakage between the ESW and non safety related service water systems. This represented a change to evaluation input parameters for the Spray Pond as described in Section 9.2.7 of the FSAR and did not represent a change or departure from a method of evaluation described in the FSAR.

The revised analysis showed that the Spray Pond would continue to perform its design basis functions with this additional water loss term. The Spray Pond would continue to provide a reliable source of cooling water to the ESW and RHRSW systems under worst case design basis accident conditions.