UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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March 17, 2008

Mr. Britt T. McKinney
Senior Vice President and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Blvd. – NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION INSPECTION REPORTS NOS. 05000387/2008006; 05000388/2008006

Dear Mr. McKinney:

On February 1, 2008, the US Nuclear Regulatory Commission (NRC) completed a team inspection at the Susquehanna Steam Electric Station. The enclosed inspection report documents the inspection results, which were discussed on February 1, 2008, with you and members of your staff.

This inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations and the conditions of your license. Within these areas, the inspection involved examination of selected procedures and representative records, observations of activities, and interviews with personnel.

On the basis of the sample selected for review, the team concluded that the implementation of the corrective action program (CAP) was adequate in that personnel identified issues at a low threshold; generally screened and prioritized issues in a timely manner; evaluated the issues commensurate with their safety significance; and implemented corrective actions in a timely manner commensurate with the safety significance.

The team identified four findings of very low safety significance (Green). These findings were determined to involve violations of regulatory requirements. However, because each of the violations was of very low safety significance (Green) and because they were entered into your corrective action program, the NRC is treating these as Non-Cited Violations (NCVs), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC, 20555-0001, with copies to the Regional Administrator, Region I;

the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC, 20555-0001; and the NRC Resident Inspector at the Susquehanna facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mel Gray, Chief
Technical Support & Assessment Branch
Division of Reactor Projects

Docket Nos. 50-387, 50-388
License Nos. NPF-14; NPF-22

Enclosure: Inspection Report Nos. 05000387/2008006; 05000388/2008006
w/ Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-387, 50-388
License No:	NPF-14, NPF-22
Report No:	05000387/2008006, 05000388/2008006
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	769 Salem Boulevard – NUCSB3 Berwick, PA 18603-0467
Dates:	January 14 – February 1, 2008
Team Leader:	B. Norris, Senior Project Engineer, Division of Reactor Projects
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Approved by:	Mel Gray, Chief Technical Support & Assessment Branch Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000387/2008-006, 05000388/2008-006; 01/14/2008 - 02/01/2008; Susquehanna Steam Electric Station; Biennial Baseline Inspection of the Identification and Resolution of Problems; Corrective Action Program, Simulator Fidelity, and Procedure Quality.

This team inspection was performed by five NRC regional inspectors and one resident inspector. Four findings of very low safety significance (Green) were identified during this inspection and determined to be Non-Cited Violations (NCVs). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Identification and Resolution of Problems

The team concluded that the implementation of the corrective action program (CAP) at Susquehanna was adequate in that personnel identified issues at a low threshold and used a single entry-point system to document the problems by the initiation of an Action Request (AR). About 20 percent of the ARs were considered to be conditions adverse to quality (CAQ) and sub-classified as a Condition Report (CR). However, the team identified several ARs that should have been classified as CAQs; as a result, CRs were not written and corrective actions were not timely. The team identified two findings of very low significance related to the AR process that had current performance cross-cutting aspects in problem identification because the issues were not categorized commensurate with their safety significance. Notwithstanding these two findings, the team concluded that in general Susquehanna personnel screened and prioritized CRs in a timely manner using established criteria.

The team also concluded that Susquehanna personnel properly evaluated the issues commensurate with their safety significance; and generally implemented corrective actions in a timely manner, commensurate with the safety significance. The team noted that Susquehanna reviewed and applied industry operating experience lessons learned. Audits and self-assessments added value to the corrective action process. On the basis of interviews conducted during the inspection, workers at the site expressed freedom to enter safety concerns into the CAP.

a. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green: The NRC identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because, in the 1990s, Susquehanna failed to adequately evaluate a deviation from the Boiling Water Reactor Owner's Group Emergency Procedure Guidelines / Severe Accident Guidelines (BWROG EPG/SAG), which resulted in one of the emergency operating procedures (EOPs) being inadequate. Specifically, Caution #1 in the BWROG EPG/SAG warned the operators that reactor pressure vessel (RPV) level instrumentation may be unreliable if the drywell temperatures exceeded RPV saturation temperature. The purpose of the Caution was to give the operators a chance to evaluate the validity of the RPV level instrumentation to avoid premature entry into the RPV flooding contingency procedure. Susquehanna did not adequately evaluate the deviation, and the Susquehanna EOPs did not use a Caution statement; but instead, changed the caution to a procedural step, which directed the operators to transition directly to the RPV flooding procedure.

  The performance deficiency is more than minor because it is associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the EOP could have directed entry into the RPV flooding procedure unnecessarily which would have restricted the use of suppression pool cooling and required other actions that would have complicated the operators' response to the event. The finding was determined to be of very low safety significance because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events. (Section 4OA2.a.3 (a))

- Green: The NRC identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify that an inconsistency between the procedures and the design basis for suppression pool (SP) cooling was a condition adverse to quality (CAQ), which resulted in corrective actions not being taken in a timely manner. Specifically, in January 2006, a Condition Report (CR) identified an inconsistency between an assumption in the Final Safety Analysis Report (FSAR) for the design basis accident and the emergency operating procedures (EOPs) regarding the timing for the implementation of SP-cooling. At the time of the inspection, the inconsistency had not been resolved because Susquehanna did not recognize that it impacted current plant operations. This performance deficiency has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because Susquehanna did not identify that the inconsistency documented in the CR should have been categorized as a CAQ, commensurate with its safety significance. [P.1(a)]

  The performance deficiency is more than minor because it is associated with the Design Control attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the EOPs provided direction that, under some accident conditions, would affect the availability and/or capability of the SP cooling system to perform its safety function. The finding screened out as having very low safety significance because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events. (Section 4OA2.a.3 (b))

- Green: The NRC identified a Non-Cited Violation of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," because the Susquehanna simulator did not accurately model reactor pressure vessel (RPV) level instrumentation following a design basis accident loss of coolant accident (DBA LOCA). Specifically, an analysis performed in 1994 to determine if the observed simulator response during a large break LOCA was consistent with the expected plant response, was based on an overly conservative assumption that the drywell would experience superheated conditions, which would cause RPV water level instrumentation reference leg flashing and a subsequent loss of all RPV level indication. The expected plant response, as stated in the analysis, was incorrect; in that a LOCA would not always cause a loss of all RPV level instruments. As a result, the simulator modeling was incorrect.

  The performance deficiency is more than minor because it is associated with the Human Performance attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the modeling of the Susquehanna simulator introduced negative operator training that could affect the ability of the operators (a mitigating system) to take the appropriate actions during an actual event. The finding was determined to be of very low safety significance because it is not related to operator performance during requalification, it is related to simulator fidelity, and it could have a negative impact on operator actions. (Section 4OA2.a.3 (c))

- Green: The NRC identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify that a setpoint error in the operating procedures for safety-related systems was a condition adverse to quality (CAQ), resulting in the procedures not being corrected in a timely manner. The setpoint for the low pressure injection permissive interlock in the RHR and CS systems had been changed in 1999 as part of a modification. However, the setpoint was not changed in the system operating procedures and operator aids. When this issue was identified by Susquehanna staff in 2006, the setpoint error in the procedure was not screened as a CAQ, which resulted in the procedures not being revised for 17 months after the issue was identified in an Action Report. This performance deficiency has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because Susquehanna did not identify that a setpoint error in operating procedures for safety-related systems was a CAQ, commensurate with its safety significance. [P.1(a)]

  The performance deficiency is more than minor because it is associated with the Procedure Quality attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the incorrect setpoint reference in the procedure impacted the reliability of operator response to the event in that it could delay operator actions or result in misoperation of equipment. The finding screened out as having very low safety significance because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events. (Section 4OA2.a.3 (e))

b. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution (PI&R) (Biennial - IP 71152B)

a. Assessment of the Corrective Action Program

1. Inspection Scope

The inspection team reviewed the procedures describing the corrective action program (CAP) at the Susquehanna Steam Electric Station. Susquehanna used a single-point entry system and identified problems by the initiation of an Action Request (AR). The AR would then be sub-classified depending on the information provided; for example, as WO for a maintenance Work Order, as CPG for assignment to the Central Procedure Group, or as CR for a Condition Report. ARs were sub-classified as CRs for conditions adverse to quality (CAQ), such as plant equipment deficiencies, industrial or radiological safety concerns, or other significant issues. The CRs were subsequently screened for operability and reportability, categorized by significance (1 to 3), assigned a level of evaluation, and issued for resolution.

The team reviewed CRs selected across the seven cornerstones of safety in the NRC's Reactor Oversight Process (ROP) to determine if problems were being properly identified, characterized, and entered into the CAP for evaluation and resolution. The team selected items from the maintenance, operations, engineering, emergency preparedness, physical security, radiation safety, training, and oversight programs to ensure that Susquehanna was appropriately considering problems identified in each functional area. The team used this information to select a risk-informed sample of CRs that had been issued since the last NRC PI&R inspection, which was conducted in February 2006.

The team selected ARs from other sub-classifications, to determine if Susquehanna had appropriately classified these items as not needing to be a CR. The team also reviewed operator log entries, control room deficiency lists, operator work-around lists, operability determinations, engineering system health reports, completed surveillance tests, and current temporary configuration change packages. In addition, the team interviewed plant staff and management to determine their understanding of and involvement with the CAP at Susquehanna. The CRs, and other documents reviewed, and the key personnel contacted, are listed in the Attachment to this report.

The team considered risk insights from the NRC's and Susquehanna's risk analyses to focus the sample selection and plant tours on risk-significant components. The team determined that the five highest risk-significant systems at Susquehanna were emergency service water, emergency diesel generators, residual heat removal service water, station black-out diesel generator, and reactor core isolation cooling. For the risk-significant systems, the team reviewed a sample of the applicable system health reports, work requests and engineering documents, plant log entries, and results from surveillance tests and maintenance tasks.

The team reviewed CRs to assess whether Susquehanna adequately evaluated and prioritized the identified problems. The CRs reviewed encompassed the full range of Susquehanna's causal evaluations, including root cause analyses (RCA – to determine the cause and prevent recurrence), apparent cause evaluations (ACE – to obtain a basic understanding of the cause), and evaluations (to determine if a problem exists). The review included the appropriateness of the assigned significance, the scope and depth of the causal analysis, and the timeliness of the resolutions. For significant conditions adverse to quality, the team reviewed the effectiveness of the corrective actions to prevent recurrence. The team observed meetings of the CR Screening Team – in which Susquehanna personnel reviewed new CRs for prioritization, and evaluated preliminary corrective action assignments, analyses, and plans. The team also attended meetings of the Corrective Action Review Board (CARB) – where senior managers reviewed selected evaluations, effectiveness reviews, and extension requests.

The team reviewed equipment operability determinations, reportability assessments, and extent-of-condition reviews for selected problems. The team assessed the backlog of corrective actions in the maintenance, engineering, and operations departments, to determine, individually and collectively, if there was an increased risk due to delays in implementation of corrective actions. The team further reviewed equipment performance results and assessments documented in completed surveillance procedures, operator log entries, and trend data to determine whether the evaluations were technically adequate to identify degrading or non-conforming equipment.

The team reviewed the corrective actions associated with selected CRs to determine if the actions addressed the identified causes of the problems. The team reviewed CRs for significant repetitive problems to determine if previous corrective actions were effective. The team also reviewed Susquehanna's timeliness in implementing corrective actions. The team reviewed the CRs associated with selected non-cited violations (NCVs) and findings to determine if Susquehanna properly evaluated and resolved these issues.

2. Assessment

(a) Identification of Issues

In general, the team considered the identification of equipment deficiencies at Susquehanna to be adequate. There was a low threshold for the identification of individual issues, 23,000 ARs were written per year, and about 4,000 of those were sub-classified as CRs. The housekeeping and cleanliness of the plant was generally good; the general cleanliness of the plant enhanced the ability of personnel to more easily identify equipment deficiencies and monitor equipment for worsening conditions.

Notwithstanding, during a tour of the facility, the inspectors observed that high density concrete shield blocks were stacked on pallets in the vicinity of the Unit 1 recirculation motor generator sets. The blocks were pre-staged for work during the upcoming refueling outage, and were in a heavily trafficked area of the turbine building. There was a painted warning on the floor, near the pallets, that the floor loading should not exceed 400 pounds per square foot (psf). When the inspectors asked whether the weight of the blocks was within the rated floor load limit, it was determined that this condition had not been identified and documented as acceptable. Initially, Susquehanna personnel concluded that the blocks exceeded the posted limit and moved the pallets to reduce the floor loading. Subsequently, Susquehanna weighed the pallets and blocks and determined that they did not exceed the allowable floor loading. Based on this evaluation the inspectors concluded the missed identification of this issue was minor. The issue was documented in CR 954950.

The team also identified that several ARs were not classified as CRs, commensurate with the safety significance, as required by their procedure (NDAP-QA-0702, "Action Request and Condition Report Process"). The result was that the issues did not go to the Screening Team, did not receive the necessary management attention, and were not corrected in a timely manner (CR 957319). In addition, ARs are not normally trended to allow the identification of an adverse change in performance. With the exception of the first example, the below are considered procedure violations of minor significance due to no impact on the related equipment. As such, these issues are not subject to enforcement action, in accordance with the NRC's Enforcement Policy.

Examples include:

- AR/CPG/OPS 751412, initiated February 2, 2006, identified that the Low Pressure Injection Permissive setpoint was not changed in the residual heat removal (RHR) and core spray (CS) operating procedures. The setpoint was changed in 1999, as part of a modification; the procedures were not changed until July 2007. (See Section 4OA2.a.3(d) for additional details.)

- AR/OPS/CSHIFT 777335, initiated July 25 2006, identified that an operator started the suppression pool (SP) filter pump contrary to the procedure. The AR was closed with no documented corrective actions taken.

  The safety significance is that the operator did not operate the safety-related system in accordance with the licensee's written procedures and the Technical Specifications (TS). The documentation of corrective actions should have included a determination of the affects of starting of the pump, and counseling of the operator on the requirement to follow procedures.

- AR/CPG 810513, initiated September 16, 2006, identified that the wrong valve numbers were listed for the emergency service water (ESW) system valves for the "E" EDG. As of the inspection, the procedure had not been changed.

  The safety significance is that operators may not have been able to use the licensee's written procedure to align the ESW system in support of the operation of the swing "E" EDG in a timely manner.

- AR/CPG/I&C 938054, initiated December 10, 2007, identified that a functional testing and calibration procedure for the RHR service water radiation monitor could not be performed, as written. As of the inspection, corrective actions had not been taken.

an inconsistency between the procedures and the design basis for SP cooling was a CAQ, which resulted in corrective actions not being taken for two years to the time of the inspection. Although the inconsistency was identified in 2006, Susquehanna personnel did not recognize that the issue impacted current plant operations; as a result, the issue was not scheduled for resolution in a timely manner. The team noted that, although Susquehanna had classified the issue as a CR, it was considered to be "NAQ" – not a CAQ – and was not scheduled for evaluation until the EPU had been approved. Refer to Section 4OA2.a.3(b) for a detailed discussion of the finding.

(b) Prioritization and Evaluation of Issues

The team determined that Susquehanna's performance in this area was adequate. Notwithstanding the above discussion of some ARs not being classified as CRs, the station appropriately reviewed those CRs that went to the Screening team and properly classified them for significance. The discussions about specific topics at the Screening meetings were detailed, and there were no classifications or immediate operability determinations with which the team disagreed. The team considered the contributions of the CARB to add value to the CAP process. One CARB review was noted to be particularly insightful with respect to the quality of the causal analysis for CR 773046. The CR identified problems with the closing of CRs by the nuclear training department without completing all the required actions. The team did not identify any items in the operations, engineering, or maintenance backlogs that were risk significant, individually or collectively. In addition, the quality of the causal analyses reviewed was generally of adequate technical detail and scope to identify causal factors and develop effective corrective actions. The team noted that the RCA for the NCV from the last PI&R inspection related to scaffolding was effective in that there had not been significant recurrences of inadequate scaffold installations since the evaluation was completed.

With regard to operability evaluations, the team observed that, an operability determination for the PAM level instruments, conducted in response to an inconsistency between the FSAR and EOPs, determined that the level instruments would be operable. (The inconsistency between the FSAR and the EOPs is described in detail in section 4OA2.a.3(b).) During follow-up discussions, the inspectors were told by operations and engineering personnel that all of the PAM instrumentation together functioned to provide the needed indications to the operators, and that the RPV level indications were not needed after the initial entry into the EOPs. This was not consistent with the requirements for the operability of each individual function of the PAM, as detailed in TS 3.3.3.1. Although subsequent discussions with the Susquehanna staff determined that the most (if not all) of the PAM RPV level instruments would indicate post-LOCA, the initial operability determination and statements during the inspection did not consider that the PAM level instruments are required to be operable post-accident regardless of whether EOPs have been entered. This issue was related to the performance deficiencies discussed in findings 4OA2.a.3(a), (b) and (c), and is not identified as an additional finding. The issue was entered into the CAP as AR/CR964836.

(c) Effectiveness of Corrective Actions

No findings of significance were identified in the area of effectiveness of corrective actions. The team determined that the effectiveness of corrective actions at Susquehanna was generally good. The control of scaffolds was a significant problem during the last PI&R inspection; the team noted that oversight of scaffolds has improved, but station personnel continue to identify examples where the scaffold does not appear to be built in accordance with the procedure. In addition, the team identified weaknesses in the scaffold procedure, such as allowing the installer to approve deviations from the approved construction. During the inspection, the procedure was revised, and plans were developed for engineering to review all current deviations.

3. Findings

(a) Failure to Adequately Evaluate a Deviation from BWROG EPG/SAG Resulted in an Inadequate Procedure

Introduction: The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because Susquehanna failed to adequately evaluate a deviation from the Boiling Water Reactor Owner’s Group Emergency Procedure Guidelines / Severe Accident Guidelines (BWROG EPG/SAG), which resulted in one of the Emergency Operating Procedures (EOPs) being inadequate.

Description: On January 5, 2006, AR/CR 739371 was initiated to document an inconsistency between the EOPs and assumptions in the Final Safety Analysis Report (FSAR) regarding the initiation of suppression pool cooling. Specifically, it was identified that the assumptions used in evaluating SP temperature response for the most limiting design basis accident (DBA) loss of coolant accident (LOCA) did not appear to be consistent with direction provided in the EOPs.

During this inspection, the team noted that the Susquehanna EOPs were not consistent with the BWROG EPG/SAG. Specifically, BWROG EPG/SAG, Revision 2, Caution #1, warned the operators that reactor pressure vessel (RPV) level instrumentation may be unreliable if the temperatures near the instrument sensing lines exceeded RPV saturation temperature. The EPG Bases stated that the purpose of Caution #1 was to give the operators a chance to evaluate the validity of the RPV level instrumentation, in order to avoid premature entry into the RPV flooding contingency procedure before it was appropriate to do so. Susquehanna did not adequately evaluate the deviation from the generic guidance in the EPG/SAG with respect to the caution. The Susquehanna EOPs did not use a Caution statement, which would have allowed the operators the opportunity to evaluate the level instrumentation; but instead, changed the caution to a procedural step which directed the operators to transition directly to the RPV Flooding procedure. Specifically, EO-100-103-1, “Primary Containment Cooling,” step DWT-3, directed the operators to transition to contingency procedure EO-000-114-1, “RPV Flooding,” when drywell temperature exceeded RPV saturation temperature.

The evaluation for the deviation was not completed in accordance with the requirements of procedure NDAP-QA-0330, “Symptom Oriented EOP and EP-DS Program and Writer’s Guide.” The procedure required that all deviations be evaluated to determine if the deviation was technically justified and appropriate. Susquehanna documented that the deviation was a minor “difference” from the generic guidelines in 50.59 Safety Evaluation NL-92-019 (October 29, 1998) and 50.59 Screen 5059-01-976 (July 3, 2002). The evaluation was based on an overly conservative assumption that all RPV level instrumentation would be lost after a DBA LOCA. The reviews did not evaluate the potential adverse consequences associated with the deviation, including the potential impact on the SP cooling safety function. Immediate corrective actions included the initiation of an informational Night Order to the control room operators explaining the issue, and the cessation of all simulator scenarios that involve the use of EO-100-103-1 until the issue is resolved.

The performance deficiency is the failure to adequately evaluate a deviation from the BWROG EPG/SAG, which resulted in one of the EOPs being inadequate for use by the operators in the event of a DBA LOCA. Specifically, under some accident conditions, the EOPs would have unnecessarily directed entry into RPV flooding which would have limited the availability of SP cooling and complicated the operators’ response to the event.

Analyses: This performance deficiency is more than minor because it is associated with the Procedure Quality (EOP) attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the EOP could have directed entry into the RPV flooding procedure unnecessarily which would have restricted the use of suppression pool cooling and required other actions that would have complicated the operators’ response to the event. The inspectors performed a review of the finding in accordance with NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP),” Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” and determined that the finding screened out as having very low safety significance (Green), because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events.

Enforcement: 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states, in part, that activities affecting quality shall be prescribed by documented procedures appropriate to the circumstances and that the activities shall be accomplished in accordance with the procedures. Contrary to the above, Emergency Operating Procedure EO-100-103-1, “Primary Containment Cooling,” was inadequate, in that it directed the operators to transition directly to the RPV Flooding procedure when RPV level instruments may have been available, which resulted in limiting the availability of SP cooling. However, because the finding was of very low safety significance (Green) and has been entered into the CAP (AR/CR 962881), this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy.

(NCV 05000387/2008006-01; 05000388/2008006-01 – Failure to Adequately Evaluate a Deviation from BWROG EPG/SAG Resulted in an Inadequate EOP)

(b) Failure to Identify and Correct Inconsistencies Between the FSAR and the EOPs

Introduction: The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify that an inconsistency between the emergency operating procedures and the design basis for SP cooling was a CAQ, which resulted in corrective actions not being taken for two years to the time of the inspection. Although the inconsistency was identified in 2006, Susquehanna personnel did not recognize that the issue impacted current plant operations; as a result, the issue was not scheduled for resolution in a timely manner. The assumption in the FSAR for the DBA LOCA stated that SP cooling would be implemented ten minutes after entry into the EOPs. The EOPs would not have allowed initiation of SP cooling for an extended period of time.

Description: On January 5, 2006, AR/CR 739371 was initiated to document an inconsistency between the EOPs and design basis assumptions for the SP cooling response. The problem was identified during Susquehanna's review in support of the extended power uprate (EPU) project. Specifically, Susquehanna Engineering identified that the assumptions used in evaluating SP temperature response for the most limiting LOCA did not appear to be consistent with direction provided in the EOPs. The team noted that, although Susquehanna personnel had classified the issue as a CR, they did not recognize that the issue impacted current plant operations. Therefore, it was considered to be "NAQ" – not a condition adverse to quality – and was not scheduled for evaluation until the EPU had been approved.

The Susquehanna FSAR, Section 6.2.1.1.3, stated that the maximum SP temperature would result from a reactor recirculation suction line break. The drywell pressure and temperature response analyses assumed that RHR heat exchangers were activated about ten minutes after entry into the EOPs to remove energy from the drywell by cooling the SP. The CR identified that, in the event of a DBA LOCA, the EOPs would direct operators to implement the RPV flooding procedure (EO-000-114) to maintain adequate core cooling, and this required that all available RHR flow be used to flood the RPV up to the steam lines. The initiator's concern was that this would delay establishing flow through a RHR heat exchanger for SP cooling, because of the unique design of the RHR system at Susquehanna, and therefore would be inconsistent with the accident analyses assumptions. In addition, the CR stated that it was assumed in the EOPs that all RPV water level indications would be unreliable and therefore unavailable for this scenario. Susquehanna personnel informed the team that they had not evaluated the issues documented in the CR, at the time it was initiated, because they had assumed that they were only associated with EPU and not current plant operation. Immediate corrective actions included the start of an evaluation during the inspection of the identified inconsistency for SP cooling, and additional guidance to the operators.

The performance deficiency is the failure to properly categorize the inconsistency between the FSAR and the EOPs as a CAQ, which resulted in the deficiency not being corrected in a timely manner commensurate with its safety significance.

Analyses: The performance deficiency is more than minor because it is associated with the Design Control attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, in the event of a DBA LOCA, SP cooling would not be initiated within the time frame assumed in the FSAR, which could affect the capability of the system to perform its safety function consistent with the design basis. The inspectors performed a review of the finding in accordance with IMC 0609, and determined that the finding screened out as having very low safety significance (Green) because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events.

This performance deficiency has a cross-cutting aspect in the area of Problem Identification and Resolution (PI&R), Corrective Action Program (CAP), because Susquehanna did not identify that the inconsistency documented in the CR should have been categorized as a CAQ, commensurate with its safety significance. [P.1(a)]

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that conditions adverse to quality shall be promptly identified and corrected. Contrary to the above, Susquehanna failed to identify that the nonconformance identified in AR/CR 739371, January 2006, was a CAQ; this resulted in the condition not being corrected for over two years. However, because the finding was of very low safety significance (Green) and has been entered into the corrective action program (AR/CR 959670), this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy.

(NCV 05000387/2008006-02; 05000388/2008006-02 – Failure to Identify and Correct Inconsistencies Between the FSAR and the EOPs)

(c) Failure to Accurately Model the Simulator for RPV Water Level Instrumentation

Introduction: The NRC identified a Green NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," because the Susquehanna plant-referenced simulator did not accurately model RPV level instrument response following a DBA LOCA. Specifically, the RPV level instruments in the simulator were programmed to fail high after a LOCA, and the expected plant response is that the instruments should indicate properly.

Description: As part of the team's follow-up on the issues in AR/CR 739371, the inspectors questioned the concern stated in the CR, that the operators would need to enter the RPV flooding procedure during a DBA LOCA due to a loss of valid RPV level instrumentation. The inspectors reviewed the Susquehanna specific EOPs and supporting documents, and determined that the Susquehanna EOP Plant Specific

Technical Guideline (PSTG) description of the expected response of the RPV level instrument response to LOCA events, was based on analysis, EC-SIMU-1001, “Evaluation of Simulator Level Instrument Response to Large LOCA,” dated May 4, 1994. The analysis was performed to determine if the observed simulator response during a large break LOCA (RPV level instrumentation off-scale high) was consistent with the expected plant response. The analysis assumed that the drywell would experience superheated conditions, which would cause RPV water level instrumentation reference leg flashing and a subsequent loss of all RPV level indication. The analysis concluded that the simulator response reasonably predicted the expected actual plant response during a large break LOCA event. The expected plant response, as stated in the analysis, was incorrect; in that a LOCA would not always cause a loss of all RPV level instruments.

On January 29, 2008, the inspectors observed two scenarios in the simulator to evaluate the response to a DBA LOCA, with all safety systems available. The inspectors observed that the RPV level instruments did indicate off-scale high shortly after the initiation of the event, consistent with the analysis. The inspectors questioned the basis of the analysis; specifically, why Susquehanna believed that the level instruments would not be available after a DBA LOCA event. Subsequently, Susquehanna determined that the RPV level instrument reference legs were not expected to routinely flash during a DBA LOCA, and that the analysis had been based on an overly conservative assumption that the drywell would always reach superheated conditions post-LOCA. Immediate corrective actions included the initiation of an informational Night Order to the control room operators explaining the issue, and the cessation of all simulator scenarios that involve the use of EO-100-103-1 until the issue is resolved.

The performance deficiency is that Susquehanna did not ensure that the plant referenced simulator accurately modeled the expected plant response for RPV level instrumentation after a DBA LOCA, resulting in negative training of the licensed operators.

Analyses: This performance deficiency is more than minor because it is associated with the Human Performance attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the incorrect modeling of the Susquehanna plant referenced simulator introduces negative operator training that could affect the ability of the operators (a mitigating system) to take the appropriate actions during an actual event. The simulator training conditioned the operators to expect the level instruments to be unavailable during events that cause drywell temperatures to reach or exceed RPV saturation temperature. As a result, during an actual event, the operators could prematurely transition into the RPV flooding procedure when the RPV level instruments should be providing valid indication. The inspectors evaluated the finding in accordance with IMC 0609, Appendix I, “Licensed Operator Requalification Significance Determination Process.” The finding was determined to be of very low safety significance (Green) because it is not related to operator performance during requalification, it is related to simulator fidelity, and could have a negative impact on operator actions.

Enforcement: 10 CFR 55.46(c)(1), "Plant Referenced Simulators," states, in part, that a plant referenced simulator must demonstrate expected plant response to normal, transient, and accident conditions. Contrary to the above, as of January 2008, the Susquehanna plant referenced simulator did not accurately demonstrate the actual expected plant response of the RPV water level instrumentation following a DBA LOCA, which could result in negative operator training. However, because the finding was of very low safety significance (Green) and has been entered into the CAP (AR/CR 962881), this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy**.**

(NCV 05000387/2008006-03; 05000388/2008006-03 – Failure to Accurately Model the Simulator for RPV Water Level Instrumentation)

(d) Failure to Identify and Correct a Setpoint Error in the RHR and CS Operating Procedures

Introduction: The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify that a setpoint error in the operating procedures for safety-related systems was a CAQ, resulting in the procedures not being corrected in a timely manner. Specifically, in February 2006, Susquehanna personnel identified an incorrect setpoint for the low pressure injection permissive interlock in the RHR and CS systems operating procedures and associated "hard cards"; however, the procedures were not revised until July 2007 due to the issue being screened as low priority and not a condition adverse to quality (CAQ).

Description: On February 11, 2006, an AR was written to identify that the low pressure injection permissive setpoint in the RHR and CS operating procedures, and the associated operator "hard cards," was incorrect. The correct setpoint is 420 pounds per square inch gage (psig), but the procedures still had the previous setpoint of 436 psig. The setpoint had been changed in 1999 as part of a modification. The procedures were not revised until July 16, 2007, 17 months after the deficiency was identified in an AR. In addition, the inspectors noted that the setpoint in the procedures (436 psig) was not within the allowable tolerance (407-433 psig) listed in the Susquehanna TS, Section 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation."

When the AR was initiated, it was sub-classified as AR/CPG/OPS; that is, assigned to the Central Procedures Group and identified as an Operations procedure. It was not recognized that deficient operating procedures for safety-related systems may be a CAQ and that the AR should have been classified as a Condition Report. The affected section in the procedures was the verification of the response of the systems to an automatic initiation signal. For example, the Unit 1 RHR procedure OP-149-001, "RHR System," Section 2.2, noted that "No operator action is required unless an automatic action failed to occur ... At ≈436 psig decreasing Reactor pressure, RHR INJ OB ISO [injection outboard isolation] HV-151-F015A & B OPEN." If the valves did not open at the specified pressure in the procedure and "hard card," the operator may have diverted their attention unnecessarily and attempted to open the valve manually, even though the interlock would not have been satisfied (420 psig) and the valve would not open in accordance with the plant design.

The pressure switches were changed in 1999, as part of a Unit 1 plant modification (Design Change Package (DCP) 97-9075); Unit 2 switches were changed by DCP 97-9076. The modification replaced the existing pressure switches with Barton pressure indicating switches, because of improved accuracy. The low pressure injection permissive interlock prevents the CS and RHR injection valves from opening until reactor pressure has decreased to the RHR and CS systems design pressure, to prevent over pressurization of the RHR and CS systems. The DCP identified the specific RHR and CS operating procedures as needing to be changed. Immediate corrective actions included the initiation of a new CR to evaluate the other pending procedure changes to determine if their priority should be revised.

The performance deficiency involved a failure to identify and correct a CAQ, the incorrect setpoint, in a timely manner commensurate with its safety significance. The inspectors concluded this action was untimely because the modification process would have revised these procedures prior to the modification being accepted by operations personnel.

Analysis: The performance deficiency is more than minor because it is associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the incorrect setpoint reference in the procedure impacted the reliability of operator response to the event in that it could delay operator actions or result in misoperation of equipment. The inspectors performed a review of the finding in accordance with NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP),” Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings.” The inspectors determined that the finding screened out as having very low safety significance (Green), because it was not a design deficiency, did not result in an actual loss of safety function, and did not screen as potentially risk significant due to external initiating events

This performance deficiency has a Cross-Cutting aspect in the area of PI&R, CAP, because Susquehanna did not identify that a setpoint error in operating procedures for safety-related systems was a CAQ, commensurate with its safety significance. [P.1(a)]

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” states, in part, that conditions adverse to quality shall be promptly identified and corrected. Contrary to the above, from 1999, when the pressure switches were replaced and the setpoint was changed, until 2006, when AR 751412 was written, Susquehanna had failed to identify that the setpoint was wrong for the low pressure injection permissive interlock in the operating procedures for RHR and CS. Subsequently, on February 11, 2006, when Susquehanna personnel initiated and approved AR 751412, they failed to identify that the stated deficiency was a CAQ, which resulted in untimely corrective actions. Susquehanna considered this to be a procedure change and not a CAQ, and classified the AR as a CPG versus a CR. As such, the procedures were not changed until July 16, 2007, 17 months after the condition was identified and eight years after the setpoint was changed in the plant. Because this finding is of very low safety significance (Green), and was entered into the Susquehanna CAP (AR/CR 956917) this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy.

(NCV 05000387/2008006-04; 05000388/2008006-04 – Failure to Identify and Correct a Setpoint Error in the RHR and CS Operating Procedures)

b. Assessment of the Use of Operating Experience

1. Inspection Scope

The team reviewed a sample of operating experience (OE) issues for applicability to Susquehanna, and for the associated actions. The documents were reviewed to ensure that underlying problems associated with the issues were appropriately considered for resolution. The team also reviewed how Susquehanna considered OE for applicability in causal evaluations.

Prior to the start of the inspection, the inspectors noted a potential negative trend in the number of issues associated with reactivity management. In accordance with the Inspection Procedure, the inspectors increased the scope of the review to determine if there was an adverse trend in the area of reactivity management over the past five years. The inspectors reviewed select ARs and CRs associated with the control rod drive system, control rod problems, human performance issues, and the spent fuel pool; the inspectors review included how Susquehanna had incorporated applicable OE for these specific systems and human performance issues into the CAP. The inspectors interviewed selected licensee staff.

2. Assessment

In general, OE was effectively used at the station. The inspectors noted that OE was reviewed during the causal evaluation process and incorporated, as appropriate, into the development of the associated corrective actions. The inspectors noted that OE was frequently used in work packages and pre-job briefs. The team did not identify any significant deficiencies within the sample reviewed. The team did not identify a negative trend nor any significant problems with the control of activities associated with reactivity management.

3. Findings

No findings of significance were identified in the area of operating experience.

c. Assessment of Self-Assessments and Audits

1. Inspection Scope

The team reviewed a sample of departmental self-assessments, CAP trend reports, and Quality Assurance (QA) audits, including QA's most recent audit of the CAP. The team also reviewed the latest internal assessment of the safety culture at Susquehanna, conducted in October 2006. The reviews were performed to determine if problems identified through these evaluations were entered into the CAP system, and whether the corrective actions were properly completed to resolve the deficiencies. The effectiveness of the audits and self-assessments was evaluated by comparing audit and self-assessment results against self-revealing and NRC-identified findings, and observations during the inspection.

2. Assessment

The team considered the quality of the audits and self-assessments to be thorough and critical. ARs were initiated for issues identified by QA and the self-assessments. The Susquehanna 2006 "Comprehensive Cultural Assessment" Report consisted of a safety culture survey and interviews. The cultural assessment report identified some weaknesses at the station, which were entered into the CAP. The team did not identify any results that were inconsistent with Susquehanna's conclusions.

3. Findings

No findings of significance were identified in the area of audits and self-assessments.

d. Assessment of Safety Conscious Work Environment

1. Inspection Scope

To evaluate the safety conscious work environment (SCWE) at Susquehanna, during interviews and discussions with station personnel, the team assessed the workers willingness to enter issues into the CAP and to raise safety issues to their management and/or to the NRC. The inspectors also interviewed the Employee Concerns Program (ECP) representative to determine if employees were aware of the program and had used it to raise concerns. The team reviewed a sample of the ECP files to ensure that issues were entered into the corrective action program, as appropriate.

2. Assessment

Based on interviews, observations of plant activities, and reviews of the ARs and ECP, the inspectors determined that the site personnel were willing to raise safety issues and document them in ARs. Individuals actively utilized the AR system, as evidenced by the number and significance of issues entered into the program. The inspectors noted that ARs were written by a variety of personnel, from workers to managers. ECP evaluations were thorough and appropriate actions were taken to address issues.

3. Findings

No findings of significance were identified related to the SCWE at Susquehanna.

4OA6 Meetings, Including Exit:

On February 1, 2008, the team presented the inspection results to Mr. B. McKinney, Senior Vice President, and to other members of the Susquehanna staff, who acknowledged the findings. The team confirmed that no proprietary information reviewed during the inspection was retained.

ATTACHMENT: Supplemental Information

In addition to the documentation that the team reviewed (listed in the Attachment), copies of information requests given to the licensee are in ADAMS, under accession number ML080430585.

ATTACHMENT - SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:

M. Adelizzi, Risk Engineer
N. D'Angelo, Manager, Station Engineering
C. Gannon, Vice President, Nuclear Operations
T. Gorman, Project Manager, Design Engineering
R. Hoffman, Manager, Nuclear Fuels & Analysis
B. McKinney, Chief Nuclear Officer
I. Missien, Project Manager, System Engineering
B. O'Rourke, Senior Engineer, Nuclear Regulatory Affairs
R. Pagodin, General Manager, Nuclear Engineering
R. Paley, General Manager, Plant Support
A. Price, Supervisor, Corrective Action & Assessment
M. Rochester, Employee Concerns Representative
G. Ruppert, Manager, Maintenance
R. Schechterly, Operating Experience Coordinator
R. Sgarro, Manager, Nuclear Regulatory Affairs
M. Sleigh, Security Manager
B. Stitt, Operations Training
T. Tonkinson, Supervisor, Maintenance Support
D. Weller, Maintenance Foreman
L. West, Supervisor, Central Procedure Group

Nuclear Regulatory Commission:

M. Gray, Branch Chief, Technical Support & Assessment
F. Jaxheimer, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed:

05000387/2008006-01 05000388/2008006-01	NCV	Failure to Adequately Evaluate a Deviation from BWROG EPG/SAG Resulted in an Inadequate EOP	(Section 4OA2.a.3 (a))
05000387/2008006-02 05000388/2008006-02	NCV	Failure to Identify and Correct Inconsistencies in the Licensing Basis and the EOPs	(Section 4OA2.a.3 (b))
05000387/2008006-03 05000388/2008006-03	NCV	Failure to Accurately Model the Simulator for RPV Water Level Instrumentation	(Section 4OA2.a.3 (c))
05000387/2008006-04 05000388/2008006-04	NCV	Failure to Identify and Correct a Setpoint Error in the RHR and CS Operating Procedures	(Section 4OA2.a.3 (d))

LIST OF DOCUMENTS REVIEWED

Procedures:

BWROG EGP/SAG and Appendix B Bases, Revision 2
Design Considerations Applicability Sheet Number 42, Emergency Plan, Revision 1
EO-000-102, RPV Control, Revision 2
EO-000-114-1, RPV Flooding, Revision 5
EO-100-103-1, Primary Containment Control, Revision 9
EP-AD-014, Surveillance Testing of Emergency Communications Equipment, Revision 10
EP-AD-015, Review, Revision, and distribution of the SSES Emergency Plan, Revision 11
ME-0RF-161, Control of Fuel Pool Cleanout Activities, Revision 5
ME-0RF-163, Fuel Pool Cleanout – Energy Solutions – Dose Rate Profiling of Irradiated Hardware and Liners, Revision 4
MFP-QA-1220, Engineering Change Process Handbook, Revision 2
MI-VL-009, Operation of Leak Rate Monitors, Surge Tank Assemblies and 1035 psig Test Pumps, Revision 3
MT-AD-504, Scaffold Erection, Review and Inspection, Revisions 9 & 10
MT-GM-018, Freeze Sealing of Piping, Revision 15
MT-GM-050, Limitorque Type SMB-000 through SMB-4 Operator Maintenance, Revision 12
NASP-QA-202, Independent Technical Review Program, Revision 2
NASP-QA-401, Internal Audits, Revision 9
NASP-QA-700, Performance Assessment Process, Revision 0
NDAP-00-0109, Employee Concerns Program, Revision 10
NDAP-00-0708, Corrective Action Review Board, Revision 4
NDAP-00-0710, Station Trending Program, Revision 1
NDAP-00-0745, Self-Assessment, Benchmarking and Performance Indicators, Revision 7
NDAP-00-0751, Significant Operating Experience Report (SOER) Review Program, Revision 3
NDAP-00-0752, Cause Analysis, Revisions 3 and 4
NDAP-00-0753, Common Issue Analysis, Revision 0
NDAP-00-0778, Performance Improvement Program, Revision 2
NDAP-QA-0103, Audit Program, Revision 9
NDAP-QA-0330, PSTG and Emergency Procedures, Revision 8
NDAP-QA-0330, Symptom Oriented EOP and EP-DS Program and Writer's Guide, Revision 3
NDAP-QA-0412, Leakage Rate Test Program, Revision 10
NDAP-QA-0702, Action Request and Condition Report Process, Revision 20
NDAP-QA-0703, Operability Assessments and Requests for Enforcement Discretion, Revision 12
NDAP-QA-0720, Station Report Matrix and Reportability Evaluation Guidance, Revision 13
NDAP-QA-0725, Operating Experience Review Program, Revision 11
NDAP-QA-0726, 10CFR50.59 and 10CFR72.48 Implementation, Revision 10
NDAP-QA-1220, Engineering Change Process, Revision 2
NTP-QA-53.1, Susquehanna Fire Brigade Training Program, Revision 15
ODCM-QA-001, ODCM Introduction, Revision 3
ODCM-QA-002, ODCM Review and Revision Control, Revision 4
ODCM-QA-003, Effluent Monitor Setpoints, Revision 3
ODCM-QA-004, Airborne Effluent Dose Calculations, Revision 4
ODCM-QA-005, Waterborne Effluent Dose Calculation, Revision 3

ODCM-QA-006, Total Dose Calculation, Revision 2
ODCM-QA-007, Radioactive Waste Treatment Systems, Revision 2
ODCM-QA-008, Radiological Environmental Monitoring Program, Revision 11
ODCM-QA-009, Dose Assessment Policy Statements, Revision 2
ON-145-004, RPV Water Level Anomaly, Revision 13
OP-024-001, Diesel Generators, Revision 49
OP-024-004, Transfer and Test Mode Operations of Diesel Generator E, Revision 26
OP-149-001, RHR System, Revisions 31 and 32
OP-151-001, Core Spray System, Revisions 27 & 28
SE-124-007, Unit 1 Division 1 Diesel Generator LOCA LOOP Test, Revision 15
SE-259-044, LLRT of RHR Containment Spray Penetration Number X-39A, Revision 11
SOP-054-B03, Quarterly ESW Flow Verification Loop B, Revision 7
SSES-EPG, SSES Plant Specific Technical Guideline, Revision 9

Audits:

666178, Corrective Action, November 2006 – February 2007
667966, QA Internal Audit Report, Fuel Management, Revision 0
691277, QA Internal Audit Report Access Authorization and Fitness for Duty, Revision 0
706249, Operations Training and Qualification Programs, May – June 2007
718607, QA Internal Audit Report, Engineering, Revision 0
744333, Operations, November – December 2007
792034, QA Internal Audit Report, Security, Revision 0
NEIP Audit of Susquehanna Quality Assurance, June 2006

Self-Assessments:

2006 Comprehensive Cultural Assessment, September – October 2006
CA&A Functional Unit Excellence Plan, 1st, 2nd, and 3rd Quarters 2007
CAA-06-01, Site Wide Self-Assessment, December 2006
CAA-06-05, Self-Assessment Program Performance, February 2006
CAA-06-08, Decrease in CR Generation Identified by Trend Report, November 2006
Focused Self Assessment, MOV Program Self-Assessment, October 2007
Maintenance Implementing Procedures Adequacy for Qualified, Inexperienced Employees, June 2007
Multi-Utility Joint Audit Program Initiative, March – April 2007
NTG Focused Self-Assessment of Operator Training Programs, June 2007
OPS-06-02, Determine the Status of Operator Fundamentals, February 2006
OPS-06-03, Operations Focused Se-f Assessment, July 2006
Pre-PI&R Focused Self-Assessment, September 2007
QA Organization Effectiveness Self-Assessment, October 2006
QA-06-01, Operations QA Audit Preparation Gap Analysis for QC, May – July 2006
SEC-06-01, Analyses of SSES Security Procedures and Physical Security Plan, Revision 0

Action Requests (* denotes an AR/CR generated as a result of this inspection):

478369	724467	741707	759209	779830	810391	843985	873741	896685	941677
524893	724717	741908	759216	780144	810513	845441	873919	897250	941810
542157	726672	741943	759827	780155	811239	849935	874227	898909	947160
545804	728295	742191	760281	780778	811429	851918	875597	899429	954950*
549328	728936	742318	760526	780992	811996	853358	875976	900301	954970*
554362	730852	742342	760526	781644	812948	854681	876021	900720	954972*
554598	730944	742427	762497	782321	813844	855266	876427	901262	954975*
555140	730947	742676	763050	782344	815268	855268	877419	903439	954990*
555263	737236	742966	763128	783655	816097	856997	877727	904689	955072*
555562	738555	743043	763397	784730	816710	858269	877743	908163	955073*
557348	738575	744975	764145	784882	817720	858578	878165	911601	955111*
565795	738634	744979	764738	784890	818082	859082	878326	912213	955130*
575128	738653	745221	764953	785561	818154	859440	879080	912476	955150*
578943	738907	745248	765421	785791	820344	859794	879847	915167	955151*
584400	738999	745462	767566	786149	820380	859839	880331	915620	955761*
591033	739262	745773	767567	786224	820989	860299	880573	916453	955780*
594366	739371	746658	768301	786564	820995	860551	880702	916463	956339*
594887	739371	747077	768502	786735	821006	861162	880806	916873	956344*
595165	739386	747438	768821	786768	821064	861366	881210	917196	956431*
604009	739419	749294	768920	787850	822996	861415	881219	918392	956696*
604296	739579	749341	769304	788616	823908	862474	881225	918549	956914*
610978	739625	749832	769867	788621	824522	864090	881236	919470	956917*
615707	739713	750140	769870	788879	824895	865286	882318	927046	957319*
623914	739737	750232	770453	789971	825107	865423	883987	928515	957484*
623949	740043	751212	771319	791115	825750	865804	886209	929461	957637*
635924	740073	751412	771876	791329	826452	865924	887048	930075	958769*
647827	740303	751433	771961	792158	826870	866930	887067	930571	959670*
655735	740477	751444	773046	793381	827023	867534	888310	931113	961655
666405	740538	752341	773409	794995	827966	867747	889683	932590	962390
668871	740658	752347	774453	795583	828626	867881	889966	936060	962881*
669732	740668	752582	774475	796640	828744	868251	891288	936250	963061*
677145	740723	753392	774509	797517	829065	868259	891733	936370	963065*
687080	740802	753664	774549	799890	829502	868828	891795	936631	963698*
688300	740804	753869	775285	802254	835002	868874	892142	937123	963861*
691108	740825	753990	775718	802539	837153	869819	892152	938054	964512*
693936	740946	755360	776112	802563	837180	869824	892528	938698	964514*
699781	740948	756094	776171	802572	839753	870968	893090	938722	964836*
723483	740955	756415	776769	802697	841169	871013	893157	939516	965167*
723976	740988	756804	776918	805698	841885	872039	893290	939780	
724102	741041	757530	777335	806710	842663	872056	895147	941290	
724165	741321	757979	777723	809503	842920	873026	896455	941401	
724374	741457	758337	778124	809702	843144	873683	896505	941626	

Maintenance Work Requests (SPWO):

099065	099364	766396	766413	767284	768234	862569
099115	448229	766401	766416	767490	768618	862578
099120	473889	766406	766496	767506	818282	866262
099259	570758	766411	767283	767532	862503	866284

Non-Cited Violations and Findings Reviewed:

NCV 2005005-01, Inadequate FME Exclusion Procedural Instructions Associated with EDG Work
FIN 2005009-01, Fire Brigade Drill Program Not Consistent with Regulatory Guidance and Industry Standards
NCV 2006002-01, Equipment Hatch Plugs are Not Watertight as Indicated in FSAR
FIN 2006002-02, Incomplete Corrective Actions Contribute to CRD Flow Control Failure
NCV 2006003-01, Inadequate Procedures Resulted in Motor Operated Valve Failures
NCV 2006003-02, Failure to Identify Material Degradation which Resulted in the Failure of the "C" ESW Pump Breaker
NCV 2006003-03, Inadequate Procedure Results in Elevated Reactor Coolant System Leakage
NCV 2006003-04, Inadequate Design Review of PRDNMS Modification Resulted in a Reactor Scram
NCV 2006003-05, Ineffective Corrective Actions to Assure Training and Qualification of Workers as Required by 10CFR50, Appendix B, Criterion XVI
NCV 2006004-01, Inadequate Risk Assessment
NCV 2006005-01, Inadequate Work Instructions for the Disassembly and Inspection of Check Valves
NCV 2006005-02, Inadequate Evaluation of EPA Breaker Failures
NCV 2006006-01, Failure to Identify Scaffolding that Affected the Safety-Related RHR Discharge Pressure Instrument Tubing Input to ADS
NCV 2006009-01, Safeguards Information
Licensee Identified NCV 2007002, U2 Div II Core Spray Pump Room (a High Radiation Area) Was Not Posted and Was Open
Licensee Identified NCV 2007002, U1 HPCI Failed a Surveillance Due to the Failure to Perform Preventive Maintenance
NCV 2007003-01, Failure to Take Timely Corrective Actions for an "E" EDG Jacket Water Leak
FIN 2007003-02, Failure to Maintain Occupational Radiation Exposure ALARA during Reactor Water Cleanup Pipe Replacement Activities
FIN 2007003-03, Failure to Maintain Occupational Radiation Exposure ALARA during Outage ISI of Reactor Pressure Vessel
NCV 2007003-04, Violation of 10CFR71.5 for Inadequately Secured Transport of Condensate Pump Motors
NCV 2007003-05, Violation of 10CFR71.5 for Inadequately Accounting for Activity in a Shipment of Irradiated Fuel Channels
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LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
AR	Action Request
BWROG	Boiling Water Reactor Owners' Group
CAP	Corrective Action Program
CAQ	Condition Adverse to Quality
CARB	Corrective Action Review Board
CFR	Code of Federal Regulations
CPG	Central Procedure Group
CR	Condition Report
CS	Core Spray
DBA	Design Basis Accident
DCP	Design Change Package
ECCS	Emergency Core Cooling System
ECP	Employee Concerns Program
EOP	Emergency Operating Procedures
EPG/SAG	Emergency Procedure Guidelines / Severe Accident Guidelines
EPU	Extended Power Uprate
FSAR	Final Safety Analysis Report
IMC	NRC Inspection Manual Chapter
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PAM	Post-Accident Monitoring
PI&R	Problem Identification and Resolution
psig	pounds per square inch
PSTG	Plant Specific Technical Guidelines
QA	Quality Assurance
RCA	Root Cause Analysis
RHR	Residual Heat Removal
ROP	Reactor Oversight Program
RPV	Reactor Pressure Vessel
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
TS	Technical Specifications