May 21, 2001

Mr. Robert G. Byram
Senior Vice President and
Chief Nuclear Officer
PPL Susquehanna, LLC
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, Pennsylvania 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION
NRC INSPECTION REPORT 05000387/2001-004, 05000388/2001-004

Dear Mr. Byram:

On March 2, 2001, the NRC completed a team inspection of the Susquehanna Steam Electric Station, Units 1 and 2. The enclosed report documents the inspection findings that were discussed on March 2, 2001, and April 6, 2001, with Messrs. G. Jones and B. Shriver, and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the team identified four issues of very low safety significance (Green). Three of the four issues were determined to involve violations of NRC requirements. However, because of their very low safety significance, and because they were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Station.

In accordance with 10CFR2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000387, 05000388
License Nos. NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2001-004, 05000388/2001-004

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	05000387, 05000388
License Nos:	NPF-14, NPF-22
Report No:	05000387/2001-004, 05000388/2001-004
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Post Office Box 35 Berwick, PA 18603
Dates:	February 12 - March 2, 2001
Inspectors:	A. Della Greca, Senior Reactor Inspector, Team Leader, DRS F. Arner, Reactor Inspector, DRS J. Carrasco, Reactor Inspector, DRS (part time) S. Chaudhary, Senior Reactor Inspector, DRS M. Maley, Reactor Inspector (Trainee) , DRS J. Richmond, Resident Inspector, DRP J. Leivo, Engineering Contractor
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000387-01-04 and 05000388-01-04, on 2/12-3/2/2001; PPL Susquehanna, LLC; Susquehanna Steam Electric Station, Units 1 & 2; safety system design & performance capability.

The inspection was conducted by region-based inspectors, a resident inspector, and a contractor. The inspection identified four Green findings, three of which were non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The team found that the reactor core isolation cooling (RCIC) pump in-service test acceptance criteria was non-conservative because it would not have ensured that the system design function was maintained. This was determined to be of very low risk significance (Green) by the significance determination process (SDP) phase 1 screening because an actual loss of the system safety function had not occurred. The failure to establish adequate test acceptance criteria for the RCIC pump was considered a non-cited violation of 10CFR50, Appendix B, Criterion XI, Test Control. (Section 1R2.b.1)

- Green. The team determined that during an anticipated transient without scram (ATWS) loss-of-offsite power (LOOP) scenario, the standby liquid control (SLC) system would not satisfy the ATWS rule requirement in that one or both SLC pump relief valves could lift and inhibit full flow to the reactor. The team determined this issue to be of very low risk significance (Green) through phase I of the SDP screening process. This conclusion was based upon the fact that the licensee’s engineering assessment determined that, even with both relief valves lifting, sufficient boron solution would be injected in the reactor to maintain the integrity of the fuel, reactor pressure vessel, and containment barriers. In addition, the event itself has a low occurrence probability. The failure by the licensee to ensure that the required ATWS rule flow rate would be injected into the reactor vessel was considered a non-cited violation of 10CFR50.62. (Section 1R21.b.2)

- Green. The team found that the licensee had neither tested nor developed test procedures that verified the ability of the safety-related motor operated valve (MOV) thermal overloads (TOLs) bypass relay contacts to perform their bypass function. The team also found that the TOL setpoints had not been selected to ensure that the valves would be able to perform their safety-related function. The team determined that the licensee’s failure to test the TOL bypasses periodically was of very low risk significance (Green) by the SDP screening process. This determination was based upon the results of a subsequent PPL analysis and test conclusions that the current sizing of all TOLs provided reasonable assurance of operability of the affected valves. The failure by the licensee to verify the integrity of the bypass circuit was considered an additional example of a non-cited violation of 10CFR50, Appendix B, Criterion XI, Test Control. (Section 1R21.b.3)

- Green. The team found that the as-found relief setpoints of the Unit 1 SLC pump discharge valves were outside the specified tolerance 20 of the 27 times the in-service test (IST) of the valves was conducted. The failure rate of the Unit 2 valves was similarly high. The team determined that the recurrence of the setpoint drift outside the valve performance criteria was of very low risk significance (Green) by the SDP screening process. This conclusion was based on the fact that the licensee's engineering assessment determined that even with both relief valves lifting, sufficient boron solution would be injected in the reactor to maintain the integrity of the RPV, fuel, and containment barriers. (Section 1R21.b.4)

Report Details

1. REACTOR SAFETY

Cornerstone: Mitigating Systems

1R02 Evaluations of Changes, Tests, or Experiments (IP71111.02)

a. Inspection Scope

The team reviewed selected safety evaluations (SEs) performed by PPL. The SEs were selected from a list of changes relating to the SLC and RCIC systems and from other plant changes implemented during the last year. The selection took into consideration the safety significance of the change, the risk to the structures, systems, and components affected, and the impact on the three reactor safety cornerstones (initiating events, mitigating systems, and barrier integrity).

The review was conducted to verify that the changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and test and experiments not described in the UFSAR, were reviewed and documented by the licensee in accordance with 10CFR50.59. The review also verified that any safety issues pertinent to the changes, tests and experiments had been properly resolved. Additionally, the team verified that the changes, tests, and experiments did not require prior NRC approval or a license amendment. The team conducted discussions with cognizant engineering personnel, as required, and evaluated supporting technical information, including calculations, analyses, and design requirements.

The inspectors also reviewed a sample of changes, tests and experiments for which PPL determined that a safety evaluation was not required. This review was performed to verify that PPL's threshold for performing safety evaluations was consistent with the requirements of 10CFR50.59. Lastly, the team verified that the problems identified with the implementation of the safety evaluation program were entered into the corrective action program.

b. Findings

No findings of significance were identified

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team selected the reactor core isolation cooling (RCIC) and the standby liquid control (SLC) systems for its review of the design and performance capability of safety systems at the Susquehanna Steam Electric Station. The two systems were selected because of their risk significance in event mitigation and core damage prevention. The primary function of the RCIC system is to provide makeup water to the reactor vessel during shutdown and isolation from the feedwater system. The primary function of the SLC system is to inject a neutron absorbing solution into the reactor and shut it down in the event that the control rods do not insert following an anticipated plant transient. The Inspection Procedure used for this effort was IP 71111, Attachment 21.

The team reviewed selected portions of the SLC and RCIC design and licensing basis documents, including applicable sections of the Updated Final Safety Analysis Report (UFSAR), the plant Technical Specifications (TS) and the design basis documents (DBDs). This review was performed to determine whether the system and component functional requirements during normal, abnormal and accident conditions were being met. The review also verified that: (1) the system design bases were in accordance with the licensing commitments and regulatory requirements; and (2) the design documents, such as drawings and design calculations, were correct. The documents reviewed included engineering analyses, calculations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument set points.

For selected mechanical and electrical calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used, and that there were adequate technical bases to support the conclusions. When appropriate, the team performed independent calculations to evaluate the document adequacy. For selected plant modifications, the team verified that the ability of the systems to perform their design functions was not adversely affected by the change.

The team also reviewed selected portions of the UFSAR, plant TS, and design documents of interfacing systems, including main steam and electrical power supplies. For these systems, the team examined piping and instrumentation drawings, electrical schematics, and configuration baseline documents and assessed the capability of the supporting systems to satisfy the design functions of the SLC and RCIC systems.

The team reviewed various documents and plant procedures to verify that the RCIC and SLC systems were operated and maintained consistent with the design and licensing bases. The operational readiness and material condition of the selected systems were assessed by reviewing appropriate documents, including operating procedures, operator logs, component maintenance history records, preventive maintenance records, surveillance test procedures and results, calibration records, and system health reports. The review also included applicable portions of the emergency operating procedures, vendor documents, portions of the design-bases documents that discussed the operation and maintenance of the systems, and selected event reports related to corrective maintenance and operation of the systems. The team also interviewed responsible PPL personnel, including licensed and non-licensed operators, the system engineer, and maintenance and instrumentation and control personnel, regarding the operation and performance of the selected systems and components.

Plant walkdowns of the SLC and RCIC systems were performed to verify that the physical installation of the systems and components was consistent with design bases document assumptions, design drawings and installation specifications. During these walkdowns the team examined the design and condition of major components, including pumps, turbines, and valves. The team also evaluated piping and pipe supports, system instrumentation, valve positions, applicable portions of AC and DC electrical switchgear, DC batteries, heat tracing, operator aids, area heating and ventilation systems, and storage of transient equipment and combustibles.

Finally, the team reviewed the licensee's effectiveness in identifying problems associated with the standby liquid control and the reactor core isolation cooling systems. The team also reviewed a sample of event reports related to the selected systems to evaluate the adequacy and timeliness of the corrective actions resulting from the identified problems. For selected event reports the team reviewed the adequacy of the operability determinations and verified the completion of the corrective actions.

b. Findings

.1 Reactor Core Isolation Cooling System (RCIC)

The team found that the RCIC pump inservice test (IST) acceptance criteria, included in attachment A of quarterly surveillance test procedure SO-150 (250)-002, could have allowed the pump performance to degrade well below the condition where the system was still capable of supporting its design function. NRC information notice 97-90, "Use Of Non-Conservative Acceptance Criteria In Safety Related Pump Surveillance Tests," had already alerted licensees of this potential deficiency in test acceptance criteria.

To ensure the system could still perform its design function, PPL performed an operability assessment. In a preliminary calculation, PPL estimated that the latest system test performance data showed the RCIC system was capable of achieving the design flow-rate of 600 gallons per minute (gpm) against the first main steam safety relief valve setting plus 1% tolerance. The team performed an independent calculation and determined that the current performance of the RCIC pumps and existing turbine speed limiter settings was adequate to ensure the system design function would be met. The team determined that a 2% degradation from the current level of pump performance could prevent the Unit 1 RCIC system from achieving its design flow rate. However, existing surveillance test procedure acceptance criteria was non-conservative because it allowed pump performance degradation up to a nominal 10% from vendor curve test performance. This degradation level would result in the inability of the RCIC system to achieve the design flow-rate of 600 gpm against the first main steam safety valve setting. Therefore, the team concluded that corrective actions were required to ensure that the combination of turbine control system maximum speed setting, main steam safety relief valve settings, and allowable pump surveillance test acceptance criteria, would continue to ensure the capability of the system to achieve its design function.

This issue was considered to be more than minor because the failure to have adequate acceptance criteria had a credible impact on safety. The existing test criteria did not ensure that the RCIC system, a mitigating system, would be capable of performing its safety function at all times. Because the inadequate test criteria had not resulted in an actual loss of this mitigating system safety function, the issue was determined to be of very low significance (Green) and was screened out in phase I of the significant determination process (SDP). The inadequate RCIC system test acceptance criteria was contrary to 10 CFR50, Appendix B, Criterion XI, "Test Control," which requires tests be performed in accordance with written procedures which incorporate the requirements and acceptance limits contained in applicable design documents. PPL entered this issue into their corrective action program and issued condition report (CR) 315019. PPL planned to develop test criteria that would ensure that both the design and IST requirements would be satisfied. Due to the overall low risk significance, this violation of 10CFR50, Appendix B, Criterion XI, was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000387; 05000388/2001-004-01)**

During the inspection of this issue, the team determined that PPL had prepared calculations and was planning to request a license amendment to increase the SRV relief setting tolerance from 1% to 3%. As a result of the teams finding, the licensee delayed their plans until they are able to evaluate the issue and develop the required actions. PPL initiated CR 313946 for this effort.

.2 Standby Liquid Control (SLC) System

The team found that inadequate design margin existed with the standby liquid control pump discharge relief valve settings. This resulted in the system not capable of meeting the equivalent flow rate required by 10 CFR 50.62, “Anticipated Transient Without Scram (ATWS).

Background

The control rod drive (CRD) system provides the primary means to control reactivity, as required by 10CFR50, Appendix A. The standby liquid control system was part of the original plant design and provided an independent and diverse (from the CRD system) method for shutting down the reactor. Its specific purpose was to provide shutdown capability, particularly in the event of an anticipated transient without scram (ATWS), i.e., in the event of an anticipated operational occurrence that requires reactor shutdown, but is not followed by an insertion of all the control rods. The SLC system does so by injecting into the reactor a neutron absorbing solution that is capable of achieving and maintaining sub-criticality. The system included two redundant pumps, each capable of performing the design function.

In 1984 the NRC issued the ATWS Rule that added more stringent injection rate requirements to the system. Specifically, paragraph (c) (4) of 10CFR50.62 requires, in part, that each boiling water reactor must have a standby liquid control system with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate that the resulting reactivity control was at least equivalent to that resulting from the injection of 86 gallons per minute (gpm) of 13 weight percent sodium pentaborate decahydrate (boron) solution.

PPL Compliance with the ATWS Rule

In calculation M-SLC-002, Revision 0, “Boron Concentration To Meet ATWS Rule,” PPL determined that, to meet the ATWS rule, the flow-rate should be 82.4 gpm and that the concentration of the sodium pentaborate solution should be at least 13.6 weight percent. This boron concentration became the licensing basis and was included in section 3.1.7 of the Susquehanna technical specifications. The formula used by the licensee to derive the required flow rate and boron concentration was included in General Electric’s topical report NEDE-31096-P, “Anticipated Transient Without Scram; Response To ATWS Rule, 10CFR50.62.” This report was reviewed by the NRC and the results were included in an appropriate safety evaluation report. To achieve the desired flow-rate, PPL implemented a modification that revised the individual SLC pump start logic to a simultaneous start of both pumps.

The team’s review of the system design documents determined that the licensee’s decision, in 1985, to operate both pumps concurrently caused the system pressure losses in the pump discharge lines to increase significantly. These losses were the result of the increased fluid velocity in the common injection line, as the flow rate doubled from 41.2 gpm to 82.4 gpm. In calculation M-SLC-004, dated September of 1987, “Calculation Of Maximum ATWS Injection Pressure,” PPL determined that the maximum discharge pressure at the SLC pumps was 1276 psig. This value was based on the setpoint of the first SRVs (1076 psig), the system friction losses for two pump operation, and the elevation losses. In 1992 and 1993 the licensee revised the calculation to incorporate the effects of a power uprate. The licensee found that the revised discharge pressure was 1319 psig. The increase in required pump discharge pressure was due to a 30 psig increase in the relief setting of the first SRV setpoint, which was changed from 1076 psig to 1106 psig, and to an increase in calculated core flow value. The increase in required pump discharge pressure also required PPL to increase the setting of the SLC pumps discharge relief valve up to 1400 psig - 0% + 3%. The relief valves were intended to protect the system from over-pressure. Their setpoints, however, must be sufficiently high to prevent their opening during system pressure pulsations due to the pump action.

Design Evaluation

During the review of a vendor report, EC-PUPC-1009, Revision 0, “Evaluation Of Susquehanna ATWS Performance For Power Uprate Conditions,” accepted by PPL on May 16, 1994, the team identified discrepancies in the maximum expected vessel dome pressures that were assumed in the calculations of record and in the results of the analysis for two specific ATWS transients. Specifically, the team found that for a main steam isolation valve (MSIV) closure transient, the analysis indicated that, at the time the SLC system would be manually initiated, the reactor vessel dome pressures were as high as 1133 psig. Similarly, for a loss of offsite power (LOOP) transient, the dome pressure at various times during the event were on the order of 1200 psig. The much higher pressure calculated for the LOOP transient event is due to the loss of power to the containment instrument gas compressors and the resulting loss of gas required to open the SRVs. Although each SRV is equipped with a gas accumulator, the amount of gas available in each accumulator is sufficient for only a few actuations (one actuation according to the UFSAR).

Based on the above, the team concluded that the maximum reactor vessel pressure of 1106 psig assumed by the licensee in the pump discharge pressure calculation was non-conservative. The team further concluded that the SLC pump discharge relief valves could lift at least during one of the transient scenarios, the loss of offsite power. The lifting of SLC pump discharge relief valves would cause the sodium pentaborate solution to be recycled back to the pump suction and, therefore, prevent the system from meeting the equivalent flow capacity required by 10CFR50.62.

As a result of the team's observations PPL initiated CR 316780. In the CR the licensee characterized the issue as a non-compliance and attributed the condition to "an oversight in the performance of the power uprate ATWS analysis and also in the design of the SLC system." Subsequently, following the inspection, on March 20, 2001, PPL completed and issued Revision 0 of a safety assessment that evaluated the condition "in accordance with guidance provided in [Generic Letter] GL 91-18..." This document, titled "Safety Assessment For Standby Liquid Control Injection During An ATWS With Loss of Normal AC Power Event," will be included in the Safety Significance section of the CR and will provide some of the bases for the resulting conclusions and corrective actions.

The assessment addressed various issues, including the Susquehanna SLC system design evolution, safety function of the system, availability of redundant equipment, potential compensatory actions, and impact of the current configuration on core damage frequency. The licensee also conducted simulation analyses to evaluate the impact on solution flow to the reactor in the event that the relief valve of one or both pumps lifted. Based on the results of these analyses, the licensee concluded that, "There is reasonable assurance that the SLCS will be able to inject into the reactor pressure vessel (RPV) in response to an ATWS/LOOP event and bring the RPV into a hot shutdown condition, while maintaining RPV integrity, fuel integrity and containment integrity."

The team's review of the licensee's March 20, 2001, assessment found their approach acceptable, but disagreed with their conclusions, in section 6.1, page 14, that "...the SLC pump discharge relief valves would not lift during an ATWS/LOOP event." PPL's conclusions were based on their calculation that the SLC pump discharge pressure, with two pumps operating, would be 1395 psig and on the assumption that the relief valve setpoint was in the as left setting of 1410-1420 psig. The team's disagreement was based on the following:

- In the pump discharge calculation the licensee failed to include the current 1% tolerance (12 psig) in the setting of the SRV relief setpoint and a margin for pressure pulsations that are typical with a positive displacement pump. As indicated in section 3.5.3 of PPL's assessment, General Electric originally had applied a margin of 124 psig, but reduced it to 75 psig during the power uprate. This new value was based on 30 psig identified in the ATWS topical report and 3% (42 psig) inaccuracy in the setting of the relief valve setpoint. If these two values (12 psig and 30 psig) are added to the PPL calculated maximum pump discharge pressure, the new maximum anticipated pressure would be 1437 psig and, therefore, higher than the as-left relief setpoint of the pump relief valves.

- As shown in the table on page 16 of PPL's assessment, historically the as-found setpoint of the "A" relief valve of both Units was considerably lower than the as-left setting.

Based on the above, the team concluded that, under the postulated ATWS/LOOP conditions, the probability of at least one valve lifting was high. The team also concluded that, under the same conditions, the requirements of the ATWS Rule were not being met. The ATWS Rule requires "...that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution." Contrary to this, during ATWS/LOOP conditions, the SLC flow-rate would not have achieved this requirement. Therefore, the identified plant configuration was in violation of 10CFR50.62. This issue was considered to be more than minor because the failure of the SLC system, a mitigating system, had a credible impact on safety. The failure of the system to achieve the flow rates specified in the ATWS Rule could reduce its ability to perform its accident mitigating function and result in significant damage to the fuel, RPV, and containment protective barriers. PPL entered this issue into their corrective action program and issued CR 316780 and planned to develop appropriate actions to correct the deficiency.

The team determined this issue to be of very low risk significance (Green) through the phase I significance determination process. This conclusion was based on the fact that the licensee's assessment determined that, even with both relief valves lifting, sufficient boron solution would be injected in the reactor to maintain the integrity of all three barriers. Thus, the issue was a design deficiency which did not affect the operability of the mitigating SLC system and was screened out green. In addition, the event itself, i.e., an ATWS concurrent with a LOOP, had a low occurrence probability of approximately 1E-7. Due to the overall low risk significance, this violation of 10CFR50.62 was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368).
(NCV 05000387; 05000388/2001-004-02)

Section 3.1.7 of the Technical Specifications (TS) requires both SLC subsystems to be operable in plant operating Modes 1 and 2. The section also describes the conditions for operability, the actions required if the operability conditions are not met, and the time allotted to restore the system to operability. Condition A of the TS requires that if the concentration of the sodium pentaborate (boron) in solution is less than 13.6 weight percent concentration must be restored in 72 hours. This concentration was based on the requirements of 10CFR50.62 and the ability of the system to inject boron into the reactor at the rate of 82.4 gallons per minute. The actions required by the TS for Conditions B (one SLC subsystem inoperable) and C (two SLC subsystems inoperable) are needed to ensure that the required flow rates are achieved.

In addressing the team's finding regarding the ATWS/LOOP condition described above, the licensee maintained that the requirements of 10CFR50.62 were beyond the design basis of the plant. For instance, in CR 316780, under Event Information, the licensee stated, "Hence, SLC injection is not assured under these specific Beyond Design Basis Conditions." As a result of this understanding, under the Operability section of the CR, the licensee considered the system operable stating that the SLC system "...is capable of performing it's Tech. Spec. design basis function...," and, under Reportability Determination section, the licensee considered the issue not reportable stating that, "...this condition does not require immediate or prompt reportability. The condition describes a 'beyond design basis' concern... SLC is fully capable of performing its Tech. Spec. design basis function and is operable." Therefore, PPL determined that the actions specified in TS 3.1.7 were not applicable and did not report the issue in accordance with 10CFR50.72 or 50.73.

Further review of PPL's position regarding the design and licensing bases of the SLC system was required at the end of the inspection period. This review will focus on PPL's conclusion regarding the functional requirement of the SLC system with respect to the ATWS rule, the applicability of TS section 3.1.7 to the ATWS rule, and on PPL's decision not to report the failure to meet the requirements of the ATWS rule.
(URI 05000387; 05000388/2001-004-03)

Following the inspection, on March 26, 2001, the licensee informed the team that they had initiated action to modify the design of the SLC system of Unit 2, currently in a refueling shutdown mode. The modification, No 318267, will change the flange of both pumps with higher rated ones (from 1400 psig to 1500 psig) and will increase the setting of the relief valves to 1500 psig. The licensee intends to perform the same modification, No. 318280, on the Unit 1 system during an upcoming outage of sufficient duration.

.3 <u>Thermal Overload Bypass Circuits for Motor Operated Valves</u>

The team found that PPL failed to periodically test the TOL bypass circuits of safety related motor operated valves (MOVs) as specified by regulatory guide 1.106 and stipulated in UFSAR section 8.1.

Section C of Regulatory Guide (RG) 1.106, Revision 1, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," states that, to ensure that safety-related valves will perform their function, licensees should implement either of the two following regulatory positions: (1) bypass the thermal overloads (TOLs), or (2) establish the trip setpoint of the TOL device with all uncertainties resolved in favor of completing the safety-related function. For bypass schemes, the RG indicates that: (a) the TOLs could either be continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or (b) the TOLs could be normally in force and bypassed only under accident conditions. The RG further states that, "The bypass initiation system circuitry should conform to the criteria of Sections 4.1, 4.2, 4.3, 4.4, 4.5, 4.10, and 4.13 of IEEE Std 279-1971, 'Criteria for Protection Systems for Nuclear Power Generating Stations,' and should be periodically tested."

PPL opted to use a continuous bypass scheme as described by (a) above. Specifically, the bypass circuit used by PPL consists of a normally-closed (NC) relay contact that is in parallel with the normally-closed TOL contact. When the MOV is being tested, the TOL is temporarily placed in force by removing the bypass. The bypass removal is accomplished by the operator who manipulates an appropriate selector switch in the control room, energizes the associated relay, and opens the NC bypass contact. The switch is returned to its normal position after completion of the testing.

The team's review of the valves within the scope of the inspection determined that the licensee had never tested nor developed test procedures that verified the ability of the contacts to perform their bypass function, as stipulated in RG 1.106. In section 8.1 of the UFSAR the licensee stated that they complied with the RG 1.106, Revision 1, recommendations. The team further determined that this deficiency affected all safety-related MOVs of both Units.

Discussions with responsible licensee personnel indicated that operators relied on the status of indicating lights to determine the status of the TOL bypass condition. Specifically, the operators confirm restoration of the bypass circuit by confirming that a status light, energized by a normally-open (NO) contact on the same bypass relay during the MOV testing, extinguishes when the test is complete and the selector switch returned to its normal position. The team noted that the verification process used by the operators was acceptable for confirming that the bypass switch had been returned to its normal position but was inadequate for assuring contact status since, (i) the light used a different contact on a different pole of a multi-pole (four or eight-pole) relay; (ii) the light contact was normally-open and closed on relay energization, whereas the bypass contact was normally-closed and opened on energization; and (iii) individual contacts could have developed high resistance without being observed by the licensee.

The significance of item (ii), above, was that a degraded relay spring potentially could prevent a NC contact from staying closed but typically would not prevent a NO contact from closing during relay energization. Conversely, a NC contact that was not fully closed would open further during energization. Regarding high contact resistance (item (iii) above), the team noted that the auxiliary relay does not use wiping action but relies on spring pressure to maintain contact closure.

The team considered this issue to be more than minor because the failure of a bypass circuit could have credible impact on safety. Specifically, the failure could prevent the affected valve and the associated safety-related system from performing its mitigating and/or isolation function. Because the licensee relied on the bypass circuits to remain closed and to ensure that the safety-related valves would perform their safety function, they had sized the TOLs to protect the motor during testing rather than to assure completion of the MOV safety function under dynamic accident conditions. With undersized TOLs, an undetected failure of the bypass circuit could result in the MOV motion stopping prematurely and prevent the affected MOV from performing its safety function.

To address the team's finding, the licensee issued CR 314020 and evaluated the TOL size of all MOVs with the TOL bypass feature. For this evaluation, the licensee used the guidance of IEEE Standard 741-1990, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," Appendix B, "Guidelines for Selection of Overload Protection for Valve Actuator Motor Circuits." Where the TOL size did not meet the IEEE Standard guidelines, the licensee used an alternate methodology that used calculated dynamic torque based on the specific valve required thrust. The licensee reported that, except for the TOL of the steam supply valve to the Unit 1 RCIC turbine (HV 150F045), their immediate evaluation provided reasonable assurance that, without TOL bypass in place, the MOVs would complete their accident mitigating function. The team evaluated the review criteria used by the licensee and the results of the analysis and found them reasonable. For HV 150F045, PPL conducted field measurements of the bypass relay contact resistance and found that the contact was closed per design.

The licensee's failure to periodically test the TOL bypass circuits of safety-related valves as specified by RG 1.106 and stipulated in UFSAR section 8.1 was contrary to 10CFR50, Appendix B, Criterion XI, "Test Control," which requires "...written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." The need and the basis for testing safety-related logic circuits was also discussed in NRC Generic Letter 96-01, dated January 10, 1996. As stated previously, PPL entered this issue into their corrective action program (CR 314020) and planned to develop appropriate actions to ensure availability of affected valves.

The team's assessment of the finding through phase I of the significance determination process concluded that the issue was of very low significance (Green). This conclusion was based on the results of the licensee's analysis and conclusion that current sizing of TOLs provided reasonable assurance of operability of the affected valves in the mitigating systems. The issue did not represent an actual loss of a mitigating system safety function. Due to the overall low risk significance, this violation of 10CFR50, Appendix B, Criterion XI, was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000387; 05000388/2001-004-04)**

.4 Operations, Maintenance, and Testing

The team determined that PPL had failed to take adequate corrective action to preclude the recurrence of SLC pump discharge pressure relief valve setpoint drift.

PPL conducts ASME inservice test (IST) of the SLC pump discharge pressure relief valves, PSV-1(2)48-F029A(B) every 24-months. Based on a table provided by PPL, the team found that, during the last 20 years, the as-found relief setpoint of the Unit 1 valves was outside the specified tolerance 20 of the 27 times the IST was conducted. The failure rate of the Unit 2 valves was similarly high. As specified in the PSV test procedure, the current valve relief setpoint should be 1400 psig, with a tolerance of - 0 psig to +42 psig. The as-found data indicated that the relief valves generally lifted well outside the set limits. Occasionally, the relief setpoint was found to be as low as 800 psig and as high as 1600 psig.

The team's review of selected condition reports and work orders determined that PPL's corrective actions had not been adequate to prevent recurrence. PPL's cause determination, when documented, typically identified the condition as an expected setpoint drift. Therefore, the corrective actions were generally limited to cleaning the valve seat and re-adjusting the lift setpoint.

The team determined that this issue was more than minor because the failure of the relief valves to perform within the specified design limits had a credible impact on safety. A drift of the setpoint in either direction could prevent the SLC system from performing its accident mitigating function. Specifically, a drift of the setpoint in the positive direction could have resulted in an inadequate protection of the system integrity (the SLC system design pressure rating was 1400 psig); a drift of the setpoint in the negative direction could have resulted in the associated relief valve lifting early and prevented the pumps from injecting the required amount of boron into the reactor.

The team determined that this issue was of very low safety significance (Green) and screened out in phase I of the significance determination process. This conclusion was based on the fact that, as stated in section 1R21 of this report, the licensee's assessment determined that, even with both relief valves lifting, sufficient boron solution would be injected in the reactor to maintain the integrity of the RPV, fuel, and containment barriers. Therefore, the issue did not represent an actual loss of the SLC system safety function.

4. OTHER ACTIVITIES (OA)

4OA6 Meetings, Including Exit

.1 Management Meeting

On March 2, 2001, the team presented the preliminary inspection results to Messrs. G. Jones and B. Shriver, and other members of licensee management. The licensee acknowledged the inspection findings presented. On April 6, 2001, based on the review of the standby liquid control system data and engineering analysis provided by the licensee after the team left the site, the team conducted a telephone exit meeting with Messrs. G. Jones and B. Shriver, and other members of licensee management, to present the results of the additional review.

(1) SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

PPL Susquehanna, LLC (PPL)

J. Akus	Electrical Design Engineering
R. Anderson	General Manager, Operations
K. Browning	Mechanical Design Engineering
G. Butler	Supervisor, System Analysis
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D. Filchner	Licensing
G. Jones	Vice President, Engineering
D. Kapuschinsky	Operations Engineering
J. Lubinski	Electrical Design Engineering/GL-89-10 MOV Program
G. Machalich	Licensing
C. Markley	NSE Supervisor
J. Meyer	Licensing
M. Mjaatvedt	Nuclear Technology
R. Mullock	Modification Engineer
R. Pagodin	Manager, Nuclear Technology
D. Roth	Site Supervisor, Quality Assurance
R. Saccone	Manager, Operations
B. Shriver	Vice President, NSO
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A. White	Instrumentation & Control Design Engineering
A. Wrape	General Manager, Nuclear Engineering

Nuclear Regulatory Commission (NRC)

L. Doerflein	Chief, Systems Branch
S. Hansell	Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000387; 05000388/2001-004-03	URI	Inclusion of SLC Design Modifications for ATWS Rule in the Design Bases of the Plant

Opened/Closed

05000387; 05000388/2001-004-01	NCV	Non-Conservative Test Acceptance Criteria for RCIC System Surveillance
05000387; 05000388/2001-004-02	NCV	Failure to Meet ATWS Rule in the Design of SLC System
05000387; 05000388/2001-004-04	NCV	Failure to Test the Motor Operated Valve Thermal Overload Bypass Circuit

(2) LIST OF DOCUMENTS REVIEWED

Design Drawings
E-1, Single Line Diagram - Station, Rev. 25.
E-9, Sh 19 & 45, Single Line Meter & Relay Diagrams, 480 V Motor Control Centers
E-11, Sh 1 & 3, Single Line Meter & Relay Diagrams, 125 & 250 Vdc System
E-26, Sh 2, Schematic Meter & Relay Diagram, 125 Vdc Distribution Panels
E-154, Sh 3-12, 14-17, & 39, Schematic Diagram, RCIC System
E-166, Sh 1 & 2, Schematic Diagram, Standby Liquid Control System Unit 1, Rev. 20.
E-185 Sh 3, Schematic Diagram, Bypass Indication System Unit 1, Rev. 10.
M1-E51-90 Sh 3-5 & 9, Elementary Diagram, RCIC System
M-108, Sh 1 & 2, P&ID, Condensate & Refueling Water Storage
M-125 Unit 1 P&ID, Reactor Bldg. Instrument Air
M-1(2)48, Standby Liquid Control (SLC) System P&ID
M-1(2)49, Reactor Core Isolation Cooling (RCIC) System P&ID
M-1(2)50, RCIC Turbine and Pump P&ID
M-151, Unit 1 P&ID, Residual Heat Removal
FF-129010, Sh. 8101, RCIC Process Diagram
Pump Curves 31227, 31225,31224 and 31228 for RCIC
J-449, Rev. 6, Unit 1 NSSS Loop Diagram RCIC

Engineering Calculations
E-AFS-002, 480 Vac Molded Case Breaker 1B217061; 1B236081
EC-EOPC-0519, SABRE Calculations to Support Technical Basis of IPE and ATWS EOP
EC-EOPC-0505, Evaluation of SLC Boron Flow while Injecting with RCIC
EC-SBOR-0506, Station Blackout (SBO) Required Coping Duration
EC-SBOR-0501, SBO Coping Assessment
EC-RISK-0528, Risk Significant Systems, Structures, and Components for Maintenance Rule
EC-RISK-1054, SSC Availability Performance Criteria for Maintenance Rule
EC-002-0514, 125 Vdc Utilization Voltage and Load Profile for Ckt 1D614, Brkr 07, ED1, ED2
EC-004-0500, SSES Unit 1 & 2 Voltage Study, Rev. 4
EC-004-0502, Oper of Class 1E Distr System at Degraded Grid Undervoltage Relay Setpoint
EC-006-0001, Cable Sizing Calculations for 480 Vac and KW Loads and 460 Vac Motors
EC-006-0002, MCC Control Circuits Voltage Drop Calculation
EC-006-0500, 480 Vac Molded Case Breaker
EC-006-0502, 250 Vdc Breaker Settings
EC-006-0504, Attachment B1, LOCA Voltage Drop, pages 26 and 63
EC-037-0012, The Prevention Of Vortex Formation In The Condensate Storage Tank
EC-037-1002, RCIC CST Suction Low Level Transfer Allowable Value and Setpoint
EC-050-0515, T.S. Change For RCIC Surveillance Testing
EC-050-0535, RCIC Turb Stm Supply Line Break Detection Isolation setpoints for Power Uprate
EC-050-0544, Limit Switch Settings and Torque Switch Settings for HV-149F007
EC-050-0553, RCIC Steam Flow Reduction For Start Up Maximum Predicted RCIC Pump/Turbine Speed During RCIC System Startup
EC-050-0554, RCIC Surveillance Test Acceptance Criteria For High Press. Test, Power Uprate
EC-050-1002, RCIC Pump Performance And Turbine Steam Demand Models
EC-050-1005, Power Uprate Program - RCIC Electronic Overspeed Trip Setpoint SSH 15052/1.
EC-053-0002, SLC Line Pressure Drop Determination
EC-053-0503, Standby Liquid Control System Pressure Drop
EC-053-0507, Calculation Of Maximum ATWS Injection Pressure

EC-053-0508, PSV-C41-F029A,B Setpoint Tolerances
EC-053-0519, Setpoint Calculation for LSHL-C41-1N600
EC-053-1001, Determination of Design Basis for SLC Accumulators
EC-053-1002, Impact of Depressurized SLC Accumulators
EC-059-1036, Bases For ECCS & RCIC FSAR NPSH Calculations
EC-083-0540, HPCI/RCIC Piping Area Ambient Temperature Setpoints
EC-083-0603, HPCI/RCIC Piping Area Differential Temperature Setpoints
EC-083-0615, RCIC Room Differential Temperature Setpoints
EC-083-0618, Ambient Temp Switches Allowable Value, Trip Setpoint and Process Setpoint
EC-083-0631, Differential Temp Switches Allowable Value, Trip Setpoint and Process Setpoint
EC-088-0002, Cable Size for 240 Vdc Motor
EC-088-0503, Attachment B, Available Start Torque - Safety Function, pp. 17-26, 43, 44, 46-52.
EC-088-0505, Class 1E, 250 Vdc System First Minute and 4 Hour Voltage Drop Calculation EE5
EC-088-0506, Class 1E 250 Vdc Batteries - Battery Sizing and Battery Charger Sizing
EC-088-0522, 250 Vdc Batt. 1D650 Station Blackout Discharge Calculation EE1 250 Vdc Unit 1
EC-088-1005, Battery 1D650 Load Profile
EC-088-1007, Battery 2D650 Load Profile
EC-PUPC-1009, Evaluation Of Susquehanna ATWS Performance For Power Uprate Conditions
EC-THYD-1046, Bubble Transport During Standby Liquid Control Air-Sparge Operation
EC-VALV-0570, Design Basis For Priority 2 MOVs (Selected RCIC Valves)
ICC-PSL-E51-1N006, RCIC Pump Suction Pressure Low

<u>Design Basis Documents</u>
BWROG EPGs/SAGs, Appendix B,
DBD-041, Design Basis Document for RCIC System
DBD-042, Design Basis Document for SLC System
GDS-08, PPL Design Standard for SBO
NPE-89-004, PPL Technical Report for SBO Coping Assessment
NRC SER, dated 10-18-88, PPL Compliance with ATWS Rule
NRC SER, dated 9-12-88, BWROG EOP Guidelines, Revision 4, NEDO-31331
PLA-4308, PPL Response to NRC RAI on Deviations from BWROG EOP Guidelines
General Electric Document 22A1354AY, Data Sheet, Reactor Core Isolation Cooling System

<u>Engineering and Design Change Requests</u>
DCP 85-3098B, ATWS-SBLC Control Circuit Modification, 9/18/88.
DCP 98-9050, RCIC Topaz Inverter (ES-14901) Replacement, 10/21/99.
ECO 97-6027, Heat Damaged Cables (RCIC, MSIVs), 5/12/97.
PCWO 104354, Perform Replacement of RCIC Pump Suction Pressure Switch, 8/3/2000.

<u>Condition Reports</u>
CR 96-0686, Unit 1 "A" SLC Pump Failed Quarterly Flow Test
CR 97-0162, Excessive Actions Required to Cope with Loss of Condensate Transfer System
CR 98-0611, Review of Test Data Form 1980 to present for Unit 1 and 2 SLC PSVs
CR 98-1488, PSV-148-F029A and B Failed Pressure Test
CR 245526, PSV-148-F029B Failed Pressure Test
CR 250859, PSV-148-F029A and B Failed Pressure Test
CR 251120, PSV-148-F029B Failed Pressure Test

CR 313929, Discrepancy In FSAR Table 9.3-11, SLC System Operating Pressure
CR 313946, Calculation EC-050–0554 Does Not Adequately Document The Basis For The RCIC High Pressure Surveillance Test
CR 314020, Testing of motor operated valve thermal overload bypass circuit
CR 314056, Setpoint Of PCV RCIC Cooling Water Control Valve, PCV F015
CR 314062, FSAR and DBD errors re: electrical separation and single failure of SLC circuits
CR 314065, PCV Setting For SLC Sparging Air
CR 315634, Low setting of RCIC steam supply valve HV 150F045 thermal overload device
CR 315019, RCIC Pumps (1/2P203) May Not Be Able To Achieve The Design Developed Head As Stated In The FSAR.
CR 316042, Unit 1 RCIC Turbine Exhaust Pressure Greater Than Design Value
CR 316043, Operator Training Module Identifies Wrong Pressure for RCIC Cooling Water Valve
CR 316201, Unit 2 RCIC Steam Admission Valve (F045) Initial Opening Does Not Conform to FSAR Description

Station Procedures

AR-107-001, CRD, SLC, and Drywell Sump Alarm Response Instruction
AR-108-001, RCIC Alarm Response Instruction
CL-150-0011, Unit 1 RCIC Electrical Check-off List
CL-150-0012, Unit 1 RCIC Mechanical Check-off List
CL-150-0013, Unit 1 RCIC Containment Check-off List
CL-153-0011, Unit 1 SLC Electrical Check-off List
CL-153-0012, Unit 1 SLC Mechanical Check-off List
CL-153-0013, Unit 1 SLC Containment Check-off List
EO-000-113, Level/Power Control Procedure
EO-100-033, RCIC Operating Guidelines during SBO
EO-100-103, EOP Flow Chart for Primary Containment Control
EO-100-113, EOP Flow Chart for Level/Power Control
ES-1(2)50-001, RCIC Turbine Isolation and Trip Bypass
ES-1(2)50-002, Boron Injection with RCIC
IC-150-001, RCIC Turbine Control System Calibration
JPM-50-EO-003-101, Bypass of RCIC Temperature Isolation Signals for ES-150-001
JPM-50-EO-003-102, Bypass of RCIC High Turbine Exhaust Trip for ES-150-001
JPM-50-EO-005-102, Connect SLC Storage Tank to RCIC for ES-150-002
MT-153-001, Unit 1 SBLC Explosive Valve Removal and Replacement
MT-153-003, SBLC Accumulator Maintenance
MT-GE-008, 480 Volt and Under Circuit Breaker High Current Testing
MT-GM-005, Safety/Relief Valve Setting
OP-150-001, RCIC System Operating Instruction
OP-153-001, Standby Liquid Control System
SE-153-301, Standby Liquid Control System Functional Leak Test
SI-150-311, Quarterly Calibration of RCIC System Turbine Exhaust Diaphragm High Pressure Channels PSH-E51-1N012A, B, C, D
SI-150-315, Quarterly Calibration of Condensate Storage Tank Low Level Channels LSL-E51-1N035A & E.

LIST OF DOCUMENTS REVIEWED (Cont.)

SI-150-317, Quarterly Calibration of RCIC Steam Leak Detection Logic A (Div. 1) Temperature and Differential Temperature Channels
SO-150-002, Quarterly RCIC Flow Verification/Completed Test 1-31-01
SO-150-005, 24 Month RCIC Flow Verification
SO-1(2)53-002, 24 Month SLC Initiation and Injection Demonstration
SO-153-004, Quarterly SLC Pumps Flow Verification
SO-250-002, Quarterly RCIC Flow Verification/Completed Test 2-10-01
TP-149-079, Unit 1A RHR Heat Exchanger Performance Test
TP-150-004, RCIC Turbine Overspeed Trip Test
TP-153-007, SLC Two Pump Injection into reactor Vessel, for ATWS Modification

Safety Evaluations
NL-00-005, Unit 1A RHR Heat Exchanger Performance Test
NL-00-011 ON-155-007 - Loss of CRD System flow
NL-99-005, FSAR Section 8.3.2.1.1.5 Change
NL-99-014, LDCN 2817 - Standby Gas Treatment System
NL-99-015, LDC 2960 - Change out of Control Rod Drive Handling System
NL-99-037, Residual Heat Removal Service Water RHRSW Emergency Service Water ESW Corrosion Monitoring
DCP 214699, Diesel Generator Building Fire detection System Upgrade
DCP 221598, Mitigation of Recirc. Pump Induced Vibration for SPDCB220-1
DCP 221778, Unit 2 CRD Pump Discharge Stop Check Valve
DCP 97-9053, Local Keep Fill Pressure Indication on HPCI Pump Discharge
99-3033E, Suppression Pool level Div. I
208986, Unit 1 Zone I Secondary Containment Damper Solenoid Valve Replacement

50.59 Screens
195287, Motor for Reactor Building - Chilled Water Recirc. Pump
212044, Card two pieces for mother board and daughter board
217744, Male and Female Rod ends
226711, Battery cells, 125 vdc
225390, Valve, pressure relief
232921, Valve, solenoids, three way, two positions
238160, A gasket, graphite pressure seal for anchor darling 4" 900# c. s. check valves
246910, Stem, piece two, ASTM a 276-410H for a gate valve
96-0113, R.R. Pump Motor (R.R. 98-031)
248280, R.R. motor lead assembly, SIL 484
252026, Stud, Bonnet for 20" pacific gates valves
264987, Differential pressure transmitters
276335, Levering a device w/guide tube (after 1982)

Miscellaneous
Quarterly System Health Report, RCIC and SLC Systems
Maintenance Rule Basis Document, RCIC and SLC Systems
Operator Work-Around Log
Inservice Test Trend Data for PSV-1(2)48-F029A,B
Inservice Test Trend Data for SLC Flow from SO-1(2)53-004
Specification E-1012 Table 1-2, Physical Separation Required Between Separation Groups
Instrument Calibration Sheet, RCIC Static Inverter (Remote Shutdown Panel), Loop C2879-01

LIST OF DOCUMENTS REVIEWED (Cont.)

Instrument Calibration Sheet, SLC Storage Tank Level, Loop No. L-C41-1N001
General Electric MPL C41-C001, VPF #3159-108-1, VPF # 3159-71(1)-2, VPF #3159-72(1)-1,
VPF #3159-71-2, VPF #3159-72-1 [Certified test data for SLC Pump and motor].
Control Scheme Test Data Sheet for Scheme 1Q1304 (E-154 Sheet 5), completed 8/5/81.
Work Authorization U40124 for replacement/testing of TOLs for MOV HV-E41-1F059, 3/31/95.
Work Authorization RTPM 206956 for test/inspection of TOLs for MOV HV-E41-1F001, 12/7/00.
Work Order 315963 confirming contact 2 of bypass relay 95-E51A was closed, 2/27/01.
TRA Strip Chart For STDP70, Plot Of Unit 1 RCIC Injection Event, 7-1-99
SIL No. 336, Surveillance Testing Recommendations For HPCI And RCIC Systems
Masoneilan Regulator Co. Drawing No. 414 Regulator, Modified D–Z4
Masoneilan No. 414 Reducing Valve Instructions (Instruction No. EY4140E, Rev B 8/83
3159-71(1)-2, Standby Liquid Control Pumps NPSH Test Report
Component Data Sheet for PCV14811A (PPL NIMS)

(3) LIST OF ACRONYMS

AC or ac	Alternating Current
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulation
CR	Condition Report
CRD	Control Rod Drive
DBD	Design Basis Document
DC or dc	Direct Current
GE	General Electric
GL	Generic Letter
gpm	Gallons Per Minute
IEEE	Institute of Electrical and Electronics Engineers
IST	Inservice Test
LOOP	Loss of Offsite Power
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
P&ID	Piping and Instrumentation Diagram
PPL	PPL Susquehanna, LLC
psia	Pounds Per Square Inch Absolute
psig	Pounds Per Square Inch Gage
PSV	Pressure Safety Valve
NC	Normally Closed
NO	Normally Open
NRC	Nuclear Regulatory Commission
RCIC	Reactor Core Isolation Cooling
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SDP	Significancy Determination Process
SE	Safety Evaluation
SLC	Standby Liquid Control System
SRV	Safety Relief Valve
TOL	Thermal Overload
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item