ABSTRACT

In April 1981, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0776) regarding the application of the Pennsylvania Power & Light Company (the applicant and/or licensee) and the Allegheny Electric Cooperative, Inc. (co-applicant) for licenses to operate the Susquehanna Steam Electric Station, Units 1 and 2, located on a site in Luzerne County, Pennsylvania.

Supplement 1 to NUREG-0776 was issued in June 1981 and addressed several outstanding issues. Supplement 2 was issued in September 1981 and addressed additional outstanding issues. Supplement 2 also contains NRC staff responses to the comments made by the Advisory Committee on Reactor Safeguards in its report dated August 11, 1981. Supplement 3 was issued in July 1982 and addressed five items that remained open and closed them out. On July 17, 1982, Operating License NPF-14 was issued to allow Unit 1 operation at power levels not to exceed 5% of rated power. Supplement 4 was issued in November 1982 and discusses the resolution of several license conditions. On November 12, 1982, Operating License NPF-14 was amended to remove the 5% power restriction, thereby permitting full-power operation of Unit 1. Supplement 5 was issued in March 1983 and addressed several issues that required resolution before licensing operating of Unit 2. Supplement 6 was issued in March 1984 and addressed the remaining issues that required resolution before licensing operation of Unit 2 and closed them out. On March 23, 1984, Operating License NPF-22 was issued to allow Unit 2 operation at power levels not to exceed 5% of rated power.

This supplement to NUREG-0776 addresses those issues which required resolution prior to allowing Unit 2 operation at power levels exceeding 5% rated power.

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In April 1981, the staff of the Nuclear Regulatory Commission (NRC) (the staff) issued its Safety Evaluation Report (SER) (NUREG-0776) regarding the application of the Pennsylvania Power & Light Company (PP&L) (the applicant and/or licensee) and the Allegheny Electric Cooperative, Inc. (the co-applicant) for licenses to operate Susquehanna Steam Electric Station, Units 1 and 2. In June 1981, the staff issued Supplement 1 to NUREG-0776, which documented the resolution of several outstanding issues in further support of the licensing activities. In September 1981, the staff issued Supplement 2 to NUREG-0776, which addressed the open items identified in the SER and Supplement 1. In July 1982, the staff issued Supplement 3 to NUREG-0776, which addressed all remaining open issues from previous supplements and closed them out. On July 17, 1982, Operating License NPF-14 was issued for Unit 1. Operation was restricted to fuel loading and low-power testing at levels not to exceed 5% rated power. In November 1982, the staff issued Supplement 4 to NUREG-0776, which addressed the resolution of several Unit 1 license conditions that had been met. On November 12, 1982, Amendment 5 to Operating License NPF-14 was issued removing the 5% power restriction, thus allowing Unit 1 operation at power levels not to exceed 100% rated power. In March 1983, the staff issued Supplement 5 to NUREG-0776, which addressed several issues that required resolution before Unit 2 could be licensed for operation. In March 1984, the staff issued Supplement 6 to NUREG-0776, which addressed the remaining issues the required resolution before licensing operation of Unit 2. On March 23, 1984, Operating License NPF-22 was issued for Unit 2. Operation was restricted to fuel loading and low-power testing at levels not to exceed 5% rated power.

Each section containing issues addressed in this report, Supplement 7 to NUREG-0776, is numbered and titled to correspond to the sections of NUREG-0776 and its earlier supplements where they are previously discussed. This report addresses the remaining issues that require resolution before Unit 2 can be licensed for full power operation and closes them out.

Copies of this report are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Osterhout Free Library, 71 South Franklin Street, Wilkes Barre, PA 18701. Copies of this report also are available for purchase from the sources indicated on the inside front cover.

The NRC project manager for Susquehanna is Mr. Robert L. Perch. Mr. Perch may be contacted by writing to the Division of Licensing, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

The methodology that the licensee utilized for this design review was divided into a two phase approach as discussed below. Phase 1, the "identification phase", consisted of identifying the following key items:

(1) Plant safety functions
(2) Control systems
(3) Power supplies and sensors to the control systems
(4) Power supplies and sensors common to control systems.

For these key items, Control System Identification Diagrams (CSID) were generated to document the information and to assist in further analysis. Power supply and sensor commonality was determined using the CSIDs. A second diagram, the Commonality Diagram (CD), was generated to show the control systems that were affected by each common power supply or sensor failure.

Phase 2, the "analysis phase", consisted of the analyses of the failure of these common power supplies and sensors with respect to their associated control systems. The control system failures were analyzed with respect to the following criteria:

(1) Plant response as per Chapter 15
(2) Plant conditions within operator and safety system capabilities
(3) Reanalysis or modifications required to correct any problems not covered by the first two criteria.

The methodology employed in the analysis phase was based upon Failure Modes and Effects Analysis (FMEA). This technique was performed on each common power supply and sensor to determine the effect of the failure on the control system and on plant performance.

A total of ten power supply and sensor commonalities for Unit 1 and eleven for Unit 2 were identified and analyzed. Of these commonalities, all were of the power supply type, except one which was of the sensor type. The review identified one commonality which required a detailed analysis concerning the loss of a 125V DC bus (1D635 for Unit 1 and 2D635 for Unit 2).

The control systems affected by this power supply failure in Unit 1 are the Reactor Feedwater, and the Pressure Regulator and Turbine/Generator (T/G) Control Systems. In addition to these, the recirculation runback circuitry is affected in Unit 2. The conditions that required a detailed analysis however, were specifically limited to the Feedwater Flow Control and Reactor Feedwater Pump Turbine (RFPT) control subsystems worst case failures. The loss of these power supplies did not generate conditions outside the boundary of the Chapter 15 safety analyses for the Pressure Regulator and T/G Control Systems or the Recirculation Runback Control System (Unit 2 only). The sequence of events for Unit 1 (Unit 2 is similar) that result from the loss of power supply 1D635 for the Feedwater System is as follows:

7 INSTRUMENTATION AND CONTROL

7.7 Control Systems Not Required for Safety

7.7.1 General Discussion

Common Electrical Power Sources or Sensor Malfunctions Causing Multiple Control System Failures

During the Instrumentation and Control Systems Branch (ICSB) review of the Susquehanna Steam Electric Station (SSES) Final Safety Analysis Report (FSAR), the staff noted that the analysis reported in Chapter 15 is intended to demonstrate the adequacy of the safety systems in mitigating anticipated operational occurrences and accidents, including those related to control systems. Based on the conservative assumptions made in defining these design basis events and the review performed, it was likely that the Chapter 15 analyses adequately bounded events initiated by a single control system failure. However, to assure that the Chapter 15 analyses adequately bound events caused by multiple control system malfunctions due to failures of shared power supplies, sensors or sensor lines, the staff requested that the licensee perform a review to determine what, if any, design changes or operator actions would be necessary to assure that these malfunctions would not complicate the event beyond the FSAR analysis.

High Energy Line Breaks and Consequential Control System Failures

If control system are exposed to the environment resulting from the rupture of reactor coolant lines, steam lines or feedwater lines, the control systems may malfunction in a manner which could cause consequences to be more severe than assumed in the FSAR safety analyses.

The staff requested Pennsylvania Power & Light Company (PP&L) to perform a review to determine what, if any, design changes or operator actions would be necessary to assure that these multiple control system malfunctions would not complicate the event beyond the FSAR analysis. In response to this concern, PP&L initiated a review to determine whether High Energy Line Breaks (HELBs) could have an effect on multiple controls systems and to investigate the impact of failure of the applicable systems on the FSAR Chapter 15 analysis.

7.7.2 Specific Findings

Common Electrical Power Sources or Sensor Malfunctions Causing Multiple Control System Failures

By letters dated October 14, 1983 and February 27, 1984 from N. W. Curtis (PP&L) to A. Schwencer (NRC), the licensee provided reports that presented the results of a design review, evaluation and plant walkdown addressing this concern for Unit 1 and Unit 2 respectively.

The first RETRAN run was performed simulating the loss of one feedwater flow element. This run indicated that the reactor water level would rise to 53.3 inches in 50 seconds and then become stable. While this level is below the 54 inch Level 8 setpoint, it is close enough that normal instrument drift could cause trips. Therefore, a second RETRAN run was performed so that the effects of the Level 8 trip could be examined. The computer code was modified to force a trip at 53.3 inches and to force a minimum feedwater injection rate of 25%.

The licensee stated that this simulation was over conservative in that the transient run had a steadily increasing water level due to the 25% assumed feedwater injection rate, when in actuality, upon a RFPT B trip, the false feedwater flow vs. steam flow mismatch is corrected and the feedwater controller will attempt to control reactor water level to the controller setpoint. Even with a feedwater pump running, the controller has the ability to terminate feedwater injection. Actual feedwater injection will terminate at approximately 70 to 90 seconds after the turbine trip due to a feedwater controller setback which was not modelled by the RETRAN code.

The results of the RETRAN simulated transient run indicate that the event is, in fact, bounded by the Chapter 15 safety analysis for thermal limit considerations. Therefore, the staff has concluded that the safety limits of Chapter 15 are not violated, and in addition, the resulting conditions are within the capabilities of the plant operators and safety systems. All of the remaining control system commonalities were determined to be either bounded by the results of the Chapter 15 safety analyses or did not impact plant safety.

The staff requested the licensee to identify all significant non-safety related multiple control system events caused by failures of shared sensor impulse lines. The licensee stated in a letter dated April 12, 1984 from N. W. Curtis to A. Schwencer, that based on the analysis performed, no significant non-safety related multiple control system events were caused by failures of shared sensor impulse lines. The staff then requested the licensee to verify that for each failed shared power supply, sensor and sensor impulse line, or the subsequent multiple control system failures, redundant safety-related systems are available (i.e., unaffected by the event) to mitigate the effects of the event. The intent was to assure that the consequences of the event can be mitigated given a single failure within the system used to mitigate the event. The licensee stated in the April 12 letter that for each multiple control systems failure event analyzed, redundant safety systems are available to mitigate the event and are unaffected by the multiple control system failure event. Furthermore, the licensee stated in the April 12 letter that with the exception of the feedwater level 8 trip, no credit was taken in the analysis for non-safety related equipment to mitigate the effects of these failures. Since the level 8 trip is used to terminate the feedwater controller failure in FSAR Chapter 15, the licensee stated that the level 8 trip can be used to mitigate the effects of various multiple control system failure events when they are analyzed against the feedwater controller failure event in Chapter 15. The level 8 trip has been incorporated into the Susquehanna Technical Specifications and its use to mitigate the effects of the feedwater controller failure event was found to be acceptable by the staff during the Operaing License (OL) review (FSAR Question 211.139).

a. While operating at 100% reactor power, the plant experiences a loss of 1D635. The feedwater flow signal from the B train instrumentation powered by 1D635 (Flow Transmitter FT1N002B and SRU 6) changes to zero due to the loss of 1D635. Since the feedwater flow signals from trains A, B and C are summed, the total feed flow signal changes from 100% feed flow to 67% feed flow subsequent to receiving the erroneous zero signal from the B train. This introduces a mismatch between steam flow, which is still at 100%, and feed flow which is at 67%.

b. In response to this steam flow, feed flow mismatch, the Feedwater Flow Control System sends a signal to the three RFPT's to increase feed flow to make up for the erroneous 33% decrease in flow. Actual feed flow at this point would be approximately 135%.

c. Since actual feed flow is significantly greater than that required, the increase in reactor vessel level <u>may</u> reach the Level 8 (high level) trip set point.

d. If the Level 8 trip set point is reached, a trip signal will be sent to RFPTs A, B, and C and the T/G. RFPTs A and B and the T/G trip. RFPT C fails to trip because its trip circuit was disabled upon loss of 1D635.

Based on the assumption that the Level 8 setpoint is reached due to excessive feedwater demand, it was found that the resulting conditions were not explicitly addressed by the Chapter 15 safety analyses. Chapter 15 status that the plant response to a Level 8 condition, initiated by excess feedwater flow, should include the trip of all RFPTs and the T/G. Since the conditions generated subsequent to the failure of RFPT C to trip are not known, it could not be determined if the plant system capabilities were within the bounds governed by the existing safety analyses.

However, it was evident that the operator retained the ability to take manual control of RFPT C to mitigate the effects of its continued operation. The operator would have been alerted to the using reactor vessel level by the Level 7 alarm. This condition, therefore, appeared to be within the capabilities of the operator. To provide a further analysis of this event, the licensee utilized a RETRAN computer code to simulate the event.

It should be noted that the NRC staff and their technical assistance consultants at Argonne National Laboratory have concluded that the use of the RETRAN computer code to perform licensing basis calculations is acceptable (with the understanding that the generic review of RETRAN is not complete, and the acceptability of all reviews is predicated on the anticipated successful completion of this generic review), that the selection of options and input to RETRAN provide a reasonable and adequate representation of the thermohydraulics, and that the results of these calculations can determine an acceptable set of input and initial conditions for the critical power ratio calculations.

In conclusion, the licensee stated the plant conditions that result from these multiple control system failures do not exacerbate the conditions that result from the events analyzed in Chapter 15 from a 10CFR100 guidelines perspective. In each case, the worst case event combinations are bounded by the radiological consequences currently provided for each Chapter 15 event.

High Energy Line Breaks and Consequential Control System Failures

By letters dated October 14, 1983 and April 2, 1984 from N. W. Curtis (PP&L) to A. Schwencer (NRC), PP&L provided reports that presented the results of a design review, evaluation and plant walkdown addressing this concern for Unit 1 and Unit 2 respectively.

The methodology that was utilized by the licensee for this review was designed to meet the following objectives:

(1) to identify potential HELB which could impact two or more control systems either by pipewhip, jet impingement, or the resultant harsh environment.

(2) to analyze the effects of the HELBs on the components/cables which comprise the control systems and to determine the impact of the specific component failures on the control systems.

(3) For simultaneous malfunctions of control systems due to a single HELB, determine if the combined failures are bounded by the Chapter 15 analyses and are within the capabilities of operators and safety systems.

A two phase approach was used as part of this methodology. Phase 1, the "identification phase", consisted of identifying the following terms:

(1) Plant safety function
(2) Control system components and cables
(3) Control system components and cable locations
(4) HELBs common to control system components/cables

Phase 2, the "analysis phase", consisted of the analysis of the multiple control system failures as a result of a single HELB. The control system failures were analyzed with respect to the following criteria:

(1) Plant response as per Chapter 15
(2) Plant conditions within operator and safety system capabilities
(3) Reanalysis or modifications required to correct any problems not covered by the first two criteria.

The methodology employed in the analysis phase was based upon Failure Modes and Effects Analysis (FMEA). The FMEA technique was used to generate failure effects information on each control system as it pertains to the specific HELB.

The licensee performed the HELB study using the guidelines noted above. The results of the study indicated that all postulated events satisfy the criteria for infrequent events, i.e., that the dose consequences do not exceed 10% of the 10 CFR 100 criteria.

A total of 24 HELB/multiple control system commonalities were identified. Of these, one was located inside primary containment, one in the reactor building outside primary containment, and 22 in the turbine building.

Because high energy lines (main steam, feedwater, and condensate) are located in almost every area of the turbine building, over 20 multiple control system/HELB interactions were identified. The most severe interaction was in a plant area adjacent to the control structure. This area contains a majority of the cable, routed from the sensors in the turbine to the control structure for the turbine/generator control, feedwater control, recirculation flow control, and reactor manual control systems. All of this cable would be affected by a jet from a 20 inch feedwater line longitudinal break. This pipe is the inlet to the feedwater heater. The initial pressure is assumed to be 400 psia based on the feedwater pump suction pressure requirements and the condensate pump discharge pressure. The turbine building is a large structure which is relatively open. This provides free communication of air and, following a major steam or feedwater line break, would result in a harsh environment (100°F) for a majority of the turbine building areas. Because of this, the licensee used the "sacrificial" approach, where all components and cables are assumed to fail in their worst mode due to harsh environment following a main steam or feedwater line break.

A postulated break of a main steam or feedwater line represents the largest steam or liquid lines outside of containment and provides the envelope evaluation relative to this type of occurrence in the turbine building. The break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the four main steam lines. The feedwater system break is less severe than the main steam line break in terms of reactor response. The consequences of the main steam line break which envelops all of the HELB/multiple control system interactions listed by the licensee are as follows:

(1) Largest steam line circumferentially breaks at a location downstream of the outermost isolation valve in the turbine building.

(2) Flow from the upstream portion is limited by the flow restrictor upstream of the inboard isolation valve.

(3) Flow from the downstream side is limited by the total area of the three unbroken lines.

(4) MSIVs start to close at 0.5 seconds on a high steam flow signal and are fully closed at 5.5 seconds.

(5) Reactor vessel level rises due to rapid depressurization and increased void formation.

(6) Recirc pumps trip on high reactor vessel pressure signal.

(7) Reactor scrams on high reactor vessel level or MSIV closure.

(8) Reactor feed pumps trip due to termination of steam flow to pump turbines following MSIV closure.

(9) Safety relief valves cycle to maintain vessel pressure at approximately 1100 psi.

(10) Turbine trips on MSIV closure or high reactor vessel level.

(11) Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1100 psi.

(12) RCIC and HPCI would initiate on low water level (RCIC considered unavailable, HPCI assumed single failure and therefore not available).

(13) Operator initiates ADS. Vessel depressurizes rapidly.

(14) Low pressure ECCS systems initiated. Reactor fuel uncovered partially.

(15) Core effectively reflooded and clad temperature heatup terminated. No fuel failure.

Following this event, none of the components located in the turbine building for the T/G control system, recirc control system, and feedwater flow control system are required to operate and there is no adverse affect on plant safety.

The staff requested the licensee to verify that for each HELB event and its consequential control system failures, redundant safety related systems are available (i.e., unaffected by the event) to mitigate the effects of the event. The intent was to assure that the consequences of the event can be mitigated given a single failure within the system used to mitigate the event. The licensee stated in a letter dated April 12, 1984 from N. W. Curtis to A. Schwencer that the conditions that resulted from the failure of multiple control systems due to HELBs were analyzed against each event in Chapter 15 to determine if these resultant conditions in combination with the conditions described in each specific Chpater 15 event were within the response capabilities of the plant safety systems.

In each case, the redundant safety systems that were available to mitigate the Chapter 15 event were unaffected by the additional failures of the control systems due to HELBs. Furthermore, the licensee stated in the April 12, 1984 letter that with the exception of the feedwater level 8 trip, no credit was taken in the analysis for non-safety related equipment to mitigate the effects

of these failures. Since the level 8 trip is used to terminate the feedwater controller failure in FSAR Chapter 15, the level 8 trip can be used to mitigate the effects of various HELBs when they are analyzed against the feedwater controller failure event in Chapter 15. The level 8 trip has been incorporated in the Susquehanna technical specifications and its use to mitigate the effects of the feedwater controller failure event was found to be acceptable by the staff during the operating license review (FSAR Question 211.139).

In conclusion the licensee stated that the plant conditions that result from the HELBs do not exacerbate the conditions that result from the events analyzed in Chapter 15 from a 10 CFR 100 guidelines perspective. In each case, the worst case event combinations are bounded by the radiological consequences currently provided for each Chapter 15 event.

7.7.3 Summary

Common Electrical Power Sources or Sensor Malfunctions Causing Multiple Control System Failures

Based on our review which indicates that the radiological consequences of the worst case multiple control system failure event is bounded by the radiological consequences currently provided for each Chapter 15 event, the staff finds that the conclusions of the analyses of the anticipated operational occurence and accidents as presented in Chapter 15 have been used to confirm that plant safety is not dependent on the response of the control systems. The staff concludes that multiple failures of control systems as a consequence of a failure of shared power supplies, sensors or sensor impulse lines will not result in plant conditions more severe than those bounded by the Chapter 15 safety analysis.

Therefore, License Conditions 2.C.(25)(a) for Unit 1 and 2.C.(10)(a) for Unit 2 of the Susquehanna facility (operating License NPF-14 for Unit 1 and NPF-22 for Unit 2) have been acceptably resolved.

However, it should be noted that the final resolution of this concern is predicated on the anticipated successful completion of the generic review of the RETRAN computer code utilized in the licensee's study. This generic review is being conducted by the staff and their technical consultants at Argonne National Laboratory. Although this review is not complete, enough progress has been made to date so that along with the information submitted by the licensee, adequate basis has been established to perform the review for this analysis.

High Energy Line Breaks and Consequential Control System Failures

Based on our review of the licensee's study which indicates that the radiological consequences of the worst case event combinations are bounded by the radiological consequences currently provided for each Chapter 15 event, the staff finds that the HELB concern is resolved. Therefore, License Conditions 2.C.(25)(b) for Unit 1 and 2.C.(10)(b) for Unit 2 of the Susquehanna facility (Operating License NPF-14 for Unit 1 and NPF-22 for Unit 2) have been acceptably resolved.

March 12, 1984	Letter from applicant transmitting a special report on fire protection.
March 13, 1984	Letter from applicant concerning clarification of Emergency Operations Facility Operation.
March 13, 1984	Letter from applicant transmitting the Monthly Operation Report for February 1984.
March 13, 1984	Letter from applicant concerning conformance to Regulatory Guide 1.97.
March 14, 1984	Letter from applicant concerning Generic Letter No. 82-33.
March 15, 1984	Letter from applicant concerning Human Engineering Discrepancies - Unit 2 Control Room.
March 15, 1984	Letter from applicant responding to request for additional information - Unit 2 SQRT Program.
March 15, 1984	Letter from applicant concerning feedwater check valve analysis.
March 16, 1984	Letter from applicant concerning notification of Unit 2 construction completion.
March 16, 1984	Letter from applicant concerning Final Safety Analysis Report Revision to Chapters 6, 7 and 18.
March 20, 1984	Letter from applicant concerning certification of Unit 2 Technical Specifications.
March 22, 1984	Letter from applicant transmitting the annual financial report.
March 23, 1984	Letter to applicant transmitting 2 copies of Supplement No. 6 to the Safety Evaluation Report Related to Operation of Susquehanna Steam Electric Station, Units 1 and 2 (NUREG-0776).
March 23, 1984	Letter to applicant transmitting Facility Operating License NPF-22 for Susquehanna Steam Electric Station, Unit 2. The license is restricted to 5% of full power pending Commission approval for 100% power.
March 28, 1984	Letter from applicant transmitting the Annual Personnel Monitoring Report.
March 29, 1984	Letter from applicant transmitting an amended response to Final Safety Analysis Report Question 110.57.

April 2, 1984	Letter from applicant concerning evaluation of high energy line breaks on control systems study for Unit 2.
April 5, 1984	Letter to applicant transmitting 20 copies of Supplement No. 6 to the Susquehanna Steam Electric Station Safety Evaluation Report - NUREG-0776.
April 6, 1984	Letter from applicant concerning major modification to initial test program for Unit 2.
April 10, 1984	Letter from applicant transmitting the Monthly Operating Report for March 1984.
April 10, 1984	Letter from applicant concerning milestone dates for Unit 2.
April 10, 1984	Letter from applicant transmitting proposed Amendment 38 to License No. NPF-14 and Proposed Amendment 1 to License No. NPF-22.
April 10, 1984	Letter from applicant transmitting proposed Amendment No. 2 to License no. NPF-22.
April 12, 1984	Letter from applicant transmitting a response to NRC letter, dated November 8, 1983.
April 18, 1984	Representatives from NRC and Pennsylvania Power & Light Company met in Bethesda, Maryland to discuss Main Steam Line - High Radiation Setpoint Technical Specification Change request. (Summary issued April 30, 1984)
April 24, 1984	Letter to applicant concerning feedwater check valve analysis.
April 24, 1984	Letter to applicant concerning Susquehanna Units 1 and 2 Annual Emergency Preparedness Exercise.
April 27, 1984	Letter to applicant concerning proposed Transco Gas Pipeline near the Susquehanna Site.
April 27, 1984	Letter from applicant transmitting one signed copy of Amendment 3 to Indemnity Agreement B-90.
May 1, 1984	Letter to applicant concerning Staff Review of Susquehanna DCRDR Summary Recommendations for the Resolution of Human Engineering Discrepancies.
May 1, 1984	Letter from applicant concerning Final Safety Analysis Report changes for radiation source terms and shielding.