EVALUATION OF THE EFFECTS OF HIGH ENERGY LINE BREAKS ON CONTROL SYSTEMS

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1.0 INTRODUCTION

On December 20, 1983, Pennsylvania Power and Light Company (PP&L) requested EDS Nuclear, Inc. (EDS) to assist in responding to the Safety Evaluation Report (SER) item concerning the failure of multiple non-safety grade control systems due to high energy line breaks (HELBs) and the subsequent effect on plant safety for the Susquehanna Steam Electric Station. The non-safety grade control systems specified by PP&L to be included in this evaluation includes the following.

1. Reactor Manual Control System (RMCS).
2. Recirculation Flow Control System (RECIRC FCS).
3. Reactor Feedwater Control System (RFCS).
4. Pressure Regulator and Turbine Generator Control System (T/G CS).
5. Traversing In-Core Probe Control System (TIP).
6. Reactor Water Cleanup Control System (RWCUCS).
7. Refueling Interlock Control System (RICS).
8. Rod Block Monitor System (RBMS).
9. Nuclear Pressure Relief Control System (PRCS).

This SER item is a result of the NRC's review of the safety analyses described in Chapter 15 of the SSES Final Safety Analysis Report. The NRC stated that the Chapter 15 analysis adequately bounds the "design basis events" assuming the failure of a single control system. However, the NRC requested additional information concerning whether multiple control system failures initiated by a single high energy line break would still be bounded by the Chapter 15 analyses.

To comply with this request, EDS performed the analysis described herein with the following objectives in mind:

- To identify potential high energy line breaks (HELBs) which could impact two or more control systems either by pipe whip, jet, or the resultant harsh environment.

- 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.

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- For simultaneous malfunctions of control systems due to a single high energy line break, determine if the combined failures are bounded by the Chapter 15 analyses and are within the capabilities of operators and safety systems.

EDS utilized a two-phased approach to achieve these objectives. The first phase consisted of identifying the control system components and associated cables and determining their physical plant location. Then, where two or more control system components are located in close proximity, determine if there is a potential for a high energy line break.

The second phase involved analyzing the HELB/Control System interactions using Failure Modes and Effects Analysis (FMEA). Using this approach, the effects of the component failures on the individual control systems and ultimately the combined control system failures on plant performance was determined. The FMEAs were then analyzed to determine the safety implications for the failure of these control systems.

This report documents the result of this analysis. The methodology employed is described generally in Section 3.0 and in greater detail in Appendix A. A summary of results is presented in general terms in Section 4.0 and in detail in Appendices B through F. References are provided in Section 5.0. An Executive Summary is provided in Section 2.0 which highlights the salient results of this project.

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2.0 EXECUTIVE SUMMARY

The purpose of this analysis is to evaluate if the effects of a single high energy line break could damage components of any of the two or more control systems in such a manner as to escalate the event beyond that analyzed in Chapter 15 of the FSAR.

2.1 Methodology

The project was divided into two-phases--the Identification Phase and the Analysis Phase. In the Identification Phase, key plant safety functions were identified using FSAR Chapter 15. The control systems that could affect these safety functions were identified by PP&L and are described in FSAR Chapter 7.7, "Control Systems Not Required for Safety." The components that are associated with these control systems were then identified. For these key items--safety functions, control systems, and sensors--Control System Identification Diagrams (CSID) were generated to document this information and to assist in further analysis.

The physical plant location of the control system components and associated cables was determined. The control system component locations were then matched with the plant areas where high energy lines exist to identify the HELB/multiple control system interactions.

A second diagram--Commonality Diagram (CD)--was generated to show the control systems and their associated components that were affected by each common HELB.

In the Analysis Phase, Failure Modes and Effects Analysis (FMEA) was performed on each common HELB to determine the effect of each component failure on the control system and on plant performance. Analysis was then performed using the FMEA results to determine the following:

1. Impact on plant safety including plant response as per Chapter 15.

2. If the plant conditions were within operator and safety system capabilities as per Chapter 15.

For those conditions that did not meet the criteria of items (1) and (2), recommendations for plant modifications or Chapter 15 reanalysis would be provided.

2.2 Results

A total of twenty four high energy line break (HELB)/multiple control system commonalities were identified. Of these, one was located inside primary containment, one in the reactor building outside primary containment, and twenty two in the turbine building.

As a result of this comprehensive evaluation, none of the multiple control system failures following a HELB adversely impact plant performance or prevent the mitigation of the initiating event. The conditions which result from the pipe rupture events are bounded by the Chapter 15 analyses. Therefore, no plant modifications or Chapter 15 reanalysis is required. Each of these commonalities and the resulting consequences are discussed in detail in Section 4.0 and Appendices A through F of this report.

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3.0 METHODOLOGY

As indicated in the introduction, the methodology that was utilized by EDS for this project was designed to meet the following objectives:

- To identify potential high energy line breaks (HELBs) which could impact two or more control system either by pipe whip, jet, or the resultant harsh environment.
- 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.
- For simultaneous malfunctions of control systems due to a single high energy line break, determine if the combined failures are bounded by the Chapter 15 analyses and are within the capabilities of operators and safety systems.

The methodology employed to achieve these objectives is summarized in this section. A detailed description of this methodology is contained in Appendix A, "Technical Procedures for the Performance of the Analysis".

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

- Plant safety functions
- Control System components and cables
- Control System component and cable locations
- 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:

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

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3.1 Identification Phase

The first part of the identification phase consisted of determining how the non-safety grade control systems specified by PP&L (see Introduction) could impact plant safety. In order to accomplish this, it was first necessary to identify those plant safety functions that are required to be met during the various modes of plant operation. The safety functions were generated using Chapter 15, Appendix 15A of the FSAR and are defined here in Appendix B for each plant operating state. The plant operating modes and safety functions were added to the Control System Identification Diagrams (CSIDs). The CSIDs are included in Appendix C.

Subsequent to this, the control system components/cables were identified and located in the plant. The criteria for component/cable selection included the following:

1. Control system components include "initiating type" components such as position switches, solenoid valves, flow elements, temperature elements, pressure and level sensing devices, instrument switches, transmitters, controllers, E/P converters, vibration instrumentation, and related devices.

2. Mechanical equipment such as valves, pumps, turbines, etc. are not considered as part of the control systems.

3. Control system instruments located in the control structure where no high energy lines exist are not considered. However, if a control system component sends a signal outside of the control structure, the cable is identified and located.

4. In most cases, instruments which provide only indication, position status, alarm, or input to a process computer are excluded.

5. Instrument tubing is included as part of the control system.

6. Cables were traced from the control system component to its terminal end, either to a piece of equipment (valve operator, pump, etc.) or to the control structure.

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When the control system components were selected and listed, the locations were then determined with the aid of the instrument index, instrument location drawings, raceway drawings, and plant location drawings.

In order to verify the component locations, a plant walkdown was conducted. Prior to the walkdown, the high energy piping drawings were reviewed to provide general information concerning which components are located in close proximity to each other.

3.1.1 Reactor Building Outside Containment HELB/Control System Interaction Identification

To determine if a control system component located in the reactor building outside primary containment was affected by a high energy line break, the information presented in the SSES Jet Impingement Report (references 5.1.3) was utilized. This report details the break locations and jet pressure vs distance data for all postulated HELBs in the reactor building but is limited in scope to only safety grade equipment as targets. In addition, the areas with harsh environments following HELBs are identified and illustrated on Figures F-5 through F-12 of Appendix F. The harsh environment information is based upon the SSES FSAR Section 3.11. Harsh environment is defined as an elevated temperature, pressure, humidity, or radiation condition which is a consequence of a HELB.

Areas with multiple control system component/cables were compared to the HELB areas to determine where commonalities exist. Only one HELB/multiple control system interaction was identified in plant area 25-5, a 4-inch reactor water cleanup (RWCU) system HELB affecting TIP and RWCU control system components (see Figure D-1) of Appendix D. No other commonalities exist in the Reactor Building outside containment due to pipe whip, jet, or harsh environment.

3.1.2 Reactor Building Inside Containment HELB/Control System Interaction Identification

After the location of all control system components were determined, it was found that relatively few are located inside primary containment. The components are part of the TIP, Recirculation and Feedwater Flow Control Systems.

Therefore, the "sacraficial" approach was chosen. For any HELB inside containment, it is assumed that all components, cables, or instrument tubing considered will fail in its worse mode. This is a conservative approach which bounds all other HELB/Control System interactions which could exist inside containment. This commonality is illustrated in Figure D-2 of Appendix D and the results are discussed in Section 4.0 of this report.

3.1.3 Turbine Building HELB/Control System Interactions

A large number of control system components and cabling is located in the turbine building. However, no pipe whip-jet impingement studies or harsh environmental analyses have been performed due to the lack of safety grade equipment located within this structure. Therefore, this task was performed in the following manner:

- The high energy piping in the turbine building was traced on the Plant Design Drawings. The systems listed in Table F-1 of Appendix F were included. High energy piping is defined as having a maximum operating temperature of 200°F or higher or a maximum operating pressure exceeding 275 psig during full power operation. To be classified as high energy, the system must operate more than 2 percent of the time to perform its intended function. Lines 1-inch NPS or smaller are also excluded. The turbine building areas which contain high energy lines are listed in Table F-2, Appendix F. The turbine building areas are shown on Figures F-1 through F-4 in Appendix F.

- The turbine building high energy piping drawings were then compared to the control system component/cable locations. Where high energy piping was in close proximity to components/cables of more than one control system, circumferential or longitudinal pipe breaks were postulated according to the MEB 3-1 criteria (reference 5.1.9).

- HELBs were assumed only at terminal ends and at intermediate pipe fittings or weld attachments. Only one HELB was postulated at any one time and full power operation was assumed.

- Circumferential pipe ruptures are displaced laterally by a minimum distance equal to one pipe diameter relative to each other producing a flow area equal to the cross sectional flow area of the pipe. Circumferential HELBs initiate pipe whip movement within the plane of the pipe geometry in the direction of the jet reaction. The pipe is assumed to form a hinge at its nearest rigid support, anchor or penetration and incapacitate any non-safety control components within the arc produced. The arc will have a radius equal to the distance from the break to the nearest support.

- Longitudinal breaks are the result of an axial split with a circular area equal to one pipe diameter. Break locations are oriented at two diametrically opposed points on the piping circumference such that a jet reaction causes out of plane bending of the piping configuration (i.e., pipe whip in the direction of the jet reaction). An example of a longitudinal pipe break at an elbow is shown on Figures F-13 and F-14.

- The jet resulting from a HELB is assumed to expand uniformly proceeding in a straight path, at a half angle not exceeding 10 degrees. Components affected by a jet are only considered if they are within the range of the jet.

- As a result of this task, 22 HELB/control system interactions were identified as being affected by pipe whip or fluid jet. These interactions are summarized in Table F-3 of Appendix F.

3.2 Analysis Phase

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.

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Using the information from the CDs, the overall effect of the component as a result of a HELB was determined with respect to control system and plant performance. The results of this part of the analysis were documented on FMEA forms as contained in Appendix E.

The FMEAs were used to determine if the plant conditions generated following a specific HELB impacted on plant safety and were within the capabilities of operators and safety systems. The conditions generated as per the FMEAs were compared with Chapter 15 analysis for verification of plant response, operator response, and safety system response. For those plant conditions in which plant safety was impacted without appropriate Chapter 15 analysis and operator and safety system capability verifications, system modifications or Chapter 15 analysis recommendations would be provided. This information was then summarized in the results section of this report.

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4.0 SUMMARY OF RESULTS

The purpose of this effort was to determine if the failure of common components for non-safety grade control systems could impact on SSES plant safety. The results of this effort are divided into two major areas:

1. Identification of key elements
   - o Control systems that could impact plant safety and the components of these control systems
   - o Common HELBs which would affect these control systems

2. Analysis of control system failure
   - o Referenced to the FSAR - Chapter 15
   - o Within capabilities of operator and safety systems
   - o Recommendations for reanalysis or modification if required

This section provides a summary of the results determined by EDS with respect to each of the two major areas. A more detailed item-by-item listing of the results is contained in Appendices B through F.

4.1 Identification

Based on the plant safety functions for each plant operating mode as described in Chapter 15 and the control systems described in Chapter 7.7, "Control System Not Required for Safety," the control systems that could impact plant safety were determined by PP&L to be the following:

1. Reactor Manual Control System (RMCS)
2. Recirculation Flow Control System (RECIRC FCS)
3. Reactor Feedwater Control System (RFCS)
4. Pressure Regulator and Turbine Generator Control System (T/G CS)
5. Traversing In-Core Probe Control System (TIP)
6. Reactor Water Cleanup Control System (RWCUCS)
7. Refueling Interlock Control System (RICS)
8. Rod Block Monitor System (RBMS)
9. Nuclear Pressure Relief Control System (PRCS)

It should be emphasized that this list includes those control systems that could impact plant safety. Actual determination of those control systems that, in fact, do impact plant safety would be accomplished during the analysis phase of the project. Documentation of the safety functions and control systems is contained in the Control System Identification Diagrams (CSIDs) in Appendix C. The components considered in this evaluation and their location are listed in Tables C-1 to C-7.

Upon completion of the component and sensor identification, HELB/control system commonality was determined. Commonality Diagrams (CDs) were generated to show commonality between those control systems identified. A total of three bounding commonalities were determined. These commonalities form the basis for the analysis phase of the project. The CDs are contained in Appendix D.

4.2 Analysis

The analysis of the control systems that are affected by a common HELB was accomplished using Failure Modes and Effects Analysis (FMEA), then analyzing the overall impact of each system FMEA on the plant. The FMEAs were generated for each control system as it pertains to the common HELB. The detailed results of each FMEA are contained in Appendix E. The worst single failures are assumed to occur as defined for each specific event in the FSAR Chapter 15 analysis. The control system component failures as a result of a HELB are not considered as a single failure. They are a result of the initiating accident event.

Based on the FMEAs, the detailed analysis of these control systems was performed. None of the commonalities which were identified as a result of this evalution have an adverse affect on plant safety following the specific HELB. Therefore, further safety analysis or plant modifications are not warranted. Each commonality is described here in detail.

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4.2.1 High Energy Line Break Inside Containment

A spectrum of high energy line breaks (from .08 ft^2 to a complete recirc line rupture) has been analyzed and reported in the SSES FSAR. The break types include steam and liquid process system lines. Any break within this category will result in a reduction of reactor coolant inventory. The high energy line breaks can be placed into two categories according to the plant response: Main steam line break (MSLB) which results in an initial reactor vessel level increase and all other breaks which result in reactor vessel level and subsequent pressure decrease. The events which occur following these HELBs are summarized as follows:

4.2.1.1 MSLB Between Reactor Vessel and Flow Restrictor

- Reactor Scram initiated by high drywell pressure or MSIV closure signal
- Diesel generators are signalled to start on high drywell pressure
- MSIV closure on high steam flow signal terminates blowdown from the unaffected steamlines, but continues on the reactor vessel side of the ruptured line
- HPCI, LPCI, and CS are signalled to start on high drywell pressure
- Turbine trip due to high reactor level or MSIV closure
- Recirc pump trip (RPT) due to turbine stop valve or turbine control valve closure signal or high reactor vessel pressure
- Reactor feed pump turbines tripped due to high reactor vessel level or MSIV closure

- Reactor water level rises rapidly due to void formation and reaches the steam nozzles. The core is not uncovered by this event.

- The reactor vessel depressurizes rapidly to drywell pressure and the blowdown ceases

- Low pressure ECCS systems are initiated and the reactor refloods to the level of the steamline nozzles and ECCS flow will spill into the drywell

4.2.1.2 Other HELBs Inside Containment

- Reactor Scram initiated by high drywell pressure, reactor vessel low water level, or MSIV closure signal

- MSIV closure on low reactor vessel level 2

- Turbine trip due to MSIV closure or manual initiation

- Recirc pumps trip on turbine stop or control valve closure signal or low reactor water level

- Reactor feed pumps tripped on MSIV closure

- HPCI, LPCI, and CS are signalled to start on high drywell pressure or low reactor vessel level

- Auto depressurization sequence begins if the following is satisfied: low reactor water level and high drywell pressure and at least 1 LPCI pump or 1 CS loop (2 pumps per loop) are running

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For smaller breaks which do not rapidly depressurize the reactor vessel, HPCI is utilized to maintain reactor vessel inventory. When assisted by the ADS, LPCI also provides protection against small breaks; in the event of HPCI falure. If a scram does not occur automatically, the operator would manually scram the reactor and initiate an orderly shutdown according to operating procedures.

Following a HELB inside containment control system components for the recirc control, TIP control, and feedwater flow control systems which are identified as being located inside containment (with the exception of the feedwater control system components which are located outside containment but have instrument lines inside) are assumed to fail in their worst possible mode; using a sacrificial approach.

The recirc control system components consist of control switches for the pump suction, discharge and bypass valves. The suction and discharge valves are assumed to fail opened therefore, not restricting recirc flow. The failure mode of the bypass valves is assumed closed. The valve failure mode has no effect on plant safety, because the recirc pumps are tripped automatically by a signal generated either turbine stop or control valve closure, low reactor water level, or high vessel pressure. This instrumentation and the breakers which open to trip the pumps are located outside containment and out of the influence of the HELB.

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The feedwater control system instrument lines inside containment are for reactor vessel level and main steam line flow instrumentation located outside containment. They are used for automatic 1 and 3 element feedwater flow control. In addition, an instrument line for reactor vessel pressure is located inside containment. However, this signal is used only as an input to the display computer, performance monitoring computer, and high pressure alarm. It is not needed for feedwater flow operation.

For most HELBs inside containment, the MSIVs close thereby terminating steam flow to the RFPTs and stopping feed flow. The feedwater system is not required as reactor level can be maintained above the top of active fuel (TAF) with HPCI, RCIC, or LPCI operation.

For smaller breaks which do not rapidly depressurize the reactor vessel, the operator could take manual control of the feedwater control system utilizing other safety-grade vessel level instrumentation to maintain reactor water inventory while initiating an orderly shutdown. Therefore, these feedwater control system components will have no effect on plant safety for this event.

The traversing in-core probe control system indexing mechanisms are located inside primary containment and are assumed to fail as is following this event. This results in a loss of control capability of the indexing mechanisms, but the TIP probe (if inserted at the time of the HELB) can still be retracted to its normal chamber shield position and containment integrity would be maintained.

The results of this HELB/multiple control system interaction are summarized in the FMEA tables in Appendix E and commonality Diagram D-2, Appendix D.

4.2.2 Reactor Water Cleanup System Line Break in the Reactor Building (Outside Containment)

Following a RWCU system line break, the flow rate from the upstream side increases to the critical flow for the break area, reaches a steady state value, and then decreases as the isolation valves close. This HELB does not depressurize the reactor vessel or release sufficient inventory to result in a unit trip.

The event is successfully terminated when the RWCU containment isolation valves are automatically closed by a high flow or high differential flow signal generated by the safety grade Leak Detection System. The Leak Detection System is not part of the reactor water cleanup control system.

The operator can detect the break by high RWCU area temperature alarms or high flow indications. The plant can then be shutdown safely using normal or emergency shutdown procedures.

Components and associated cables for the reactor water cleanup and TIP control systems are affected by a whip or jet from any of several 4 inch reactor water cleanup system line breaks in plant area 25-5 (in the circulation space outside of the reactor water cleanup pump room on elevation 749 ft). The breaks could occur in lines 4"-DBB-122, 4"-EBC-101, 4"-EBC-103, 4"-EBC-104. Typically, this interaction could occur from a break at locations one through eight on pipe 4"-DBB-122 as described in the SSES "Jet Impingement Report - Reactor Building" beginning on page 238 (Reference 5.1.3). The cables/components affected by this interaction are assumed to fail in their worst mode as described on the FMEA tables in Appendix E. The commonality is shown on Diagram D-1, Appendix D.

The TIP guide tube assemblies and purge valve are affected. This may result in the shear and ball valves inoperative, unable to perform TIP operation, and inability to control the purge valve. These failures have no consequence following a RWCU system line break, because the TIP system is not required to function following this line break and does not inhibit its isolation.

The RWCU control system instrumentation affected by this HELB is utilized for normal automatic operation of the RWCU system. Therefore, these failures have no impact following the HELB when the system is incapacitated. The line break is automatically isolated by the safety-grade single failure proof leak detection system.

4.2.3 High Energy Line Breaks In the Turbine Building

Because high energy lines (main steam, feedwater, and condensate) are located in almost every area in the turbine building, over twenty multiple control system/HELB interactions were identified. The most severe interaction was in plant area 8-3 adjacent to the control structure. This area contains a majority of the cabling routed from the sensors in the turbine to the control structure for the T/G Control, Feedwater Control, Recirculation Flow Control, and Reactor Manual Control Systems. All of this cabling would be affected by a jet from a 20 inch feedwater line longitudinal break. The longitudinal break occurs on the underside of the elbow, propelling the pipe upward. The whip is terminated when the pipe strikes a wall. An illustration of this interaction is provided in Figures F-13 through F-14 in Appendix F. This pipe is the inlet to feedwater heater 1-3. The initial pressure is assumed to be 400 psia based on the feedwater pump suction pressure requirements and the condensate pump discharge pressure. The jet pressure is high near the break and dissipates with distance. Since all the trays are structurally attached, it is reasonable to assume that the trays near the break which fail due to whip and jet also cause the failure of the more distant trays. If a cable tray is hit by the jet, the cables within it are assumed to fail.

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The turbine building is a large structure which is relatively open. It is not highly compartmentalized, with many of the elevations separated by only gratings. Large open equipment hatches are common. 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 where control system components are located. Because of this the "sacrificial" approach was also utilized in the turbine building. This includes the components from the area 8-3 commonality and the remaining components located in other turbine building areas. These components/cables are assumed to fail in their worst mode following a main steam or feedwater line break due to harsh environment.

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 severence of one of the four main steam lines. The feedwater system pipe break is less severe than the main steam line break, in terms of reactor response.

The sequence of events following a main steam and feedwater HELB is given in the following sections. The consequences of these breaks are more severe and envelope the HELB/multiple control system interactions listed in Table F-3, Appendix F. Commonality Diagram D-3 illustrates this interaction (see Appendix D).

4.2.3.1 Main Steam Line Break in the Turbine Building

- Largest steam line circumferentially breaks at a location downstream of the outermost isolation valve in the turbine building
- Flow from the upstream portion is limited by the flow restrictor upstream of the inboard isolation valve
- Flow from the downstream side is limited by the total area of the three unbroken lines

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- MSIVs start to close at 0.5 seconds on a high steam flow signal and are fully closed at 5.5 seconds
- Reactor vessel level rises due to rapid depressurization and increased void formation
- Recirc pumps trip on high reactor vessel pressure signal
- Reactor scrams on high reactor vessel level or MSIV closure
- Reactor feed pumps trip due to termination of steam flow to pump turbines following MSIV closure
- Safety relief valves cycle to maintain vessel pressure at approximately 1100 psi
- Turbine trips on MSIV closure or high reactor vessel level
- 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.
- RCIC and HPCI would initiate on low water level (RCIC considered unavailable, HPCI assumed single failure and therefore not available).
- Operator initiates ADS. Vessel depressurizes rapidly
- Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
- 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, reactor manual control system, and feedwater flow control system are required to operate and their failure has no adverse affect on plant safety as shown in the FMEA tables in Appendix E.

The closing of the MSIVs effectively shuts down the turbine generator and the feedwater pumps. Therefore, any failure in their respective control systems is acceptable.

The reactor manual control system is utilized to manually control the movement of rods during power operation. It is entirely independent of the reactor scram system. The reactor will be scrammed successfully by the RPS. The RMCS is not required and its failure has no adverse affect on plant safety.

The recirc pumps will be tripped on a high reactor vessel pressure signal following the MSIV closure. The instrumentation and controls necessary to perform this trip are located in the reactor building, separated from the harsh environment in the turbine building. The recirc pump trip could also be generated by the safety grade signal generated by the turbine stop or control valve closure, however, this instrumentation is assumed to fail due to the harsh environment created by the HELB.

4.2.3.2 Feedwater Line Break in the Turbine Building

- One feedwater line breaks
- Feedwater line check valves isolate the reactor from the break

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- At low low reactor water level RCIC would initiate, HPCI would initiate, MSIV closure would initiate, reactor scram would initiate and recirculation pumps would trip

- The safety relief valves would open and close and maintain the reactor vessel pressure at approximately 1100 psig.

- Normal reactor cooldown procedure would be established

Similar to the steam line break, the MSIVs are closed and the recirc pumps are tripped however, the initiating signal is low low reactor vessel level. The secondary effects of the MSIV closure are turbine trip and reactor feed pump turbine trip. RCIC and/or HPCI would be sufficient to maintain reactor vessel level.

As with the main steam line break, the instruments and controlling functions for the non-safety control system components located in the turbine building are not required to operate. The failure of these components in their worst mode following a feed line break has no adverse effect on plant safety.

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5.0 REFERENCES

The following is the list of references used during this project:

5.1 General Information

1. Susquehanna Steam Electric Station, Units 1 and 2, Final Safety Analysis Report, Pennsylvania Power and Light Company, Volumes 1-17, Revision 32, 12/82.

2. EDS Report No. 02-0160-1102, Rev. 1, "SSES Control System Power Supply and Sensor Malfunction Study", dated April 8, 1982.

3. SSES 1 and 2, "Jet Impingement Report Inside Containment and Reactor Building", 8856-M-193.

4. SSES 1 Instrument Index 76/B, Report Date 03/17/82.

5. SSES 1 Circuit Schedule Sorted by SUS Rev. 51, dated 1/83.

6. SSES 1 and 2, Numbering System Description, E-41, Rev. 6.

7. SSES 1 and 2, Standard Symbols for Electrical Diagrams, E-42, Rev. 4.

8. SSES 1 and 2, Terminal Block and Nameplate Details, E-43, Rev. 11.

9. U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" (with BTP APCSB 3-1) and Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" (with BTP, MEB 3-1).

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5.2 System Descriptive References

A. Reactor Feedwater Control System

1. Susquehanna 1, Operations and Maintenance Instructions, Feedwater Control System, General Electric (GEK-93592A), April, 1981.

2. Design Specification Feedwater/Level System, Document No. 22A1367, Rev. 8.

B. Recirculation Flow Control System

1. Susquehanna 1, Operations and Maintenance Instructions, Recirculation Flow Control System, General Electric (GEK-73590), February, 1979.

2. Design Specification Reactor Recirculation System, Document Numbers 22A2916AJ Rev. 6 and 234A9302AE.

C. Reactor Water Cleanup System

1. Susquehanna 1, Operation and Maintenance Instructions, Reactor Water Cleanup System, General Electric (GEK-73608), February 1979.

2. Design Specification Reactor Water Cleanup System, Document Numbers 22A1428AF and 234A9312AE.

D. Pressure Regulator and T/G Control System

1. Generator Protection, General Electric (GEK-75512A), November, 1980.

2. Basic Functions of Electrohydraulic Control (EHC) System, Nuclear (Boiling Water Reactor) Units, General Electric (GEK-17911).

3. Protection System - Electrohydraulic Control, Basic Functions, General Electric, (GEK-11366).

4. Introduction to the Basic Elements of Control Systems for large Steam Turbine-Generators, General Electric, (GET-3096C).

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E. Traversing Incore Probe Control System

1. SSES Plant System Course, TIP Study Guide No. 40, Rev. 0.

2. Operation and Maintenance Instructions, Indexing Mechanism 791E241G4 (GEK-73601A), February, 1981.

F. Reactor Manual Control System

1. Susquehanna 1 and 2, Operation and Maintenance Instructions, Reactor Manual Control System, General Electric (GEK-73596A), April, 1981.

2. Design Specification CRD Hydraulic System, Document Numbers 234A9303AE and 22A1342 Rev. 11.

G. Nuclear Pressure Relief System

1. Susquehanna 1, Operation and Maintenance Instructions, Automatic Depressurization System, General Electric (GEK-73602), February, 1979.

2. Design Specification Nuclear Boiler System, Document No. 22A2925 Rev. 6 and 234A9301AE.

5.3 System Drawings

5.3.1 General Reference

High Energy Piping Drawings - Reactor Building

Drawing Number	Revision
HE-M-25-5	0
HE-M-25-9	0
HE-M-25-13	0
HE-M-25-14	0
HE-M-25-16	0
HE-M-25-17	0
HE-M-25-18	0
HE-M-25-19	0
HE-M-28-1	0
HE-M-28-2	0

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Drawing Number	Revision
HE-M-28-3	0
HE-M-28-5	0
HE-M-28-12	0
HE-M-28-13	0
HE-M-28-14	0
HE-M-28-15	0
HE-M-29-2	0
HE-M-29-3	0
HE-M-29-12	0
HE-M-26-2	0
HE-M-26-3	0
HE-M-26-4	0
HE-M-26-5	0
HE-M-26-6	0
HE-M-26-7	0
HE-M-26-9	0
HE-M-26-10	0
HE-M-26-11	0
HE-M-26-12	0
HE-M-26-14	0
HE-M-26-15	0

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Instrument Location Drawings

E-103485-1, Rev. 11
-2, Rev. 10
-3, Rev. 18
-4, Rev. 20
-5, Rev. 12
-6, Rev. 8
-7, Rev. 4

E-103486-2, Rev. 11
-3, Rev. 12
-4, Rev. 8
-5, Rev. 6
-6, Rev. 6
-7, Rev. 3
-8, Rev. 8
-12, Rev. 6
-14, Rev. 5

E-103487-1, Rev. 7
-2, Rev. 9
-3, Rev. 19
-4, Rev. 17
-5, Rev. 11
-6, Rev. 12
-7, Rev. 10

E-103488-1, Rev. 20
-2, Rev. 10
-3, Rev. 16
-4, Rev. 15
-5, Rev. 9
-6, Rev. 4

E-103489-1, Rev. 16
-2, Rev. 9
-3, Rev. 10
-4, Rev. 14
-5, Rev. 9
-6, Rev. 1
-7, Rev. 5

E-103461-1, Rev. 7
-2, Rev. 8
-3, Rev. 6

E-103462-1, Rev. 9
-2, Rev. 5
-3, Rev. 6
-4, Rev. 10

E-103463-1, Rev. 6
-2, Rev. 3
-3, Rev. 5
-4, Rev. 9

E-103464-1, Rev. 10
-2, Rev. 4
-3, Rev. 7
-4, Rev. 3

E-103465-1, Rev. 8
-2, Rev. 7
-3, Rev. 9

E-103466-1, Rev. 7
-2, Rev. 8
-3, Rev. 6
-4, Rev. 9

E-103467-1, Rev. 9
-2, Rev. 7
-3, Rev. 5
-4, Rev. 7

E-103468-1, Rev. 12
-2, Rev. 6
-3, Rev. 7
-4, Rev. 5

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Instrument Location Drawings
(continued)

E-103469-1, Rev. 8
-2, Rev. 6
-3, Rev. 7
-4, Rev. 8

E-103470-1, Rev. 18
-2, Rev. 9
-3, Rev. 7
-4, Rev. 5
-6, Rev. 3

E-103471-1, Rev. 12
-2, Rev. 5
-3, Rev. 8
-4, Rev. 6
-5, Rev. 3

E-103472-1, Rev. 9

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Raceway Drawings

E-107595-1, Rev. 12
-2, Rev. 12
-3, Rev. 12

E-107596-1, Rev. 9
-2, Rev. 9
-3, Rev. 8
-4, Rev. 8

E-107597-1, Rev. 11
-2, Rev. 9
-3, Rev. 9
-4, Rev. 9

E-107598-1, Rev. 14
-2, Rev. 15
-3, Rev. 17
-6, Rev. 11

E-107599-1, Rev. 12
-2, Rev. 11
-3, Rev. 12

E-107600-1, Rev. 10
-2, Rev. 8
-3, Rev. 12
-4, Rev. 8

E-107601-1, Rev. 12
-2, Rev. 7
-3, Rev. 11
-4, Rev. 14

E-107602-1, Rev. 15
-2, Rev. 18
-3, Rev. 19
-4, Rev. 14

E-107603-1, Rev. 18
-2, Rev. 18
-3, Rev. 16
-4, Rev. 11
-5, Rev. 12

E-107604-1, Rev. 14
-2, Rev. 19
-3, Rev. 16
-4, Rev. 12
-5, Rev. 8
-6, Rev. 4

E-107605-1, Rev. 16
-2, Rev. 18
-3, Rev. 19
-4, Rev. 16

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Raceway Drawings
(continued)

E-107606-1, Rev. 16
-2, Rev. 18
-3, Rev. 26
-4, Rev. 18
-5, Rev. 18
-6, Rev. 19
-7, Rev. 13
-8, Rev. 14
-9, Rev. 18
-10, Rev. 14
-11, Rev. 8
-12, Rev. 4
-13, Rev. 3
-14, Rev. 13
-15, Rev. 1
-16, Rev. 15
-17, Rev. 11
-18, Rev. 4
-19, Rev. 2
-20, Rev. 1
-21, Rev. 1

E-107626-1, Rev. 19
-2, Rev. 24
-3, Rev. 21
-4, Rev. 17
-5, Rev. 13
-6, Rev. 11
-7, Rev. 8
-8, Rev. 5
-9, Rev. 7
-10, Rev. 6

E-107627-2, Rev. 23
-3, Rev. 21
-4, Rev. 16
-5, Rev. 7
-6, Rev. 6
-7, Rev. 6
-8, Rev. 14
-9A, Rev. 8
-9B, Rev. 3
-10, Rev. 3

E-107628-1, Rev. 16
-2, Rev. 21
-3, Rev. 22
-4, Rev. 20
-5, Rev. 16
-6, Rev. 13
-8, Rev. 6
-9, Rev. 8

E-107629-1, Rev. 26
-2, Rev. 22
-3, Rev. 21
-4, Rev. 20
-5, Rev. 10
-6, Rev. 11
-7, Rev. 5
-8, Rev. 7
-9, Rev. 5

E-107630-1, Rev. 19
-2, Rev. 16
-3, Rev. 17
-4, Rev. 20
-5, Rev. 15
-6, Rev. 16
-7, Rev. 5
-8, Rev. 8

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Plant Design Drawings - Turbine Building

E-106151 Sheet 1 Rev. 12
E-106151 Sheet 2 Rev. 19
E-106151 Sheet 3 Rev. 22
E-106151 Sheet 4 Rev. 3
E-106151 Sheet 7 Rev. 27
E-106151 Sheet 8 Rev. 15
E-106151 Sheet 9 Rev. 3
E-106151 Sheet 10 Rev. 23
E-106151 Sheet 11 Rev. 0

E-106152 Sheet 1 Rev. 12
E-106152 Sheet 2 Rev. 9
E-106152 Sheet 3 Rev. 18
E-106152 Sheet 4 Rev. 6
E-106152 Sheet 6 Rev. 7
E-106152 Sheet 8 Rev. 22
E-106152 Sheet 9 Rev. 2
E-106152 Sheet 10 Rev. 18
E-106152 Sheet 11 Rev. 1

E-106153 Sheet 1 Rev. 24
E-106153 Sheet 2 Rev. 13
E-106153 Sheet 3 Rev. 19
E-106153 Sheet 4 Rev. 6
E-106153 Sheet 7 Rev. 8
E-106153 Sheet 8 Rev. 26
E-106153 Sheet 9 Rev. 2
E-106153 Sheet 10 Rev. 18
E-106153 Sheet 11 Rev. 3

E-106154 Sheet 1 Rev. 30
E-106154 Sheet 2 Rev. 10
E-106154 Sheet 3 Rev. 16
E-106154 Sheet 4 Rev. 16
E-106154 Sheet 7 Rev. 6
E-106154 Sheet 10 Rev. 36
E-106154 Sheet 11 Rev. 4
E-106154 Sheet 12 Rev. 7

E-106155 Sheet 1 Rev. 20
E-106155 Sheet 2 Rev. 15
E-106155 Sheet 3 Rev. 27
E-106155 Sheet 4 Rev. 2
E-106155 Sheet 8 Rev. 18
E-106155 Sheet 9 Rev. 2
E-106155 Sheet 10 Rev. 34
E-106155 Sheet 11 Rev. 1

E-106156 Sheet 1 Rev. 21
E-106156 Sheet 2 Rev. 21
E-106156 Sheet 3 Rev. 19
E-106156 Sheet 4 Rev. 4
E-106156 Sheet 6 Rev. 7
E-106156 Sheet 7 Rev. 19
E-106156 Sheet 8 Rev. 29
E-106156 Sheet 9 Rev. 0
E-106156 Sheet 10 Rev. 25
E-106156 Sheet 11 Rev. 2

E-106157 Sheet 1 Rev. 27
E-106157 Sheet 2 Rev. 22
E-106157 Sheet 3 Rev. 27
E-106157 Sheet 4 Rev. 4
E-106157 Sheet 9 Rev. 3
E-106157 Sheet 10 Rev. 28
E-106157 Sheet 11 Rev. 1

E-106158 Sheet 1 Rev. 22
E-106158 Sheet 2 Rev. 14
E-106158 Sheet 3 Rev. 14
E-106158 Sheet 4 Rev. 8
E-106158 Sheet 6 Rev. 12
E-106158 Sheet 8 Rev. 18
E-106158 Sheet 10 Rev. 21

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Plant Design Drawings - Turbine Building
(Continued)

E-106159	Sheet 1 Rev. 33
E-106159	Sheet 2 Rev. 11
E-106159	Sheet 3 Rev. 19
E-106159	Sheet 4 Rev. 10
E-106159	Sheet 5 Rev. 6
E-106159	Sheet 6 Rev. 20
E-106159	Sheet 7 Rev. 5
E-106159	Sheet 8 Rev. 42
E-106159	Sheet 9 Rev. 11
E-106159	Sheet 10 Rev. 25
E-106159	Sheet 11 Rev. 2
E-106160	Sheet 1 Rev. 49
E-106160	Sheet 2 Rev. 19
E-106160	Sheet 3 Rev. 15
E-106160	Sheet 4 Rev. 12
E-106160	Sheet 5 Rev. 7
E-106160	Sheet 6 Rev. 13
E-106160	Sheet 7 Rev. 6
E-106160	Sheet 8 Rev. 42
E-106160	Sheet 9 Rev. 11
E-106160	Sheet 10 Rev. 4
E-106160	Sheet 11 Rev. 3
E-106161	Sheet 1 Rev. 42
E-106161	Sheet 2 Rev. 20
E-106161	Sheet 3 Rev. 16
E-106161	Sheet 4 Rev. 19
E-106161	Sheet 5 Rev. 14
E-106161	Sheet 6 Rev. 9
E-106161	Sheet 7 Rev. 18
E-106161	Sheet 8 Rev. 40
E-106161	Sheet 9 Rev. 15
E-106161	Sheet 10 Rev. 42
E-106161	Sheet 11 Rev. 14
E-106161	Sheet 12 Rev. 7
E-106162	Sheet 1 Rev. 32
E-106162	Sheet 4 Rev. 1
E-106162	Sheet 7 Rev. 7
E-106162	Sheet 8 Rev. 27
E-106162	Sheet 10 Rev. 11
E-106162	Sheet 12 Rev. 8
E-106162	Sheet 15 Rev. 16
E-106162	Sheet 16 Rev. 0
E-106162	Sheet 17 Rev. 2
E-106162	Sheet 19 Rev. 1

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5.3.2 System Specific Drawings

A. Reactor Feedwater Control System

Bechtel	M-106
	M-127 Sh. 1 and 2
	M-141
	E-121 Sh. 1-4
	E-179 Sh. 1
	E-126 Sh. 1-2
	E-128 Sh. 6
	J-427 Sh. 3
	E-127 Sh. 6

B. Recirculation Flow Control System

Bechtel	M-103
	M-105
	M-106
	M-115
	M-140
	M-143
	E-129 Sh. 17
	E-151 Sh. 1-8, 12
	J-105 Sh. 11
	J-115 Sh. 3
	J-406 Sh. 2

C. Reactor Water Cleanup Control System

Bechtel	M-144
	M-145
	E-164 Sh. 1-2
	E-165 Sh. 1-8
	E-170 Sh. 1
G.E.	8856-M1-933-153 Sh. 2,3

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D. Pressure Regulator and T/G Control

Bechtel	M-101
	M-105
	M-119
	E-114 Sh. 3-6
	E-120 Sh. 1-9
	E-121 Sh. 1-3
	E-122 Sh. 1-4
G.E.	8856-M2J-39 Sh. 1-7
	8856-M2J-6 Sh. 1-39
	8856-M2J-5 Sh. 1-9

E. Traversing In-Core Probe Control System

Bechtel	E-177 Sh. 4
G.E.	791E413AE Sh. 1-5

F. Reactor Manual Control System

Bechtel	M-146
	M-147
	E-158 Sh. 1-3
	E-169 Sh. 2-4

G. Nuclear Pressure Relief Control System

Bechtel	M-141
	M-142
	E-180 Sh. 1-3