15.0 ACCIDENT ANALYSES

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events. The calculational results contained in Appendix 15E are applicable to Cycle 1 of both units. Some of the accident analyses are cycle dependent and must be performed for each reload core. This chapter 15 contains the reload analysis results. The results of these reload analyses for the current cycles are documented in Appendices 15C and 15D for Units 1 and 2, respectively.

Appendix 15E contains information and analytical results for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system).

15.0.1 ANALYTICAL OBJECTIVE

The spectrum of postulated initiating events is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence; the limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines, without undue risk to the public health and safety.

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this report are discussed in individual categories as required by Reference 15.0-1. The results of the events are summarized in Tables 15C.0-1 and 15D.0-1 for the current cycles for Units 1 and 2. Table 15E.0-1 contains results of analyses that are for non-limiting events for the initial cycles for Units 1 and 2. Each event is assigned to one of the following applicable categories:

1. Decrease in Core Coolant Temperature:

   Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage.

2. Increase in Reactor Pressure:

   Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core-moderator thereby increasing core reactivity and power level which threaten fuel cladding due to overheating.

3. Decrease in Reactor Core Coolant Flow Rate:

   A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

4. Reactivity and Power Distribution Anomalies:

   Transient events included in this category are those which cause rapid increases in power which are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator increasing core reactivity and power level.

5. Increase in Reactor Coolant Inventory:

   Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

6. Decrease in Reactor Coolant Inventory:

   Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

7. Radioactive Release from a Subsystem or Component:

   Loss of integrity of a component which contains radioactivity is postulated.

8. Anticipated Transients Without Scram:

   In order to determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation plus multi-single active component failures (SACF) situation is postulated.

15.0.3 EVENT EVALUATION

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized based upon operating plant history for the transient event. Events for which inconclusive data exists are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of the following frequency groups:

1. Incidents of moderate frequency - these are incidents that may occur during a calendar year to once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational transient."

2. Infrequent incidents - these are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an "abnormal (unexpected) operational transient."

3. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a "design basis (postulated) accident."

4. Normal operation - operations of high frequency are not discussed here but are examined along with (1), (2), and (3) in the nuclear systems operational analyses in Appendix 15A.

15.0.3.1 .1 Unacceptable Results for Incidents of Moderate Frequency (Anticipated(Expected) Operational Transients)

The following are considered to be unacceptable safety results for incidents of moderate frequency (anticipated operational transients):

1. A release of radioactive material to the environs that exceeds the limits of 10CFR20.

2. Reactor operation induced fuel cladding failure.

3. Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes.

4. Containment stresses in excess of that allowed for the transient classification by applicable industry codes.

15.0.3.1.2 Unacceptable Results for Infrequent Incidents (Abnormal (Unexpected) Operational Transients)

The following are considered to be unacceptable safety results for infrequent incidents (abnormal operational transients):

1. Release of radioactivity which results in dose consequences that exceed a small fraction of 10CFR 50.67.

2. Fuel damage that would preclude resumption of normal operation after a normal restart.

3. Generation of a condition that results in consequential loss of function of the reactor coolant system.

4. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

1. Radioactive material release which results in dose consequences that exceed the guideline values of 10CFR 50.67.

2. Failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited.

3. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

4. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required.

15.0.3.2 Sequence of Events and Systems Operations

Each transient or accident is discussed and evaluated in terms of:

1. A step-by-step sequence of events from initiation to final stabilized condition.

2. The extent to which normally operating plant instrumentation and controls are assumed to function.

3. The extent to which plant and reactor protection systems are required to function.

4. The credit taken for the functioning of normally operating plant systems.

5. The operation of engineered safety systems that is required.

6. The effect of a single failure or an operator error on the event.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

For each event, the effect of single failures and/or operator errors is discussed. A plant operational analysis was performed prior to the initial startup of the units (see Appendix 15A). Although this information is historical in nature, it provided initial independent evaluation of the adequacy of systems as they related to the events under study.

Most events evaluated are already the results of single equipment failures or single operator errors that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operators errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs.

15.0.3.2.1.2 Initiating Event Analysis

1. The undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow)

   or

2. The undesired starting or stopping of any single component

   or

3. The malfunction or maloperation of any single control device

   or

4. Any single electrical component failure

   or

5. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

1. Those actions that could be performed by one person.

2. Those actions that would have constituted a correct procedure had the initial decision been correct.

3. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of single operator errors are as follows:

1. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

2. The selection and complete withdrawal of a single control rod out of sequence.

3. An incorrect calibration of an average power range monitor.

4. Manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indication.

15.0.3.2.1.3 Single Active Component Failure or Single Operator Failure Analysis

1. The undesired action or maloperation of a single active component

   or

2. Any single operator error where operator errors are defined as in Subsection 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4 describes the various fuel failure mechanisms. Avoidance of safety limits 1 and 2 (Subsection 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion has been established to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (see Reference 15.0-3). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than the safety limit established for the current cycle. The reactor steady state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

The methods for modeling and analyzing Units 1 and 2 are described in References 15.0-8 through 15.0-12. Determination of the steady-state operating limit is accomplished as follows:

1. The change in critical power ratio (ΔCPR) is calculated for each event.

2. The ΔCPR value is then added to the safety limit CPR value to result in the event based MCPR. The current cycle MCPR safety limits are given in Tables 15C.0-3 and 15D.0-3 for Units 1 and 2.

The operating limit MCPR is the maximum value of the event MCPRs calculated from the transient analysis. A set of plots of the MCPR Operating Limits (MCPROLs) as a function of core flow and as a function of core power is prepared. Separate plots are prepared that consider core exposure, operability of Recirculation Pump Trip, operability of Main Turbine Bypass, average scram speed, and single loop operation. These plots are prepared prior to the start of a new cycle and issued in the form of a Core Operating Limits Report (COLR). The COLR is prepared in accordance with the SSES Technical Specifications. The COLR for the current cycle of each unit is contained within the Technical Requirements Manual for each unit (see FSAR section 16.3).

Maintaining the CPR operating limit at or above this operating limit assures that the safety limit CPR is never violated for incidents of moderate frequency.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4 and Section 6.3.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general the limiting events analyzed within this section have values for input parameters and initial conditions as specified in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2. These tables include the current conditions for power uprate. Analyses which assume data inputs different than the power uprate values are designated accordingly in the appropriate event discussion. Table 15E.0-2 provides the initial conditions used for the analysis of the non-limiting events for the initial cycle for Units 1 and 2.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for most of the transient safety analyses at a core flow of 108 Mlbs/hr and a power given in Tables 15C.0-2 and 15D.0-2. However to assure that thermal margins are maintained over the entire power/flow operational space, the anticipated operational occurrences were analyzed over a range of power and flow conditions for the current cycles. In addition, single loop operation was analyzed for each of the anticipated operational occurrences and accidents. It was determined that for each anticipated event and the ASME overpressure analysis, the two loop results bound the results from single loop operation. Explicit analyses of LOCA and the pump seizure in single loop operation were also performed.

Figure 15E.0-1 is a typical power/flow map for a BWR. Power/flow maps for the current cycles for Units 1 and 2 are included in their COLR. The COLR for the current cycle of each Unit is contained within the Technical Requirements Manual for each Unit (see FSAR Section 16.3).

Referring to Figure 15E.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (105%, rod line A-D'), the lower bound is the zero power line H'-J', the right bound is the rated pump speed line A-H', and the left bound is either the minimum pump speed line D-J or the natural circulation line D'-J'.

The power/flow map, A-D'-J'-H-A, represents the acceptable operational constraints for anticipated operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100% , the power/flow map is truncated by the line B-C on Figure 15E.0-1 and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the maximum operating power level has to be limited, such as point F, the operating bounds would be F-M-D’-J’-L-K-F, provided that the MCPR operating limit is not violated by operating in this space. Similarly, if operating limitations are imposed by the analysis of a transient with an initial operating basis at point A, the power/flow boundary for 100% licensed power would be B-C-D'-J'-J-L-K-B. This power/flow boundary would be truncated by the MCPR operating limit, (for which there is no direct correlation to a line on the power/flow boundary), and by the constraints imposed by the safety analysis.

Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis.

This boundary may be truncated by the licensed power and other operating restrictions.

Certain localized events are evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event.

15.0.3.3.4 Results

A summary of the results of analytical evaluations are provided for each event. In addition, critical parameters are shown in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2.

15.0.3.5 Barrier Performance

This section primarily evaluates the performance of the Reactor Coolant Pressure Boundary (RCPB) and the Containment System during transients and accidents.

During transients that occur with no release of coolant to the containment only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

Containment integrity is maintained so long as internal pressures remain below the maximum allowable values. The design internal pressures are as follows:

Drywell (primary containment)	53 psig
Suppression Chamber (primary containment)	53 psig
Secondary Containment	7 in. H_2O

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the FSAR where the mechanical design features of systems and components are described. Design basis accidents are used in determining the sizing and strength requirements of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals either that the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

15.0.3.6 Radiological Consequences

In this chapter, the consequences of radioactivity released during the three types of events: a) incidents of moderate frequency (anticipated operational transients), b) infrequent incidents (abnormal operational transients), and c) limiting faults (design basis accidents) are considered. For all events whose consequences are limiting a detailed quantitative evaluation is presented. For non-limiting events a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults (design basis accidents) two quantitative analyses are considered:

1. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of bounding the event and determining the adequacy of the plant design to

meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

2. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis."

Results for both are shown to be within NRC guidelines.

Atmospheric Dispersion Parameters

Short-term site-specific X/Q's were calculated as described in Section 2.3. For the conservative case, the 0.5 percentile X/Q's were used in the dose calculations. The resultant offsite doses are conservative. For the realistic case, 50 percentile X/Q's were used. The valves are given in Table 2.3-92 and 2.3-105 for Units 1 and 2.

15.0.4 Nuclear Safety Operational Analysis (NSQA) Relationship

Appendix 15A is a comprehensive system-level, qualitative FMEA, relative to all the events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-reference of protective actions is an integral part of this chapter and the appendices.

Contained in Appendix 15A is a summary table which classifies events by frequency only (i.e., not just within a given category such as Decrease in Core Coolant Temperature).

15.0.5 REFERENCES

15.0-1 United States Nuclear Regulatory Commission Regulation Guide 1.70 Revision 2 (Preliminary), September 1975, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants, Light Water Reactor Edition."

15.0-2 "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973, (NEDO-10959 and NEDE-10958).

15.0-3 NUREG-0800, U.S. Nuclear Regulatory Commission Standard Review Plan, Section 4.4 Thermal and Hydraulic Design

15.0-4 Deleted

15.0-5 Deleted

15.0-6 Deleted

15.0-7 Deleted

15.0-8 XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis,” Exxon Nuclear Company, March 1983.

15.0-9 XN-NF-80-19(P)(A) Volume 3 Revision 2, “Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,” Exxon Nuclear Company, January 1987.

15.0-10 XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” Exxon Nuclear Company, June 1986.

15.0-11 XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, “XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,” Exxon Nuclear Company, February 1987.

15.0-12 ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation, August 1990.

Chapter 15.0

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Table 15.0-1

Chapter 15.0

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Table 15.0-1A

Table 15.0-2

Table 15.0-3

Table 15.0-4

Table 15.0-5

Chapter 15.0

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Table 15.0-1A

Table 15.0-2

Table 15.0-3

Table 15.0-4

Table 15.0-5

Chapter 15.0

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Table 15.0-1A

Table 15.0-2

Table 15.0-3

Table 15.0-4

Table 15.0-5

Chapter 15.0

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Appendix 15C, 15D, and/or 15E

Table 15.0-1A

Table 15.0-2

Table 15.0-3

Table 15.0-4

Table 15.0-5

Chapter 15.0

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Table 15.0-1A

Table 15.0-2

Table 15.0-3

Table 15.0-4

Table 15.0-5

Chapter 15.0

All Figures have been moved to
Appendix 15C, 15D, and/or 15E

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FIGURE 15.0-1, Rev. 54

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FIGURE 15.0-2, Rev. 54

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FIGURE 15.0-3A, Rev. 54

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Chapter 15.0

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Appendix 15C, 15D, and/or 15E

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FIGURE 15.0-3B, Rev. 54

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15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E contains information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.1.1 LOSS OF FEEDWATER HEATING

FANP NRC approved methods (Reference 15.1-2) have identified this event as a limiting event, and therefore, it has been evaluated for the current cycles for Units 1 and 2.

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater (Although applicable to some BWRs, this capability does not apply to Susquehanna).

The first case produces a relatively gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100^{o}F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control. However, the automatic flow control mode has been disabled for the SSES Units. For this reason, the loss of feedwater heating for current cycles has been analyzed only for the reactor in the manual control mode.

15.1.1.1.2 Frequency Classification

The probability of this event is considered low enough to be categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency.

This event is analyzed under worst case conditions of a 100°F loss and full power although a reduction of feedwater temperature of 100°F at high power has never been reported.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Tables 15C.1.1-1 and 15D.1.1-1 list the sequence of events for this transient for the reactor in the manual control mode for Units 1 and 2.

15.1.1.2.1.1 Identification of Operator Actions

If no automatic recirculation runback occurs, the reactor operator is to reduce power according to off-normal reactor operating procedures. The operator monitors the core for instabilities and monitors operating conditions versus the Power/Flow map. The operator then examines the operation of the feedwater heaters and takes necessary corrective action and resumes normal operation. If reactor scram occurs, although it is not predicted, the operator is directed to perform those actions listed in Table 15E.1.1-1.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that the plant instrumentation and controls, plant protection and reactor protection systems functioned normally.

The average power range monitor (APRM) provides the alarm to the operator, but no protection system trip is expected or required to mitigate the predicted consequences of this event.

Required operation of Engineered Safeguard Features is not expected for either of these transients.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The APRM alarm alerts the operator, however, the reactor requires no automatic trip. Therefore, single failures are not expected to result in a more severe event than analyzed. See Appendix 15A.

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The loss of feedwater heating is a relatively slow transient and has been conservatively analyzed by determining the final steady-state reactor operating condition assuming no operator or control system action were to occur. The analysis method is described in Reference 15.1-2. It has been shown that the method conservatively bounds the current fuel assembly designs.

Startup test data for the Susquehanna reactors show that a loss of feedwater heaters results in a drop in feedwater temperature on the order of 40°F, (Reference 15.1-3) which is considerably less than the 100°F drop in temperature assumed for the analysis performed here.

Therefore the analysis results reported herein will bound the expected transient conditions if this anticipated operational occurrence were to occur.

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2 and 15D.0-2 for the current cycles for Units 1 and 2.

The plant was analyzed at power levels of 100% and 50%. For the 100% power condition, a range of flows were analyzed. For the 50% power condition, the flow was assumed to be 62 Mlbs/hr.

Since this transient is relatively slow it can be conservatively analyzed by determining the change in the core thermal conditions based on initial and final steady-state conditions.

The transient is simulated by running an initial condition steady-state 3-D calculation and determining the MCPR and the corresponding eigenvalue for this state-point. The calculation is then repeated. The xenon distribution is kept the same as that determined for the initial state-point. The sub-cooling is changed to correspond to the value that would be obtained if the feedwater temperature decreased 100°F. The reactor power is increased until the eigenvalue matches the initial steady-state value. The MCPR is determined for this second condition and the ΔCPR is determined by taking the difference between this MCPR and the MCPR from the initial state-point.

15.1.1.3.3 Results

In manual mode, no compensation is provided by core flow and thus the power continues to increase. Vessel steam flow increases and the initial system pressure increase is small. The increased core inlet subcooling aids core thermal margins. ΔCPR's determined for the current cycles for Units 1 and 2 are given in tables 15C.0-1 and 15D.0-1.

The maximum steady-state power for this event is less than the 122% high neutron flux analytical setpoint. It was also determined that for the current cycles for this event, the LHGR power transient limit was not violated.

The thermal margin was also determined for reduced powers for two loop operation and single loop operation. The results indicate that the ΔCPRs for two loop operation are more limiting than for single loop operation and therefore can be used to establish the MCPR operating limits for single loop operation for this event.

15.1.1.3.4 Considerations of Uncertainties

The magnitude of the feedwater temperature change assumed was more severe than what is believed possible for the Susquehanna Units. Since the analysis conservatively assumed steady-state conditions at the end of the transient, realistic physics and thermal-hydraulic parameters were used.

15.1.1.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.1.2 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

This event has been identified as a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

15.1.2.1.2 Frequency Classification

This event is an incident of moderate frequency.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

With excess feedwater flow the water level rises to the high-level reference point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Tables 15C.1.2-1 and 15D.1.2-1 list the sequence of events for this transient for Units 1 and 2. Figures 15C.1.2-1 and 15D.1.2-1 show the changes in important variables during this transient for the current cycles for Units 1 and 2.

15.1.2.2.1.1 Identification of Operator Actions

(1) Observe that the high level feedwater pump trip has terminated the failure event.

(2) Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal.

(3) Identify causes of the failure and report all key plant parameters during the event.

15.1.2.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high water level tripping of the main turbine, turbine stop valve scram trip initiation, recirculation pump trip (RPT), and low water level initiation of the Reactor Core Isolation Cooling system (RCIC) and the High Pressure Coolant Injection system (HPCI) to maintain long-term water level control following the trip of the feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Tables 15C.1.2-1 and 15D.1.2-1, the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 set point. At this point in the logic a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature."

However, high moisture levels entering the turbine will be detected by high levels in the turbine's moisture separators which results in a trip of the unit.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. See Appendix 15A for further discussion.

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of the SSES Units 1 and 2. The methods for modeling and analyzing this event for Unit 1 and Unit 2 are described in References 15.1-5 and 15.1-6.

The nonlinear computer simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

a. An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks.

b. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

c. The steam lines are modeled using pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transient.

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, and pressure, are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

g. The control systems and reactor protection system are modeled.

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2 and 15D.0-2 for the current cycles for Units 1 and 2.

The transient model for the SSES Units 1 and 2 was initialized and executed for this event at one or more exposure steps for the current cycles. The initialization includes both the physics and thermal-hydraulic input that is exposure dependent. The Feedwater Controller Failure is analyzed for each of these exposures to determine the most limiting conditions for the cycles. The analyses are also performed over a range of power levels for Unit 1 and Unit 2. The flow was held constant at 108 Mlbs/hr. In general, the limiting initial condition for this event is full flow of 108 Mlbs/hr. If there is reason to believe that the limiting initial flow condition is other than full flow, additional analyses are performed at lower flows.

The analyses also consider the following:

1. Steam bypass and Recirculation Pump Trip operable,

2. Steam bypass inoperable and Recirculation Pump Trip operable,

3. Steam bypass operable and Recirculation Pump Trip inoperable.

4. Realistic Scram Insertion Time and Maximum Allowable Scram Insertion Time.

The analysis is performed using relief/safety valve setpoints corresponding to the “safety mode.”

The initiating event for this transient is the failure of the feedwater control system causing a step change of feedwater flow from its initial steady-state value to the maximum value of full power feedwater flow.

The results of these analyses are used to establish the MCPR operating limits as a function of power. These analyses are performed prior to the start of each cycle for SSES Units 1 and 2.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in Figures 15C.1.2-1 and 15D.1.2-1 for the current cycles of SSES Units 1 and 2. The high water level turbine trip and feedwater pump trip are initiated at the times shown in Tables 15C.1.2-1 and 15D.1.2-1 for Units 1 and 2. Scram occurs simultaneously from stop valve closure, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. If the turbine bypass system is operable, it opens to limit peak pressure in the steam line and the reactor pressure vessel. Peak pressures are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2. These pressures are well below the design limit of 1375 psig. Therefore the nuclear system process barrier pressure limit is not endangered.

The ΔCPRs for this transient are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2.

The bypass valves subsequently close to re-establish pressure control in the vessel during shutdown. The water level will gradually drop to the Low Level isolation reference point, activating the RCIC/HPCI systems for long term level control.

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response ((e.g., relief setpoints, scram stroke time (realistic and maximum allowable), and analytical setpoints for reactor protection system trips)). Plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual, therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.3 PRESSURE REGULATOR FAILURE - OPEN

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for power uprate conditions and the results are applicable to the current cycles for Units 1 and 2.

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 129% of rated flow. The maximum steam flow controller setting is 125% of the maximum flow through the turbine control valves, which is 103%, (turbine control valves fully open). Therefore failure of the controller to its maximum results in steam flow demand of 129% of rated flow. The percent flows listed above are based on a rated power of 3441 MWt.

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine control valves can be fully opened and the turbine bypass valves can be opened until the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

Tables 15C.1.3-1 and 15D.1.3-1 list the sequence of events for this transient for Units 1 and 2. Figures 15C.1.3-1 and 15D.1.3-1 show the transient behavior of important system parameters for Units 1 and 2.

15.1.3.2.1.1 Identification of Operator Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer operation to the backup controller in time to prevent the full transient. If the reactor scrams as a result of the isolation caused by low pressure sensed prior to the main turbine inlet (861 psig) in the run mode, the following is the sequence of operator actions expected during the course of the event. Once isolation occurs the pressure will increase to a point where the relief valves open. The operator should perform those scram actions indicated in Table 15E.1.1-1, and enter the appropriate Emergency Operating Procedure.

Prior to reactor restart, the operator should complete the scram report and initiate a maintenance work authorization of the pressure regulator.

15.1.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems except as described below.

Initiation of HPCI and RCIC system functions will occur when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized. If these events occur, they will follow some time after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization. Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single failure proof for initiation of MSIV closure.

Reactor scram sensing, originating from limit switches on the main steam line isolation valves, is designed to be single failure proof. It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated. See Appendix 15A for a further discussion.

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described in Reference 15.1-3 was used to simulate this event.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator output to a high value, which causes the turbine control valves to open fully. The model initiates the opening of the bypass valves by a trip signal (at the time the controller fails). The valve then opens based on an input table of valve position versus time. Since the controlling and backup regulator outputs are gated by a high value gate, the effect of such a failure in the backup regulator would be exactly the same. A regulator failure with approximately 130% of rated steam flow was simulated as a worst case.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steam line isolation initiation. This is within the specification limits of the valve and tends to aggravate the results of the analysis.

For purposes of the analysis the MSIV isolation on low main steam line pressure was conservatively set at its analytical value of 825 psig instead of the nominal value of 861 psig.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Tables 15C.0-2 and 15D.0-2.

15.1.3.3.3 Results

Figures 15C.1.3-1 and 15D.1.3-1 show the response of important nuclear system variables for this transient for Units 1 and 2. The turbine inlet pressure decreases to the low pressure trip setpoint and initiates trip of the MSIV's. Closure of the MSIV's initiates scram, and stops subsequent loss of steam to the main turbine and the feedwater turbines.

Reactor low turbine pressure trip limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier. After the rapid portion of the transient is complete the nuclear system safety/relief valves operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system and because the safety/relief valves operate only to relieve the pressure increase caused by decay heat, the nuclear system process barrier is not threatened by high internal pressure for this pressure regulator malfunction.

The ΔCPR was determined for this event for the power uprate condition and found to be on the order of 0.01 which indicates that this event does not significantly reduce core thermal margins, as noted above, and therefore does not need to be analyzed for each cycle.

15.1.3.3.4 Consideration of Uncertainties

If the maximum flow limiter were set higher or lower than normal, a faster or slower loss in nuclear steam pressure would result. The rate of depressurization may be limited by the bypass capacity. For example, the turbine control valves will open to the valves wide-open state admitting slightly more than the rated steam flow, and with the limiter in this analysis set at 125%, (130% rated steam flow), we would expect approximately 25% steam flow to be bypassed. This is essentially the maximum bypass flow and a faster rate of depressurization due to a pressure regulator failure is therefore not possible.

Depressurization rate has a proportional effect upon the voiding action of the core. If it is large enough, the sensed vessel water level trip set point (L8) may be reached initiating a turbine and feedwater pump trip early in the transient. Turbine trip will initiate reactor scram and shut down the reactor. Thermal margins will be better than a typical turbine trip event because of the power reduction initially experienced due to increased core voids in this event. Since the pressure regulator failure continues to signal the bypass to remain fully open, the turbine inlet pressure will drop below the low pressure isolation setpoint and the expected transient signature will conclude with an isolation of the main steam lines.

15.1.3.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed. Peak pressure in the bottom of the vessel is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure is shown in Figures 15C.1.3-1 and 15D.1.3-1 for Units 1 and 2.

15.1.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual, therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.4 INADVERTENT SAFETY/RELIEF VALVE OPENING

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle.

Inadvertent opening of a safety/relief valve can lead to two possible events. First, the valve may "open" and "re-close". This event has no significant effect on plant operation. Second, the valve may "open" and stick in the "open" position. This is the more limiting case and results in the plant transient discussed below.

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. Opening and closing circuitry at the individual valve level (as opposed to groups of valves) is subject to a single failure impact. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident but due to a lack of a comprehensive data basis, it is being analyzed as an incident of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-1 lists the sequence of events for this transient.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must "re-close" the valve as soon as possible and check that the reactor and T-G output return to normal.

15.1.4.2.2 Systems Operation

In this transient, the core performance analysis assumes normal functioning of plant instrumentation and controls, specifically the pressure regulator and level control systems. Additionally, normal operation of relief valve discharge line temperature sensors and the suppression pool temperature sensors provides operator information as the basis for initiating a timely plant shutdown.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in Chapter 15. In addition, a detailed discussion of such effects is given in Appendix 15A.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

It was determined that this event is not limiting from a core performance standpoint. Therefore a qualitative presentation of results is described below.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level of 3510 Mwt (4032 Mwt was used for dose analyses) when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 928,800 lbs/hr.

15.1.4.3.3 Qualitative Results

The opening of a safety/relief valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and therefore the safety limit margin is unaffected.

Continued maximum steam flow to the suppression pool will be terminated by operator action.

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following capability. RCPB and containment design limits are not exceeded.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

The activity released to the suppression pool chamber can be contained for some period of time. It is, therefore, assumed that the activity airborne above the suppression pool will be released under controlled conditions. The operator can choose to release the activity after decay has reduced the amount of activity to levels where the offsite dose consequence is minimal. For example, consider the case of a stuck open relief valve event with full MSIV closure which represents an upper bound on steam released to the suppression pool during an operational transient. The activity from this postulated event is released through the containment purge system at an assumed time of eight hours after the blowdown is complete (approximately 24 hours after the transient begins).

The containment airborne activity is discharged via the SGTS, with an assumed filter efficiency of 99 percent of the iodine activity. For this bounding example, the airborne activity above the suppression pool and the activity released to the environs are presented in Tables 15.1.-2 and 15.1-3 respectively. The offsite and control room radiological doses are given in Table 15.1.-4 for the maximum expected (realistic) and the design basis reactor coolant source terms. In both cases, the resultant doses are a small fraction of 10CFR20 limits detailed description of the control room model is provided in Appendix 15B. The input and assumptions used in the analysis are provided in Table 15.1-5.

15.1.5 SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT IN A PWR

This event is not applicable to BWR plants.

15.1.6 INADVERTENT RHR SHUTDOWN COOLING OPERATION

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

If the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from maloperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from maloperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15E.1.6-1.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation since the nuclear system pressure is too high to permit operation of the shutdown cooling mode of RHR.

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by maloperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

15.1.6.2.3 Effect of Single Failures and Operator Action

No single failures can cause this event to be more severe. If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, scram will terminate the power increase before thermal limits are reached and the operator will perform these actions in Table 15E.1.1-1. (See Appendix 15A for details.)

15.1.6.3 Core and System Performance

The increased subcooling caused by maloperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise would be terminated by a high flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here.

15.1.6.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed, therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Since this event does not result in any barrier failures, no analysis of radiological consequences is required for this event.

15.1.7 REFERENCES

15.1-1 Deleted

15.1-2 ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating in Boiling Water Reactors," Advanced Nuclear Fuels Corporation, September 2005.

15.1-3 PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis", Pennsylvania Power & Light, Issue Date: July 1992

15.1-4 Deleted

15.1-5 XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads" Exxon Nuclear Company, June 1986.

15.1-6 ANF-913(P)(A), Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

TABLE 15.1-1

SEQUENCE OF EVENTS FOR INADVERTENT
SAFETY RELIEF VALVE OPENING

TIME (SEC.)	EVENT
0	Opening of 1 safety relief valve, reaches full flow and remains open throughout the event.
1,200	Reactor scrammed on high suppression pool temperature. Technical Specification limit of 110°F. Closure of all MSIVs. Two loops of RHR suppression pool cooling placed into service.
5,200	Reactor depressurization initiated on high suppression pool temperature Technical Specification limit of 120°F.
59,200	Reactor depressurized to 14.7 psia, terminating blowdown through safety relief valve.

TABLE 15.1-2

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY ABOVE SUPPRESSION POOL (curies)

Isotope	Realistic	Design Basis
I-131	5.40E-01	2.22E+00
I-132	6.19E+00	2.54E+01
I-133	3.81E+00	1.54E+01
I-134	1.54E+01	6.19E+01
I-135	5.87E+00	2.38E+01
Kr-83m	2.53E+00	1.02E+01
Kr-85m	4.55E+00	1.83E+01
Kr-85	1.49E-02	6.01E-02
Kr-87	1.49E+01	6.01E+01
Kr-88	1.49E+01	6.01E+01
Kr-89	9.69E+01	3.91E+02
Xe-131m	1.12E-02	4.51E-02
Xe-133m	2.16E-01	8.72E-01
Xe-133	6.11E+00	2.46E+01
Xe-135m	1.94E+01	7.81E+01
Xe-135	1.64E+01	6.61E+01
Xe-137	1.12E+02	4.51E+02
Xe-138	6.63E+01	2.67E+02

TABLE 15.1-3

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY RELEASED TO THE ENVIRONS (curies)

Isotope	Realistic	Design Basis
I-131	5.24E-03	2.15E-02
I-132	5.55E-03	2.28E-02
I-133	2.91E-02	1.18E-01
I-134	2.75E-04	1.11E-03
I-135	2.53E-02	1.03E-01
Kr-83m	1.22E-01	4.92E-01
Kr-85m	1.32E+00	5.30E+00
Kr-85	1.49E-02	6.00E-02
Kr-87	1.90E-01	7.66E-01
Kr-88	2.11E+00	8.51E+00
Xe-131m	1.10E-02	4.42E-02
Xe-133m	1.94E-01	7.83E-01
Xe-133	5.84E+00	2.35E+01
Xe-135m	6.87E-09	2.76E-08
Xe-135m	8.90E+00	3.59E+01
Xe-138	4.20E-09	1.69E-08

TABLE 15.1-4

SAFETY RELIEF VALVE OPENING EVENT
RADIOLOGICAL DOSES (REM-TEDE)

Source Terms	EAB (REM-TEDE)	CRHE (REM-TEDE)
Realistic	1.87E-04	2.02E-04
Design Basis	7.54E-04	8.20E-04

Table 15.1-5

INADVERTENT SAFETY/RELIEF VALVE OPENING

Parameter	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source Term from Postulated Accident		
Core Thermal Power Level (MWt)	4032	4032
Fuel Damaged	None	None
Noble gas release rate (µCi/sec @ 30 min decay)	403,200	100,000
Iodine carryover fraction reactor water to steam (percent)	2% NWC 8% HWC	2% NWC 8% HWC
Radioiodine Chemical Species	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic
II. Data and Assumptions Used to Estimate Activity Released		
Main Steam Mass Release (lbs)	3.5E+06	3.5E+06
Time for Release (seconds)	57,700	57,700
Turbine Design Flow Rate 9Lbm/hr)	1.69E+07	1.69E+07
Release Timing	Instantaneous release	Instantaneous release
SGTS Filter Bed Depth (inches)	8	8
SGTS Filter Bed Efficiency (percent)	99	99
III. Data and Assumptions Used to Evaluate Control Room Doses		
CRHE Volume (ft^3)	518,000	518,000
Control Room Free Air Volume (ft^3)	110,000	110,000
CRHE Isolation	No isolation assumed	No isolation assumed
CRHE Air Intake Flow	5229 to 6391 cfm	5229 to 6391 cfm
CRHE Ingress/Egress Flow	10 cfm	10 cfm
CRHE Unfiltered Inleakage	500 cfm	500 cfm
IV. Dispersion Data		
Site Boundary (meters)	549	549
X/Q's for Site Boundary	Table 2.3-92 (50 percentile)	Table 2.3-92 (50 percentile)
X/Q's for CRHE	Appendix 15B	Appendix 15B
V. Dose Data		
Method of Dose Conversion	Appendix 15B	Appendix 15B
Dose Conversion Factors	Appendix 15B	Appendix 15B
Doses	Table 15.1-4	Table 15.1-4

Chapter 15.1

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Chapter 15.1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Chapter 15.1
Figures Moved to Appendix 15C, 15D, and/or 15E

FIGURE 15.1-1, Rev. 54

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Chapter 15.1

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Chapter 15.1
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FIGURE 15.1-2, Rev. 54

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Chapter 15.1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.1 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.1-3, Rev. 54

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Chapter 15.1

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FIGURE 15.1-4, Rev. 54

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FIGURE 15.1-5, Rev. 49

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15.1-6, Rev. 49

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15.2 INCREASE IN REACTOR PRESSURE

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E contains information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However, the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.2.1 PRESSURE REGULATOR FAILURE - CLOSED

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate setpoints to create proportional error signals that produce each regulator output. The output of both regulators feeds in a high gate value. The regulator with the highest output controls the main turbine control valves. The lowest pressure setpoint gives the largest pressure error and thereby the largest regulator output. The backup regulator is set 3 psi higher giving a slightly smaller error and a slightly smaller effective output of the controller.

It is assumed for the purposes of this transient analysis that a single failure occurs which erroneously causes the controlling regulator to close the main turbine control valves and thereby increases reactor pressure. If this occurs, the backup regulator is ready to take control as described in Subsection 15.2.1.2.1.

15.2.1.1.2 Frequency Classification

This event is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

Postulating a failure of the primary or controlling pressure regulator in the closed mode as discussed in Subsection 15.2.1.1.1 will cause the turbine control valves to close momentarily. The pressure will increase, because the reactor is still generating the initial steam flow. The backup regulator will reopen the valves and re-establish steady-state operation above the initial pressure equal to the setpoint difference of 3 psi.

15.2.1.2.1.1 Identification of Operator Actions

The operator will verify that the backup regulator assumes proper control. However, this action is not required as discussed below in Subsection 15.2.1.2.3.

15.2.1.2.2 Systems Operation

Normal plant instrumentation and control is assumed to function. This event requires no protection system or safeguard systems operation.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control. If the backup pressure regulator fails at this time, the turbine control valves (TCVs) will close in the servo or normal operating mode. Since the TCV closure is not a fast closure, there is no direct scram on closure. The reactor pressure will increase to the point that a flux or a pressure scram is initiated to shut down the reactor. Under these conditions the Recirculation Pump Trip (RPT) will occur if initiated by high dome pressure. Analyses have been performed for a failed pressure regulator (closed) with the backup pressure regulator out of service. The analyses were for a range of operating conditions and determined operational limits. These limits for the backup pressure regulator out of service are in the COLR.

15.2.1.3 Core and System Performance

The disturbance is mild, similar to a pressure setpoint change and no significant reductions in fuel thermal margins occur. This transient is much less severe than the generator and turbine trip transients described in Subsections 15.2.2 and 15.2.3.

15.2.1.3.1 Mathematical Model

Only qualitative evaluation is provided.

15.2.1.3.2 Input Parameters and Initial Conditions

Only qualitative evaluation is provided.

15.2.1.3.3 Results

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, in less than 2 seconds or so, due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip setpoints.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, and work characteristics). Plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.1.5 Radiological Consequences

Since this event does not result in any additional fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.2.2 GENERATOR LOAD REJECTION

The generator load rejection with the steam bypass system operable is a non-limiting event, and therefore, it is not explicitly analyzed each cycle. The analysis of this event described below was performed for the initial cycles for Units 1 and 2.

The generator load rejection with the steam bypass system failed is a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (T-G) rotor. Closure of the main turbine control valves will cause a sudden reduction in steam flow which results in an increase in system pressure and a reactor scram.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection With or Without Bypass

This event is categorized as an incident of moderate frequency.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection - Turbine Control Valve Fast Closure with Bypass Operable

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15E.2.2-1.

15.2.2.2.1.2 Generator Load Rejection with Failure of Bypass

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in Table 15C.2.2-1 and 15D.2.2-1 for the current cycles for Units 1 and 2.

15.2.2.2.1.3 Identification of Operator Actions

The operator should perform those actions listed in Table 15E.1.1-1.

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass Operable

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than P_{bypass}, where P_{bypass} = 26% power for Unit 1 and Unit 2. In addition, a recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

Same as Subsection 15.2.2.2.2.1 except that failure of the main turbine bypass valves is assumed for the entire transient. A number of other conservative assumptions are made when analyzing this event. These are listed in Subsection 15.2.2.3.2.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion.

An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The generator load rejection with bypass operable was analyzed for the initial cycle for Units 1 and 2 using the computer model described in References 15.2-5 and 15.2-10. The generator load rejection with bypass inoperable is a limiting event and is analyzed for each cycle. Commencing with Unit 2 Cycle 13 and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.2-11 and 15.2-12.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Tables 15C.0-2 and 15D.0-2.

The generator lockout relays initiate turbine control valve fast closure upon detection of load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 second. If the valves were to operate in the partial arc mode at powers above 90%, three turbine control valves would be fully open and one valve would be partially open. Therefore, closure time for the valves would be approximately 0.15 seconds. In the full arc mode at powers above 90%, four turbine control valves are approximately 50% to 60 % open and their closure time would be less than 0.15 seconds. Operation is currently in the full arc mode and is within the analysis performed.

Auxiliary power would normally be independent of any turbine-generator overspeed effects and continuously supplied at rated frequency since automatic fast transfer to auxiliary power supplies normally occurs. This is what is assumed for this analysis.

The reactor is operating in the manual flow-control mode when load rejection occurs. The SSES Units do not use automatic flow-control.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

The analyses for the current cycles for power levels above P_{bypass} for Units 1 and 2 were performed for the following conditions:

1. A range of power, flow, and exposures to establish the most limiting initial conditions.

2. Scram speed: Realistic and Maximum Allowable Scram times based on the Core Operating Limits Report.

3. Recirculation Pump Trip: Operable and Inoperable

4. Bypass: Failed.

5. Beginning with Unit 2 Cycle 13 and Unit 1 Cycle 15, the SRVs are assumed to open based on their safety valve pressure set points plus a 3% calibration tolerance. At least two SRVs at the lowest pressure settings are assumed to be out of service for EPU conditions.

Because of the similarity of the sequence of events between the generator load rejection without bypass and the turbine trip without bypass, the two events are analyzed as a single event with the additional assumptions described in Subsection 15.2.3 Turbine Trip.

The analysis of this event for power levels at P_{bypass} or less are described in Subsection 15.2.3.2.2.3, Turbine Trip at Low Power with Failure of Bypass.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15E.2.2-1 shows the results of the generator trip from rated power. The peak neutron flux and the peak average heat flux are given as a percent of rated power in Table 15E.0-1.

The MCPR does not significantly decrease below its initial value.

These results are less severe than the generator load rejection with failure of the bypass and are presented as typical results that are applicable to Units 1 and 2. The analyses were based on the initial cycle conditions for Units 1 and 2. These analyses have not been performed for the current cycles for Units 1 and 2.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

Figures 15C.2.2-1 and 15D.2.2-1 show the response of the key variables versus time for the generator load rejection without bypass for the current cycles for Units 1 and 2. Tables 15C.0-1 and 15D.0-1 provide peak neutron flux and peak heat flux for Unit 1 and Unit 2 for this event.

15.2.2.3.4 Consideration of Uncertainties

Typically, the actual full stroke closure time of the turbine control valve is 0.15 seconds. Clearly the less time it takes to close, the more severe the pressurization effect. For these analyses, it was assumed that the TSVs and the TCVs begin closing simultaneously. In the analysis of this event, the closure characteristics of these valves are modeled. The closure time of the turbine control valves will be dependent on the initial power level assumed for the event. However, in all cases the closure time determined by the model will be equal to 0.1 second (if the TSVs close first) or less than 0.1 second, (if the TCVs close first).

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram rod insertion time (realistic and maximum allowable), and analytical setpoints for reactor protection system setpoints). Plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection with Bypass Operable

Peak pressure remains within the normal safety range, and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressures for Unit 1 and Unit 2 are given in Tables 15C.0-1 and 15D.0-1. These peak nuclear system pressures are below the nuclear barrier transient pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.3 TURBINE TRIP

The turbine trip with the steam bypass system operable is a non-limiting event, and therefore, it is not explicitly analyzed each cycle. The analysis of this event described below was performed for the initial cycles for Units 1 and 2.

The turbine trip with the steam bypass system failed is a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

The turbine trip with the steam bypass system failed is defined as closure of the turbine stop valves followed almost immediately by closure of all turbine control valves with coincident failure of the turbine bypass valves to open. The generator load rejection without bypass is defined as the rapid closure of all of the turbine control valves followed by the closure of all of the turbine stop valves with coincident failure of the turbine bypass valves to open. The analysis of the generator load rejection without bypass and the turbine trip without bypass is performed as a single event by conservatively assuming simultaneous closure of the TCV's and TSV's (no time delays between the start of closure of the valves). The results of this analysis will bound the two events and a single set of results for the current cycles are reported in appendices 15C and 15D.

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lock out, loss of control fluid pressure, low condenser vacuum and reactor high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a by-product of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. In order to get an accurate event- by-event frequency breakdown, this type of division of initiating causes is required.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip with Bypass Operable

Turbine trip with bypass operable at high power produces the sequence of events listed in Table 15E.2.3-1. For the initial cycles of Units 1 and 2.

15.2.3.2.1.2 Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in Tables 15C.2.2-1 and 15D.2.2-1 for the current cycles for Units 1 and 2.

15.2.3.2.1.3 Identification of Operator Actions

The operator must perform those actions illustrated in Table 15E.1.1-1 .

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip with Bypass Operable

All plant control systems maintain normal operation unless specifically designated to the contrary.

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system.
Turbine stop valve closure initiates recirculation pump trip (RPT) thereby reducing the jet pump drive flow as the recirculation pumps coast down.

The pressure relief system, which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed.

It should be noted that below P_{bypass}, where P_{bypass} = 26% power for Unit 1 and Unit 2, a main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. The recirculation pump trip is also bypassed at power levels below P_{bypass}. All other protection system functions remain functional as before and credit is taken for those protection system trips.

15.2.3.2.2.2 Turbine Trip with Failure of the Bypass

Same as Subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.2.3 Turbine Trip at Low Power with Failure of the Bypass

Same as Subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

It should be noted that below P_{bypass}, the main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. The EOC-RPT is disabled below P_{bypass}. All other protection system functions remain functional as before and credit is taken for those protection system trips.

However, since it is assumed that the turbine bypass system fails and since the scram trip derived from the closure of the turbine stop valves is bypassed at power levels below P_{bypass}, this event may set the thermal limits for power levels below P_{bypass}. It therefore has been analyzed for the current cycles for Units 1 and 2.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than P_{bypass}

Mitigation of pressure increase is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than P_{bypass}

Same as Subsection 15.2.3.2.3.1 except RPT and stop valve closure scram trips are normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will also continue to function and scram the reactor should a single failure occur. However, to assure that thermal margins are maintained in the event of failure of the steam bypass system, this event is analyzed (See Subsection 15.2.3.3.3.3).

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in References 15.2-5 and 15.2-10 was used to simulate the turbine trip with bypass event. Commencing with Unit 2 Cycle 13 and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.2-11 and 15.2-12.

15.2.3.3.2 Input Parameters and Initial Conditions

The turbine trip with bypass operable and recirculation pump trip operable is a non-limiting event and was performed for the initial cycle for Units 1 and 2. The analyses of these non-limiting events used the plant conditions in Table 15E.0-2. The analyses of the turbine trip with failure of the bypass events have been performed, unless otherwise noted, with plant conditions tabulated in Table 15C.0-2 and 15D.0-2.

Turbine stop valves full stroke closure time is slightly greater than 0.1 second. A closure time of 0.1 seconds is used for the turbine stop valves in the simulation of this event. Note that the turbine control valves may be partially closed (depending on the initial power level), and since both the turbine control valves and the turbine stop valve closing characteristics are modeled, the turbine control valves may close slightly before the turbine stop valves. Therefore, the cessation of steam flow will occur in slightly less than or equal to 0.1 second.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below P_{bypass}.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry that trips the recirculation pumps.

Beginning with Unit 2 Cycle 13 and Unit 1 Cycle 15, the SRVs are assumed to open based on their safety valve pressure set points plus a 3% calibration tolerance. At least two SRVs at the lowest pressure settings are assumed to be out of service for EPU conditions.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip with Bypass Operable

A simulation of the turbine trip with the bypass system operating normally was performed for the initial core for Units 1 and 2 for conditions prior to power uprate. The results of this analysis are presented in Figure 15E.2.3-1.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited by the stop valve scram and the RPT system. Peak neutron flux and fuel surface heat flux are given in Table 15E.0-1 as a percent of rated power for the initial cycles for Units 1 and 2 for this event.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated as described in Subsection 15.2.3. The neutron flux and heat flux versus time are shown in Figure 15C.2.2-1 and 15D.2.2-1 for the current cycle for Units 1 and 2.

Peak neutron flux, peak average heat flux and ΔCPRs for this event are given in Tables 15C.0-1 and 15D.0-1 for Unit 1 and Unit 2.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

Below P_{bypass} the turbine stop valve closure and turbine control valve closure scrams are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by either high vessel pressure or high neutron flux. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief setpoints are reached.

For the analyses of this event, the opening of the safety/relief valves is assumed to occur based on the safety valve setpoints. At least two SRVs at the lowest pressure settings are assumed to be out of service for EPU conditions. Analyses are performed with the recirculation pump trip operable and inoperable. A power level P_{bypass} is analyzed for a number of initial core flows. Peak pressures are expected to slightly exceed the pressure safety valve setpoints and will be significantly below the RCPB transient limit of 1375 psig.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) In addition to using realistic scram speeds, the analyses were also performed using the slowest allowable control rod scram motion.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for over-pressure protection.

(4) Setpoints of the safety/relief valves include errors (high) for all valves.

(5) The analyses were performed at various cycle exposures to assure that the most limiting neutronic conditions are analyzed.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip with Bypass Operable

Since the turbine trip with the failure of the bypass is more severe, the turbine trip with bypass operable has not been analyzed for the current cycles for Units 1 and 2. The results of the analysis described below are based on the initial core for Units 1 and 2 and are shown in Figure 15E.2.3-1.

Peak pressure in the bottom of the vessel is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure and the pressure at the bottom of the vessel are given in Table 15E.0-1 for this event. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. Peak nuclear system pressure for the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig.

The peak pressure within the vessel for this event is given in Tables 15C.0-1 and 15D.0-1 for the current cycles for Units 1 and 2.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

The analysis of this event described in Subsection 15.2.3.3.3.3 used conservative assumptions to maximize the thermal transient. It was assumed that the safety/relief valves would open at their nominal safety setpoints plus 3%. At least two SRVs at the lowest pressure settings assumed to be out of service for EPU conditions. The analyses show that the reactor is scrammed on high pressure before any of the operable safety valves open. Based on this conservative analysis it is concluded that the pressure in the reactor vessel will be no higher than the lowest pressure setting for the operable safety valves. This pressure is below the 1375 psig reactor coolant pressure boundary transient limit.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.4 MSIV CLOSURES

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle. However, since the SSES Units have been licensed for EPU conditions, the MSIV closure events were analyzed to confirm that they remain non-limiting.

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

Various steam line and nuclear system malfunctions, or operator actions, can initiate main steam isolation valve (MSIV) closure. Examples are low steam line pressure, high steam line flow, high steam line radiation, low water level or manual action.

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All Main Steam Isolation Valves

This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not the byproduct of another transient, only the following contribute to the frequency: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum, etc.; and finally, equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all the other MSIVs depending on reactor conditions. If this occurs, it is also included in this category. During the main steam isolation valve closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines are less than 90%* open (except for interlocks which permit proper plant startup). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One Main Steam Isolation Valve

This event is categorized as an incident of moderate frequency. One MSIV may be closed at a time for testing purposes; this is done manually. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 80% when this occurs, a high flux or high steam line flow scram may result, (if all MSIVs close as a result of the single closure, the event is considered as a closure of all MSIVs).

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15E.2.4-1 lists the sequence of events for Figure 15E.2.4-1.

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*Changed to 85% with no significant impact on transient results.

15.2.4.2.1.1 Identification of Operator Actions

Table 15E.1.1-1 lists the sequence of operator actions expected during the course of the event.

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steam Isolation Valves

MSIV closures initiate a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the protection system.

The pressure relief system which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.2.2 Closure of One Main Steam Isolation Valve

The closure of a single MSIV at any given time will not initiate a reactor scram. This is because the valve position scram trip logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by initiation of the reactor scram via MSIV position switches and the protection system. Relief valves also operate to limit system pressure. All of these aspects are designed to single failure criterion and additional single failures would not alter the results of this analysis. Closure of one MSIV plus a single active component failure (the second MSIV) results in a situation no worse than the analysis of the four closed MSIVs.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 20 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1375 psig. The design basis and performance of the pressure relief system is discussed in Chapter 5.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in References 15.2-11 and 15.2-12 was used to simulate these transient events shown in Figure 15E.2.4-1 for EPU conditions.

15.2.4.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2.

The main steam isolation valves close in 3 to 5 seconds. A conservative 2-second closure time, is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90%* open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

Valve closure indirectly causes a trip of the main turbine and generator.

Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pump and initiate the HPCI and RCIC systems.

15.2.4.3.3 Results

15.2.4.3.3.1 Closure of All Main Steam Isolation Valves

Figure 15E.2.4-1 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 100% of rated steam flow for the initial cycles for Units 1 and 2. Peak neutron flux occurs within a few seconds of the start of the main steam line isolation valve closure. At this time, the nonlinear valve closure becomes a strong effect and the conservative scram characteristic assumption has not yet allowed credit for the full shutdown of the reactor.

Since credit is taken for the valve position switch scram for this event, this analysis for EPU conditions confirms that this event is non-limiting from either ΔCPR or pressure boundary considerations.

15.2.4.3.3.2 Closure of One Main Steam Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedures require an initial power reduction to approximately 80 to 90% of design conditions in order to avoid a high flux scram, a high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3-second closure of one main steam isolation valve during 100% rated power conditions, the steam flow disturbance raises vessel pressure and reactor power. This event is non-limiting from a Δ CPR or pressure boundary condition even at EPU conditions. The event was conservatively analyzed at EPU conditions with credit taken for the high pressure scram.

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*Changed to 85% with no significant impact on transient results.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in Subsection 15.2.4.3.3.1.

15.2.4.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For examples:

(1) Slowest allowable control rod scram motion is assumed.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for overpressure protection.

(4) Setpoints of the safety/relief valves are assumed to be 3% higher than the valve's nominal setpoint.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Isolation Valves

As shown in Table 15E.2.4-1, the nuclear system safety valves begin to open shortly after the start of isolation. The SRVs close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom is given in Table 15E.0-1 and is clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is also given in Table 15E.0-1.

15.2.4.4.2 Closure of One Main Steam Isolation Valve

If closure of the valve occurs at an unacceptably high operating power level, a flux or pressure scram will result; therefore, no significant effect is imposed on the RCPB. The main turbine bypass system will continue to regulate system pressure via the other three "live" steam lines.

15.2.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.5 LOSS OF CONDENSER VACUUM

PPL NRC approved methods (Reference 15.2-7) have identified this event as non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions which can cause a loss of condenser vacuum due to some single equipment failure are designated in Table 15E.2.5-1.

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15E.2.5-2 lists the sequence of events for Figure 15E.2.5-1.

15.2.5.2.1.1 Identification of Operator Actions

The operator must perform those actions illustrated in Table 15E.1.1-1.

15.2.5.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. Tripping functions incurred by sensing main turbine condenser vacuum pressure are designated in Table 15E.2.5-3.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to a general increase in reactor power level. Mitigation of power increase is accomplished by the protection system initiation of scram. Failure of the integrity of the condenser unit itself is considered to be an accident situation and is described in Subsection 15.7.1.

Single failures will not affect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single failure proof.

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

The computer model described in References 15.2-5 and 15.2-10 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with the plant conditions tabulated in Table 15E.0-2 unless otherwise noted. Turbine stop valves full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below P_{bypass}, where P_{bypass} = 26% power for Unit 1 and for Unit 2 rated power level.

The analysis presented here is a hypothetical case with a conservative 2.0 inches Hg per second vacuum decay rate. Thus, the bypass system is available for several seconds since the bypass is signaled to close at a vacuum level of about 10 inches Hg less than the stop valve closure.

15.2.5.3.3 Results

Under this hypothetical vacuum decay condition, the turbine bypass valve and main steam isolation valve closure would follow main turbine and feedwater turbine trips about 12 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam isolation valve closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Figure 15E.2.5-1 shows the transient expected for this event. It is assumed that the plant is initially operating at 105% of rated steam flow conditions (prior to power uprate). Peak neutron flux and peak average fuel surface heat flux are given in Table 15E.0-1 as a percent of rated value. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the main steamline isolation valves and bypass valves. While these are the major events occurring, other resultant actions will include scram (from stop valve closure) and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is directly dependent upon the rate at which the vacuum pressure is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problems produces a very slow rate of loss of vacuum (minutes, not seconds); (See Table 15E.2.5-1). If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs will occur.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

Other uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for over-pressure protection.

(4) Setpoints of the safety/relief valves are assumed to be at the upper limit of the Technical Specifications for all valves.

15.2.5.4 Barrier Performance

Peak nuclear system pressure at the vessel bottom is given in Table 15E.0-1 and is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure is also given in Table 15E.0-1. A comparison of these values to those for Turbine Trip with Bypass Failure at high power shows the similarities between these two transients. The primary differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.6 LOSS OF AC POWER

PPL NRC approved methods (Reference 15.2-7) have identified this event as non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Auxiliary Power Transformer

Causes for interruption or loss of the auxiliary power transformer power can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high current operation as well as operator error which trips the transformer breakers.

15.2.6.1.1.2 Loss of All Grid Connections

Loss of all grid connections can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities will cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

15.2.6.1.2 Frequency Classification

15.2.6.1.2.1 Loss of Auxiliary Power Transformer

This transient disturbance is categorized as an incident of moderate frequency.

15.2.6.1.2.2 Loss of All Grid Connections

This transient disturbance is categorized as an incident of moderate frequency.

15.2.6.2 Sequence of Events and Systems Operation

15.2.6.2.1 Sequence of Events

15.2.6.2.1.1 Loss of Auxiliary Power Transformer

Table 15E.2.6-1 lists the sequence of events for Figure 15E.2.6-1.

15.2.6.2.1.2 Loss of All Grid Connections

Table 15E.2.6-2 lists the sequence of events for Figure 15E.2.6-2.

15.2.6.2.1.3 Identification of Operator Actions

The operator should maintain the reactor water level by use of the RCIC and/or HPCI system, control reactor pressure by use of the relief valves and verify that the turbine dc oil pump is operating satisfactorily to prevent turbine bearing damage. Also, he should verify proper switching and loading of the emergency diesel generators.

Table 15E.1.1-1 lists the sequence of operator actions expected during the course of the events when no immediate restart is assumed.

15.2.6.2.2 Systems Operation

15.2.6.2.2.1 Loss of Auxiliary Power Transformer

This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the auxiliary transformer) provide the following simulation sequence:

(1) The recirculation pumps are tripped at a reference time, t=0, with normal coastdown times.

(2) At approximately 2 seconds, independent MSIV closure and scram are initiated due to loss of power to MSIV logic and actuator solenoids.

(3) At approximately 4 seconds, feedwater pump trips are initiated.

Operation of the HPCI and RCIC system functions are not simulated in this analysis. Their operation occurs at some time beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

15.2.6.2.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.2.2.1 with the following additional concern.

The loss of all grid connections is another feasible, although improbable, way to lose all auxiliary power. This event would add a generator load rejection to the above sequence at time t = 0. The load rejection immediately forces the turbine control valves closed, causes a scram, and initiates recirculation pump trip (RPT) (already tripped at reference time t = 0).

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the auxiliary power transformer in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to turbine trip occurring after the reactor scram has occurred. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single failure criteria and as such no change in analyzed consequences is expected.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The computer model described in References 15.2-5 and 15.2-10 was used to simulate this event.

Operation of the RCIC or HPCI systems is not included in the simulation of this transient, since startup of these pumps does not permit flow in the time period of this simulation.

15.2.6.3.2 Input Parameters and Initial Conditions

15.2.6.3.2.1 Loss of Auxiliary Power Transformer

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15E.0-2 and under the assumed systems constraints described in Subsection 15.2.6.2.2.

15.2.6.3.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.3.2.1.

15.2.6.3.3 Results

The results described below are for the initial cycles for Units 1 and 2. These events are non-limiting and have not been analyzed for the current cycles.

15.2.6.3.3.1 Loss of Auxiliary Power Transformer

Figure 15E.2.6-1 shows graphically the simulated transient. The initial portion of the transient is similar to the loss-of-feedwater transient. At 2 seconds MSIV's start to close and the reactor is scrammed. The feedwater turbines trip off at about 4 seconds.

The RHR system, in the shutdown cooling mode, is initiated to dissipate the heat. Sensed level drops to the RCIC and HPCI initiation setpoint at approximately 32 seconds after loss of auxiliary power.

There is no significant increase in fuel temperature or decrease in the operating MCPR value, fuel thermal margins are not threatened and the design basis is satisfied.

15.2.6.3.3.2 Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in Subsection 15.2.2. Figure 15E.2.6-2 shows graphically the simulated event. Peak neutron flux and peak fuel surface heat flux are given in Table 15E.0-1 as a percent of the initial value. There is no significant increase in fuel temperature.

15.2.6.3.4 Consideration of Uncertainties

The most conservative characteristics of protection features are assumed. Any actual deviations in plant performance are expected to make the results of this event less severe.

Operation of the RCIC or HPCI systems is not included in the simulation of the first 50 seconds of this transient. Startup of these pumps occurs in the latter part of this time period but these systems have no significant effect on the results of this transient.

Following main steam line isolation and RHR initiation the reactor pressure is expected to increase until the safety/relief valve set point(s) (5) are reached. At this time the valves operate in a cyclic manner to discharge the decay heat to the suppression pool.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Auxiliary Power Transformer

The consequences of this event do not result in any significant temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.6.4.2 Loss of All Grid Connections

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their set points. The pressure in the dome is limited to the value given in Table 15E.0-1, which is well below the vessel pressure limit of 1375 psig.

15.2.6.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.7 LOSS OF FEEDWATER FLOW

The loss of feedwater flow has been identified as non-limiting, and therefore, it was not explicitly analyzed each cycle. However, since the SSES Units have been licensed for EPU conditions, the loss of feedwater flow was analyzed to confirm that it remains non-limiting.

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as the high vessel water level (L8) trip signal.

15.2.7.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.2.7.2 Sequence of Events and Systems Operation

15.2.7.2.1 Sequence of Events

Table 15E.2.7-1 lists the sequence of events for Figure 15E.2.7-1.

15.2.7.2.1.1 Identification of Operator Actions

The operator should ensure RCIC and HPCI actuation so that water inventory is maintained in the reactor vessel and monitor reactor pressure.

Table 15E.1.1-1 lists the sequence of operator actions expected during the course of the event.

15.2.7.2.2 Systems Operation

Loss of feedwater flow results in a proportional reduction of vessel inventory causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. The reactor protection system responds within 1 second after this trip to scram the reactor. The low level (L3) scram trip function meets the single failure criterion.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

The nature of this event, as explained above, results in a lowering of vessel water level. Key corrective efforts to shut down the reactor are automatic and designed to satisfy single failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient.

The potential exists for a single relief valve failing to close once it is opened. This would result in a complete depressurization of the reactor. This is discussed in Subsection 15.1.4. Either the HPCI or RCIC system is capable of maintaining adequate core coverage and will provide inventory control.

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer model described in References 15.2-11 and 15.2-12 was used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2.

15.2.7.3.3 Results

The results of this transient simulation are shown in Figures 15E.2.7-1 through 4. Feedwater flow terminates at approximately 5 seconds. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure. Water level continues to drop until the vessel level (L3) scram trip set point is reached whereupon the reactor is shut down. The recirculation system is tripped and HPCI and RCIC operation are initiated due to vessel water dropping to the (L2) trip. Note, for this simulation only the RCIC was assumed to operate. Prior to reaching L1, water level starts increasing due to the RCIC flow entering the vessel. MCPR remains considerably above the safety limit since increases in heat flux are not experienced since the water level does not reach L1, the MSIVs do not close and system pressure remains low.

15.2.7.3.4 Considerations of Uncertainties

End-of-cycle scram characteristics are assumed.

This transient is most severe from high power conditions, because the rate of level decrease is greatest and the amount of stored and decay heat to be dissipated are highest.

Operation of the HPCI system is not included in the simulation of this transient. Operation of the RCIC is included in the simulation of the transient.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel is given in Table 15E.0-1 and is below the ASME Code limit of 1375 psig for the RCPB. Vessel dome pressure is also given in Table 15E.0-1. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.8 FEEDWATER LINE BREAK

(Refer to Subsection 15.6.6)

15.2.9 FAILURE OF RHR SHUTDOWN COOLING

Normally, in evaluating component failure considerations associated with the RHR Shutdown Cooling mode operation, active pumps or instrumentation (all of which are redundant for safety system portions of the RHR aspects) would be assumed to be the likely failed equipment. For purposes of worst case analysis, the single recirculation loop suction valve to the redundant RHR loops is assumed to fail. This failure would, of course, still leave two complete RHR loops for LPCI, suppression pool, and containment cooling minus the normal RHR Shutdown Cooling loop connection. Although the suction valve could be manually manipulated open, it is assumed failed indefinitely.

If it is now assumed that the SACF criteria is applied, the plant operator has one complete RHR loop available with the further selective worst case assumption that the other RHR loop is lost.

Recent analytical evaluations of this event have required additional worst case assumptions. These included:

(1) loss of all offsite ac power

(2) utilization of safe shutdown equipment only

(3) operator involvement no earlier than 10 minutes after coincident assumptions.

These accident-type assumptions would change the initial incident (malfunction of RHR suction valve) from a moderate frequency incident to a classification in the design basis accident status. However, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 102% of rated Thermal Power when a long-term loss of offsite power occurs, causing multiple safety-relief valve actuations (see Subsection 15.2.6 Loss of AC Power) and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power, an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. The operator then establishes a shutdown cooling path for the vessel through the ADS valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event. However, for the following reasons it could be considered an infrequent incident:

(1) Only a few RHR valves have failed in the shutdown cooling mode in BWR total operating experience.

(2) The set of conditions evaluated is for multiple failures as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15E.2.9-1.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are identical to those described in Table 15E.1.1-1 resulting in an isolation. The operator then proceeds to do the following:

(1) at approximately 10 minutes into the transient, initiate RPV shutdown depressurization at 100°F/hr by manual actuation of the safety/relief valves;

(2) at approximately 15 minutes into the transient, initiate suppression pool cooling (again for purposes of this "worst case" analysis, it is assumed that only one RHR heat exchanger is available);

(3) when the reactor pressure vessel is depressurized to approximately 100 psig, opens the RHR shutdown cooling system isolation valves. However, it is assumed that a failure occurs and the operator cannot open one of the isolation valves on the RHR suction line and the normal RHR shutdown cooling path is not established;

(4) selectively opens safety/relief valves (ADS) to complete blowdown and floods the vessel up through the safety/relief valves thereby establishing a closed cooling path as described in the notes for Figure 15E.2.9-1 and off normal procedures.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation credit is taken for the plant and reactor protection systems and/or the ESF utilized.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (Loss of Division Power) has already been analyzed in this event. Therefore, no single failure or operator error can make the consequences of this event any worse.

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action. The initial conditions used for evaluation of failure of RHR Shutdown Cooling are given in Table 15E.2.9-2.

15.2.9.3.2 Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using other, normal shutdown cooling equipment. In cases where both of the RHR shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15E.2.9-2). An evaluation has been performed assuming the worst single failure that could disable the RHR shutdown cooling valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The evaluation assures that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the safety function can be accomplished, assuming a worst-case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 15.2-1 and Figure 15E.2.9-1).

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. Even if it is additionally postulated that all of the ADS or relief valve discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell.

The systems have suitable redundancy in components such that, for onsite electrical power operation (assuming offsite power is not available) and for offsite electrical power operation (assuming onsite power is also not available), the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 psia, 200°F) conditions.

15.2.9.3.2.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCI systems, together with the nuclear boiler pressure relief system.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by transient event 15.2.6 (loss of A-C power), which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure and temperature are reduced to and maintained at saturated conditions at approximately 100 psig. The reactor vessel is depressurized by manually opening selected safety/relief valves. Reactor vessel makeup water is automatically provided via the RCIC/HPCI systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC, HPCI and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

15.2.9.3.2.2 Approximately 100 psig to Cold Shutdown

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

(1) the vessel is at 100 psig and saturated conditions;

(2) a worst-case single failure is assumed to occur (i.e., loss of a division of emergency power); and

(3) there is no offsite power available.

In the event that the RHR shutdown suction line is not available because of single failure, the first action to be taken will be for personnel to gain access and effect repairs. For example, if a single electrical failure caused a suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. If for some reason the normal shutdown cooling suction line cannot be repaired, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

The RHR shutdown cooling line valves are in two divisions (Division 1 = the outboard valve, and Division 2 = the inboard valve) to satisfy containment isolation criteria. For evaluation purposes, the worst-case failure is assumed to be the loss of a division of emergency power, since this also prevents actuation of one shutdown cooling line valve. Engineered safety feature equipment available for accomplishing the shutdown cooling function includes (for the selected path):

ADS	(DC Division 1 and DC Division 2)
RHR Loop A	(Division 1)
RHR Loop B	(Division 2)
HPCI	(DC Division 2)
RCIC	(DC Division 1)
Core Spray A	(Division 1)
Core Spray B	(Division 2)

For failures of Division 1 or 2, the following systems are assumed functional:

(1) Division 1 Fails, Division 2 Functional:

Failed Systems	Functional Systems
RHR Pumps A & C	HPCI
CS Loop A	ADS
RCIC	RHR Loop B
	CS Loop B
	RHR Pumps B & D

(2) Division 2 Fails, Division 1 Functional:

Failed Systems	Functional Systems
RHR Pumps B & D	CS Loop A
CS Loop B	RCIC
HPCI	RHR Loop A
	ADS
	RHR Pumps A &C

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15E.2.9-1. If the assumed single failure is Division 1, the safety function is accomplished by establishing one of the cooling loops described as Activity C1 of Figure 15E.2.9-1.

Using the above assumptions and following the depressurization transient shown in Figure 15E.2.9-3, the suppression pool temperature is shown in Figure 15E.2.9-4.

15.2.9.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described below.

15.2.9.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.10 REFERENCES

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15.2-6 Deleted

15.2-7 Deleted

15.2-8 Deleted

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Chapter 15.2

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Appendix 15C, 15D, and/or 15E

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Table 15.2-6

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-1, Rev. 54

AutoCAD Figure 15_2_1.doc

Chapter 15.2

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FIGURE 15.2-2, Rev. 54

AutoCAD Figure 15_2_2.doc

Chapter 15.2

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-3, Rev. 54

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FIGURE 15.2-4, Rev. 54

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Chapter 15.2

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Appendix 15C, 15D, and/or 15E

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-5, Rev. 54

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Chapter 15.2

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FIGURE 15.2-6, Rev. 54

AutoCAD Figure 15_2_6.doc

Chapter 15.2

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-7, Rev. 54

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Chapter 15.2

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-8, Rev. 54

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FIGURE 15.2-9, Rev. 54

AutoCAD Figure 15_2_9.doc

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FIGURE 15.2-10, Rev. 54

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FIGURE 15.2-11, Rev. 54

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Chapter 15.2

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-12, Rev. 54

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-13, Rev. 54

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-14, Rev. 54

AutoCAD Figure 15_2_14.doc

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FIGURE 15.2-15, Rev. 54

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FIGURE 15.2-11-1, Rev. 54

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FIGURE 15.2-11-2, Rev. 54

AutoCAD Figure 15_2_11_2.doc

Chapter 15.2

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.2 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.2-11-3, Rev. 54

AutoCAD Figure 15_2_11_3.doc

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E has information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the variable values do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However, the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.3.1 RECIRCULATION PUMP TRIP

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Recirculation pump motor operation can be tripped off by design for intended reduction of other transient core and RCPB effects as well as randomly by unpredictable operational failures. Intentional tripping will occur in response to:

(1) Reactor vessel water level L2 set point trip.

(2) TCV fast closure or Stop Valve closure.

(3) Failure to scram high pressure set point trip.

(4) Motor branch circuit over-current protection.

(5) Motor overload protection.

(6) Suction block valve not fully open.

Random tripping will occur in response to:

(1) Operator error.

(2) Loss of electrical power source to the pumps.

(3) Equipment or sensor failures and malfunctions which initiate the above intended trip response.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of One Recirculation Pump

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of Two Recirculation Pumps

This transient event is categorized as one of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

15.3.1.2.1.1 Trip of One Recirculation Pump

Table 15E.3.1-1 lists the sequence of events for Figure 15E.3.1-1.

15.3.1.2.1.2 Trip of Two Recirculation Pumps

Table 15E.3.1-2 lists the sequence of events for Figure 15E.3.1-2.

15.3.1.2.1.3 Identification of Operator Actions

15.3.1.2.1.3.1 Trip of One Recirculation Pump

Since no scram occurs for trip of one recirculation pump, no immediate operator action is required, unless the current power/flow condition is in either stability Region I or II of the power/flow maps for Units 1 and 2, (see COLR, FSAR Section 16.3). If the reactor is in either of these regions, the operator must take immediate action to avoid possible instability. Otherwise, as soon as possible, the operator must verify that no operating limits are being exceeded, and reduce flow of the operating pump to conform to the single pump flow criteria. Also, the operator must determine the cause of failure prior to returning the system to normal and follow the restart procedure.

15.3.1.2.1.3.2 Trip of Two Recirculation Pumps

If the reactor scrams with the turbine trip resulting from reactor water level swell, the operator should regain control of reactor water level through RCIC operation, monitoring reactor water level and pressure control after shutdown. When both reactor pressure and level are under control, the operator should secure both HPCI and RCIC as necessary. The operator must also determine the cause of the trip prior to returning the system to normal and perform those actions illustrated in Table 15E.1.1-1.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of One Recirculation Pump

Tripping a single recirculation pump requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.2.2 Trip of Two Recirculation Pumps

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems.

Specifically, this transient takes credit for vessel level (L8) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine stop valves. High system pressure is limited by the pressure relief valve system operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

15.3.1.2.3.1 Trip of One Recirculation Pump

Since no corrective action is required, other than that described in Subsection 15.3.1.2.1.3.1, no additional effects of single failures need be discussed. If additional SAEF or SOE are assumed (for envelope purposes the other pump is assumed tripped) then the following two pump trip analysis is provided.

15.3.1.2.3.2 Trip of Two Recirculation Pumps

Table 15E.3.1-2 lists the vessel level (L8) trip event as the first response to initiate corrective action in this transient. The level (L8) is intended to prohibit moisture carryover to the main turbine. Multiple level sensors are used to sense and detect when the water level reaches the L8 set point. At this point, a single failure will neither initiate nor impede a turbine trip signal. Turbine trip signal transmission circuitry, however, is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature." However, high moisture levels entering the turbine can cause vibration and trip the turbine via turbine supervisory instrumentation.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation.

15.3.1.3 Core and System Performance

15.3.1.3.1 Mathematical Model

The nonlinear, dynamic model described in Reference 15.3-4 was used to simulate this event for the initial cycles of Unit 1 and Unit 2.

15.3.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions (prior to power uprate) tabulated in Table 15E.0-2.

Pump motors and pump rotors are simulated with minimum specified rotating inertias.

15.3.1.3.3 Results

15.3.1.3.3.1 Trip of One Recirculation Pump

Figure 15E.3.1-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 5.7 seconds. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 143% of normal diffuser flow and 72% of rated core flow. MCPR remains approximately at the Operating Limit, thus the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram.

15.3.1.3.3.2 Trip of Two Recirculation Pumps

Figure 15E.3.1-2 shows graphically this transient with minimum specified rotating inertia. MCPR remains unchanged at the Operating Limit. No scram is initiated directly by pump trip. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip thereby shutting down the main turbine and feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. Subsequent events, such as initiation of RCIC and HPCI systems occurring late in this event, have no significant effect on the results.

15.3.1.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than expected under actual plant conditions. Actual pump and pump-motor drive line rotating inertias are expected to be somewhat greater than the minimum design values assumed in this simulation. Actual plant deviations regarding inertia are expected to lessen the severity as analyzed. Minimum design inertias were used as well as the least negative void coefficient since the primary interest is in the flow reduction.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Recirculation Pump

Figure 15E.3.1-1 results indicate a basic reduction in system pressures from the initial conditions. Therefore, the RCPB barrier is not threatened.

15.3.1.4.2 Trip of Two Recirculation Pumps

The high water level trip (L8) trips the turbine which causes the system pressure to increase.

The results shown in Figure 15E.3.1-2 indicate peak pressures stay well below the 1375 psig limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

There are no radiological consequences for a trip of one recirculation pump.

While the consequence of the trip of two recirculation pumps does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.2 RECIRCULATION FLOW CONTROL FAILURE-DECREASING FLOW

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

Three causes of a control failure are:

(1) Failure of an individual loop of recirculation motor generator set speed control logic (ICS) or positioning control of an individual scoop tube actuator can result in a rapid flwo decrease in only one recirculation loop.

(2) A Failure pf the common ICS logic inputting to the #1 and #2 Runback Limiters can generate a minimum speed demand signal to both recirculation flow control loops.

(3) A failure of the common ICS logic inputting to the Rundown logic can generate a decreasing speed demand bias signal maximum 15% to both recirculation flow control loops.

15.3.2.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Failure of One Speed Control Loop Including Failure of the Redundant Control Processor (CP) Pair

The sequence of events for this transient is similar to, and can not be more severe than that listed in Table 15E.3.1-1 for the trip of one recirculation pump.

15.3.2.2.1.2 Identification of Operator Actions

As soon as possible, the operator must verify that no operating limits are being exceeded. In particular, the operator must determine if the current power/flow condition is in operational regions I or II of the power/flow maps for Units 1 and 2, (Figures 15C.0-1 and 15D.0-1). If the reactor is in either of these regions, the operator must take immediate action to avoid possible instability. If any other operating limits are being exceeded, corrective actions must be initiated. Also, the operator must determine the cause of the trip prior to returning the system to normal.

15.3.2.2.2 Systems Operation

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip if it occurs. This is true for single and both pump speed controller failure events.

15.3.2.2.3 The Effect of Single Failures and Operator Errors

The single failure and operator error considerations for these events are the same as discussed in the section on recirculation pump trips, Subsection 15.3.1.2.3. Failure of two MG-sets and thus a double RPT or the common failure of digital recirculation pump speed controllers and thus a two RPT situation would be the envelope cases for additional SEF or SOE.

15.3.2.3 Core and System Performance

15.3.2.3.1 Mathematical Model

Since this event is less severe than the recirculation pump trips discussed in Subsection 15.3.1, this event was analyzed qualitatively.

15.3.2.3.2 Input Parameters and Initial Conditions

See Subsection 15.3.2.3.1.

15.3.2.3.3 Results

15.3.2.3.3-1

The ICS design for recirculation pump speed control incorporates a Signal Failure/Control System Fault feature to avoid the potential for uncontrolled reactivity excursions due to failed ICS hardware, interruption of control signal propagation, and self detected diagnostic faults. If detected, these faults result in a Scoop Tube Lock. Separation between recirculation loops ‘A’ and ‘B’ has been maintained within the ICS structure. Additional layers of redundancy and separation of functions exists within ICS such that single-failure criteria are maintained in most aspects. In the unlikely occurrence of a common failure resulting in complete loss of the control processing pairs within the Integrated Control System for both recirculation pump speed control, a zero demand signal will be established and both reactor recirculation pumps will go to minimum speed. This transient can never be more severe than the simultaneous trip of both recirculation pumps, evaluated in Subsection 15.3.1.3.3.2

15.3.2.3.3-2

In case of failure of one speed control loop, the scoop tube positioners are designed so that the flow change rate limit is determined by the individual stroking rate. The MG Set speed reduction is limited to less than approx. 25% per second due to the inherent design characteristic, mostly as a result of the systems mechanical inertia (e.g. scoop tube posititioner response and physical inertia of the MG set). This case is similar to the trip of one recirculation pump, described in Subsection 15.3.1.3.3.1, and is less severe than the transient that results from the simultaneous trip of both recirculation pumps.

15.3.2.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than otherwise expected. These analyses, unlike the pump trip series, will be unaffected by deviations in pump, pump motor and driveline inertias since it is the flow controllers that cause rapid recirculation decreases.

15.3.2.4 Barrier Performance

The barrier performance considerations for these events are the same as discussed in the section on recirculation pump trips.

15.3.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.3 RECIRCULATION PUMP SEIZURE

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a recirculation pump is considered as a design basis accident event. The analysis has been conducted with consideration of one and two loop operation.

In order to ensure compliance with the acceptance criteria for a design basis accident, a more conservative criterion, MCPR, is used to analyze this event. This approach assigns an initial MCPR for the event such that MCPR is always above the SLMCPR. By maintaining MCPR above the SLMCPR localized dryout within the fuel assembly is avoided, and fuel damage will not occur. If fuel damage does not occur, then the acceptance criteria regarding dose for a design basis accident are met.

Refer to Section 5.1 for specific mechanical considerations and Chapter 7 for electrical aspects.

The seizure event postulated would not be the mode failure of such a device. Safe shutdown components (e.g., electrical breakers, protective circuits) would preclude an instantaneous seizure event.

15.3.3.1.1 Identification of Causes

The case of recirculation pump seizure represents the extremely unlikely event of nearly instantaneous stoppage of the pump motor shaft of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the stopped rotor.

15.3.3.1.2 Frequency Classification

This event is considered to be a limiting fault.

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

For a pump seizure from two loop operation, the sequence of events for Unit 1 is given in Table 15C.3.3-1 and in Table 15D.3.3-1 for Unit 2. Figures 15C.3.3-1 and 15D.3.3-1 show the response of key variables following a pump seizure from two loop operation for Units 1 and 2, respectively. For single loop operation, the sequence of events for Unit 1 are given in Table 15C.3.3-2 and in Table 15D.3.3-2 for Unit 2. Figures 15C.3.3-3 and 15D.3.3-3 show the response of key variables following a pump seizure from single loop operation for Units 1 and 2, respectively.

15.3.3.2.1.1 Identification of Operator Actions

If the reactor were to scram, the operator would perform the actions listed in Table 15E.1.1-1. If necessary, the operator would regain control of reactor water level through HPCI and/or RCIC operation or by restart of a feedwater pump; and must monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection systems. The seizure of a recirculation pump results in a reactor water level swell. If the swell is of sufficient magnitude, the high vessel water level (L8) trip would be reached initiating a turbine trip and reactor scram. The analysis of the pump seizure assumes that neither a reactor scram occurs nor that the water level gets high enough to initiate a L8 trip.

Operation of safe shutdown features, though not included in this simulation, is expected to be utilized to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2. A trip due to high water level (L8) for this event is not expected (Figures 15C(d).3.3-1 and 15C(d).3.3-3).

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The pump seizure accidents from single loop and two loop operation were analyzed using the methods and model described in References15.3-1, 15.3-2 and 15.3-5.

15.3.3.3.2 Input Parameters and Initial Conditions

For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft.

The analysis for pump seizure from single loop operation was performed for an initial power level of approximately 2652 MWt and 52 Mlbs/hr core coolant flow. For the analysis with two loop operation, the initial conditions for Units 1 and 2 are given in Figures 15C.3.3-1 and 15D.3.3-1.

Also, the reactor is assumed to be operating at thermally limited conditions for each of the initial conditions analyzed. Note that the pump seizure occurs at 0.5 seconds as shown in Figures 15C.3.3-1 through 15C3.3-4 and 15D.3.3-1 through 15D.3.3.-4.

15.3.3.3.3 Results

The results of the analysis of the pump seizure accident are shown in Figures 15C.3.3-1 through 15C.3.3-4 and 15D.3.3-1 through 15D.3.3-4 for Units 1 and 2, respectively. The core coolant flow drops rapidly. The water level shows a small increase but falls back to its initial value. The power and heat flux all fall below their initial values as does the reactor dome pressure.

The assumed initial MCPR is set at a value to assure that the limiting MCPR reached during the transient does not fall below the SLMCPR for the pump seizure accident for either two loop or single loop operation. Figures 15C.3.3-2 and 15D.3.3-2 show the change in the MCPR with time for the pump seizure accident for two loop operation for Units 1 and 2, respectively. Figures 15C.3.3-4 and 15D.3.3-4 show the change in the MCPR with time for the pump seizure accident for single loop operation for Units 1 and 2, respectively.

To account for uncertainties in the model, the delta CPR determined from these analysis, Figures 15C.3.3-2, 15C.3.3-4, 15D.3.3-2 and 15D.3.3-4, were adjusted by approximately 14% for model uncertainties and 10% based on experience with pressurization events. An additional 0.05 was added to the resulting delta CPR to ensure that the limits being established will bound future 24 month cycles of ATRIUM-10 fuel. The assumed initial MCPR limits are shown in Table 15C.0-4 and 15D.0-4 for a range of SLMCPRs for Units 1 and 2.

15.3.3.3.4 Considerations of Uncertainties

Considerations of uncertainties are included in the methods of analysis described in References 15.3-1, 15.3-2 and 15.3-5.

15.3.3.4 Barrier Performance

The maximum pressures reached in the reactor coolant system for this accident are given in Tables 15C.0-1 and 15D.0-1. These pressures are within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 Radiological Consequences

Fuel damage is not expected for the pump seizure accident for either two loop or single loop operation. The SRVs are not expected to open during the accident; therefore, no reactor coolant will be released from the reactor to the primary containment.

While the consequences of the pump seizure accident does not result in fuel failure, it may result in the discharge of normal coolant activity to the suppression pool if SRVs are used to control reactor pressure following the accident. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.4 RECIRCULATION PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered a design basis accident event. It has been evaluated as a very mild accident in relation to other design basis accidents such as the LOCA. The analysis has been conducted with consideration to a single or two loop operation.

Refer to Chapter 5 for specific mechanical considerations and Chapter 7 for electrical aspects.

The shaft shearing event postulated certainly would not be the mode failure of such a device. Safe shutdown components (e.g., electrical breakers protective circuits) would preclude an instantaneous seizure event.

This postulated event is bounded by the more limiting case of recirculation pump seizure. Quantitative results for this more limiting case are presented in Subsection 15.3.3.

15.3.4.1.1 Identification of Causes

The case of recirculation pump shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the shaft-rotor condition.

15.3.4.1.2 Frequency Classification

This event is considered to be a limiting fault.

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one recirculation pump as discussed in Subsection 15.3.4.1.1 will cause the core flow to decrease rapidly resulting in water level swell in the reactor vessel. If the vessel water level reaches the high water level setpoint (Level 8), main turbine trip and feedwater pump trip will be initiated. Subsequently, reactor scram and the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by HPCI and RCIC flow.

15.3.4.2.1.1 Identification of Operator Actions

If the reactor were to scram, the operator would perform actions listed in Table 15E.1.1-1. If necessary, the operator would regain control of reactor water level through HPCI and/or RCIC operation or by restart of a feedwater pump; and he must monitor reactor water level and pressure control after shutdown.

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (Level 8) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the pressure relief system operation.

Operation of HPCI and RCIC systems is expected in order to maintain adequate water level control.

15.3.4.2.3 The Effect of Single Failures and Operator Errors

Effects of single failures in the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2.

Assumption of SEF or SOE in other equipment has been examined and this has led to the conclusion that no other credible failure exists for this event. Therefore the bounding case has been considered.

15.3.4.3 Core and System Performance

The severity of the pump seizure event is described in Subsection 15.3.3 and the pump seizure is more severe than the breakage of the recirculation pump shaft. This can be easily demonstrated by consideration of those two events as discussed in subsection below. Since this event is less limiting than the event described in Subsection 15.3.3, only qualitative evaluation is provided. Therefore no discussion of mathematical model, input parameters, and consideration of uncertainties, etc., is necessary.

15.3.4.3.1 Qualitative Results

If this extremely unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a trip of the main and feedwater turbines. Subsequently, A scram is initiated due to turbine trip. Since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, the threat to thermal limits is no more severe than described in Subsection 15.3.3. Additionally, the bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

The severity of this pump shaft break event is bounded by the pump seizure event (see Subsection 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump seizure event, the loop flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 Radiological Consequences

Since this accident is no more severe than the pump seizure accident the radiological consequences will be no more severe than those described in Subsection 15.3.3.5.

15.3.5 REFERENCES

15.3-1 NEDE-24011-P-A-14, June 2000, and U.S. Supplement, NEDE-24011-P-A-15, September 2005 and Amendment 26 “General Electric Standard Application for Reactor Fuel”.

15.3-2 NEDC-24154P-A, Revision 1, February 2000, Qualification of the 1-Dimensional Core Transient Model (ODYN) for BWRs” (Supplement 1-Volume 4).

15.3-3 Deleted

15.3-4 Linford, R.B., ”Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor”, April 1973 (NEDO 10802).

15.3-5 NEDC-32084P-A, Revision 2, July 2002 “TASC-03A – Computer Program for Transient Analysis of a Single Channel”.

Chapter 15.3

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Chapter 15.3

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Chapter 15.3

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Chapter 15.3
Figures Moved to Appendix 15C, 15D, and/or 15E

FIGURE 15.3-1, Rev. 54

AutoCAD Figure 15_3_1.doc

Chapter 15.3

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Appendix 15C, 15D, and/or 15E

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.3 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.3-2, Rev. 54

AutoCAD Figure 15_3_2.doc

Chapter 15.3

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.3 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.3-3, Rev. 54

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15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E contains information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.4.1 ROD WITHDRAWAL ERROR - LOW POWER

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the second most reactive rod (assuming most reactive rod is fully withdrawn) during refueling. The probability of the initial causes alone is considered low enough to warrant its being categorized as an infrequent incident, since refueling system interlocks and administrative controls will prevent an inadvertent Rod Withdrawal Error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal

During refueling operations both refueling system interlocks and administrative controls which ensure control rods are latched provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Movement With Control Rod Removed

Fuel movement and other core alterations with control rods removed will be controlled by the Technical Specifications. The Technical Specification requirements, the associated refueling interlocks, and administrative controls which ensure control rods are latched sufficiently minimize the possibility of loading fuel into a cell containing no control rod, moving the refueling platform over the core, and withdrawing additional control rods when there is uncontrolled fuel in the core.

15.4.1.1.2.3 Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the velocity limiter, does not physically permit upward removal of the control rod without simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

15.4.1.1.2.4 Identification of Operator Actions

As discussed above, refueling system interlocks and administrative controls are required to prevent this event. The administrative controls require that the operator verifies that all fully inserted control rods are latched.

15.4.1.1.2.5 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other Single Equipment Failure (SEF) or Single Operator Error (SOE), the necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is prevented by a combination of system design and administrative controls, the core and system performances were not analyzed. SDM calculations and tests ensure that the core remains subcritical with the highest worth control rod fully withdrawn. (See Subsection 4.3.2 for a description of the methods and results of the shutdown margin analysis.) Additional reactivity insertion due to control rod withdrawal is prevented by interlocks (See Subsection 7.6.1a.1) and administrative controls. As a result, no radioactive material is ever released from the fuel, making it unnecessary to assess any radiological consequences.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event and does not result in any change in the core pressure or temperature.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

Note: Unit 2 RSCS removed.

The event is defined as: while operating below the low power setpoint and coincident with a failure or bypass of the RWM the operator makes a procedural error and withdraws an out of sequence control rod of maximum worth. The probability of initial causes or errors of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the failure of the RWM system, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator ignorance of any alarm annunciations prior to safety system actuation. Whenever the RWM is inoperable or bypassed, there is a Technical Specification requirement that a second operator verify that the correct control rod withdrawal sequence is followed.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RWM plus procedural requirements prevent the operator from selecting and withdrawing an out-of-sequence control rod.

The purpose of the RWM is to control rod patterns during startup, such that only specified rod sequences and relative positions are allowed over the operating range from all control rods inserted to approximately 10% of rated core power. The sequences effectively limit the potential amount and rate of reactivity increase during a Control Rod Drop Accident. The RWM is designed to act as a backup to operator control of the rod sequences. Therefore if the RWM is inoperable or bypassed the Technical Specifications require that a second operator verify that any subsequent rod selection and withdrawal is in accordance with the specified rod sequence.

In the unlikely event that the RWM fails to prevent an out-of-sequence control rod from being withdrawn in the reactor startup range, fuel failure will not occur as shown by generic analyses performed by General Electric in Reference 15.4-10. Protection is provided by the IRM upscale scram function and/or APRM scram which are both single failure proof designed systems.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error followed by another SEF or SOE, the necessary safety actions are taken (e.g., rod blocks) prior to any limit violation.

15.4.1.2.3 Core and System Performance

The performance of the RWM and procedural requirements prevent erroneous selection and withdrawal of an out-of-sequence control rod. The core and system performance is not affected by such an operator error.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics of temperature and pressure.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.2 ROD WITHDRAWAL ERROR - AT POWER

This event has been identified as a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.4.2.1 Identification of Causes and Frequency Classifications

15.4.2.1.1 Identification of Causes

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and withdraws the maximum worth control rod until the Rod Block Monitor (RBM) System inhibits further withdrawal or the control rod is fully withdrawn.

15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this event is classified and analyzed as an incident of moderate frequency. Starting with U2C14 and U1C16 ARTS has been implemented and credit for the RBM is taken.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for the Rod Withdrawal Error (RWE) transient, as analyzed with conservative RWE assumptions, is presented in Tables 15C.4.2-1 and 15D.4.2-1. It is assumed that the operator takes no mitigating actions during the course of this event. Following the event, the operator will re-insert the control rod to reduce core power to rated conditions.

15.4.2.2.2 System Operations

The focal point of this event is localized to a small portion of the core. A discussion of the event follows below.

While operating in the power range in a normal mode of operation (except as noted in Subsection 15.4.2.3.2), the reactor operator makes a procedural error and withdraws the maximum worth control rod until either the RBM system inhibits further withdrawal or the rod is completely withdrawn.

Under most normal operating conditions no operator action is required since the transient which would occur would be very mild.

If the rod withdrawal error is severe enough, the rod block monitor (RBM) system will sound alarms, and block further withdrawal of the control rod.

The ARTS RBM is designed to block control rod withdrawal if localized thermal power in the vicinity of the control rod exceeds a predetermined trip set point. There are three reactor power ranges (low, intermediate and high) that have set points established that define their range. Each thermal power range has a localized thermal power trip set point to block control rod movement. The trip set points for each thermal power range are set by analysis to assure that the SLMCPR (point of boiling transition) and LHGR limit (centerline melt and 1% plastic strain limit) are not reached before the control rod movement is blocked. The power ranges and trip set points are specified in the COLR.

15.4.2.2.3 Effect of Single Failure and Operator Errors

The effect of operator errors has been discussed above. Termination of this event is assured by the RBM system or complete withdrawal of the control rod.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

For this transient the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibrate (i.e., that both the neutron flux and heat flux are in phase). Making use of the above assumption, this transient is calculated using a steady-state three-dimensional coupled nuclear-thermal-hydraulics computer program. All spatial effects are included in the calculation.

Commencing with Unit 2 Cycle 13 and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.4-14, 15.4-17, and 15.4-18.

The primary output from this code, in addition to the basic nuclear parameters, is: the variation of the linear heat generation rate (LHGR); the variation of the minimum critical power ratio (MCPR); the total reactor power; and the variation of the in-core instrumentation output during the transient.

The analytical methods and assumptions which are used in evaluating the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided.

The conservative assumptions are:

(1) The assumed rod withdrawal error is a continuous withdrawal of the maximum worth rod.

(2) The core is assumed to be operating at rated conditions. (See Tables 15C.0-2 and 15D.0-2 for Units 1 and 2).

(3) The reactor is presumed to be devoid of all xenon. This insures that the amount of excess reactivity which must be controlled by the movable control rods is maximized.

(4) Furthermore, it is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits, (MCPR limit and LHGR limit), in the fuel bundles in the vicinity of the rod to be withdrawn. It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate operator action or by numerous operator errors.

(5) It is assumed that the operator makes a procedural error and withdraws the maximum worth control rod until either the RBM system inhibits further withdrawal or the rod is completely withdrawn.

(6) In addition to the above conditions, it is possible that as the reactor power increases there could be a loss of pressure control which would result in a higher power level if the steam bypass is inoperable versus having it operable. Therefore, this event is analyzed for the conditions of the steam bypass operable and inoperable. Loss of pressure control is assumed to occur for steam flows greater than steam passing capability of the Turbine Control Valves.

The conservative assumptions indicated above provide a high degree of assurance that the transient as analyzed bounds all RWEs which could possibly occur.

With the implementation of ARTS the functioning of the RBM is now credited in the RWE analysis.

15.4.2.3.2.1 RBM System Operation

With ARTS implementation the RBM has three power dependent trip levels (rod withdrawal permissives removed). The trip levels are automatically varied with reactor power to protect against fuel damage. The initial RBM signal is normalized to a fixed (constant) reference signal. The trip levels are set at a fixed level above the reference and will vary as step functions of core power. This will allow longer rod withdrawals at low powers where thermal margins are higher and allow only short rod withdrawals at high power.

The ARTS based RBM uses an improved LPRM assignment. As opposed to the flow biased RBM system, this improved LPRM assignment provides readily predictable behavior and will limit thermal margin reduction during rod withdrawals without restricting rod withdrawals on the basis of core power level.

For each power range, (low, intermediate, and high) Allowable Values and Nominal trip set points are established by analysis to assure that the SLMCPR and LHGR limit are not reached before the control rod movement is blocked. The analysis for the rod withdrawal error also establish the MCPR limits when the RBM is inoperable. These set points and limits are included in the COLR.

15.4.2.3.3 Results

The ∆CPRs determined for this event for steam bypass operable are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2, respectively. The increase in the LHGR during the event was also determined for the condition of the steam bypass operable and inoperable. In general, for this event, the increase in the LHGR is less than the PAPT (Protection Against Power Transient) Limit for both Units 1 and 2. If, for a particular condition (e.g. bypass inoperable), the analytical results were to indicate that the PAPT Limit would be exceeded, a reduction in the normal steady state LHGR Limit would be established for that condition and would be recorded in the COLR. Maintaining the LHGR less than the PAPT Limit assures that the plastic strain limit of 1% for the cladding is not exceeded.

15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions which assure that this event has been conservatively analyzed have been previously discussed in Subsection 15.4.2.3.2.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 7% and the changes in pressure are negligible. If there is a loss of pressure control and the steam bypass system is inoperable the increase in core power is less than 14%. The increase in system pressure is small, (i.e., considerably less than that of the limiting transient for system overpressure evaluations) for this event, and therefore represents no threat to the Reactor Coolant Pressure Boundary.

15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This event is covered with evaluation cited in Subsections 15.4.1 and 15.4.2.

15.4.4 ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15E.4.4-1 lists the sequence of events for Figure 15E.4.4-1.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern as necessary for new power level following idle loop start.

(2) Determine that the idle recirculation pump suction and discharge bypass valves are open, the discharge valve is closed, and the scoop tube in the idle loop is in the starting position, if not, place them in this configuration.

(3) Readjust flow of the running loop downward to less than half of rated flow.

(4) Determine that the temperature difference between the two loops is no more than 50°F apart.

(5) Start the idle loop pump and allow pump speed and drive flow to reach a settled state.

(6) Open the discharge valve and slowly adjust pump speed.

(7) Readjust power, as necessary, to satisfy plant requirements per standard procedure.

Note: The time to do above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. In particular, credit is taken for high flux scram to terminate the transient. No ESF action occurs as a result of the transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level. Corrective action first occurs in the high flux trip and being part of the reactor protection system, it is designed to single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow so quickly after the postulated failure.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The nonlinear dynamic model described in Reference 15.4-15 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed unless otherwise noted with plant conditions tabulated in Table 15E.0-2.

One recirculation loop is idle and filled with cold water (100°F). Normal procedure when starting an idle loop with one pump already running requires heating the idle recirculation loop to within 50°F of core inlet temperature prior to loop startup.

The active recirculation loop is operating with about 80% of normal rated diffuser flow going across the active jet pumps.

The core is receiving 38% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

Reactor power is 55% of rated (prior to power uprate). Normal procedures require startup of an idle loop at a lower power.

The idle recirculation pump suction valve is open, but the pump discharge valve is closed.

The idle pump fluid coupler is at a setting which approximates 50% generator speed demand.

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15E.4.4-1. Shortly after the pump begins to move, a surge in flow from the standard jet pump diffusers causes the core inlet flow to rise sharply.

A short-duration neutron flux peak reaches the flow referenced APRM flux set point at 10 seconds and reactor scram is initiated. The neutron flux peak is given in Table 15E.0-1. Surface heat flux follows the slower response of the fuel and its peak value is also given in Table 15E.0-1. Nuclear system pressures do not increase significantly above initial. The water level does not reach the high set point.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for an initial power level that is much higher than that expected for the actual event. The much slower thermal response of the fuel mitigates the effects of the rather sharp neutron flux spike and even in this high range of power, no threat to thermal limits is possible.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15E.4.4-1.

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.5 RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

This event is a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

An unlikely Integrated Control System failure that results in the occurrence of a common failure of both speed control loops may in turn result in a speed increase of both recirculation pumps. The ramp rate of increase in the speed of the recirculation pumps is assumed to be slow.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Tables 15C.4.5-1 and 15D.4.5-1 list the sequence of events for the recirculation flow controller failure for Units 1 and 2.

15.4.5.2.1.1 Identification of Operator Actions

Initial action by the operator will include:

(1) Transfers flow control to manual and reduces flow to minimum.

(2) Identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The operator should perform those activities listed in Table 15E.1.1-1.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system, except for the APRM flow biased scram. The MG set electrical and mechanical stop design feature is also not credited. Operation of engineered safeguards is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a gradual rise in reactor power level. Corrective action occurs from either the high flux trip or the high pressure trip and, being part of the reactor protection system, these trips are designed to meet the single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow soon after the postulated failure.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

Since the transient is gradual, it has been demonstrated that a quasi-steady state analysis will yield acceptable results. A quasi-steady state analysis was performed for Unit 1 and Unit 2.

15.4.5.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2.

For this event a number of different Power/Flow conditions are analyzed. Commencing with Unit 2 Cycle 13, and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.4-14, 15.4-17, 15.4-18 and 15.4-19.

The Unit 1 and Unit 2 quasi – steady state analyses were performed for steam bypass operable and for it inoperable. The initial MCPR and LHGR were determined, using the approved three dimensional nodal simulation methods. The maximum core flow is determined based on the change in recirculation pump speed at different power levels.

For steam bypass operable, a flow excursion is assumed to occur that is equal to or greater than a normal rate of increase in reactor coolant flow and that the flow increases to that corresponding to the flow at maximum pump runout. It is also assumed that this flow excursion passes through the 100% power/ 100% flow statepoint. The fuel assembly power distributions are determined at a number of flow/power statepoints along the flow excursion path, including the condition of maximum core flow, using the three dimensional simulator. Using these power distributions, the MCPR for each statepoint is determined. The change in the MCPR along the flow path is used to establish the flow dependent CPR operating limits for Units 1 and 2. The fuel assembly power distributions are also used to determine whether or not the LHGR needs to be limited at lower powers and flows to prevent the LHGR limits from being exceeded at some point during the excursion.

For steam by-pass inoperable, steam flow will exceed the capacity of the turbine control valves and the power and pressure will rise until the reactor protection system trips on either high power or high steam dome pressure. The three dimensional simulator is run with the setpoints power and pressure set at their reactor protection system trip setpoints. The resulting MCPR and LHGRs are used as before to determine the ΔCPR and whether or not the LHGR limit is exceeded.

15.4.5.3.3 Results

The nuclear system pressure increase is limited by the high pressure analytical trip setpoint and operation of the safety/relief valves which are set to open at the nominal relief valve setpoints.

The peak neutron flux rise approaches the high neutron flux analytical trip setpoint. Since the transient is relatively slow, the change in heat flux is essentially the same as the change in neutron flux.

The ΔCPRs for this event are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2. The change in the LHGR was determined for this event and it was less than the Protection Against Power Transients (PAPT) Limit.

15.4.5.3.4 Considerations of Uncertainties

The analysis addresses uncertainties by conservatively setting the rate of increase in recirculation pump speed, using the maximum allowable Technical Specification scram insertion time, and using analytical set points for the high neutron flux trip and the high pressure trip. The conservative assumptions used in the quasi-steady state methodology account for any uncertainties.

15.4.5.4 Barrier Performance

This transient results in an increase in reactor vessel pressure slightly above the high pressure analytical trip setpoint and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTIONS

Not applicable to BWRs.

15.4.7 MISPLACED BUNDLE ACCIDENT

References 15.4-14, 15.4-17, and 15.4-18 describe the methodology used for cycles starting with U2C13 and U1C15. Bounding analyses have been performed for the ATRIUM-10 fuel for this event using this methodology. These analyses determined that for this event, for all expected operational conditions, less than 0.1% of the fuel rods will fail. Since less than 0.1% of the fuel rods in the core will fail, the radiological release will be less than a small fraction of that permitted by 10CFR Part 50.67. Also Design Bases analysis shows that the control rod drop accident analyzed in Subsection 15.1.9 is bounding. Therefore the analysis will not be performed for future Unit 1 and Unit 2 cycles fueled with ATRIUM-10 fuel.

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

One of the events discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the initial core loading. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to be overlooked and also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following core loading.

Another possible event is to misload the fuel assembly by rotating it either 90° or 180° from its proper orientation. For this event to occur, two operator errors will have to occur. The assembly will have to have been placed in its proper location but oriented incorrectly. The subsequent verification of the core loading will have to have overlooked the incorrect orientation of the fuel assembly handle.

15.4.7.1.2 Frequency of Occurrence

It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle.

Similarly the placement of an assembly with the incorrect orientation is assumed to be in a location(s) that yield the largest change in LHGR and ΔCPR.

Neither of these events have occurred at the SSES Units. These events are categorized as infrequent incidents.

15.4.7.2 Sequence of Events and Systems Operation

The postulated sequence of events for the misplaced bundle accident (MBA) and the Rotated Bundle Accident (RBA) is presented in Table 15.4-12 for Units 1 and 2.

Fuel loading errors, undetected by in-core instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed, and therefore, no corrective operator action or automatic protection system functioning occurs.

15.4.7.2.1 Effect of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three Single Equipment Failure (SEF) or Single Operator Error (SOE) or two SEF or SOE for the Rotated Bundle Accident) and there are no further operator errors which can make the event results any worse.

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

The methodology used for the bounding analyses applied to U2C13 U1C15 and subsequent cycles is described in References 15.4-14, 15.4-17, and 15.4-18.

A three-dimensional steady-state BWR simulator model is used to calculate the core performance resulting from the misplaced bundle accident. For the analysis of the rotated bundle accident, the change in local peaking in the bundle due to the change in water gap width is modeled using a two dimensional lattice code in conjunction with a three-dimensional steady-state BWR simulator.

15.4.7.3.2 Input Parameters and Initial Conditions

Misloaded Fuel Bundle

By placing a misloaded fresh fuel assembly face adjacent to other fresh assemblies, excessive power peaking occurs in the mislocated and surrounding bundles near the middle and through to the end of cycle. At the beginning of a cycle the worst mislocation will be caused by placing a misloaded once burned assembly, which is at or near peak reactivity, adjacent to as many other once burned assemblies as possible. In addition, the limiting once burned assembly is assumed to be misloaded adjacent to the limiting MCPR assembly at Beginning of Cycle (BOC). The results of both instances are lower CPRs and higher LHGRs for the mislocated and immediately surrounding bundles.

The analysis is performed by examining the cycle at separate exposure steps throughout the cycle to determine the most severe consequences of either a misloaded fresh fuel assembly or a misloaded once burned assembly. The analysis also considers the effect of fuel designs with axially varying enrichments and/or gadolinia loadings.

At each exposure step examined, the core MCPR and the limiting LHGR with the mislocated bundle is compared to the MCPR and the LHGR of the properly loaded core to determine the ΔCPR and whether or not the LHGR limit is violated. This determination is made with nominal operating control rod patterns. No credit is taken for the ability of the core monitoring system to detect a power distribution anomaly due to a mislocated bundle.

Rotated Bundle

Another possibility for misloading a fuel bundle is to load the bundle rotated by 90° or 180° from its correctly loaded orientation.

Due to the location of the channel spacer buttons, a rotated bundle will be tilted towards the adjacent fuel bundles in the same control cell. When the bundle is rotated and inserted into the core, these buttons contact the top guide and push the top of the bundle toward the center of the control cell. The tilt increases the size of the inter-assembly water gaps along the sides of the bundle adjacent to the core top guide and decreases the size of the water gaps along the sides of the fuel bundle adjacent to the control rod. This change in water gap size changes both the reactivity of the fuel bundle and the local pin power distribution within the bundle.

The change in pin power distribution will affect the MCPR and the LHGR for this assembly and its neighbors.

The bounding analyses performed for the Rotated Bundle took into consideration variations of pin power distributions as a function of water gap size and variations in fuel assembly lattice design including enrichment and gadolinia loadings.

To analyze this event the change in pin power distributions are determined assuming a conservatively wide water gap exists for the rotated fuel bundle. For fuel with axially varying enrichments and/or gadolinia loadings, the individual lattices are analyzed. The change in pin power distribution is evaluated at separate exposure steps throughout the cycle. The effect of change in pin power distribution on MCPR and LHGR are evaluated using the 3D core simulation program.

Since the Rotated Bundle is considered an infrequent incident, some fuel damage is permitted provided that the resulting radiological consequences will be a small fraction of the 10FR100 limits. To ensure that these dose limits are met, the analysis methodology established the criteria that at least 99.9% of the fuel rods in the core will avoid boiling transition and the cladding of these fuel rods will not exceed the 1% plastic strain criterion.

15.4.7.3.3 Results

Misloaded Fuel Bundle and Rotated Bundle

The Mislocated Bundle and the Rotated Bundle bounding analyses confirmed that for all normal operational conditions with ATRIUM-10 fuel, 99.9% of the fuel rods in the core avoid boiling transition and the fuel rods will not exceed the 1% plastic strain criteria.

15.4.7.3.4 Considerations of Uncertainties

In order to assure the conservatism of the analysis of the mislocated bundle, major input parameters are taken as a worst case, i.e., the bundle is placed in the location with the highest bundle power in the core and the bundle is operating on design thermal limits.

For the analysis of the rotated bundle it is assumed that the bundle is placed in the location with the highest LHGR and/or the lowest CPR in the core and the bundle is operating on design thermal limits

This assures that the ΔCPR and the LHGR are the upper bounds for the mislocated and the rotated bundle errors.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a very mild and highly localized event. No perceptible change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

Analyses confirmed that for all normal operating conditions with ATRIUM-10 fuel, 99.9% of the fuel rods in the core avoid boiling transition and the fuel rods will not exceed the 1% plastic strain criteria for both the Misloaded Bundle and the Rotated Bundle events. Any radiological doses resulting from these events would be a small fraction of the 10CFR50.67 regulatory dose limits. The regulatory dose consequences from these events are bounded by the control rod drop event given in Subsection 15.4.9.

15.4.8 SPECTRUM OF ROD EJECTION ASSEMBLIES

Not applicable to BWRs.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the control rod drive system assemblies. The control rod drive housing support assemblies are described in Chapter 4.

15.4.9 CONTROL ROD DROP ACCIDENT (CRDA)

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident is the result of a postulated event in which a high worth control rod is inserted in-sequence into the core. Subsequently, it becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later optimum moment, the control rod suddenly falls free and drops out of the core. This results in the insertion of large positive reactivity to the core and causes a localized power excursion.

A more detailed discussion is given in Reference 15.4-1.

15.4.9.1.2 Frequency Classification

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant; but, if postulated to occur, it has consequences that include the potential for the release of radioactive material from the fuel.

15.4.9.2 Sequence of Events and System Operation

15.4.9.2.1 Sequence of Events

Before the control rod drop accident (CRDA) is possible, the sequence of events presented in Tables 15.4-1 must occur. No operator actions are required to terminate this transient.

15.4.9.2.2 Systems Operation

Note: Unit 2 RSCS removed.

The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The RWM restricts control rod patterns during startup, such that only specified rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% of rated core power. The sequences effectively limit the potential amount and rate of reactivity increase during a Control Rod Drop Accident. The RWM is designed to act as a backup to operator control of the rod sequences. Therefore if the RWM is inoperable or bypassed, the Technical Specifications require that a second operator verify that any subsequent rod selection and withdrawal is in accordance with the specified rod sequence.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Other plant instrumentation and controls are assumed to function normally. The main condenser is assumed to isolate on main steam line high radiation (i.e., isolation of mechanical vacuum pump and steam jet air ejectors occurs). No credit is taken for other plant instrumentation and controls.

15.4.9.2.3 Effect of Single Failures and Operator Errors

Systems mitigating the consequences of this event are the RWM and APRM scram. The RWM in conjunction with the Technical Specification requirement that a second operator verify the control rod sequence and withdrawal if the RWM is bypassed essentially provides a redundant system and therefore together provide single failure protection. The APRM scram system is designed to single failure criteria. Therefore, termination of this transient within the limiting results discussed below is assured.

No operator error (in addition to the one that initiates this event) can result in a more limiting case since the reactor protection system will automatically terminate the transient.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

A method for determining the energy deposition in the UO_2 fuel in the fuel bundles surrounding the dropped control rod is described in References 15.4-14 and 15.4-16. The necessary input data for this method are:

Delayed Neutron Fraction
Doppler Coefficient
Bundle Peaking Factors
Worth of the Dropped Control Rod.

The method allows one to determine the number of fuel bundles that have an energy deposition of 170 cal/gm or more. Assemblies which have this amount or more of energy deposited in the fuel are assumed to fail. The method also allows one to determine the maximum energy deposition in any fuel bundle. This value must be less than 280 cal/gm.

Commencing with Unit 1 Cycle 15 and Unit 2 Cycle 13, the methods for modeling and analyzing this event are described in References 15.4-14, 15.4-17, and 15.4-18. These methods and models were used to determine the maximum control rod worth, Doppler coefficient, delayed neutron fraction, and fuel bundle peaking factors that are needed to assess the effects of the control rod drop accident.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop accident is assumed to be at the point in cycle which results in the highest control rod worth, to contain no xenon, and to be in a hot startup condition. The highest control rod worth is assumed to occur at one of the following core conditions:

a. Peak hot zero power reactivity
b. Peak hot excess reactivity
c. Maximum control rod density

The control rod sequence and rods withdrawn are shown in Tables 15C.4.9-2 and 15D.4.9-2 for the current cycles for Units 1 and 2. Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods. Control rod density of up to 50%("black and white" rod pattern), which nominally occurs at the hot-startup condition, is also assumed. This assumption ensures that withdrawal on the next rod results in the maximum increment of reactivity.

Reference 15.4-6 limits the maximum enthalpy for the control rod drop accident to 280 calories per gram. Therefore the maximum incremental rod worths are maintained at very low values so that a postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition.

15.4.9.3.3 Results

The radiological evaluations are conservatively based on the assumed failure of 2000 fuel rods. The number of rods which exceed the damage threshold is less than 2000 for all plant operating conditions.

The results of the compliance-check calculations, are shown in the Tables 15C.4.9-2 and 15D.4.9-2 for the current cycles for Units 1 and 2. These tables show the maximum incremental rod worth, the number of fuel rods that exceed an energy deposition of 170 cal/gm, the number of fuel rods that are assumed to have failed and the maximum energy deposition in the fuel bundles surrounding the dropped control rod. The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 2000 pins for the radiological evaluation is conservative.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance was not made for this accident since this is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.5 Radiological Consequences

Two cases are analyzed for the DRDA. The first case follows the guidance in Regulatory Guide 1.183 and is applicable when the plant is at power. Rods with 170 cal/gm or more energy deposited in the fuel are assumed to fail. No fuel melting is expected to occur at this energy deposition value. However, this analysis conservatively assumes that 0.77% of the fuel within a failed fuel rod melts. This fuel melt assumption is intended to ensure compatibility with the same assumption made in GE's Topical Report NEDO-31400A, which evaluated the elimination of certain main steam radiation monitor safety functions (Reference 15.4-7). The second case potentially occurs during low power operation with the mechanical vacuum pump (MVP) running. At low power with fewer than 30 rods failing, main steam line dose rates may be too low to be reliably sensed by the Main Steam Line Radiation Monitors to generate a trip signal for the mechanical vacuum pump. Failure to trip the vacuum pump would result in an unfiltered release of fission products to the environment from the turbine building vent stack.

Two separate radiological analyses are provided for each of these cases:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR50.67 guidelines. This analysis is referred to as the "Design Basis Analysis."

(2) The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis."

A schematic of the leakage path is shown in Figure 15.4-1.

Specific parametric values used for the design basis and the realistic analyses are presented in Table 15.4.-11.

15.4.9.5.1 Design Basis Analysis

The design basis analysis is based on Regulatory Guide 1.183 (Reference 15.4-6). The RADTRAD Computer Program (Reference 15.4-8) is used to evaluate the radiological consequences for the design basis analysis.

It is assumed that 10 percent of the iodines and noble gases and 12 percent of the Cs and Rb contained in the gap of the fuel rods that experience cladding damage are released from the fuel. The release from the fuel melting is assumed to be 100% of the noble gases and 50% of the iodines within the region that melts per Regulatory Guide 1.183, Appendix C.1. Solids released from the melted fuel are in accordance with the fractions shown in Regulatory Guide 1.183, Table 1.

Activity released from the fuel is assumed to instantaneously mix in the reactor coolant. Of this activity, 100% of the noble gases, 10% of the radioiodine, and 1% of the remaining nuclides are assumed to reach the condenser and turbine. Of the activity reaching the condenser and turbine, 100% of the noble gases, 10% of the iodine, and 1% of the particulate nuclides are available for release to the environment. The turbine and condenser leak to the atmosphere at 1% per day for 24 hours, after which time, the leakage is assumed to terminate. No credit is taken for holdup in the turbine building.

IF the mechanical vacuum pump is running, it removes activity from the main condenser at the rate of 1212%/day. The rate is based on a realistic condenser volume of 195,000 ft^3 and extraction of non-condensable gases at 75 cfm by the mechanical vacuum pump. Radioiodine chemical species released from the main condenser/turbine is 97% elemental and 3% organic iodine.

The activity airborne in the condenser is presented in Tables 15.4-2 and 15.4-3 for the cases with 2000 failed rods and 30 failed rods, respectively. The activity releases to the environment are presented in Tables 15.4-4 and 15.4-5 for 2000 failed rods and 30 failed rods, respectively.

15.4.9.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD Computer Program (Reference 15.4-8) is used to evaluate the radiological consequences for the realistic analysis.

The following assumptions are used in calculating fission product activity release from the fuel:

a) The reactor has been operating at design power for 3 years until 30 minutes prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 minutes of the departure from design power. The 30 minute time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations. 2000 fuel rods are conservatively assumed to be damaged.

b) An average of 1.8 percent of the noble gas, cesium and rubidium activity and 0.32 percent of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 15.4-9).

c) The fraction of other solid fission product activity available for release from the fuel is negligible.

d) The fission products produced during the nuclear excursion are neglected.

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

a) All of the noble gas activity and 2% of the iodine, cesium and rubidium activity released from the damaged fuel rods is conservatively assumed to be immediately available for release from the condenser.

b) The reactor water sample line is isolated by a Main Steam Line Radiation Monitor high high radiation signal prior to the release of any activity to the reactor building.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. A partition factor of 140 is assumed in condenser for iodine, cesium and rubidium activity. By using the above conditions, the activity airborne in the condenser is presented in Tables 15.4-6 and 15.4-7.

The following assumptions and conditions are used to evaluate the activity released to the environment:

a) The leak rate out of the condenser is 0.5 percent per day of the combined condenser and turbine free volume.

b) The activity released from the condenser becomes airborne in the turbine building. The turbine building ventilation rate is seven air changes per day.

c) No filtration or plateout of iodines occurs in the building prior to release to the atmosphere.

Based on the above assumptions, the fission product release to the environment is presented in Tables 15.4-8 and 15.4-9.

15.4.9.5.3 Results

Offsite

The calculated radiological doses at the site boundary and low population zone for the design basis and realistic cases with 2000 failed fuels rods and 30 failed fuel rods are presented in Table 15.4-10. The doses are well within the 10CFR50.67 dose limits and the Regulatory Guide 1.183 acceptance criteria.

Control Room

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.4-11. The radiological exposure to the control room personnel for the design basis case is given in Table 15.4-10. The doses are well within the 10CFR50.67 dose limits.

15.4.10 REFERENCES

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15.4-2 C. J. Paone, "Bank Position Withdrawal Sequence," September 1976 (NEDO-21231).

15.4-3 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," July 1972 Supplement 1 (NEDO-10527).

15.4-4 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," January 1973 Supplement 2 (NEDO-10527).

15.4-5 "GE BWR Generic Reload Application for 8x8 Fuel" (NEDO-20360).

15.4-6 USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000.

15.4-7 General Electric Topical Report, NEDO-31400A, Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Reactor Scram Function of the Main Steam Line Radiation Monitor", October 1992.

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15.4-18 XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” Exxon Nuclear company, June 1986.

15.4-19 EMF-CC-043(P) Rev. 9, “XCOBRA Code Theory and User’s Manual,” Framatome ANP, Inc., November 2003.

TABLE 15.4-1

SEQUENCE OF EVENTS FOR CONTROL ROD DROP ACCIDENT

APPROPRIATE ELAPSED TIME	EVENT
	Reactor is operating at rod density pattern of approximately 50%.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the Banked Position Withdrawal Sequence (BPWS).
	Decoupled control rod sticks in the fully inserted or in an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1 second	Reactor goes on a positive period and initial power increase is terminated by the Doppler effect.
<1 second	APRM 120% power signal scrams the reactor.
<5 seconds	Scram terminates the accident.

TABLE 15.4-2

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
I-131	5.30E+03	5.28E+03	5.26E+03	5.13E+03	4.81E+03	3.72E+03	3.95E+02
I-132	7.78E+03	5.75E+03	4.25E+03	6.96E+02	5.83E+00	1.43E-01	5.67E-04
I-133	1.10E+04	1.07E+04	1.03E+04	8.41E+03	4.90E+03	4.45E+02	4.15E-07
I-134	1.22E+04	5.51E+03	2.50E+03	2.17E+01	6.91E-05	0	0
I-135	1.05E+04	9.41E+03	8.47E+03	4.51E+03	8.36E+02	4.40E-01	0
Co-58	5.44E-05	5.44E-05	5.43E-05	5.41E-05	5.34E-05	5.18E-05	4.02E-05
Co-60	2.93E-05	2.93E-05	2.93E-05	2.92E-05	2.90E-05	2.90E-05	2.87E-05
Kr-83m	6.73E+04	4.67E+04	3.24E+04	3.62E+03	1.05E+01	4.11E-11	0
Kr-85	7.60E+03	7.60E+03	7.59E+03	7.58E+03	7.52E+03	7.52E+03	7.49E+03
Kr-85m	1.38E+05	1.18E+05	1.01E+05	3.98E+04	3.33E+03	4.83E-02	0
Kr-87	2.75E+05	1.60E+05	9.25E+04	3.51E+03	5.68E-01	0	0
Kr-88	3.83E+05	3.00E+05	2.35E+05	5.42E+04	1.08E+03	2.53E-05	0
Rb-86	1.26E-01	1.26E-01	1.26E-01	1.24E-01	1.21E-01	1.08E-01	4.10E-02
Sr-89	7.55E-02	7.54E-02	7.54E-02	7.49E-02	7.37E-02	7.08E-02	4.95E-02
Sr-90	9.62E-03	9.62E-03	9.62E-03	9.59E-03	9.53E-03	9.53E-03	9.51E-03
Sr-91	9.62E-02	8.94E-02	8.31E-02	5.35E-02	1.65E-02	8.65E-05	0
Sr-92	1.02E-01	7.91E-02	6.12E-02	1.32E-02	2.18E-04	0	0
Y-90	9.99E-05	2.02E-04	3.04E-04	8.88E-04	2.26E-03	6.22E-03	9.56E-03
Y-91	9.83E-04	1.00E-03	1.02E-03	1.10E-03	1.18E-03	1.19E-03	8.73E-04
Y-92	1.04E-03	1.70E-02	2.64E-02	2.69E-02	2.39E-03	2.39E-09	0
Y-93	7.85E-04	7.32E-04	6.83E-04	4.52E-04	1.50E-04	1.07E-06	0
Zr-95	1.41E-03	1.41E-03	1.41E-03	1.40E-03	1.38E-03	1.34E-03	1.01E-03
Zr-97	1.39E-03	1.34E-03	1.28E-03	1.00E-03	5.16E-04	2.69E-05	0
Nb-95	1.41E-03	1.41E-03	1.41E-03	1.40E-03	1.40E-03	1.39E-03	1.29E-03
Mo-99	1.86E-02	1.84E-02	1.82E-02	1.70E-02	1.43E-02	6.70E-03	9.55E-06
Tc-99m	1.64E-02	1.64E-02	1.64E-02	1.59E-02	1.43E-02	6.87E-03	9.79E-06
Ru-103	1.58E-02	1.58E-02	1.58E-02	1.57E-02	1.54E-02	1.46E-02	9.22E-03
Ru-105	1.09E-02	9.34E-03	7.99E-03	3.12E-03	2.55E-04	3.35E-09	0
Ru-106	6.30E-03	6.29E-03	6.29E-03	6.27E-03	6.22E-03	6.19E-03	5.89E-03
Rh-105	1.02E-02	1.02E-02	1.01E-02	9.57E-03	7.23E-03	1.77E-03	8.64E-09
Sb-127	1.73E-02	1.72E-02	1.70E-02	1.62E-02	1.43E-02	8.33E-03	7.72E-05
Sb-129	6.39E-02	5.44E-02	4.63E-02	1.76E-02	1.34E-03	1.29E-08	0
Te-127	1.71E-02	1.71E-02	1.71E-02	1.69E-02	1.60E-02	1.08E-02	2.61E-03
Te-127m	2.92E-03	2.92E-03	2.92E-03	2.91E-03	2.89E-03	2.88E-03	2.48E-03
Te-129	6.05E-02	5.79E-02	5.33E-02	2.71E-02	1.22E-02	9.71E-03	5.68E-03
Te-129m	1.22E-02	1.22E-02	1.22E-02	1.22E-02	1.19E-02	1.12E-02	6.57E-03
Te-131m	3.95E-02	3.86E-02	3.77E-02	3.28E-02	2.25E-02	4.26E-03	2.33E-09
Te-132	2.83E-01	2.81E-01	2.78E-01	2.63E-01	2.27E-01	1.20E-01	4.75E-04
Xe-133	1.09E+06	1.08E+06	1.08E+06	1.04E+06	9.50E+05	6.45E+05	2.10E+04
Xe-133m	3.58E+04	3.54E+04	3.49E+04	3.23E+04	2.63E+04	1.07E+04	4.22E+00
Xe-135	3.60E+05	3.42E+05	3.18E+05	2.03E+05	6.09E+04	2.60E+02	0
Xe-135m	3.10E+05	2.31E+04	3.00E+03	8.02E+02	3.82E+02	2.01E-01	0

TABLE 15.4-2 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(2000 FAILED RODS)

Cs-134	1.34E+01	1.34E+01	1.34E+01	1.33E+01	1.32E+01	1.32E+01	1.29E+01
Cs-136	4.27E+00	4.26E+00	4.25E+00	4.19E+00	4.01E+00	3.42E+00	8.65E-01
Cs-137	1.01E+01	1.01E+01	1.01E+01	1.00E+01	9.97E+00	9.97E+00	9.95E+00
Ba-139	1.43E-01	8.66E-02	5.24E-02	2.56E-03	8.13E-07	0	0
Ba-140	1.44E-01	1.44E-01	1.43E-01	1.41E-01	1.35E-01	1.15E-01	2.79E-02
La-140	1.54E-03	3.96E-03	6.34E-03	1.96E-02	4.81E-02	1.02E-01	3.24E-02
La-141	1.31E-03	1.09E-03	9.17E-04	3.17E-04	1.88E-05	5.73E-11	0
La-142	1.28E-03	8.16E-04	5.20E-04	3.49E-05	2.61E-08	0	0
Ce-141	3.31E-03	3.31E-03	3.30E-03	3.28E-03	3.21E-03	3.01E-03	1.73E-03
Ce-143	3.07E-03	3.01E-03	2.94E-03	2.59E-03	1.84E-03	4.05E-04	8.22E-10
Ce-144	2.78E-03	2.78E-03	2.78E-03	2.77E-03	2.75E-03	2.73E-03	2.56E-03
Pr-143	1.18E-03	1.18E-03	1.19E-03	1.21E-03	1.23E-03	1.19E-03	3.27E-04
Nd-147	5.33E-04	5.31E-04	5.30E-04	5.20E-04	4.95E-04	4.10E-04	7.94E-05
Np-239	3.90E-02	3.85E-02	3.80E-02	3.53E-02	2.88E-02	1.19E-02	5.65E-06
Pu-238	8.38E-06	8.38E-06	8.38E-06	8.36E-06	8.30E-06	8.30E-06	8.32E-06
Pu-239	8.90E-07	8.90E-07	8.90E-07	8.88E-07	8.84E-07	8.89E-07	8.92E-07
Pu-240	1.43E-06	1.43E-06	1.43E-06	1.43E-06	1.42E-06	1.42E-06	1.42E-06
Pu-241	3.53E-04	3.53E-04	3.53E-04	3.52E-04	3.50E-04	3.49E-04	3.48E-04
Am-241	1.87E-07	1.87E-07	1.87E-07	1.86E-07	1.86E-07	1.91E-07	2.31E-07
Cm-242	4.89E-05	4.88E-05	4.88E-05	4.86E-05	4.82E-05	4.76E-05	4.26E-05
Cm-244	2.86E-06	2.86E-06	2.86E-06	2.85E-06	2.83E-06	2.83E-06	2.83E-06

TABLE 15.4-3

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(30 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
I-131	7.95E+01	4.78E+01	2.88E+01	1.36E+00	3.98E-04	3.07E-04	3.26E-05
I-132	1.17E+02	5.21E+01	2.33E+01	1.84E-01	4.82E-07	1.18E-08	4.68E-11
I-133	1.65E+02	9.65E+01	5.63E+01	2.23E+00	4.05E-04	3.68E-05	0
I-134	1.82E+02	4.99E+01	1.37E+01	5.75E-03	0	0	0
I-135	1.57E+02	8.53E+01	4.63E+01	1.19E+00	6.90E-05	3.63E-08	0
Co-58	8.16E-07	4.93E-07	2.97E-07	1.43E-08	4.41E-12	4.28E-12	3.32E-12
Co-60	4.39E-07	2.65E-07	1.60E-07	7.73E-09	2.39E-12	2.39E-12	2.37E-12
Kr-83m	1.01E+03	4.23E+02	1.77E+02	9.60E-01	8.67E-07	0	0
Kr-85	1.14E+02	6.88E+01	4.15E+01	2.01E+00	6.21E-04	6.21E-04	6.18E-04
Kr-85m	2.06E+03	1.07E+03	5.52E+02	1.05E+01	2.75E-04	3.99E-09	0
Kr-87	4.13E+03	1.45E+03	5.06E+02	9.29E-01	4.69E-08	0	0
Kr-88	5.74E+03	2.72E+03	1.29E+03	1.44E+01	8.95E-05	0	0
Rb-86	1.89E-03	1.14E-03	6.88E-04	3.29E-05	9.95E-09	8.90E-09	3.39E-09
Sr-89	1.13E-03	6.83E-04	4.12E-04	1.98E-05	6.09E-09	5.84E-09	4.09E-09
Sr-90	1.44E-04	8.71E-05	5.26E-05	2.54E-06	7.87E-10	7.87E-10	7.85E-10
Sr-91	1.44E-03	8.10E-04	4.54E-04	1.42E-05	1.37E-09	7.14E-12	0
Sr-92	1.53E-03	7.16E-04	3.35E-04	3.49E-06	0	0	0
Y-90	1.50E-06	1.83E-06	1.66E-06	2.35E-07	1.87E-10	5.14E-10	7.89E-10
Y-91	1.47E-05	9.07E-06	5.57E-06	2.90E-07	9.78E-11	9.81E-11	7.21E-11
Y-92	1.56E-05	1.54E-04	1.44E-04	7.13E-06	1.97E-10	0	0
Y-93	1.18E-05	6.63E-06	3.74E-06	1.20E-07	1.24E-11	0	0
Zr-95	2.11E-05	1.28E-05	7.70E-06	3.71E-07	1.14E-10	1.10E-10	8.33E-11
Zr-97	2.09E-05	1.21E-05	7.02E-06	2.65E-07	4.26E-11	0	0
Nb-95	2.11E-05	1.28E-05	7.70E-06	3.72E-07	1.15E-10	1.15E-10	1.07E-10
Mo-99	2.78E-04	1.66E-04	9.93E-05	4.50E-06	1.18E-09	5.53E-10	7.89E-13
Tc-99m	2.47E-04	1.49E-04	8.96E-05	4.22E-06	1.18E-09	5.67E-10	0
Ru-103	2.37E-04	1.43E-04	8.63E-05	4.15E-06	1.27E-09	1.20E-09	7.61E-10
Ru-105	1.64E-04	8.46E-05	4.37E-05	8.27E-07	2.11E-11	0	0
Ru-106	9.44E-05	5.70E-05	3.44E-05	1.66E-06	5.14E-10	5.11E-10	4.86E-10
Rh-105	1.53E-04	9.22E-05	5.55E-05	2.53E-06	5.97E-10	1.46E-10	0
Sb-127	2.59E-04	1.55E-04	9.31E-05	4.30E-06	1.18E-09	6.88E-10	6.38E-12
Sb-129	9.58E-04	4.93E-04	2.53E-04	4.67E-06	1.11E-10	0	0
Te-127	2.57E-04	1.55E-04	9.35E-05	4.47E-06	1.32E-09	8.93E-10	2.15E-10
Te-127m	4.38E-05	2.65E-05	1.60E-05	7.72E-07	2.39E-10	2.38E-10	2.05E-10
Te-129	9.08E-04	5.24E-04	2.91E-04	7.19E-06	1.01E-09	8.02E-10	4.69E-10
Te-129m	1.84E-04	1.11E-04	6.69E-05	3.22E-06	9.86E-10	9.27E-10	5.42E-10
Te-131m	5.93E-04	3.50E-04	2.06E-04	8.68E-06	1.86E-09	3.52E-10	0
Te-132	4.25E-03	2.54E-03	1.52E-03	6.97E-05	1.87E-08	9.89E-09	3.92E-11
Xe-133	1.63E+04	9.81E+03	5.89E+03	2.76E+02	7.84E-02	5.33E-02	1.74E-03
Xe-133m	5.37E+02	3.20E+02	1.91E+02	8.56E+00	2.17E-03	8.81E-04	3.49E-07
Xe-135	5.40E+03	3.10E+03	1.74E+03	5.38E+01	5.03E-03	2.15E-05	0

TABLE 15.4-3 (CONTINUED)

TABLE 15.4-3

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(30 FAILED RODS)

Xe-135m	4.65E+03	2.10E+02	1.64E+01	2.12E-01	3.15E-05	1.66E-08	0
Cs-134	2.01E-01	1.21E-01	7.31E-02	3.53E-03	1.09E-06	1.09E-06	1.06E-06
Cs-136	6.41E-02	3.86E-02	2.33E-02	1.11E-03	3.31E-07	2.83E-07	7.14E-08
Cs-137	1.51E-01	9.12E-02	5.51E-02	2.66E-03	8.24E-07	8.23E-07	8.22E-07
Ba-139	2.15E-03	7.85E-04	2.86E-04	6.77E-07	0	0	0
Ba-140	2.16E-03	1.30E-03	7.83E-04	3.73E-05	1.11E-08	9.47E-09	2.30E-09
La-140	2.30E-05	3.59E-05	3.47E-05	5.20E-06	3.97E-09	8.40E-09	2.67E-09
La-141	1.96E-05	9.91E-06	5.01E-06	8.41E-08	0	0	0
La-142	1.92E-05	7.39E-06	2.84E-06	9.26E-09	0	0	0
Ce-141	4.96E-05	2.99E-05	1.81E-05	8.69E-07	2.65E-10	2.49E-10	1.43E-10
Ce-143	4.61E-05	2.72E-05	1.61E-05	6.86E-07	1.52E-10	3.34E-11	0
Ce-144	4.17E-05	2.52E-05	1.52E-05	7.34E-07	2.27E-10	2.25E-10	2.11E-10
Pr-143	1.77E-05	1.07E-05	6.50E-06	3.19E-07	1.02E-10	9.81E-11	2.70E-11
Nd-147	7.99E-06	4.81E-06	2.90E-06	1.38E-07	4.09E-11	3.39E-11	6.56E-12
Np-239	5.85E-04	3.49E-04	2.08E-04	9.34E-06	2.38E-09	9.83E-10	0
Pu-238	1.26E-07	7.59E-08	4.58E-08	2.21E-09	6.86E-13	6.86E-13	6.87E-13
Pu-239	1.34E-08	8.06E-09	4.87E-09	2.35E-10	7.30E-14	7.34E-14	7.36E-14
Pu-240	2.15E-08	1.30E-08	7.82E-09	3.78E-10	1.17E-13	1.17E-13	1.17E-13
Pu-241	5.30E-06	3.20E-06	1.93E-06	9.32E-08	2.89E-11	2.89E-11	2.88E-11
Am-241	2.80E-09	1.69E-09	1.02E-09	4.94E-11	1.54E-14	1.58E-14	1.90E-14
Cm-242	7.33E-07	4.42E-07	2.67E-07	1.29E-08	3.98E-12	3.93E-12	3.52E-12
Cm-244	4.29E-08	2.59E-08	1.56E-08	7.56E-10	2.34E-13	2.34E-13	2.33E-13

TABLE 15.4-4

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
I-131	1.10E-03	2.20E+00	4.40E+00	1.74E+01	5.05E+01	5.05E+01	5.05E+01
I-132	1.62E-03	2.75E+00	4.79E+00	9.63E+00	1.05E+01	1.05E+01	1.05E+01
I-133	2.30E-03	4.51E+00	8.86E+00	3.21E+01	7.50E+01	7.50E+01	7.50E+01
I-134	2.53E-03	3.36E+00	4.89E+00	6.14E+00	6.15E+00	6.15E+00	6.15E+00
I-135	2.18E-03	4.11E+00	7.82E+00	2.34E+01	3.75E+01	3.75E+01	3.75E+01
Co-58	1.13E-11	2.27E-08	4.53E-08	1.81E-07	5.39E-07	5.39E-07	5.39E-07
Co-60	6.10E-12	1.22E-08	2.44E-08	9.75E-08	2.91E-07	2.91E-07	2.91E-07
Kr-83m	1.40E-02	2.31E+01	3.91E+01	7.13E+01	7.53E+01	7.53E+01	7.53E+01
Kr-85	1.58E-03	3.16E+00	6.33E+00	2.53E+01	7.56E+01	7.56E+01	7.56E+01
Kr-85m	2.87E-02	5.27E+01	9.79E+01	2.61E+02	3.55E+02	3.55E+02	3.55E+02
Kr-87	5.74E-02	8.60E+01	1.36E+02	2.02E+02	2.05E+02	2.05E+02	2.05E+02
Kr-88	7.98E-02	1.40E+02	2.49E+02	5.54E+02	6.40E+02	6.40E+02	6.40E+02
Rb-86	2.63E-08	5.26E-05	1.05E-04	4.18E-04	1.23E-03	1.23E-03	1.23E-03
Sr-89	1.57E-08	3.14E-05	6.28E-05	2.51E-04	7.46E-04	7.46E-04	7.46E-04
Sr-90	2.00E-09	4.01E-06	8.01E-06	3.20E-05	9.58E-05	9.58E-05	9.58E-05
Sr-91	2.00E-08	3.85E-05	7.43E-05	2.42E-04	4.47E-04	4.47E-04	4.47E-04
Sr-92	2.13E-08	3.71E-05	6.57E-05	1.43E-04	1.63E-04	1.63E-04	1.63E-04
Y-90	2.08E-11	6.51E-08	1.73E-07	1.68E-06	1.25E-05	1.25E-05	1.25E-05
Y-91	2.05E-10	4.14E-07	8.35E-07	3.49E-06	1.12E-05	1.12E-05	1.12E-05
Y-92	2.17E-10	4.34E-06	1.38E-05	9.14E-05	1.63E-04	1.63E-04	1.63E-04
Y-93	1.63E-10	3.15E-07	6.09E-07	2.00E-06	3.79E-06	3.79E-06	3.79E-06
Zr-95	2.94E-10	5.87E-07	1.17E-06	4.68E-06	1.39E-05	1.39E-05	1.39E-05
Zr-97	2.91E-10	5.68E-07	1.11E-06	3.95E-06	8.76E-06	8.76E-06	8.76E-06
Nb-95	2.94E-10	5.87E-07	1.17E-06	4.69E-06	1.40E-05	1.40E-05	1.40E-05
Mo-99	3.87E-09	7.68E-06	1.53E-05	5.92E-05	1.63E-04	1.63E-04	1.63E-04
Tc-99m	3.43E-09	6.84E-06	1.37E-05	5.41E-05	1.55E-04	1.55E-04	1.55E-04
Ru-103	3.29E-09	6.58E-06	1.32E-05	5.25E-05	1.56E-04	1.56E-04	1.56E-04
Ru-105	2.28E-09	4.18E-06	7.75E-06	2.06E-05	2.80E-05	2.80E-05	2.80E-05
Ru-106	1.31E-09	2.62E-06	5.24E-06	2.09E-05	6.26E-05	6.26E-05	6.26E-05
Rh-105	2.12E-09	4.24E-06	8.47E-06	3.32E-05	8.89E-05	8.89E-05	8.89E-05
Sb-127	3.60E-09	7.17E-06	1.43E-05	5.58E-05	1.57E-04	1.57E-04	1.57E-04
Sb-129	1.33E-08	2.44E-05	4.51E-05	1.19E-04	1.59E-04	1.59E-04	1.59E-04
Te-127	3.57E-09	7.13E-06	1.43E-05	5.68E-05	1.66E-04	1.66E-04	1.66E-04
Te-127m	6.09E-10	1.22E-06	2.43E-06	9.73E-06	2.91E-05	2.91E-05	2.91E-05
Te-129	1.26E-08	2.47E-05	4.78E-05	1.45E-04	2.56E-04	2.56E-04	2.56E-04
Te-129m	2.55E-09	5.10E-06	1.02E-05	4.07E-05	1.21E-04	1.21E-04	1.21E-04
Te-131m	8.24E-09	1.63E-05	3.21E-05	1.20E-04	3.01E-04	3.01E-04	3.01E-04
Te-132	5.90E-08	1.17E-04	2.34E-04	9.10E-04	2.54E-03	2.54E-03	2.54E-03
Xe-133	2.27E-01	4.52E+02	9.02E+02	3.55E+03	1.02E+04	1.02E+04	1.02E+04
Xe-133m	7.46E-03	1.48E+01	2.94E+01	1.13E+02	3.07E+02	3.07E+02	3.07E+02
Xe-135	7.51E-02	1.47E+02	2.84E+02	9.23E+02	1.69E+03	1.69E+03	1.69E+03

TABLE 15.4-4 (CONTINUED)

TABLE 15.4-4

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(2000 FAILED RODS)

Xe-135m	6.46E-02	3.99E+01	4.33E+01	4.62E+01	5.07E+01	5.07E+01	5.07E+01
Cs-134	2.79E-06	5.57E-03	1.11E-02	4.45E-02	1.33E-01	1.33E-01	1.33E-01
Cs-136	8.90E-07	1.78E-03	3.55E-03	1.41E-02	4.14E-02	4.14E-02	4.14E-02
Cs-137	2.10E-06	4.19E-03	8.39E-03	3.35E-02	1.00E-01	1.00E-01	1.00E-01
Ba-139	2.98E-08	4.57E-05	7.34E-05	1.14E-04	1.16E-04	1.16E-04	1.16E-04
Ba-140	3.00E-08	5.98E-05	1.20E-04	4.75E-04	1.39E-03	1.39E-03	1.39E-03
La-140	3.20E-10	1.20E-06	3.39E-06	3.65E-05	2.71E-04	2.71E-04	2.71E-04
La-141	2.72E-10	4.94E-07	9.08E-07	2.31E-06	2.99E-06	2.99E-06	2.99E-06
La-142	2.66E-10	4.19E-07	6.87E-07	1.13E-06	1.16E-06	1.16E-06	1.16E-06
Ce-141	6.89E-10	1.38E-06	2.75E-06	1.10E-05	3.26E-05	3.26E-05	3.26E-05
Ce-143	6.40E-10	1.26E-06	2.50E-06	9.40E-06	2.39E-05	2.39E-05	2.39E-05
Ce-144	5.80E-10	1.16E-06	2.32E-06	9.25E-06	2.76E-05	2.76E-05	2.76E-05
Pr-143	2.46E-10	4.93E-07	9.87E-07	3.98E-06	1.21E-05	1.21E-05	1.21E-05
Nd-147	1.11E-10	2.22E-07	4.43E-07	1.75E-06	5.14E-06	5.14E-06	5.14E-06
Np-239	8.13E-09	1.61E-05	3.21E-05	1.24E-04	3.35E-04	3.35E-04	3.35E-04
Pu-238	1.75E-12	3.49E-09	6.98E-09	2.79E-08	8.34E-08	8.34E-08	8.34E-08
Pu-239	1.85E-13	3.71E-10	7.42E-10	2.96E-09	8.87E-09	8.87E-09	8.87E-09
Pu-240	2.98E-13	5.96E-10	1.19E-09	4.76E-09	1.42E-08	1.42E-08	1.42E-08
Pu-241	7.36E-11	1.47E-07	2.94E-07	1.18E-06	3.51E-06	3.51E-06	3.51E-06
Am-241	3.89E-14	7.77E-11	1.55E-10	6.22E-10	1.86E-09	1.86E-09	1.86E-09
Cm-242	1.02E-11	2.03E-08	4.07E-08	1.62E-07	4.85E-07	4.85E-07	4.85E-07
Cm-244	5.97E-13	1.19E-09	2.38E-09	9.53E-09	2.85E-08	2.85E-08	2.85E-08

TABLE 15.4-5

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(30 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
I-131	2.01E-02	3.15E+01	5.04E+01	7.76E+01	7.89E+01	7.89E+01	7.89E+01
I-132	2.95E-02	3.98E+01	5.76E+01	7.19E+01	7.20E+01	7.20E+01	7.20E+01
I-133	4.17E-02	6.44E+01	1.02E+02	1.53E+02	1.55E+02	1.55E+02	1.55E+02
I-134	4.60E-02	4.96E+01	6.32E+01	6.83E+01	6.83E+01	6.83E+01	6.83E+01
I-135	3.96E-02	5.90E+01	9.10E+01	1.28E+02	1.29E+02	1.29E+02	1.29E+02
Co-58	2.06E-10	3.24E-07	5.19E-07	8.02E-07	8.16E-07	8.16E-07	8.16E-07
Co-60	1.11E-10	1.74E-07	2.79E-07	4.32E-07	4.39E-07	4.39E-07	4.39E-07
Kr-83m	2.55E-01	3.34E+02	4.74E+02	5.75E+02	5.75E+02	5.75E+02	5.75E+02
Kr-85	2.88E-02	4.52E+01	7.25E+01	1.12E+02	1.14E+02	1.14E+02	1.14E+02
Kr-85m	5.21E-01	7.57E+02	1.15E+03	1.56E+03	1.57E+03	1.57E+03	1.57E+03
Kr-87	1.04E+00	1.26E+03	1.70E+03	1.93E+03	1.93E+03	1.93E+03	1.93E+03
Kr-88	1.45E+00	2.02E+03	2.97E+03	3.82E+03	3.83E+03	3.83E+03	3.83E+03
Rb-86	4.78E-07	7.50E-04	1.20E-03	1.86E-03	1.89E-03	1.89E-03	1.89E-03
Sr-89	2.86E-07	4.49E-04	7.20E-04	1.11E-03	1.13E-03	1.13E-03	1.13E-03
Sr-90	3.64E-08	5.72E-05	9.18E-05	1.42E-04	1.44E-04	1.44E-04	1.44E-04
Sr-91	3.64E-07	5.51E-04	8.61E-04	1.24E-03	1.26E-03	1.26E-03	1.26E-03
Sr-92	3.87E-07	5.35E-04	7.85E-04	1.00E-03	1.00E-03	1.00E-03	1.00E-03
Y-90	3.78E-10	9.04E-07	1.82E-06	4.29E-06	4.58E-06	4.58E-06	4.58E-06
Y-91	3.72E-09	5.90E-06	9.53E-06	1.50E-05	1.53E-05	1.53E-05	1.53E-05
Y-92	3.95E-09	5.81E-05	1.38E-04	2.99E-04	3.05E-04	3.05E-04	3.05E-04
Y-93	2.97E-09	4.51E-06	7.05E-06	1.02E-05	1.03E-05	1.03E-05	1.03E-05
Zr-95	5.34E-09	8.38E-06	1.34E-05	2.08E-05	2.11E-05	2.11E-05	2.11E-05
Zr-97	5.28E-09	8.12E-06	1.28E-05	1.91E-05	1.93E-05	1.93E-05	1.93E-05
Nb-95	5.34E-09	8.38E-06	1.34E-05	2.08E-05	2.11E-05	2.11E-05	2.11E-05
Mo-99	7.03E-08	1.10E-04	1.75E-04	2.68E-04	2.72E-04	2.72E-04	2.72E-04
Tc-99m	6.23E-08	9.77E-05	1.57E-04	2.41E-04	2.46E-04	2.46E-04	2.46E-04
Ru-103	5.99E-08	9.40E-05	1.51E-04	2.33E-04	2.37E-04	2.37E-04	2.37E-04
Ru-105	4.14E-08	6.00E-05	9.11E-05	1.24E-04	1.24E-04	1.24E-04	1.24E-04
Ru-106	2.38E-08	3.74E-05	6.00E-05	9.28E-05	9.44E-05	9.44E-05	9.44E-05
Rh-105	3.86E-08	6.06E-05	9.71E-05	1.49E-04	1.52E-04	1.52E-04	1.52E-04
Sb-127	6.55E-08	1.02E-04	1.64E-04	2.51E-04	2.55E-04	2.55E-04	2.55E-04
Sb-129	2.42E-07	3.50E-04	5.31E-04	7.18E-04	7.21E-04	7.21E-04	7.21E-04
Te-127	6.49E-08	1.02E-04	1.63E-04	2.52E-04	2.57E-04	2.57E-04	2.57E-04
Te-127m	1.11E-08	1.74E-05	2.79E-05	4.31E-05	4.38E-05	4.38E-05	4.38E-05
Te-129	2.29E-07	3.53E-04	5.53E-04	7.86E-04	7.92E-04	7.92E-04	7.92E-04
Te-129m	4.64E-08	7.28E-05	1.17E-04	1.80E-04	1.84E-04	1.84E-04	1.84E-04
Te-131m	1.50E-07	2.32E-04	3.69E-04	5.58E-04	5.67E-04	5.67E-04	5.67E-04
Te-132	1.07E-06	1.68E-03	2.68E-03	4.11E-03	4.18E-03	4.18E-03	4.18E-03
Xe-133	4.12E+00	6.46E+03	1.03E+04	1.59E+04	1.62E+04	1.62E+04	1.62E+04
Xe-133m	1.36E-01	2.12E+02	3.38E+02	5.16E+02	5.24E+02	5.24E+02	5.24E+02
Xe-135	1.36E+00	2.10E+03	3.28E+03	4.75E+03	4.80E+03	4.80E+03	4.80E+03

TABLE 15.4-5 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(30 FAILED RODS)

Xe-135m	1.17E+00	6.26E+02	6.58E+02	6.65E+02	6.65E+02	6.65E+02	6.65E+02
Cs-134	5.07E-05	7.96E-02	1.28E-01	1.97E-01	2.01E-01	2.01E-01	2.01E-01
Cs-136	1.62E-05	2.54E-02	4.07E-02	6.27E-02	6.38E-02	6.38E-02	6.38E-02
Cs-137	3.81E-05	5.99E-02	9.61E-02	1.48E-01	1.51E-01	1.51E-01	1.51E-01
Ba-139	5.42E-07	6.66E-04	9.10E-04	1.05E-03	1.05E-03	1.05E-03	1.05E-03
Ba-140	5.45E-07	8.55E-04	1.37E-03	2.11E-03	2.15E-03	2.15E-03	2.15E-03
La-140	5.82E-09	1.65E-05	3.51E-05	8.87E-05	9.49E-05	9.49E-05	9.49E-05
La-141	4.94E-09	7.11E-06	1.07E-05	1.43E-05	1.44E-05	1.44E-05	1.44E-05
La-142	4.84E-09	6.10E-06	8.45E-06	9.92E-06	9.92E-06	9.92E-06	9.92E-06
Ce-141	1.25E-08	1.97E-05	3.15E-05	4.87E-05	4.96E-05	4.96E-05	4.96E-05
Ce-143	1.16E-08	1.81E-05	2.87E-05	4.35E-05	4.42E-05	4.42E-05	4.42E-05
Ce-144	1.05E-08	1.65E-05	2.65E-05	4.10E-05	4.17E-05	4.17E-05	4.17E-05
Pr-143	4.47E-09	7.04E-06	1.13E-05	1.75E-05	1.78E-05	1.78E-05	1.78E-05
Nd-147	2.02E-09	3.16E-06	5.07E-06	7.82E-06	7.95E-06	7.95E-06	7.95E-06
Np-239	1.48E-07	2.31E-04	3.68E-04	5.62E-04	5.71E-04	5.71E-04	5.71E-04
Pu-238	3.17E-11	4.98E-08	7.99E-08	1.24E-07	1.26E-07	1.26E-07	1.26E-07
Pu-239	3.37E-12	5.29E-09	8.49E-09	1.31E-08	1.34E-08	1.34E-08	1.34E-08
Pu-240	5.42E-12	8.51E-09	1.37E-08	2.11E-08	2.15E-08	2.15E-08	2.15E-08
Pu-241	1.34E-09	2.10E-06	3.37E-06	5.20E-06	5.30E-06	5.30E-06	5.30E-06
Am-241	7.06E-13	1.11E-09	1.78E-09	2.75E-09	2.80E-09	2.80E-09	2.80E-09
Cm-242	1.85E-10	2.90E-07	4.66E-07	7.20E-07	7.33E-07	7.33E-07	7.33E-07
Cm-244	1.08E-11	1.70E-08	2.73E-08	4.22E-08	4.30E-08	4.30E-08	4.30E-08

Table 15.4-6

CONTROL ROD DROP ACCIDENT
ACTIVITY AIRBORNE IN CONDENSER (curies)
(Realistic Analysis)
(2000 Rods)

ISOTOPE	Condenser Airborne Activity As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
I-131	2.31E+00	2.26E+00	2.12E+00	1.61E+00	1.51E-01
I-132	1.87E+00	3.06E-01	2.45E-03	0	0
I-133	4.52E+00	3.70E+00	2.16E+00	1.93E-01	1.58E-10
I-134	1.10E+00	9.53E-03	3.05E-08	0	0
I-135	3.72E+00	1.98E+00	3.69E-01	1.91E-04	0
Kr-83m	5.43E+03	6.07E+02	1.77E+00	0	0
Kr-85	1.27E+03	1.27E+03	1.27E+03	1.25E+03	1.09E+03
Kr-85m	1.69E+04	6.67E+03	5.59E+02	8.00E-03	0
Kr-87	1.55E+04	5.87E+02	9.55E-02	0	0
Kr-88	3.93E+04	9.08E+03	1.82E+02	4.19E-06	0
Rb-86	2.65E-02	2.62E-02	2.55E-02	2.25E-02	7.52E-03
Xe-133	1.80E+05	1.74E+05	1.60E+05	1.07E+05	3.05E+03
Xe-133m	5.84E+03	5.41E+03	4.41E+03	1.76E+03	6.11E-01
Xe-135	5.27E+04	3.33E+04	9.81E+03	3.99E+01	0
Xe-135m	1.93E+02	3.52E-01	1.68E-01	8.73E-05	0
Cs-134	2.82E+00	2.81E+00	2.80E+00	2.75E+00	2.36E+00
Cs-136	8.96E-01	8.83E-01	8.50E-01	7.14E-01	1.58E-01
Cs-137	2.12E+00	2.12E+00	2.11E+00	2.08E+00	1.82E+00

Table 15.4-7

CONTROL ROD DROP ACCIDENT
ACTIVITY AIRBORNE IN CONDENSER (curies)
(Realistic Analysis)
(30 Rods)

ISOTOPE	Condenser Airborne Activity As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
I-131	1.26E-02	5.96E-04	1.74E-07	1.35E-07	1.43E-08
I-132	1.02E-02	8.09E-05	2.02E-10	0	0
I-133	2.47E-02	9.78E-04	1.78E-07	1.61E-08	0
I-134	6.00E-03	2.52E-06	0	0	0
I-135	2.03E-02	5.24E-04	3.03E-08	1.59E-11	0
Kr-83m	2.97E+01	1.61E-01	1.45E-07	0	0
Kr-85	6.95E+00	3.36E-01	1.04E-04	1.04E-04	1.03E-04
Kr-85m	9.24E+01	1.76E+00	4.60E-05	6.67E-10	0
Kr-87	8.46E+01	1.55E-01	7.85E-09	0	0
Kr-88	2.15E+02	2.40E+00	1.50E-05	0	0
Rb-86	1.45E-04	6.94E-06	2.10E-09	1.88E-09	7.14E-10
Xe-133	9.85E+02	4.61E+01	1.31E-02	8.89E-03	2.90E-04
Xe-133m	3.19E+01	1.43E+00	3.63E-04	1.47E-04	5.80E-08
Xe-135	2.88E+02	8.82E+00	8.06E-04	3.33E-06	0
Xe-135m	1.06E+00	9.32E-05	1.38E-08	0	0
Cs-134	1.54E-02	7.44E-04	2.30E-07	2.30E-07	2.24E-07
Cs-136	4.90E-03	2.34E-04	6.98E-08	5.96E-08	1.51E-08
Cs-137	1.16E-02	5.60E-04	1.74E-07	1.74E-07	1.73E-07

Table 15.4-8

CONTROL ROD DROP ACCIDENT
ACTIVITY RELEASED TO ENVIRONS (curies)
(Realistic Analysis)
(2000 Rods)

ISOTOPE	Activity Released to Environs As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
I-131	2.34E-04	2.33E-03	9.47E-03	3.73E-02	1.17E-01
I-132	2.32E-04	9.44E-04	1.13E-03	1.14E-03	1.14E-03
I-133	4.66E-04	4.19E-03	1.34E-02	2.54E-02	2.66E-02
I-134	1.99E-04	3.62E-04	3.64E-04	3.64E-04	3.64E-04
I-135	4.02E-04	2.85E-03	5.86E-03	6.55E-03	6.55E-03
Kr-83m	7.05E-01	2.48E+00	2.79E+00	2.79E+00	2.79E+00
Kr-85	1.28E-01	1.30E+00	5.44E+00	2.43E+01	1.76E+02
Kr-85m	1.89E+00	1.15E+01	1.91E+01	1.98E+01	1.98E+01
Kr-87	2.31E+00	5.77E+00	5.97E+00	5.97E+00	5.97E+00
Kr-88	4.68E+00	2.21E+01	2.90E+01	2.91E+01	2.91E+01
Rb-86	2.68E-06	2.70E-05	1.11E-04	4.71E-04	2.25E-03
Xe-133	1.83E+01	1.81E+02	7.25E+02	2.69E+03	6.49E+03
Xe-133m	5.95E-01	5.75E+00	2.16E+01	6.47E+01	9.32E+01
Xe-135	5.59E+00	4.35E+01	1.04E+02	1.30E+02	1.30E+02
Xe-135m	2.63E-01	2.70E-01	2.71E-01	2.71E-01	2.71E-01
Cs-134	2.85E-04	2.88E-03	1.21E-02	5.38E-02	3.86E-01
Cs-136	9.06E-05	9.09E-04	3.74E-03	1.54E-02	6.34E-02
Cs-137	2.14E-04	2.17E-03	9.08E-03	4.05E-02	2.94E-01

Table 15.4-9

CONTROL ROD DROP ACCIDENT
ACTIVITY RELEASED TO ENVIRONS (curies)
(Realistic Analysis)
(30 Rods)

ISOTOPE	Activity Released to Environs As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
I-131	6.14E-03	2.73E-02	3.41E-02	3.42E-02	3.42E-02
I-132	6.21E-03	1.51E-02	1.55E-02	1.55E-02	1.55E-02
I-133	1.23E-02	5.08E-02	6.09E-02	6.09E-02	6.09E-02
I-134	5.52E-03	8.07E-03	8.08E-03	8.08E-03	8.08E-03
I-135	1.06E-02	3.73E-02	4.17E-02	4.17E-02	4.17E-02
Kr-83m	1.90E+01	4.20E+01	4.28E+01	4.28E+01	4.28E+01
Kr-85	3.37E+00	1.52E+01	1.90E+01	1.91E+01	1.91E+01
Kr-85m	5.02E+01	1.58E+02	1.71E+02	1.71E+02	1.71E+02
Kr-87	6.28E+01	1.12E+02	1.13E+02	1.13E+02	1.13E+02
Kr-88	1.25E+02	3.34E+02	3.48E+02	3.48E+02	3.48E+02
Rb-86	7.05E-05	3.15E-04	3.95E-04	3.96E-04	3.96E-04
Xe-133	4.80E+02	2.13E+03	2.65E+03	2.66E+03	2.66E+03
Xe-133m	1.56E+01	6.81E+01	8.39E+01	8.40E+01	8.40E+01
Xe-135	1.48E+02	5.52E+02	6.32E+02	6.32E+02	6.32E+02
Xe-135m	8.10E+00	8.24E+00	8.24E+00	8.24E+00	8.24E+00
Cs-134	7.48E-03	3.36E-02	4.22E-02	4.23E-02	4.23E-02
Cs-136	2.38E-03	1.06E-02	1.33E-02	1.33E-02	1.33E-02
Cs-137	5.63E-03	2.53E-02	3.18E-02	3.18E-02	3.18E-02

Table 15.4-10

CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS
Design Basis Case

	2000 Failed Rods with Condenser Leakage REM TEDE	30 Failed Rods with MVP Running REM TEDE
Acceptance Criterion - Offsite	6.3	6.3
EAB	0.19	2.3
LPZ	0.05	0.18
Acceptance Criterion - CRHE	5.0	5.0
CRHE	0.43	1.5
CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS Realistic Case		
Acceptance Criterion - Offsite	6.3	6.3
EAB	0.00053	0.011
LPZ	0.00016	0.00097
Acceptance Criterion - CRHE	5.0	5.0
CRHE	0.0011	0.013

TABLE 15.4-11

CONTROL ROD DROP ACCIDENT – PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
I. Data And Assumptions Used To Estimate Radioactive Source Term From Postulated Accidents		
A. Reactor power level(MWt)	4032	4032
B. Number of fuel bundles in core	764	764
C. Number of ATRIUM-10 rods per fuel bundle	87.8	87.8
D. Number of fuel rods damaged by CRDA (rods)	2000	2000
E. Fuel melting in damaged rod (percent)	0.77	0.0
F. Core radial peaking factor	1.6	1.6
G. Gap activity release to reactor coolant from damaged rods	10 percent noble gases and iodines 12 percent Cs and Rb	1.8 percent noble gases and 0.32 percent iodines 1.8 percent Cs and Rb
H. Fuel activity release to reactor coolant from melted regions	100 percent noble gases 50 percent iodines 25 percent alkalis 5 percent tellurium 2 percent barium and strontium 0.25 percent noble metals 0.02 percent lanthanides 0.05 percent ceriums	NA
II. Data And Assumptions Used To Estimate Activity Released		
A. Activity in Reactor Coolant Transported to Condenser	100 percent noble gases 10 percent iodine 1 percent others	100 percent noble gases 2 percent iodine 2 percent Cs and Rb
B. Activity in the Condenser Available for Release to Environment	100 percent noble gases 10 percent iodine 1 percent others	100 percent noble gases 0.7 percent iodine 0.7 percent others
C. Leak Rate from Condenser to Environment (percent/day)	1	0.5
D. Removal Rate from Condenser to Environment with MVP running (percent/day)	1212	1212
E. Number of Rods Damaged by CRDA Needed to Cause MVP Trip and Isolation	30	30
F. Radioiodine Species Released from Condenser	97 percent Elemental 3 percent Organic	97 percent Elemental 3 percent Organic
G. Turbine Building Release Rate	No holdup credited	700 %/day
H. Accident Duration (hr)	24	24
III. Data And Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume(ft^3) B. Control room free volume(ft^3) C. Control structure air intake flow(cfm) D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	518,000 110,000 6391 10	518,000 110,000 6391 10

TABLE 15.4-11

CONTROL ROD DROP ACCIDENT – PARAMETERS TO BE TABULATED
FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency (percent)	0	0
IV. Dispersion Data		
A. Site Boundary/Low Population Zone distance(meters)	549/4827	549/4827
B. X/Q's for Site Boundary	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (50 percentile)
C. X/Q's for LPZ	Table 2.3-105 (0.5 percentile)	Table 2.3-105(50 percentile)
D. X/Q's for CRHE	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of calculation	Appendix 15B	Appendix 15B
B. Isotopic data and dose conversion factors	Appendix 15B	Appendix 15B
C. Activity in condenser	Tables 15.4-2 & 15.4-3	Tables 15.4-6 & 15.4-7
D. Activity released to environment	Tables 15.4-4 & 15.4-5	Tables 15.4-8 & 15.4-9
E. Offsite and control room doses	Table 15.4-10	Table 15.4-10

TABLE 15. 4-12

SEQUENCE OF EVENTS FOR MISLOADED BUNDLE ACCIDENT

1.	During core loading operation, bundle is placed in the wrong position.
2.	Subsequently, the bundle intended for this position is placed in the position of the previous bundle.
3.	During core verification procedure, error is not observed.
4.	Plant is brought to full power operation without detecting misplaced bundle.
5.	Plant continues to operate

SEQUENCE OF EVENTS FOR ROTATED BUNDLE ACCIDENT

1.	During core loading operation, bundle is placed in its proper location but rotated either 90° or 180° from its proper orientation.
2.	During core verification procedure this error is not observed.
3.	Plant is brought to full power operation without detecting rotated bundle.
4.	Plant continues to operate.

Chapter 15.4

All Figures and Tables have been moved to:
Appendix 15C, 15D, and/or 15E

Chapter 15.4

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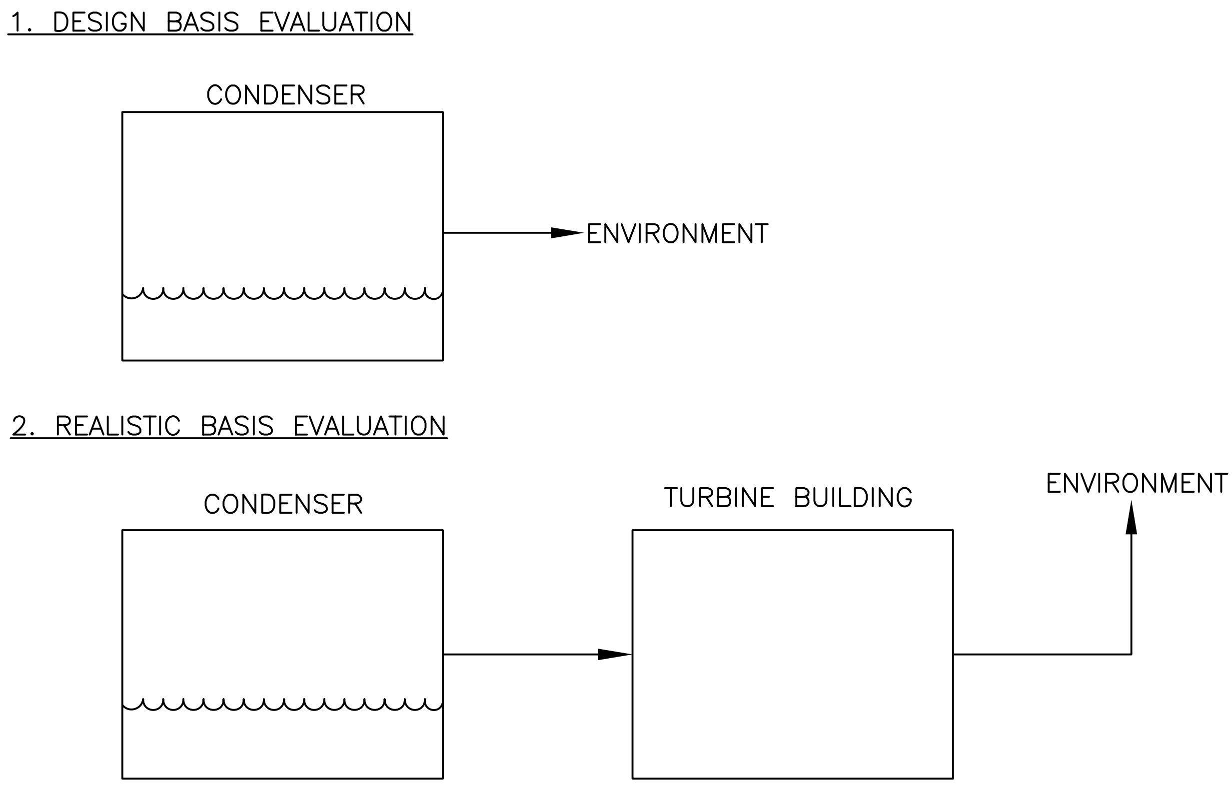

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LEAK PATH FOR CONTROL ROD
DROP ACCIDENT

FIGURE 15.4-1, Rev 55

AutoCAD: Figure Fsar 15_4_1.dwg

Chapter 15.4

All Figures have been moved to
Appendix 15C, 15D, and/or 15E

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.4 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.4-2, Rev. 54

AutoCAD Figure 15_4_2.doc

Chapter 15.4

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Appendix 15C, 15D, and/or 15E

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Chapter 15.4 Figures Moved to Appendix 15C, 15D, and/or 15E FIGURE 15.4-3, Rev. 54

AutoCAD Figure 15_4_3.doc

<u>Chapter 15.4</u>

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FIGURE 15.4-4, Rev. 54

AutoCAD Figure 15_4_4.doc

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FIGURE 15.4-5, Rev. 54

AutoCAD Figure 15_4_5.doc

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FIGURE 15.4-6, Rev. 54

AutoCAD Figure 15_4_6.doc

Chapter 15.4

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Chapter 15.4
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FIGURE 15.4-7, Rev. 54

AutoCAD Figure 15_4_7.doc

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Chapter 15.4
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FIGURE 15.4-8, Rev. 54

AutoCAD Figure 15_4_8.doc

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FIGURE 15.4-9, Rev. 54

AutoCAD Figure 15_4_9.doc

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 INADVERTENT HPCI STARTUP

This event is non-limiting, and therefore, it had not been analyzed for each cycle. However, the inadvertent HPCI startup was reanalyzed for using the methods in References 15.5-4 through 15.5-6. Based on the results of this analysis, this event is identified as non-limiting at full power EPU conditions. The results of this analysis are reported in Section 15E.

Analyses of the inadvertent HPCI Startup at lower powers have shown that this event is potentially limiting and is evaluated on a cycle specific basis.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCI system is postulated for this analysis, i.e., operator error.

15.5.1.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.5.1.2 Sequence of Events and Systems Operation

15.5.1.2.1 Sequence of Events

Table 15E.5.1-1 lists the sequence of events for Figure 15E.5.1-1.

15.5.1.2.1.1 Identification of Operator Actions

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCI has initiated, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

15.5.1.2.2 System Operation

To properly simulate the expected sequence of events the analysis of this event assumes normal functioning of plant instrumentation and controls, specifically, the pressure regulator and the vessel level control which respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this transient event.

The system is assumed to be in the manual flow control mode of operation.

15.5.1.2.3 The Effect of Single Failures and Operator Errors

Inadvertent operation of HPCI results in a mild pressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state. The effect of a single failure in the pressure regulator will aggravate the transient depending upon the nature of the failure. Pressure regulator failures are discussed in Subsections 15.1.3 and 15.2.1.

A single failure in the level control system causes level rise or fall by improper control of the feedwater system. Increasing level will trip the turbine and automatically trip the HPCI system off. This trip signature is already described in the failure of feedwater controller with increasing flow. Decreasing level will automatically initiate scram at the L3 level trip and will have a signature similar to loss of feedwater control - decreasing flow.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model described in References 15.5-4 through 15.5-6 was used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15C.0-2.

The water temperature of the HPCI system was assumed to be 40^{o}F with an enthalpy of 11 BTU/Lb.

Inadvertent startup of the HPCI system was chosen to be analyzed since it provides the greatest auxiliary source of cold water into the vessel.

For the SSES Units, the HPCI is introduced into only one of the feedwater lines. This will cause a non-symmetrical change in the inlet enthalpy. To account for the non-symmetrical introduction of colder water to the core, the HPCI flow assumed for this analysis is conservatively increased by 40% from 19% of the normal feedwater flow to 26% of normal feedwater flow.

15.5.1.3.3 Results

Figure 15E.5.1-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the feedwater sparger. Within 1 second the full HPCI flow is established at approximately 27% (19% plus 40% assumed increase) of the rated feedwater flow rate. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the core causes the neutron flux to increase to the value shown in Table 15E.0-1 for this event.

15.5.1.3.4 Consideration of Uncertainties

Important analytical factors including reactivity coefficient and feedwater temperature change have been assumed to be at the worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15E.5.1-1 indicates a slight pressure increase from initial conditions. The peak pressure is shown in Table 15E.0-1 for this event. Since the peak pressure is well below the design pressure of the RCPB, the RCPB is not threatened.

15.5.1.5 Radiological Consequences

Since no activity is released during this event, a detailed evaluation is not required.

15.5.2 Chemical Volume Control System Malfunction (or operator error)

This section is not applicable to BWR.

15.5.3 BWR Transients Which Increase Reactor Coolant Inventory

These events are discussed in Sections 15.1 and 15.2.

15.5.4 REFERENCES

15.5-1 Linford, R.B., ”Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor,” April 1973 (NEDO 10802).

15.5-2 F. Odar, “Safety Evaluation for General Electric Topical Report: Qualification of One-Dimensional Core Transient Model for Boiling Water Reactors, ”NEDO-24154, NEDE-24154-P, Vols. I, II, III, dated June 1980.

15.5-3 Deleted

15.5-4 EMF-215B(P)(A) Rev.. 0: “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,” Siemens Power Corporation, October 1999.

15.5-5 XNF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

15.5-6 ANF-913(P)(A), Volume 1 Revision 1, Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation, August 1990.

TABLE 15.5-1

THIS TABLE HAS BEEN DELETED

THIS FIGURE HAS BEEN
RENUMBERED TO 15E.5.1-1

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Figure renumbered from 15.5-1 to 15E.5.1-1

FIGURE 15.5-1, Rev. 50

AutoCAD Figure 15_5_1.doc

15.6 DECREASE IN REACTOR COOLANT INVENTORY

The analyses described in this subsection of Chapter 15 are bounding analyses and are applicable to both Unit 1 and Unit 2. The tables and figures for this subsection are immediately following the text of the subsection and are not in Appendices 15C, 15D, or 15E.

For the Instrument Line Break and the Steamline Break outside Containment, there is no fuel damage. Design basis radiological release and dose consequences are based on Technical Specification limits of iodine concentration in the reactor water. Therefore the resulting consequences are independent of the design of the fuel assemblies that comprise the core.

For the Loss of Coolant Accidents inside of Containment a conservative radiological design basis analysis is performed in accordance with NRC guidelines. This analysis bounds the current core designs for Units 1 and 2. A second radiological analysis that is based on realistic assumptions is also performed. The realistic LOCA analysis shows that no fuel failures occur and the radiological release is dependent on the various activation and corrosion products contained in the reactor coolant. As was the case for the Steamline Break, the concentration of these radionuclides in the reactor coolant is limited by the Technical Specifications and the resulting dose consequences are independent of the core design.

15.6.1 INADVERTENT SAFETY RELIEF VALVE OPENING

This event is discussed and analyzed in Subsection 15.1.4.

15.6.2 INSTRUMENT LINE BREAK

This accident is less severe than the event analyzed in Subsection 15.6.5. Quantitative results for the spectrum of LOCA events which bound this event may be found in Section 6.3.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

15.6.2.1.1.1 Event Description

A circumferential rupture of an instrument line which is connected to the primary coolant system is postulated to occur outside the primary containment but inside the secondary containment. This failure results in the release of primary system coolant to the secondary containment, until the reactor is depressurized. This event could be postulated to occur in the drywell; however, the effects would not be as significant as those from a failure in the secondary containment.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Systems Operation

15.6.2.2.1 Sequence of Events

The sequence of events for this accident is shown in Table 15.6-1.

15.6.2.2.1.1 Identification of Operator Actions

This instrument line should be automatically isolated by the excess flow check valves. The operator shall, if necessary, attempt to isolate the affected instrument line. Note, failure of certain instrument lines may result in a full scram. The following assumes there is no automatic scram. Depending on which line is broken, the operator shall determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown, and may initiate SGTS or other ventilation effluent treatment systems when directed by the appropriate Emergency Operating Procedure.

Operator action can be initiated by any one or any combination of the following:

(1) Operator comparing radiation, temperature, humidity, fluid and noise readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

(2) By annunciation of the control function, either high or low in the main control room.

(3) By a half-channel scram if rupture occurred on a reactor protection system instrument line.

(4) By a general increase in the area radiation monitor readings.

(5) By an increase in the ventilation process radiation monitor readings.

(6) By increases in area temperature monitor readings in the containment.

(7) Leak detection system actuations.

Upon receiving one or more of the above signals and having made the decision to shut down the plant, the operator should proceed to shut down the reactor in an orderly manner.

15.6.2.2.2 System Operation

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow, and suppression pool cooling capability. As a consequence of the accident, the reactor is manually scrammed and the reactor vessel cooled and depressurized over approximately a 5-hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

The initiating event is handled by a protection sequence which can accommodate additional SCF or SOE occurrences. See Appendix 15A for further discussion.

15.6.2.3 Core and System Performance

15.6.2.3.1 Qualitative Summary - Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.6.4, 15.6.5, and 15.6.6. Consequently instrument line breaks are considered to be bounded by the steamline break, Subsection 15.6.4. Details of this calculation, including those pertinent to core and system performance are discussed in detail in Subsection 15.6.4.3.

Instrument line breaks result in a slower rate of coolant loss and are bounded by the calculations referenced above. Since the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovery occurs as a result of this accident.

15.6.2.3.2 Quantitative Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and system performance standpoint than the steamline break outside containment. Similarly, instrument line breaks are considered within the spectrum considered in ECCS performance calculations discussed in detail in Subsection 6.3.3

Therefore, all information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steamline break) event may be found in Reference 15.6-13.

15.6.2.3.3 Considerations of Uncertainties

The approach toward conservatively analyzing this event is discussed in detail for a more limiting case (steamline break) in Reference 15.6-13.

15.6.2.4 Barrier Performance

The release of primary coolant through the orificed instrument line could result in an increase in secondary containment compartment pressure and the potential of isolation of the normal ventilation system.

The following assumptions and conditions are the basis for the mass loss during the 5 hour reactor shutdown period of this event:

(1) Shutdown and depressurization initiated at 10 min. after break occurs and continues for 5 hours.

(2) Normal depressurization and cooldown of reactor pressure vessel.

(3) The break is postulated in a location where, due to a cross-tie between instruments, both liquid and steam is released from the reactor pressure vessel. The primary containment penetrations/instrument lines considered are those carrying reactor coolant and listed in FSAR Table 6.2-12a. Each of these lines is provided with an excess flow check valve and there is a flow-restricting orifice installed upstream of the check valve to limit blow-down in the event of a break outside primary containment coupled with an excess flow check valve failure. The limiting configuration was identified as a line off of the RPV level instrumentation condensing chambers that has both a liquid and steam reactor nozzle feeding it. The RPV nozzles are cross connected by a ½" diameter line. The line configuration allows the steam path to bypass the flow restricting orifice. For this case, a break outside of the primary containment penetration is supplied by both a liquid source which will be restricted by a 3/8" orifice, and by a steam source which will be restricted by the ½" diameter line.

(4) Moody critical blowdown flow model (Reference 15.6-1) is applicable and flow is critical at the orifice.

The total integrated mass of fluid released into the secondary containment via the break during the blowdown is 73,713 pounds. Of this total, 23,706 pounds is steam.

Release of this mass coolant results in a secondary containment pressure which is well below the design pressure.

15.6.2.5 Radiological Consequences

Design Basis analysis shows that the event analyzed in Subsection 15.6.5 is bounding. The following describes a realistic analysis of the event. The dose consequences of the Instrument Line Break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The calculated total mass of coolant released is, 73,713 pounds. Of this 50,007 pounds is liquid and 23,706 pounds is steam. Table 15.6-2 presents the mass released as a function of time.

The reactor water iodine concentration existing at the time of the break was assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. All of the iodine activity in the steam from the flashed liquid, steam from the steam dome, and 10 percent from the remaining liquid released from the break is assumed to become airborne inside secondary containment. No credit is taken for holdup in the secondary containment. Although there will be some activation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine isotopes and noble gas activity released from the break to the environment are presented in Table 15.6-3. The dose consequences for a realistic analysis of the instrument line break are presented in Table 15.6-4. Specific values of parameters used in the analysis are presented in Table 15.6-5. The leakage path used in the evaluation is shown in Figure 15.6-1. The radiological consequences are well within 10CRF50.67 dose acceptance criteria.

15.6.3 STEAM GENERATOR TUBE FAILURE

This section is not applicable to the direct cycle BWR.

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

This event involves the postulation of a large steamline pipe break outside containment. It is assumed that the largest steamline, instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. This postulated event represents the envelope evaluation of steamline failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steamline break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards and to restrictive seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steamline is assumed to occur.

15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.4.2 Sequence of Events and Systems Operation

15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A 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 sequence of events is given in Table 15.6-6.

15.6.4.2.1.1 Identification of Operator Actions

Normally, the reactor operator will maintain reactor vessel water inventory and core cooling with the HPCI and/or RCIC system. Without operator action, HPCI and RCIC would initiate automatically on low water level (L2) following isolation of the main steam supply system (i.e., MSIV closure).

The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming HPCI failure, the operator must initiate the ADS or manual relief valve system to ensure termination of the accident without fuel damage.

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and reactor protection system action and ESF action is given in Sections 6.3, 7.3, and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All of the protective sequences for this event are capable of SCF and SOE accommodation and completion of the necessary safety action. Refer to Appendix 15A for further details.

15.6.4.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are given in References 15.6-11 and 15.6-12. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

Refer to References 15.6-11 and 15.6-12 for initial conditions.

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to References 15.6-11 and 15.6-12 for the results of this analysis.

15.6.4.3.3 Considerations of Uncertainties

Sections 6.3 and 7.3 contain discussions of the uncertainties associated with the ECCS performance and the containment isolation systems, respectively.

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the barrier performance outside primary containment can be found in Subsection 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs:

(1) The reactor is operating at the power level associated with maximum mass release.

(2) Nuclear system pressure is 1050 psia and remains constant during closure.

(3) An instantaneous circumferential break of the main steamline occurs.

(4) Isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 5.5 sec.

(5) The Moody critical flow model (Reference 15.6-1) is applicable.

Initially only steam will issue from the broken end of the steamline. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line. Rapid depressurization of the RPV causes the water level to rise resulting in a steam-water mixture flowing from the break until the valves are closed.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

(2) The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the "realistic analysis."

A schematic of the release path is shown in Figure 15.6-2.

15.6.4.5.1 Design Basis Analysis

The Design Basis Analysis is based on NRC Regulatory Guide 1.183. The dose consequences of the Main Steamline Break are determined using the calculated mass of coolant released in the time required for the MSIVs to fully close after the break occurs. The reactor was assumed to be at a hot standby condition prior to the break in order to maximize the calculated liquid release. The calculated total mass of coolant released is 97,970 pounds. Of this 84,840 pounds is liquid and

13,130 pounds is steam. These values were increased by 20% for dose analyses. There is no fuel damage as a result of this accident.

Consistent with Regulatory Guide 1.183, two cases were analyzed. In the first case, the reactor water iodine concentration existing at the time of the break was assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. In the second case, the iodine concentration prior to the break was assumed to be 4.0 micro-curies/gram dose equivalent I-131, which is the maximum short-term concentration permitted by the Technical Specifications. This concentration corresponds to an assumed pre-existing iodine spike. In each case, all iodine activity in the coolant which is released from the break is assumed to become airborne. Although there will be some activitation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine isotopes and noble gas activity released from the break and to the environment prior to isolation valve closure are presented in Table 15.6-7.

The RADTRAD Computer Program (Reference 15.6-9) is used to evaluate the radiological consequences for the design basis analysis. For the model, the mass releases are increased by a 20% margin to add conservatism.

The activity release is modeled as an instantaneous puff release per Regulatory Guide 1.194 (Reference 15.6-6). This is justified since all of the activity's is assumed to be released in the 5.5 seconds it takes for the MSIVs to close. The activity is assumed to be released as a ground level release with no holdup in the turbine building.

The specific models, assumptions and parameters used in the analyses are presented in Table 15.6-10.

15.6.4.5.2 Realistic Analysis

For the realistic analysis, it is assumed that the reactor is operating at full power prior to the break. The dose consequences are determined based on the calculated mass of coolant released in the time required for the MSIVs to fully close after the break occurs. The total integrated mass of coolant leaving the break is 40,316 pounds. (This value is increased by 20% for dose analyses). Of this quantity, 19,994 pounds is liquid and 20,322 pounds is steam. Of the 19,994 pounds of liquid released, 8,000 pounds is flashed to steam.

The activity released from the hypothetical steamline break accident is a function of the coolant activity, valve closure time, and mass of coolant released. A portion of the released coolant exists as steam prior to the blowdown, and as such does not contain the same concentration per unit of mass as does the steam generated as a consequence of the blowdown. Therefore, it is necessary to subtract the initial steam mass from the total mass released and assign to it only 8 percent of the iodine activity contained by an equivalent mass of primary coolant. The isotopic activity released to the environment is shown in Table 15.6-8.

The radiological dose consequences are calculated using the RADTRAD computer code. For the model, the mass releases are increased by a 20% margin to add conservatism, The activity release is modeled as an instantaneous puff release based on guidance from Regulatory Guide 1.194. This is reasonable since the activity is released over a very short period of time. The release is assumed to be a ground level release with no holdup in the turbine building.

The specific models, assumptions, and parameters used in the analyses are presented in Table 15.6-10.

15.6.4.5.3 Results

OFFSITE

The calculated exposures at the site boundary and low population zone for the design basis and realistic analyses are presented in Table 15.6-9. The dose consequences are well within 10CFR 50.67 guidelines.

CONTROL ROOM

A detailed description of the control room model can be found in Appendix 15B. The radiological exposure to the control room personnel for the design basis case is given in Table 15.6-9. The dose consequences satisfy the 10CFR50.67 acceptance criterion.

15.6.5 LOSS-OF-COOLANT ACCIDENTS (RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY) - INSIDE CONTAINMENT

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coupled with severe natural environmental conditions including earthquake coincidence.

The event has been analyzed quantitatively in Sections 6.2, 6.3, 7.1 and 8.3. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a loss-of-coolant accident coincident with safe shutdown earthquake plus SACF criteria requirements. The subject piping is designed to strict emergency code and standard criteria, and for severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

Representative sequences of events associated with this accident are shown in Table 6.3-1B-2 for Unit 1 and Unit 2 for ATRIUM™-10 fuel for core system performance.

Following the pipe break and scram, the MSIV will begin closing due to the loss of offsite power. The low-low water level or high drywell pressure signal will initiate HPCI, CS and LPCI systems.

15.6.5.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for the accident. However, by procedural requirement, the operator will perform the following described actions.

The operator will, after checking that all rods are inserted at time 0 plus approximately 10 seconds, determine plant condition by observing the annunciators. After observing that the Emergency Core Cooling Systems are initiated on low water level or high drywell pressure and low RPV pressure, the operator will check that the diesel generators have started and are in standby condition. The operator will also ensure primary containment isolations and ensure that reactor water level is properly maintained. Within approximately 20 minutes, the operator aligns the RHR heat exchangers for long term containment cooling. After the RHR system and other auxiliary systems are in proper operation, the operator will monitor suppression pool temperature, drywell temperature and pressure, and the hydrogen concentration in the drywell for proper activation of the recombiner, if necessary.

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe break sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steamlines upstream of the flow restrictors, and the recirculation loop pipelines. The greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any Reactor and Plant Protection System are discussed in Sections 6.2, 6.3, 7.3, 7.6, and 8.3, and Appendix 15A.

15.6.5.2.3 The Effect of Single Failures and Operator Errors

Single failures and operator errors have been considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for SCF and SOE occurrence are shown to be fully accommodated without the loss of any required safety function. See Appendix 15A for further details.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide a conservative assessment of the expected consequences of this very improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.2, 6.3, 7.3, 7.6, 8.3 and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-1B for Unit 1 and for Unit 2.

15.6.5.3.3 Results

Results of this event are given in detail in Sections 6.2 and 6.3.

The conservative, design basis AREVA analyses of the ATRIUM™-10 fuel LOCA, are described in Section 6.3.3.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed; see Sections 6.2, 6.3, 7.3, 7.6, 8.3 and Appendix 15A for details.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of the largest single primary system piping within the structure while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated loss-of-coolant accident does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2. This conclusion is valid for both Units.

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

(2) The second is based on assumptions considered to provide a realistic estimate of radiological consequences. This analysis is referred to as the "realistic analysis."

15.6.5.5.1 Design Basis Analysis

The methods, assumptions, and conditions used to evaluate this accident are in accordance with the requirements of Regulatory Guide 1.183. The RADTRAD computer code (Reference 15.6-9) is used to evaluate offsite and unprotected control room doses for this event. Specific values of parameters used in this evaluation are presented in Table 15.6-22.

15.6.5.5.1.1 Fission Product Release from Fuel

The assumptions related to the release of radioactive material from the fuel and containment as stated in Regulatory Guide (1.183) were used in this analysis.

The core inventory release fractions airborne into the primary containment, by radionuclide groups, for the gap release and early in-vessel damage phases are listed as follows.

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.2	0.25
Tellurium Metals	0	0.05	0.05
Ba, Sr	0	0.02	0.02
Noble Metals	0	0.0025	0.0025
Cerium Group	0	0.0005	0.0005
Lanthanides	0	0.0002	0.0002

The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The values used in this analysis for the release to the primary containment are provided as follows.

Phase	Onset	Duration
Gap Release	2 min	0.5 hr
Early In-Vessel Release	0.5 hr	1.5 hr

The primary containment free volume consists of the drywell free volume and the wetwell free volume.

Drywell free volume	= 239600 ft^3
Wetwell free volume	= 148590 ft^3
Total Primary containment free volume	= 388190 ft^3

For the first two hours following the event, the airborne activity released to the primary containment is assumed to only be mixed in the drywell free volume of 239600 ft^3.

The core inventory release fractions and release timing for the activity released directly into the suppression pool volume are the same as above except for noble gases which are assumed to only be present in the primary containment air. The volume of the suppression pool is 132,000 ft^3.

The activity airborne in the primary containment for the design basis case is presented in Table 15.6-11.

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway to the environment is by several different mechanisms discussed below.

a) Containment leakage: Leakage from the primary containment shell and its penetrations (excluding the main steam lines) is mixed with the air in the secondary containment, and discharged from there to the environment via the SGTS. Also, a small fraction of primary containment leakage can bypass secondary containment and be discharged to the environment without being processed by SGTS.

   Per Technical Specifications, the leakage rate for the primary containment is defined as 1% by weight of containment air per 24 hours. In accordance with Regulatory Guide 1.183, the primary containment is assumed to leak at this rate for the first 24 hours and reduced after the first 24 hours to 50% of the leak rate based on the significant reduction of the calculated internal pressure of primary containment at 24 hours. Of the 1% primary containment leakage, 15 scfh (0.0223% per 24 hours) is assumed to bypass the secondary containment and be released directly to the environment. Similar to the above the bypass leakage is reduced by 50% after 24 hours. The fraction of the primary containment leakrate which enters the secondary containment and is processed by the SCGT is the difference between the total leakrate and the secondary containment bypass leakrate or 0.9777% for the first 24 hours and 0.4889%/day thereafter.

b) Water leakage to secondary containment from Engineered Safety Feature (ESF) components outside the primary containment, such as the Emergency Core Cooling System (ECCS) and CRD:

   It is expected that during the postulated post-accident operation of the emergency systems, the total liquid leakage to the reactor building from pumps, seals, and valves will be small, i.e., on the order of gallons per hour. This leakage will be minimized by normal system tests and maintenance operations. Those ESF systems which contribute to this leakage are identified in Section 18.1.69, and the Leakage Rate Test program. The total leakage from these systems is maintained ≤2.5 gpm, in accordance with Technical Specification 5.5.2, and Section 18.1.69. Additionally, leakage from the CRD insert/withdrawal lines, as described in Section 6.2.4.3.2.3, contributes to the amount of post-accident liquid leakage from primary to secondary containment. Therefore, in order to conservatively bound the total post-accident liquid leakage for dose analysis purposes, it is assumed that 20 gpm of liquid leakage from all potential sources (ESF and CRD I/W lines) and Scram Discharge Volume contribute to the radioactive releases to the reactor building. The leakage is assumed to begin at time equal 0 and continue for the 30 day duration of the LOCA. In accordance with Regulatory Guide 1.183 with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. An analysis of the response of the suppression pool under post LOCA-accident conditions determined that the maximum bulk suppression pool water temperature does not exceed 212 °F. Therefore, per Regulatory Guide 1.183, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates." A flash fraction of 10% is conservatively used herein. The iodine species assumed available for release to the environment from this leakage are to be 97% elemental and 3% organic. Secondary containment airborne activity is processed by the SGTS prior to release to the environment.

c) Leakage from the Main Steam Isolation Valves (MSIV): The Isolated Condenser Treatment Method (ICTM) routes leakage past MSIVs to the main condenser utilizing the main steam drain lines as a pathway. In the condenser, volumetric dilution and plate-out hold up fission products until eventual release to the environment through the low pressure turbine seals.

The activity available for release from this leakage path is conservatively taken to be instantaneously released from the core and mixed in the drywell free volume (239,600 ft^3) for the first two hours. After two hours the activity is assumed to be further diluted in the drywell plus wetwell free volume (388,190 ft^3).

The MSIV leakage test limits provided in the plant Technical Specification Surveillance Requirements are ≤ 100 scfh from any one valve or ≤ 300 scfh total from the four valves. This analysis assumes one main steam line is faulted and has the 100 scfh flow. The remaining leakage is evenly split between the three non-faulted lines. The leakages are reduced by 50% after the first 24 hours based on the reduction in the drywell accident pressure.

Per a Letter from Fermi 2 to USNRC (Reference 15.6-14), the NRC states that an acceptable method for modeling the removal of aerosols, elemental and organic iodine in the main steam line piping is provided in Appendix A to AEB-98-03 (Reference 15.6-10). Credit is taken for aerosol and elemental iodine plateout in the piping based this information. Only the horizontal runs of piping are considered in determining plateout. Additionally, since aerosol plateout is a mechanistic settling process only the bottom one half of the inside surface area of the lines is applicable for plateout. Since the bottom half of a circular pipe has sides which are essentially vertical or inclined, the area for aerosol plateout is modeled as the projected area of the diameter of the pipe To account for the phenomenon that steam condensation in the piping could potentially wash out and re-evolve some of the settled aerosols an additional factor of 2 reduction in the conservatively calculated projected area is used. Therefore, the aerosol settling area is defined as one half the projected area of the diameter of the pipe.

The key parameter in the removal equations is the settling velocity of the material of interest. Reference 15.6-10, provides values for aerosol settling velocities. This analysis conservatively uses an aerosol settling velocity equal to ¼ of 10^{th} percentile value from Reference 15.6-10.

The elemental removal is based on the same methodology except that the elemental iodine deposition velocity based on information provided by J. E. Cline, in Reference 15.6-15. Re-suspension of elemental iodine is also included in the analysis.

Figure 15.6-3 shows the various leakage pathways for the LOCA.

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment and the volumetric discharge rate from the secondary containment. Upon receipt of appropriate signals, the reactor building ventilation isolation valves isolate the reactor building atmosphere in 10 seconds. This rapid closure time prevents possible uncontrolled escape of radioactivity. Upon reactor building isolation, the recirculation system is designed to circulate the reactor building air to provide a delay mechanism whereby radioisotopes are retained in the reactor building and undergo radioactive decay rather than direct escape through the SGTS. A further function of the recirculation system is to provide thorough mixing of the recirculated flow to ensure that the SGTS cannot extract an unmixed quantity of radioactivity. Any fission product removal effects in the secondary containment such as plateout, are neglected; however, the effects of decay are considered. A mixing efficiency of 50 percent has conservatively been assumed in the analysis although a higher efficiency is expected.

The performance characteristic of the SGTS will be verified by periodic tests. The system removal efficiency is designed to be in excess of 99 percent removal of all forms of iodine and 0.3 micron or larger particulates. The SGTS has a design flow of one air change per day of the secondary containment.

The activity buildup in the secondary containment and activity release to environment are presented in Tables 15.6-13 and 15.6-14 respectively.

15.6.5.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD computer code (Reference 15.6-9) is used to evaluate the radiological consequences for this event. Specific values of parameters used in the evaluation are presented in Table 15.6-22.

15.6.5.5.2.1 Fission Product Release from Fuel

GE LOCA evaluations have determined that 10x10 fuel is expected to produce lower peak cladding temperature than 9x9 fuel in a given reactor. This is reasonable since the smaller rod diameter and larger surface area decreases stored energy. Since Susquehanna specific GE analyses of the 9x9-2 fuel (using the best estimate SAFER/GESTR-LOCA methodology) showed no fuel failures Since, for EPU conditions the LOCA analysis results are comparable to those determined for recent pre-EPU LOCA analyses, it is expected that a realistic analysis at EPU conditions would result in no fuel failures. (Reference 15.6-13), no ATRIUM™-10 fuel failures would be expected in a realistic analysis.

Thus, for the realistic analysis of dose consequences, no failures are assumed. Since this accident does not result in any fuel damage, the only activity released to the drywell is that activity contained in the reactor coolant plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

The design bases coolant iodine concentrations:

I-131	0.013 μCi/gm
I-132	0.12 μCi/gm
I-133	0.089 μ Ci/gm
I-134	0.24 μCi/gm
I-135	0.13 μCi/gm

For iodine, the pre LOCA initial reactor coolant iodine concentrations are based on Improved Technical Specification equilibrium operating I-131 Dose Equivalent (DE) TEDE reactor coolant system specific activity limit of 0.2 μCi/gm. The equivalent reactor coolant iodine activity concentrations are then given as follows:

I-131	0.0479 μCi/gm
I-132	0.442 μCi/gm
I-133	0.328 μCi/gm
I-134	0.884 μCi/gm
I-135	0.479 μCi/gm

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. Measurements performed (Reference 15.6-4) at operating BWRs during reactor shutdown have been used to develop an analytical model for the prediction of iodine and noble gas spiking as a consequence of reactor scram and vessel depressurization. The spiking for the non coolant activation and other fission products isotopes was conservatively determined by using an activity spike model which increases the equilibrium activity release rates for the pertinent isotopes from the fuel by a factor of 500.

Considering that approximately 40% of the released liquid flashes to steam, it is conservatively assumed that 50% of the released iodine activity is airborne initially. The total activity airborne in the containment is presented in Table 15.6-15.

15.6.5.5.2.2 Fission Product Transport to the Environment

The leak rate from the primary containment to the secondary containment is 1.0%/day where 50% mixing is assumed to occur. The transport pathways for the released activity to reach the environment are the same as described in section 15.6.5.5.1.2. The activity buildup in the secondary containment is presented in Table 15.6-16. The integrated isotopic activity released to the environment is presented in Table 15.6-17.

15.6.5.5.3 Results

15.6.5.5.3.1 Offsite Exposure

The radiological exposures resulting from the activity released to the environment as a consequence of the LOCA have been determined for the realistic and design basis cases. The design basis doses use the 0.5 percent direction dependent X/Q's and the analytical model as described in Appendix 15B. The realistic doses use the 50 percent direction independent X/Q's and the same dose model. The design basis and realistic LOCA doses are presented in Table 15.6-18.

15.6.5.5.3.2 Control Room Doses

Control room X/Q values were calculated using the ARCON 96 computer code in accordance with Regulatory Guide 1.194 (Reference 15.6-5 and 15.6-6). NRC occupancy factors were assumed. A detailed description of the control room dose model can be found in Appendix 15B. The radiological exposure of the control room personnel for the design and realistic basis case is given in Table 15.6-21.

15.6.6 FEEDWATER LINE BREAK-OUTSIDE CONTAINMENT

In order to evaluate the plant response to large liquid process line pipe break outside containment, the failure of a feedwater line is assumed. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and in the 30” diameter feedwater header, just downstream of the reactor feedwater pumps.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been qualitatively analyzed in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3.

It is assumed that the reactor is operating at an initial power level of 4032 MWt for this analysis.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed, to strict emergency codes and standards, and to severe seismic environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-23.

15.6.6.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for this accident. However, by procedural requirements the operator will perform the following actions which are shown below for informational purposes:

(1) The operator determines that line break has occurred and evacuates the area of the turbine building.

(2) The operator is not required to take any action to prevent primary reactor system mass loss, but should ensure that the reactor is shut down and that RCIC and/or HPCI are operating normally.

(3) The operator will implement site radiation incident procedures.

(4) If possible, the operator will shutdown the feedwater system and will deenergize any electrical equipment which may be damaged by the feedwater system in the turbine building.

(5) The operator will continue to monitor reactor water level and the performance of the ECCS systems while the radiation incident procedure is being implemented and begins normal reactor cooldown measures.

(6) When the reactor pressure has decreased below 100 psi, the operator will initiate RHR in the shutdown cooling mode to continue cooling down the reactor.

The above operator procedures occur over an elapsed time of 3-4 hours.

15.6.6.2.2 Systems Operations

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS system. The reactor protection system (safety relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF systems and RCIC/HPCI systems are assumed to operate normally.

15.6.6.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general loss-of-coolant accident break spectrum considered in Section 6.3. The general single-failure analysis for loss-of-coolant accidents is discussed in detail in Subsection 6.3.3.3. For the feedwater line break outside the containment which can be isolated, either the RCIC or the HPCI can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either the HPCI or the RCIC would still provide sufficient flow to keep the core covered with water. See Appendix 15A for further description of the analysis.

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative assessment of the consequences of the postulated failure (i.e., severance) of one of the feedwater piping lines external to the containment.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either the steamline breaks outside the containment (analysis presented in References 15.6-11 and 15.6-12 and Subsection 15.6.4) or the feedwater line break inside the containment. It is qualitatively evaluated as less limiting than the design basis accident (the recirculation line break analysis presented in Subsections 6.3.3 and 15.6.5).
The RCIC and HPCI initiate on low low-water level and together restore the reactor water level to the normal elevation. The low-low-low water level for reactor isolation is not expected to be reached. The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

Sections 6.3 and 7.3 contain discussions of uncertainties associated with ECCS Performance and Containment Isolation Systems, respectively.

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power-conversion system boundary. A 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 main steamlines as described in Subsection 15.6.4. The feedwater system piping break is less severe than the main steamline break. Results of the main steamline break analysis can be found in Subsection 15.6.4.3.2.

15.6.6.5 Radiological Consequences

Two separate radiological analyses are provided for this accident.

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis” and should not exceed a small fraction (I,e., 10 percent) of 10CFR50.57 guidelines.

The dose consequences of the Feedwater Line Break represent an upper bound on feedwater release and provide the maximum steam release for this event. The feedwater line break is postulated in the 30” diameter feedwater header, just downstream of the reactor feed pumps, to maximize the discharge enthalpy. The reactor scrams on low level immediately following the break. The entire hotwell inventory is pumped through the feedwater heater strings and out of the break.

All of the feedwater released through the break is processed through the condensate demineralizer with an assumed iodine decontamination factor of ten. The activity released from the hypothetical feedwater line break accident is a function of the coolant activity, accident duration, and mass of coolant released. The calculated total mass of coolant released for the realistic and design basis accidents are 2.50 X 10^6 pounds. Of this 2.24 X 10^6 pounds is liquid and 2.60 X 10^5 pounds is steam.
For all analyses, ten percent of the iodine activity in the coolant and 100 percent of the iodine activity in the flashed steam is assumed to become airborne. Although there will be some activation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine activity released from the break and to the environment are presented in Table 15.6-25.

The specific models, assumptions, and parameters used in the analysis are presented in Table 15.6-24.

A schematic of the release path is shown in Figure 15.6-4.

15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident.

Since this event is bounded by the Main Steam Line Break, the guidelines and dose acceptance criteria of Standard Review Plan 15.6.4 are applied to this analysis. Consistent with this guidance two cases are analyzed. In the first case, the reactor water iodine concentration existing at the time of the break is assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. In the second case, the iodine concentration prior to the break is assumed to be 4.0 micro-curies/gram dose equivalent I-131, which is the maximum short-term concentration permitted by the Technical Specifications. This concentration corresponds to an assumed pre-existing iodine spike.

15.6.6.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The reactor water iodine concentration existing at the time of the break is assumed to be equal to the design basis reactor coolant values given in Table 11.1-2. The specific models, assumptions and parameters used in the evaluation are presented in Table 15.6-24. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-4.

15.6.6.5.3 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to occurrence of the break) is released from the contained piping system prior to isolation closure.

Noble gas activity in the condensate is negligible since the air ejectors remove practically all noble gas from the condenser.

15.6.6.5.4 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and partitioning and unfiltered release to the environment through the turbine building ventilation system. The release of activity to the environment is presented in Table 15.6-25. The release is assumed to take place within 2 hours of the occurrence of the break.

15.6.6.5.5 Results

Offsite

The calculated exposures at the site boundary and low population zone for the design basis and realistic analyses are presented in Table 15.6-26. For the design basis analysis, the consequences of Case 1 and case 2 are less than 10% of the 10CFR100 limits. The realistic analysis dose consequences are well within 10CFR50.67 guidelines.

Control Room

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.6-4. The radiological exposure to the control room personnel for the design basis case is given in Table 15.6-26. The calculated dose meets the 10CFR50.67 control room dose acceptance criterion.

15.6.7 REFERENCES

15.6-1 F. J. Moody, "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-2 NEDO-21143-1, "Conservative Radiological Accident Evaluation - The CØNACO3 Code."

15.6-3 Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide."

15.6-4 Brutschy, F. J., G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," NEDO-10585, August, 1972.

15.6-5 NUREG/CR-6331, ARCON96, Atmospheric Relative Concentrations in Building Wakes, Revision 1, May, 1997.

15.6-6 USNRC Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

15.6-7 Not Used

15.6-8 Not Used

15.6-9 NUREG/CR-6604, RADTRAD A Simplified Model for RADionuclideTransport and Removal And Dose Estimation, and Supplement 1, 6/8/99.

15.6-10 AEB-98-03, Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NEREG-1465) Source Term, 12/9/98.

15.6-11 NE-092-001A, Rev. 1, “SSES Power Uprate Licensing Topical Report”, and NRC letter dated November 30, 1993, from Thomas E. Murley to Robert G. Byram (PP&L), Subject: Licensing Topical Report for Power Uprate with Increased Core Flow, Rev 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC NOS. M83426 and M83427) with enclosed Safety Evaluation Report.

15.6-12 NEDC-32161P, December 1993, (General Electric Report),” Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2.

15.6-13 “Safter/GESTR-LOCA Analysis Basis Documentation for Susquehanna Steam Electric Station Units 1 and 2,” NEDC-32281P, September 1993.

15.6-14 Letter from Fermi 2 to USNRC, NRC-03-0095, Response to NRC Request for Additional Information Regarding the Implementation of Alternative Source Term, NRC Docket No. 50-341, 12/12/03.

15.6-15 J.E. Cline, MSIV Leakage Iodine Transport Analysis, Letter Report, 3/26/1991 (ADAMS Accession Number ML003683718).

TABLE 15.6-1 SEQUENCE OF EVENTS FOR INSTRUMENT LINE BREAK	
Time	Event
0	Instrument line fails.
0 - 10 min.	Identification of break attempted.
10 min.	Manual Scram
310 minutes	Reactor Vessel depressurized and break flow terminated.

TABLE 15.6-2

MASS RELEASES - INSTRUMENT LINE BREAK

Time (minutes)	Liquid Fraction		Mass Steam From Dome (lbm)
	Mass Steam (lbm)	Mass Liquid (lbm)	
0	0	0	0
10	2,036	4,195	926
70	8,736	17,190	5,046
130	12,592	27,963	6,801
190	14,532	36,452	7,532
250	15,368	43,066	7,971
310	15,589	50,007	8,117

TABLE 15.6-3

INSTRUMENT LINE BREAK
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Realistic Analysis[1]	
	EAB[2]	LPZ[3]
I-131	3.39E-01	4.52E-01
I-132	3.13E+00	4.18E+00
I-133	2.32E+00	3.10E+00
I-134	6.25E+00	8.35E+00
I-135	3.39E+00	4.52E+00
Co-58	3.49E-02	4.68E-02
Co-60	3.49E-03	4.68E-03
Sr-89	2.17E-02	2.90E-02
Sr-90	1.61E-03	2.15E-03
Sr-91	4.82E-01	6.45E-01
Sr-92	7.69E-01	1.03E+00
Zr-95	2.80E-04	3.74E-04
Zr-97	2.24E-04	2.99E-04
Nb-95	2.94E-04	3.93E-04
Mo-99	1.54E-01	2.06E-01
Tc-99m	1.96E+00	2.62E+00
Ru-103	1.33E-04	1.78E-04
Ru-106	1.82E-05	2.43E-05
Te-129m	2.80E-04	3.74E-04
Te-132	3.42E-01	4.58E-01
Cs-134	1.12E-03	1.50E-03
Cs-136	7.69E-04	1.03E-03
Cs-137	1.68E-03	2.24E-03
Ba-139	1.12E+00	1.50E+00
Ba-140	6.29E-02	8.42E-02
Ce-141	2.73E-04	3.65E-04
Ce-143	2.45E-04	3.27E-04
Ce-144	2.45E-04	3.27E-04
Pr-143	2.66E-04	3.55E-04
Nd-147	9.79E-05	1.31E-04
Np-239	1.68E+00	2.24E+00
Kr-83m	4.38E-01	4.68E-02
Kr-85m	8.46E-01	4.68E-03
Kr-85	3.02E-03	2.90E-02
Kr-87	2.27E+00	2.15E-03
Kr-88	2.72E+00	6.45E-01
Kr-89	2.72E-02	1.03E+00
Xe-131m	2.27E-03	3.74E-04
Xe-133m	4.23E-02	2.99E-04
Xe-133	1.24E+00	3.93E-04
Xe-135m	1.04E+00	2.06E-01
Xe-135	3.32E+00	2.62E+00
Xe-137	1.01E-01	1.78E-04
Xe-138	3.17E+00	2.43E-05

(1) Based on 0.2 µCi/gram dose equivalent I-131.
(2) 2 hour release.
(3) Duration of accident release.

TABLE 15.6-4

INSTRUMENT LINE BREAK INSIDE SECONDARY CONTAINMENT RADIOLOGICAL CONSEQUENCES (REALISTIC ANALYSIS)

Exclusion Area Boundary (2 hr)	Low Population Zone (duration)	Control Structure Habitability Envelope (duration)
Rem TEDE	Rem TEDE	Rem TEDE
8.38E-04 [(1)]	9.36E-05	3.36E-02

(1) 8.38E-04 = 8.38×10^{-04}

TABLE 15.6-5

INSTRUMENT LINE BREAK ACCIDENT
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

I.	Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents	Realistic Analysis
A.	Reactor power (MWt)	4032
B.	Fuel damage	None
C.	Reactor coolant activity before the accident	
	1. Iodine concentration in coolant (μCi/gm I-131 dose equivalent)	0.2
	2. Noble gas release rate for duration of the accident (μCi/sec at 30 minutes decay)	100,000
D.	Iodine carry over fraction reactor water to steam (percent)	8
II.	Data and Assumptions Used to Estimate Activity Released to the Environment	
A.	Mass releases:	Table 15.6-2
B.	Fraction of noble gases airborne from reactor steam release	100 percent
C.	Fraction of iodines airborne from reactor steam release and from coolant liquid release that flashes to steam	100 percent
D.	Fraction of iodines airborne from reactor coolant liquid release	10 percent
E.	Credit taken for holdup in secondary containment	None
F.	Iodine removal efficiency of the Standup Gas Treatment System	0 percent
G.	Plate-out inside Secondary Containment	0 percent
III.	Data and Assumptions Used To Evaluate Control Room Doses	
A.	Control structure habitability envelope free volume (ft^3)	518,000
B.	Control Room free volume (ft^3)	110,000
C.	Control Structure filtered air intake flow (cfm)	5229 – 6391
D.	Control structure unfiltered outside air filtration rate – ingress/egress (cfm)	10
E.	Control structure unidentified unfiltered outside air infiltration rate (cfm)	500
F.	Control structure filter efficiency for iodine (percent)	99
IV.	Dispersion Data	
A.	EAB and LPZ distance (meters)	549/4827
B.	X/Qs for time intervals	Table 2.3-92
C.	X/Qs for LPZ	Table 2.3-105
D.	X/Qs for Control Structure Habitability Envelope	Appendix 15B
V.	Dose Data	
A.	Method of dose calculations	Appendix 15B
B.	Dose conversion assumptions	Appendix 15B
C.	Activity released to environment	Table 15.6-3
D.	Doses	Table 15.6-4

TABLE 15.6-6

SEQUENCE OF EVENTS FOR STEAM LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	Guillotine break of one main steam line outside primary containment
~0.5	High steamline flow signal initiates closure of main steam line isolation valve.
<1.25	Reactor begins scram.
6.0	Main steam isolation valves fully closed.
~131	Safety relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1200 psig.
330	Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1200 psig.
2109	Operator initiates ADS. Vessel depressurizes rapidly.
~2300	Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
2365	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT (curies)
(Design Basis Analysis)

Isotope	Case 1 Activity (1)	Case 2 Activity (2)
I-131	2.40E+00	4.79E+01
I-132	2.21E+01	4.42E+02
I-133	1.64E+01	3.28E+02
I-134	4.42E+01	8.85E+02
I-135	2.40E+01	4.79E+02
Kr-83m	4.19E-01	4.19E-01
Kr-85m	7.29E-01	7.29E-01
Kr-85	2.41E-03	2.41E-03
Kr-87	2.36E+00	2.36E+00
Kr-88	2.45E+00	2.45E+00
Kr-89	1.56E+01	1.56E+01
Xe-131m	1.81E-03	1.81E-03
Xe-133m	3.39E-02	3.39E-02
Xe-133	9.89E-01	9.89E-01
Xe-135m	3.15E+00	3.15E+00
Xe-135	2.75E+00	2.75E+00
Xe-137	1.84E+01	1.84E+01
Xe-138	1.10E+01	1.10E+01

Notes:

1. Iodine concentration in coolant = 0.2 μCi/g dose equivalent I-131
2. Iodine concentration in coolant = 4.0 μCi/g dose equivalent I-131

TABLE 15.6-8

STEAM LINE BREAK ACCIDENT ACTIVITY RELEASED TO THE ENVIRONMENT (curies) (Realistic Analysis)

Isotope	Activity
I-131	1.45E-01
I-132	1.35E+00
I-133	9.96E-01
I-134	2.72E+00
I-135	1.46E+00
Kr-83m	4.27E-02
Kr-85m	7.45E-02
Kr-85	2.46E-04
Kr-87	2.41E-01
Kr-88	2.51E-01
Xe-131m	1.84E-04
Xe-133m	3.46E-03
Xe-133	1.01E-01
Xe-135m	3.21E-01
Xe-135	2.81E-01
Xe-138	1.12E+00

Table 15.6-9

STEAM LINE BREAK OUTSIDE CONTAINMENT
RADIOLOGICAL DOSES (rem TEDE)
Design Basis Case

Location / Dose Type	Design Basis Analysis		Realistic Analysis
	Case 1 (1)	Case 2 (2)	
Acceptance Criteria - Offsite	2.5	25	2.5
EAB	0.10	2.0	9.80E-04
LPZ	0.006	0.12	3.60E-05
Acceptance Criteria - CRHE	5.0	5.0	5
CRHE	0.05	0.93	7.20E-03

Notes:

1. Case 1: Iodine concentration in coolant = 0.2 μCi/g dose equivalent I-131
2. Case 2: Iodine concentration in coolant = 4.0 μCi/g dose equivalent I-131

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TABLE 15.6-10

STEAM LINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Engineering Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt)	4032	4032
B. Fuel damaged	None	None
C. Reactor coolant activity before the accident		
1. Iodine concentration in coolant		
Realistic (μCi/gm)	-	Table 11.1-2
Conservative case 1(μCi/gm dose-equivalent I-131)	0.2	N/A
Conservative case 2(μCi/gm dose-equivalent I-131)	4.0	N/A
2. Noble gas release rate prior to MSIV closure (μCi/sec at 30 min decay)	403,000	100,000
D. Iodine carryover fraction reactor water to steam (percent)	2	2
II. Data and assumption used to estimate activity released		
A. Isolation valve closure time (sec)	5.5	5.5
B. Coolant mass releases from break (lb_m)		
Total	97,970(2)	40,316(1)
Liquid	84,840	19,994
Steam from flashed liquid	6,480	8,000
Steam from steam dome	6,650	20,322
C. Fraction of iodine in:		
Released coolant assumed airborne (%)	100	100
Steam from flashed liquid (%)	100	100
Steam from steam dome (%)	8	8
D. Holdup in Turbine Building	No	No
III. Data and Assumptions Used to Evaluate Control Room Doses		
A. Control structure habitability envelope free volume (ft^3)	518,000	518,000
B. Control Room free volume (ft^3)	110,000	110,000
C. Control structure filtered air intake flow (cfm)	5229 – 6391	5229 – 6391
D. Control Structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Disposition Data		
A. Release height (m)	0	0
B. Boundary for SB/LPZ distance (meters)	549/4827	549/4827
C. X/Q's for SB (0-2 hrs) sec/m^3	Table 2.3-92 (0.5 percentile)	Table 2.3-119 (50 percentile)
D. X/Q's for LPZ(0-8 hrs) sec/m^3	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (50 percentile)
E. XQ's for Control Room sec/m^3(puff)	6.3E-04	1.6E-03
V. Dose Data		
A. Method of dose calculation	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.6-9	Table 15.6-9

(1) The 8000 lbm of steam from flashed liquid is included in the 19,994 lbm liquid release. The total release is the sum of the mass of liquid plus the mass of steam from the steam dome.

(2) Values as shown are increased by 20% in dose analysis.

TABLE 15.6-11
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Design Basis Accident)

Isotope	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)								
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58			4.45E+02	1.15E+03	6.74E+00	1.28E-03	9.18E-07	4.68E-13		
Co-60			2.39E+02	6.19E+02	3.64E+00	6.98E-04	5.13E-07	2.77E-13		
Kr-85	2.47E+04	7.40E+04	5.43E+05	1.48E+06	1.48E+06	1.47E+06	1.44E+06	1.40E+06	1.33E+06	1.27E+06
Kr-85m	4.36E+05	1.24E+06	8.42E+06	1.97E+07	7.76E+06	6.48E+05	9.27E+00	1.90E-09		
Kr-87	8.16E+05	2.04E+06	1.14E+07	1.80E+07	6.83E+05	1.11E+02				
Kr-88	1.19E+06	3.30E+06	2.14E+07	4.58E+07	1.06E+07	2.11E+05	4.86E-03			
Rb-86	3.40E+03	9.02E+03	2.02E+04	3.86E+04	2.25E+02	4.21E-02	2.77E-05	1.20E-11		
Sr-89			6.17E+05	1.60E+06	9.34E+03	1.78E+00	1.25E-03	6.25E-10		
Sr-90			7.87E+04	2.04E+05	1.20E+03	2.29E-01	1.69E-04	9.13E-11		
Sr-91			7.32E+05	1.76E+06	6.67E+03	3.98E-01	1.53E-06			
Sr-92			6.47E+05	1.30E+06	1.64E+03	5.26E-03				
Y-90			1.06E+03	3.70E+03	9.57E+01	5.21E-02	1.09E-04	8.50E-11		
Y-91			8.08E+03	2.11E+04	1.34E+02	2.79E-02	2.06E-05	1.04E-11		
Y-92			4.55E+04	2.11E+05	2.73E+03	5.23E-02	3.94E-11			
Y-93			5.99E+03	1.45E+04	5.63E+01	3.60E-03	1.89E-08			
Zr-95			1.15E+04	2.98E+04	1.75E+02	3.32E-02	2.37E-05	1.20E-11		
Zr-97			1.10E+04	2.72E+04	1.25E+02	1.24E-02	4.77E-07			
Nb-95			1.15E+04	2.98E+04	1.75E+02	3.36E-02	2.47E-05	1.32E-11		
Mo-99			1.50E+05	3.84E+05	2.12E+03	3.44E-01	1.19E-04	1.42E-11		
Tc-99m			1.34E+05	3.47E+05	1.98E+03	3.44E-01	1.22E-04	1.45E-11		
Ru-103			1.29E+05	3.34E+05	1.95E+03	3.70E-01	2.58E-04	1.26E-10		
Ru-105			7.64E+04	1.69E+05	3.89E+02	6.14E-03	5.94E-11			
Ru-106			5.15E+04	1.33E+05	7.82E+02	1.50E-01	1.10E-04	5.86E-11		
Rh-105			8.33E+04	2.15E+05	1.19E+03	1.74E-01	3.14E-05			
Sb-127			1.40E+05	3.59E+05	2.02E+03	3.43E-01	1.47E-04	2.71E-11		
Sb-129			4.44E+05	9.78E+05	2.19E+03	3.23E-02	2.28E-10			
Te-127			1.40E+05	3.61E+05	2.10E+03	3.83E-01	1.91E-04	5.30E-11		
Te-127m			2.39E+04	6.17E+04	3.63E+02	6.95E-02	5.08E-05	2.68E-11		
Te-129			4.72E+05	1.13E+06	3.38E+03	2.94E-01	1.72E-04	8.21E-11		
Te-129m			9.99E+04	2.58E+05	1.51E+03	2.87E-01	1.99E-04	9.49E-11		
Te-131m			3.15E+05	7.97E+05	4.08E+03	5.40E-01	7.53E-05			
Te-132			2.29E+06	5.88E+06	3.27E+04	5.45E+00	2.12E-03	3.20E-10		
I-131	1.68E+06	4.48E+06	1.17E+07	2.35E+07	1.68E+06	1.46E+06	1.11E+06	6.42E+05	2.58E+05	1.04E+05
I-132	2.37E+06	6.06E+06	1.53E+07	2.83E+07	4.54E+05	3.64E+03	3.66E-03	5.53E-10		
I-133	3.47E+06	9.18E+06	2.35E+07	4.61E+07	2.75E+06	1.48E+06	1.33E+05	1.06E+03	3.40E-01	1.09E-04
I-134	3.38E+06	6.94E+06	1.22E+07	1.12E+07	7.10E+03	2.09E-02				
I-135	3.25E+06	8.41E+06	2.08E+07	3.79E+07	1.47E+06	2.53E+05	1.31E+02	3.52E-05		
Xe-133	3.54E+06	1.06E+07	7.77E+07	2.11E+08	2.04E+08	1.86E+08	1.23E+08	5.42E+07	1.38E+07	3.50E+06
Xe-135	1.19E+06	3.68E+06	2.74E+07	7.61E+07	4.99E+07	1.50E+07	6.30E+04	1.05E+00	1.12E-08	
Cs-134	3.60E+05	9.56E+05	2.14E+06	4.11E+06	2.41E+04	4.63E+00	3.39E-03	1.83E-09		
Cs-136	1.15E+05	3.05E+05	6.83E+05	1.31E+06	7.57E+03	1.40E+00	8.80E-04	3.47E-10		
Cs-137	2.71E+05	7.19E+05	1.61E+06	3.09E+06	1.82E+04	3.48E+00	2.56E-03	1.39E-09		

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TABLE 15.6-11
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Design Basis Accident)

Isotope	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Ba-139			7.09E+05	1.11E+06	3.19E+02	1.96E-05				
Ba-140			1.17E+06	3.03E+06	1.76E+04	3.25E+00	2.03E-03	7.93E-10		
La-140			1.83E+04	6.99E+04	2.11E+03	1.11E+00	1.79E-03	8.95E-10		
La-141			8.95E+03	1.94E+04	3.96E+01	4.52E-04				
La-142			6.67E+03	1.10E+04	4.36E+00	6.28E-07				
Ce-141			2.70E+04	6.98E+04	4.08E+02	7.73E-02	5.33E-05	2.54E-11		
Ce-143			2.45E+04	6.22E+04	3.22E+02	4.41E-02	7.16E-06			
Ce-144			2.27E+04	5.87E+04	3.45E+02	6.60E-02	4.82E-05	2.57E-11		
Pr-143			9.67E+03	2.51E+04	1.50E+02	2.95E-02	2.10E-05	8.65E-12		
Nd-147			4.35E+03	1.12E+04	6.49E+01	1.19E-02	7.26E-06	2.69E-12		
Np-239			3.15E+05	8.04E+05	4.39E+03	6.92E-01	2.10E-04	1.95E-11		
Pu-238			6.84E+01	1.77E+02	1.04E+00	1.99E-04	1.47E-07	7.95E-14		
Pu-239			7.27E+00	1.88E+01	1.11E-01	2.12E-05	1.57E-08	8.53E-15		
Pu-240			1.17E+01	3.02E+01	1.78E-01	3.41E-05	2.51E-08	1.36E-14		
Pu-241			2.88E+03	7.45E+03	4.38E+01	8.40E-03	6.18E-06	3.34E-12		
Am-241			1.53E+00	3.95E+00	2.32E-02	4.48E-06	3.38E-09	1.92E-15		
Cm-242			4.00E+02	1.03E+03	6.06E+00	1.16E-03	8.43E-07	4.45E-13		
Cm-244			2.34E+01	6.06E+01	3.56E-01	6.83E-05	5.02E-08	2.72E-14		
SUM	2.21E+07	5.80E+07	2.44E+08	5.50E+08	2.81E+08	2.06E+08	1.26E+08	5.63E+07	1.54E+07	4.87E+06

TABLE 15.6-12

THIS TABLE HAS BEEN INTENTIONALLY LEFT BLANK

TABLE 15.6-13
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58			4.58E-02	3.57E-01	4.27E-01	6.66E-02	1.48E-04	1.06E-04	7.17E-05	4.85E-05
Co-60			2.47E-02	1.92E-01	2.30E-01	3.62E-02	8.28E-05	6.26E-05	4.66E-05	3.47E-05
Kr-85	8.23E-01	7.38E+00	6.84E+01	4.53E+02	2.82E+03	4.77E+03	2.53E+03	2.45E+03	2.33E+03	2.21E+03
Kr-85m	1.45E+01	1.24E+02	1.06E+03	6.02E+03	1.48E+04	2.11E+03	1.62E-02	0.00E+00	0.00E+00	0.00E+00
Kr-87	2.72E+01	2.04E+02	1.44E+03	5.52E+03	1.31E+03	3.60E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	3.98E+01	3.29E+02	2.70E+03	1.40E+04	2.02E+04	6.88E+02	8.50E-06	0.00E+00	0.00E+00	0.00E+00
Rb-86	1.16E-01	9.56E-01	3.82E+00	1.49E+01	1.57E+01	2.40E+00	4.87E-03	2.94E-03	1.52E-03	7.82E-04
Sr-89			6.36E+01	4.95E+02	5.91E+02	9.21E+01	2.02E-01	1.41E-01	9.19E-02	5.99E-02
Sr-90			8.11E+00	6.31E+01	7.57E+01	1.19E+01	2.72E-02	2.06E-02	1.54E-02	1.15E-02
Sr-91			7.54E+01	5.45E+02	4.22E+02	2.07E+01	2.47E-04	5.13E-09	0.00E+00	0.00E+00
Sr-92			6.66E+01	4.02E+02	1.04E+02	2.73E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90			1.17E-01	1.34E+00	6.14E+00	2.71E+00	1.77E-02	1.92E-02	1.54E-02	1.16E-02
Y-91			8.34E-01	6.56E+00	8.48E+00	1.45E+00	3.34E-03	2.35E-03	1.56E-03	1.04E-03
Y-92			5.81E+00	8.89E+01	1.76E+02	2.73E+00	6.38E-09	0.00E+00	0.00E+00	0.00E+00
Y-93			6.17E-01	4.49E+00	3.57E+00	1.87E-01	3.06E-06	1.18E-10	0.00E+00	0.00E+00
Zr-95			1.19E+00	9.24E+00	1.11E+01	1.72E+00	3.82E-03	2.71E-03	1.82E-03	1.22E-03
Zr-97			1.13E+00	8.43E+00	7.90E+00	6.44E-01	7.70E-05	1.59E-07	6.30E-12	0.00E+00
Nb-95			1.19E+00	9.24E+00	1.11E+01	1.74E+00	3.98E-03	2.99E-03	2.17E-03	1.56E-03
Mo-99			1.55E+01	1.19E+02	1.34E+02	1.78E+01	1.92E-02	3.20E-03	1.92E-04	1.15E-05
Tc-99m			1.38E+01	1.08E+02	1.26E+02	1.79E+01	1.96E-02	3.28E-03	1.97E-04	1.18E-05
Ru-103			1.33E+01	1.04E+02	1.24E+02	1.92E+01	4.17E-02	2.84E-02	1.78E-02	1.11E-02
Ru-105			7.87E+00	5.24E+01	2.46E+01	3.18E-01	9.58E-09	0.00E+00	0.00E+00	0.00E+00
Ru-106			5.30E+00	4.13E+01	4.95E+01	7.76E+00	1.77E-02	1.32E-02	9.71E-03	7.12E-03
Rh-105			8.57E+00	6.66E+01	7.55E+01	9.03E+00	5.07E-03	2.28E-04	1.54E-06	1.04E-08
Sb-127			1.44E+01	1.11E+02	1.28E+02	1.78E+01	2.38E-02	6.11E-03	7.54E-04	9.31E-05
Sb-129			4.57E+01	3.03E+02	1.39E+02	1.68E+00	3.69E-08	0.00E+00	0.00E+00	0.00E+00
Te-127			1.44E+01	1.12E+02	1.33E+02	1.99E+01	3.09E-02	1.20E-02	5.07E-03	3.15E-03
Te-127m			2.46E+00	1.91E+01	2.30E+01	3.61E+00	8.21E-03	6.06E-03	4.27E-03	3.00E-03
Te-129			4.86E+01	3.49E+02	2.14E+02	1.52E+01	2.77E-02	1.85E-02	1.13E-02	6.85E-03
Te-129m			1.03E+01	8.01E+01	9.59E+01	1.49E+01	3.20E-02	2.14E-02	1.30E-02	7.92E-03
Te-131m			3.25E+01	2.47E+02	2.58E+02	2.80E+01	1.22E-02	3.31E-04	9.65E-07	2.81E-09
Te-132			2.36E+02	1.82E+03	2.07E+03	2.83E+02	3.42E-01	7.23E-02	6.43E-03	5.73E-04
I-131	5.79E+01	4.81E+02	2.10E+03	8.89E+03	1.26E+04	7.44E+03	3.01E+03	1.66E+03	6.21E+02	2.35E+02
I-132	8.14E+01	6.33E+02	2.58E+03	9.66E+03	4.38E+03	4.42E+02	4.51E+01	1.06E+01	9.40E-01	8.37E-02
I-133	1.20E+02	9.86E+02	4.24E+03	1.74E+04	2.06E+04	7.56E+03	3.60E+02	2.74E+00	8.16E-04	2.46E-07
I-134	1.17E+02	7.45E+02	2.19E+03	4.23E+03	5.30E+01	1.07E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	1.12E+02	9.03E+02	3.75E+03	1.43E+04	1.10E+04	1.29E+03	3.56E-01	9.09E-08	0.00E+00	0.00E+00
Xe-133	1.18E+02	1.06E+03	9.82E+03	6.49E+04	3.98E+05	6.41E+05	2.78E+05	1.21E+05	2.93E+04	7.16E+03
Xe-135	4.08E+01	4.00E+02	3.71E+03	2.50E+04	1.44E+05	1.25E+05	9.19E+02	1.45E-02	1.27E-10	0.00E+00
Cs-134	1.23E+01	1.01E+02	4.06E+02	1.59E+03	1.68E+03	2.64E+02	5.96E-01	4.48E-01	3.32E-01	2.46E-01
Cs-136	3.91E+00	3.23E+01	1.29E+02	5.04E+02	5.28E+02	8.00E+01	1.55E-01	8.51E-02	3.74E-02	1.65E-02

TABLE 15.6-13
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Cs-137	9.22E+00	7.63E+01	3.05E+02	1.19E+03	1.27E+03	1.99E+02	4.50E-01	3.40E-01	2.54E-01	1.90E-01
Ba-139			7.30E+01	3.44E+02	2.02E+01	1.02E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140			1.21E+02	9.40E+02	1.11E+03	1.68E+02	3.28E-01	1.79E-01	7.77E-02	3.37E-02
La-140			2.06E+00	2.62E+01	1.35E+02	5.77E+01	2.89E-01	2.02E-01	9.01E-02	3.91E-02
La-141			9.22E-01	6.02E+00	2.51E+00	2.34E-02	1.64E-10	0.00E+00	0.00E+00	0.00E+00
La-142			6.87E-01	3.41E+00	2.76E-01	3.26E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141			2.78E+00	2.16E+01	2.59E+01	4.01E+00	8.61E-03	5.74E-03	3.46E-03	2.09E-03
Ce-143			2.53E+00	1.93E+01	2.04E+01	2.29E+00	1.16E-03	4.25E-05	2.05E-07	9.92E-10
Ce-144			2.34E+00	1.82E+01	2.18E+01	3.42E+00	7.79E-03	5.81E-03	4.24E-03	3.09E-03
Pr-143			9.97E-01	7.77E+00	9.48E+00	1.53E+00	3.39E-03	1.95E-03	8.78E-04	3.93E-04
Nd-147			4.48E-01	3.48E+00	4.11E+00	6.19E-01	1.17E-03	6.08E-04	2.42E-04	9.60E-05
Np-239			3.24E+01	2.49E+02	2.78E+02	3.59E+01	3.40E-02	4.40E-03	1.73E-04	6.82E-06
Pu-238			7.05E-03	5.49E-02	6.58E-02	1.03E-02	2.37E-05	1.80E-05	1.34E-05	1.00E-05
Pu-239			7.48E-04	5.83E-03	7.00E-03	1.10E-03	2.54E-06	1.93E-06	1.44E-06	1.08E-06
Pu-240			1.20E-03	9.37E-03	1.12E-02	1.77E-03	4.05E-06	3.06E-06	2.29E-06	1.71E-06
Pu-241			2.97E-01	2.31E+00	2.77E+00	4.36E-01	9.97E-04	7.55E-04	5.63E-04	4.20E-04
Am-241			1.57E-04	1.22E-03	1.47E-03	2.32E-04	5.46E-07	4.33E-07	3.48E-07	2.79E-07
Cm-242			4.12E-02	3.20E-01	3.84E-01	6.01E-02	1.36E-04	1.00E-04	7.19E-05	5.15E-05
Cm-244			2.41E-03	1.88E-02	2.25E-02	3.54E-03	8.11E-06	6.13E-06	4.58E-06	3.42E-06
SUM	7.55E+02	6.08E+03	3.54E+04	1.81E+05	6.39E+05	7.92E+05	2.85E+05	1.25E+05	3.23E+04	9.61E+03

TABLE 15.6-14
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

Isotope	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)							
	0.5 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58	0.00E+00	8.86E-03	2.28E-02	2.65E-02	2.72E-02	2.72E-02	2.72E-02	2.72E-02
Co-60	0.00E+00	4.77E-03	1.23E-02	1.43E-02	1.47E-02	1.47E-02	1.47E-02	1.47E-02
Kr-85	7.59E-01	5.01E+01	1.50E+03	1.05E+04	4.11E+04	1.11E+05	2.51E+05	4.08E+05
Kr-85m	1.30E+01	7.14E+02	1.20E+04	2.71E+04	2.85E+04	2.85E+04	2.85E+04	2.85E+04
Kr-87	2.23E+01	7.91E+02	4.33E+03	4.68E+03	4.68E+03	4.68E+03	4.68E+03	4.68E+03
Kr-88	3.48E+01	1.73E+03	2.16E+04	3.47E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04
Rb-86	2.53E-02	3.83E-01	8.70E-01	1.02E+00	1.06E+00	1.10E+00	1.12E+00	1.14E+00
Sr-89	0.00E+00	1.23E+01	3.16E+01	3.68E+01	3.77E+01	3.77E+01	3.77E+01	3.77E+01
Sr-90	0.00E+00	1.57E+00	4.03E+00	4.70E+00	4.82E+00	4.82E+00	4.82E+00	4.82E+00
Sr-91	0.00E+00	1.40E+01	3.31E+01	3.57E+01	3.58E+01	3.58E+01	3.58E+01	3.58E+01
Sr-92	0.00E+00	1.13E+01	2.25E+01	2.28E+01	2.28E+01	2.28E+01	2.28E+01	2.28E+01
Y-90	0.00E+00	2.56E-02	1.12E-01	2.02E-01	2.38E-01	2.38E-01	2.38E-01	2.38E-01
Y-91	0.00E+00	1.62E-01	4.23E-01	5.01E-01	5.15E-01	5.15E-01	5.15E-01	5.15E-01
Y-92	0.00E+00	1.39E+00	6.07E+00	6.95E+00	6.96E+00	6.96E+00	6.96E+00	6.96E+00
Y-93	0.00E+00	1.15E-01	2.73E-01	2.95E-01	2.97E-01	2.97E-01	2.97E-01	2.97E-01
Zr-95	0.00E+00	2.29E-01	5.90E-01	6.87E-01	7.04E-01	7.04E-01	7.04E-01	7.04E-01
Zr-97	0.00E+00	2.13E-01	5.23E-01	5.78E-01	5.83E-01	5.83E-01	5.83E-01	5.83E-01
Nb-95	0.00E+00	2.30E-01	5.91E-01	6.88E-01	7.06E-01	7.06E-01	7.06E-01	7.06E-01
Mo-99	0.00E+00	2.97E+00	7.55E+00	8.66E+00	8.82E+00	8.82E+00	8.82E+00	8.82E+00
Tc-99m	0.00E+00	2.67E+00	6.85E+00	7.92E+00	8.08E+00	8.08E+00	8.08E+00	8.08E+00
Ru-103	0.00E+00	2.57E+00	6.61E+00	7.69E+00	7.88E+00	7.88E+00	7.88E+00	7.88E+00
Ru-105	0.00E+00	1.40E+00	3.05E+00	3.15E+00	3.16E+00	3.16E+00	3.16E+00	3.16E+00
Ru-106	0.00E+00	1.02E+00	2.64E+00	3.07E+00	3.15E+00	3.15E+00	3.15E+00	3.15E+00
Rh-105	0.00E+00	1.66E+00	4.23E+00	4.84E+00	4.91E+00	4.91E+00	4.91E+00	4.91E+00
Sb-127	0.00E+00	2.78E+00	7.08E+00	8.16E+00	8.33E+00	8.33E+00	8.33E+00	8.33E+00
Sb-129	0.00E+00	8.13E+00	1.76E+01	1.82E+01	1.82E+01	1.82E+01	1.82E+01	1.82E+01
Te-127	0.00E+00	2.78E+00	7.15E+00	8.30E+00	8.49E+00	8.49E+00	8.49E+00	8.49E+00
Te-127m	0.00E+00	4.75E-01	1.22E+00	1.42E+00	1.46E+00	1.46E+00	1.46E+00	1.46E+00
Te-129	0.00E+00	9.02E+00	2.07E+01	2.20E+01	2.21E+01	2.21E+01	2.21E+01	2.21E+01
Te-129m	0.00E+00	1.99E+00	5.12E+00	5.96E+00	6.11E+00	6.11E+00	6.11E+00	6.11E+00
Te-131m	0.00E+00	6.20E+00	1.55E+01	1.75E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01
Te-132	0.00E+00	4.54E+01	1.16E+02	1.33E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02
I-131	1.30E+01	2.27E+02	6.86E+02	1.57E+03	4.54E+03	8.49E+03	1.17E+04	1.29E+04
I-132	1.78E+01	2.83E+02	6.10E+02	6.60E+02	6.63E+02	6.63E+02	6.63E+02	6.63E+02
I-133	2.68E+01	4.52E+02	1.29E+03	2.43E+03	3.72E+03	3.85E+03	3.85E+03	3.85E+03
I-134	2.29E+01	1.85E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02
I-135	2.48E+01	3.88E+02	9.83E+02	1.35E+03	1.43E+03	1.43E+03	1.43E+03	1.43E+03
Xe-133	1.09E+02	7.17E+03	2.12E+05	1.44E+06	5.17E+06	9.91E+06	1.32E+07	1.42E+07
Xe-135	3.67E+01	2.61E+03	7.76E+04	3.83E+05	6.16E+05	6.18E+05	6.18E+05	6.18E+05
Cs-134	2.69E+00	4.07E+01	9.24E+01	1.08E+02	1.14E+02	1.18E+02	1.23E+02	1.26E+02
Cs-136	8.57E-01	1.30E+01	2.93E+01	3.43E+01	3.58E+01	3.68E+01	3.75E+01	3.78E+01

TABLE 15.6-14
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)							
Isotope	0.5 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Cs-137	2.02E+00	3.06E+01	6.96E+01	8.16E+01	8.56E+01	8.88E+01	9.26E+01	9.51E+01
Ba-139	0.00E+00	1.10E+01	1.85E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01
Ba-140	0.00E+00	2.34E+01	5.99E+01	6.96E+01	7.12E+01	7.12E+01	7.12E+01	7.12E+01
La-140	0.00E+00	4.71E-01	2.28E+00	4.26E+00	4.98E+00	4.98E+00	4.98E+00	4.98E+00
La-141	0.00E+00	1.63E-01	3.47E-01	3.57E-01	3.57E-01	3.57E-01	3.57E-01	3.57E-01
La-142	0.00E+00	1.06E-01	1.84E-01	1.85E-01	1.85E-01	1.85E-01	1.85E-01	1.85E-01
Ce-141	0.00E+00	5.37E-01	1.38E+00	1.61E+00	1.65E+00	1.65E+00	1.65E+00	1.65E+00
Ce-143	0.00E+00	4.83E-01	1.21E+00	1.37E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00
Ce-144	0.00E+00	4.52E-01	1.16E+00	1.36E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00
Pr-143	0.00E+00	1.93E-01	4.98E-01	5.82E-01	5.98E-01	5.98E-01	5.98E-01	5.98E-01
Nd-147	0.00E+00	8.65E-02	2.22E-01	2.57E-01	2.63E-01	2.63E-01	2.63E-01	2.63E-01
Np-239	0.00E+00	6.22E+00	1.58E+01	1.81E+01	1.84E+01	1.84E+01	1.84E+01	1.84E+01
Pu-238	0.00E+00	1.36E-03	3.51E-03	4.09E-03	4.19E-03	4.19E-03	4.19E-03	4.19E-03
Pu-239	0.00E+00	1.45E-04	3.72E-04	4.34E-04	4.45E-04	4.45E-04	4.45E-04	4.45E-04
Pu-240	0.00E+00	2.33E-04	5.99E-04	6.98E-04	7.15E-04	7.15E-04	7.15E-04	7.15E-04
Pu-241	0.00E+00	5.74E-02	1.48E-01	1.72E-01	1.76E-01	1.76E-01	1.76E-01	1.76E-01
Am-241	0.00E+00	3.04E-05	7.82E-05	9.12E-05	9.36E-05	9.36E-05	9.36E-05	9.36E-05
Cm-242	0.00E+00	7.95E-03	2.05E-02	2.38E-02	2.44E-02	2.44E-02	2.44E-02	2.44E-02
Cm-244	0.00E+00	4.66E-04	1.20E-03	1.40E-03	1.43E-03	1.43E-03	1.43E-03	1.43E-03
SUM	3.27E+02	1.49E+04	3.33E+05	1.91E+06	5.90E+06	1.07E+07	1.42E+07	1.53E+07

TABLE 15.6-15
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Realistic Analysis)

	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)				
Isotope	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	8.18E+02	8.01E+02	7.56E+02	2.96E+01	3.08E+00
I-132	1.45E+03	1.57E+03	1.59E+03	8.99E-01	0.00E+00
I-133	1.84E+03	1.50E+03	8.82E+02	4.06E+00	3.71E-09
I-134	4.51E+02	3.93E+00	1.26E-05	0.00E+00	0.00E+00
I-135	1.53E+03	8.18E+02	1.53E+02	4.07E-03	0.00E+00
Xe-131m	4.98E+01	4.92E+01	4.76E+01	4.02E+01	8.87E+00
Xe-133m	2.47E+02	2.32E+02	1.94E+02	7.62E+01	2.01E-02
Xe-133	8.79E+03	8.56E+03	7.95E+03	5.41E+03	1.75E+02
Xe-135m	2.78E+02	1.46E+02	7.09E+01	1.89E-03	0.00E+00
Xe-135	7.48E+03	5.15E+03	1.76E+03	7.71E+00	0.00E+00
Xe-138	2.25E+01	4.81E-07	0.00E+00	0.00E+00	0.00E+00
Kr-83m	3.23E+02	3.32E+01	7.75E-02	0.00E+00	0.00E+00
Kr-85m	1.25E+03	4.94E+02	4.16E+01	6.03E-04	0.00E+00
Kr-87	1.11E+03	4.22E+01	6.89E-03	0.00E+00	0.00E+00
Kr-88	2.87E+03	6.64E+02	1.34E+01	3.12E-07	0.00E+00
Kr-85	3.74E+02	3.74E+02	3.74E+02	3.73E+02	3.70E+02
Co-58	6.49E-01	6.48E-01	6.43E-01	4.66E-04	0.00E+00
Co-60	6.00E-02	6.00E-02	6.00E-02	4.47E-05	0.00E+00
Sr-89	2.08E+02	2.07E+02	2.05E+02	1.47E-01	0.00E+00
Sr-90	7.78E+00	7.78E+00	7.78E+00	5.80E-03	0.00E+00
Sr-91	2.33E+03	1.50E+03	4.68E+02	1.83E-03	0.00E+00
Sr-92	2.24E+03	4.83E+02	8.07E+00	6.05E-11	0.00E+00
Zr-95	1.40E+00	1.39E+00	1.38E+00	1.00E-03	0.00E+00
Zr-97	1.44E+00	1.12E+00	5.83E-01	2.27E-05	0.00E+00
Nb-95	1.42E+00	1.42E+00	1.42E+00	1.06E-03	0.00E+00
Mo-99	7.28E+02	6.83E+02	5.77E+02	2.02E-01	0.00E+00
Tc-99m	7.67E+03	4.15E+03	1.13E+03	2.07E-01	0.00E+00
Ru-103	6.59E-01	6.56E-01	6.48E-01	4.59E-04	0.00E+00
Ru-106	9.00E-02	8.99E-02	8.98E-02	6.66E-05	0.00E+00
Te-129m	1.36E+00	1.35E+00	1.33E+00	9.34E-04	0.00E+00
Te-132	1.62E+03	1.54E+03	1.34E+03	5.27E-01	0.00E+00
Cs-134	1.57E+01	1.57E+01	1.57E+01	1.17E-02	0.00E+00
Cs-136	3.73E+00	3.68E+00	3.56E+00	2.26E-03	0.00E+00
Cs-137	8.13E+00	8.13E+00	8.13E+00	6.06E-03	0.00E+00
Ba-139	2.13E+03	1.04E+02	3.34E-02	0.00E+00	0.00E+00
Ba-140	3.03E+02	2.99E+02	2.88E+02	1.82E-01	0.00E+00
Ce-141	1.33E+00	1.32E+00	1.30E+00	9.11E-04	0.00E+00
Ce-143	1.14E+00	1.01E+00	7.19E-01	1.18E-04	0.00E+00
Ce-144	1.24E+00	1.24E+00	1.24E+00	9.16E-04	0.00E+00
Pr-143	1.28E+00	1.28E+00	1.26E+00	8.47E-04	0.00E+00
Nd-147	4.68E-01	4.60E-01	4.41E-01	2.72E-04	0.00E+00
Np-239	7.91E+03	7.34E+03	6.03E+03	1.86E+00	0.00E+00

TABLE 15.6-16
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Realistic Analysis)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)				
	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	7.70E-01	2.20E+00	3.19E+00	6.00E-01	3.19E-02
I-132	1.21E+00	3.39E+00	5.17E+00	6.06E-03	0.00E+00
I-133	1.73E+00	4.13E+00	3.72E+00	8.24E-02	3.83E-11
I-134	4.25E-01	1.08E-02	5.31E-08	0.00E+00	0.00E+00
I-135	1.44E+00	2.24E+00	6.44E-01	8.26E-05	0.00E+00
Xe-131m	3.62E-02	1.05E-01	1.59E-01	8.23E-02	2.16E-02
Xe-133m	1.81E-01	5.10E-01	7.07E-01	2.12E-01	4.27E-05
Xe-133	6.39E+00	1.84E+01	2.71E+01	1.12E+01	2.98E-01
Xe-135m	3.52E-01	4.50E-01	3.04E-01	4.10E-05	0.00E+00
Xe-135	5.71E+00	1.31E+01	8.76E+00	4.29E-02	0.00E+00
Xe-138	1.63E-02	1.02E-09	0.00E+00	0.00E+00	0.00E+00
Kr-83m	2.34E-01	7.03E-02	2.53E-04	0.00E+00	0.00E+00
Kr-85m	9.07E-01	1.04E+00	1.35E-01	1.03E-06	0.00E+00
Kr-87	8.06E-01	8.93E-02	2.24E-05	0.00E+00	0.00E+00
Kr-88	2.08E+00	1.40E+00	4.35E-02	5.33E-10	0.00E+00
Kr-85	2.71E-01	7.91E-01	1.22E+00	6.38E-01	5.59E-01
Co-58	4.71E-04	1.37E-03	2.10E-03	4.37E-06	0.00E+00
Co-60	4.35E-05	1.27E-04	1.95E-04	4.19E-07	0.00E+00
Sr-89	1.51E-01	4.38E-01	6.68E-01	1.38E-03	0.00E+00
Sr-90	5.64E-03	1.64E-02	2.53E-02	5.44E-05	0.00E+00
Sr-91	1.69E+00	3.18E+00	1.52E+00	1.71E-05	0.00E+00
Sr-92	1.63E+00	1.02E+00	2.63E-02	0.00E+00	0.00E+00
Zr-95	1.01E-03	2.95E-03	4.51E-03	9.37E-06	0.00E+00
Zr-97	1.04E-03	2.38E-03	1.90E-03	2.13E-07	0.00E+00
Nb-95	1.03E-03	3.00E-03	4.62E-03	9.89E-06	0.00E+00
Mo-99	5.27E-01	1.44E+00	1.88E+00	1.89E-03	0.00E+00
Tc-99m	5.56E+00	8.78E+00	3.70E+00	1.94E-03	0.00E+00
Ru-103	4.78E-04	1.39E-03	2.11E-03	4.30E-06	0.00E+00
Ru-106	6.52E-05	1.90E-04	2.93E-04	6.24E-07	0.00E+00
Te-129m	9.84E-04	2.86E-03	4.34E-03	8.75E-06	0.00E+00
Te-132	1.18E+00	3.26E+00	4.35E+00	4.94E-03	0.00E+00
Cs-134	1.14E-02	3.33E-02	5.12E-02	1.10E-04	0.00E+00
Cs-136	2.71E-03	7.79E-03	1.16E-02	2.12E-05	0.00E+00
Cs-137	5.89E-03	1.72E-02	2.65E-02	5.68E-05	0.00E+00
Ba-139	1.55E+00	2.21E-01	1.09E-04	0.00E+00	0.00E+00
Ba-140	2.19E-01	6.31E-01	9.38E-01	1.71E-03	0.00E+00
Ce-141	9.62E-04	2.79E-03	4.24E-03	8.53E-06	0.00E+00
Ce-143	8.27E-04	2.13E-03	2.34E-03	1.11E-06	0.00E+00
Ce-144	8.99E-04	2.62E-03	4.03E-03	8.58E-06	0.00E+00
Pr-143	9.28E-04	2.70E-03	4.11E-03	7.93E-06	0.00E+00
Nd-147	3.39E-04	9.73E-04	1.44E-03	2.55E-06	0.00E+00
Np-239	5.73E+00	1.55E+01	1.97E+01	1.74E-02	0.00E+00

TABLE 15.6-17
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Realistic Analysis)

	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)				
Isotope	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	2.45E+00	9.66E+00	2.78E+01	3.46E+01	3.90E+01
I-132	4.13E+00	1.79E+01	5.52E+01	6.60E+01	6.60E+01
I-133	5.66E+00	2.05E+01	4.75E+01	5.28E+01	5.29E+01
I-134	3.16E+00	3.97E+00	3.98E+00	3.98E+00	3.98E+00
I-135	5.07E+00	1.52E+01	2.42E+01	2.47E+01	2.47E+01
Xe-131m	1.53E-01	6.45E-01	2.04E+00	5.03E+00	1.61E+01
Xe-133m	7.65E-01	3.15E+00	9.30E+00	1.86E+01	2.42E+01
Xe-133	2.71E+01	1.13E+02	3.51E+02	8.08E+02	1.61E+03
Xe-135m	1.44E+00	3.48E+00	6.57E+00	6.84E+00	6.84E+00
Xe-135	2.43E+01	8.70E+01	1.81E+02	2.11E+02	2.11E+02
Xe-138	3.57E+00	3.58E+00	3.58E+00	3.58E+00	3.58E+00
Kr-83m	1.45E+00	2.67E+00	2.82E+00	2.82E+00	2.82E+00
Kr-85m	4.46E+00	1.24E+01	1.74E+01	1.77E+01	1.77E+01
Kr-87	5.99E+00	9.09E+00	9.22E+00	9.22E+00	9.22E+00
Kr-88	1.12E+01	2.58E+01	3.03E+01	3.04E+01	3.04E+01
Kr-85	1.15E+00	4.87E+00	1.56E+01	4.04E+01	2.12E+02
Co-58	1.94E-03	7.71E-03	2.28E-02	2.74E-02	2.74E-02
Co-60	1.79E-04	7.13E-04	2.12E-03	2.55E-03	2.55E-03
Sr-89	6.20E-01	2.47E+00	7.28E+00	8.75E+00	8.75E+00
Sr-90	2.32E-02	9.25E-02	2.74E-01	3.30E-01	3.30E-01
Sr-91	7.46E+00	2.42E+01	4.46E+01	4.64E+01	4.64E+01
Sr-92	8.63E+00	1.87E+01	2.13E+01	2.13E+01	2.13E+01
Zr-95	4.18E-03	1.66E-02	4.91E-02	5.90E-02	5.90E-02
Zr-97	4.46E-03	1.58E-02	3.48E-02	3.77E-02	3.77E-02
Nb-95	4.24E-03	1.69E-02	5.01E-02	6.03E-02	6.03E-02
Mo-99	2.19E+00	8.47E+00	2.31E+01	2.69E+01	2.69E+01
Tc-99m	2.54E+01	7.59E+01	1.27E+02	1.32E+02	1.32E+02
Ru-103	1.97E-03	7.82E-03	2.31E-02	2.77E-02	2.77E-02
Ru-106	2.69E-04	1.07E-03	3.17E-03	3.82E-03	3.82E-03
Te-129m	4.05E-03	1.61E-02	4.75E-02	5.70E-02	5.70E-02
Te-132	4.89E+00	1.90E+01	5.25E+01	6.13E+01	6.13E+01
Cs-134	4.70E-02	1.87E-01	5.55E-01	6.68E-01	6.68E-01
Cs-136	1.12E-02	4.42E-02	1.29E-01	1.54E-01	1.54E-01
Cs-137	2.43E-02	9.66E-02	2.87E-01	3.45E-01	3.45E-01
Ba-139	1.07E+01	1.66E+01	1.69E+01	1.69E+01	1.69E+01
Ba-140	9.05E-01	3.58E+00	1.04E+01	1.25E+01	1.25E+01
Ce-141	3.96E-03	1.57E-02	4.64E-02	5.57E-02	5.57E-02
Ce-143	3.47E-03	1.30E-02	3.29E-02	3.71E-02	3.71E-02
Ce-144	3.70E-03	1.47E-02	4.37E-02	5.26E-02	5.26E-02
Pr-143	3.82E-03	1.52E-02	4.49E-02	5.39E-02	5.39E-02
Nd-147	1.40E-03	5.53E-03	1.61E-02	1.91E-02	1.91E-02
Np-239	2.39E+01	9.17E+01	2.47E+02	2.86E+02	2.86E+02

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT
SUMMARY OF OFFSITE DOSES

Dose Location	Dose (Rem / TEDE)		
	Regulatory Limit	Design Basis Analysis	Realistic Analysis
<u>THYROID</u>			
2 Hour Site Boundary	25	1.20E+01(1)	1.7E-02
30 Day Low Population Zone	25	4.5E+00	4.3E-03

1. 1.07E+01 = $1.07 \times 10^{+01}$

TABLE 15.6-19

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TABLE 15.6-20

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TABLE 15.6-21 LOSS-OF-COOLANT ACCIDENT SUMMARY OF CONTROL ROOM OPERATOR DOSES (Design Basis Analysis)			
Dose	Control Room Operator Dose (Rem TEDE)		
	Regulatory Limit	Design Basis Analysis	Realistic Basis Analysis
30 Day Operator Dose	5	4.69	0.065

TABLE 15.6-22

LOSS-OF-COOLANT ACCIDENT
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
I. Data And Assumptions Used To Estimate Radioactive Source Term From Postulated Accidents		
A. Reactor power level (MWt)	4032	4032
B. Fuel damaged (percent)	100	0
C. Activity released to primary containment atmosphere	Subsection 15.6.5.5.1.1	Subsection 15.6.5.5.2.1
D. Activity released to suppression pool water	Subsection 15.6.5.5.1.1	Subsection 15.6.5.5.2.1
E. Iodine form fractions (percent)		
1. Aerosol	95	95
2. Elemental	4.85	4.85
3. Organic	0.15	0.15
II. Data And Assumptions Used To Estimate Activity Released		
A. Primary containment leak rate (percent/day)	1 (0-24 hr) 0.5 (1-30 d)	1 (0-24 hr) 0.5 (1-30 d)
B. Reactor building leak rate(percent/day)	140 (0-10 min) 0 > 10 min	140 (0-10 min) 0 > 10 min
C. Secondary containment bypass leak rate (SCFH)	15	15
D. MSIV Leakage (SCFH for 4 steam lines)	300	300
E. ESF leak rate		
1. Leakage rate inside reactor building (gpm)	20	20
2. Flashing fraction for iodine (percent)	10	10
Primary containment free volume (ft^3)		
Drywell	239600	239600
Wetwell	148590	148590
TOTAL	388190	388190
G. Reactor building free volume (ft^3)		
1. Zone 1	1,488,600	1,488,600
2. Zone 2	1,598,600	1,598,600
3. Zone 3	2,668,000	2,668,000
TOTAL Used in Analysis (Zone 1 + Zone 3)	4,156,600	4,156,600
H. Suppression pool water volume (ft^3)	132,000	132,000
I. Standby Gas Treatment System Parameters		
1. SGTS flow during drawdown (cfm)	11,110	11,110

TABLE 15.6-22

LOSS-OF-COOLANT ACCIDENT
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
2. Drawdown time to reach 0.25 inch of vacuum water gage in reactor building (minutes)	10	10
3. SGTS flow following drawdown (cfm)	4041	4041
4. SGTS filter efficiencies (percent)		
Iodine (All species)	99	99
J. Reactor Building Recirculation System Parameters		
1. Flow rate (cfm)	83,000	83,000
2. Mixing efficiency (percent)	50	50
3. Filter efficiency	0	0
4. Recirculation system actuation (seconds)	10 to 30	10 to 30
K. Post-LOCA activity concentrations in primary containment and reactor building	Tables 15.6-11, 15.6-13	Tables 15.6-15, 15.6-16
III. Data And Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume (ft^3)	518,000	518,000
B. Control room free volume(ft^3)	110,000	110,000
C. Control structure filtered air intake flow (cfm)	5229 - 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Dispersion Data		
A. Site Boundary/Low Population Zone distance (meters)	549/4827	549/4827
B. Site Boundary atmospheric dispersion factors	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (0.5 percentile)
C. Low Population Zone atmospheric dispersion factors	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (0.5 percentile)
D. Control room atmospheric dispersion factors	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of calculation	App 15B	Appendix 15B
B. Isotopic data and dose conversion factors	App 15B	Appendix 15B
C. Activity released to environment	Table 15.6-14	Table 15.6-17
D. Offsite doses	Table 15.6-18	Table 15.6-18
E. Control room doses	Table 15.6-21	Table 15.6-21

TABLE 15.6-23

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
5	Reactor scram on low water level.
<30	A low-low water reactor level RCIC and HPCI would initiate and are expected to maintain the water level above low-low-low level trip and eventually restore it to the normal elevation.
1 to 2 hours	Normal reactor cooldown procedure established.

TABLE 15.6-24

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS
FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt)	4032	4032
B. Fuel damaged	None	None
C. Reactor coolant iodine activity before the accident		
Realistic (μCi/gm)	N/A	Table 11.1-2
Conservative case 1 (μCi/gm dose-equivalent I-131)	0.2	N/A
Conservative case 2 (μCi/gm dose-equivalent I-131)	4.0	N/A
II. Data and Assumption Used to Estimate Activity Released		
A. Coolant mass releases from break (lb_m)		
Total	2.5×10^6	2.5×10^6
Liquid	2.24×10^6	2.24×10^6
Steam from flashed liquid	2.6×10^5	2.6×10^5
B. Fraction of iodine released from:		
Coolant assumed airborne (%)	100	100
Steam (%)	100	100
C. Holdup in Turbine Building	No	No
D. Iodine decontamination factor for the condensate demineralizer	10	10
III. Data And Assumptions Used to Evaluate Control Room Doses		
A. Control Structure habitability envelope free volume (ft^3)	518,000	518,000
B. Control room free volume(ft^3)	110,000	110,000
C. Control structure filtered air intake flow (cfm)	5229 – 6391	5229 – 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified outside air infiltration rate (cfm)\	500	500
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Disposition Data		
A. Boundary for SB/LPZ distance (meters)	549/4827	549/4827
B. Site Boundary X/Q	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (50 percentile)
C. LPZ X/Q	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (50 percentile)
D. CRHE X/Q	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of dose calculation	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.6-26	Table 15.6-26

TABLE 15.6-25

FEEDWATER LINE BREAK ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT (curies)

Isotope	Design Basis Analysis Case 1(1)	Design Basis Analysis Case 2(2)	Realistic Analysis
I-131	8.42E-02	1.68E+00	2.29E-02
I-132	7.78E+01	1.56E+01	2.11E-01
I-133	5.77E-01	1.15E+01	1.56E-01
I-134	1.56E+00	3.11E+01	4.22E-01
I-135	8.42E-01	1.68E+01	2.29E-01

Notes:
1. Iodine concentration in coolant = 0.2 µCi/gm dose equivalent I-131.
2. Iodine concentration in coolant = 4.0 µCi/gm dose equivalent I-131.

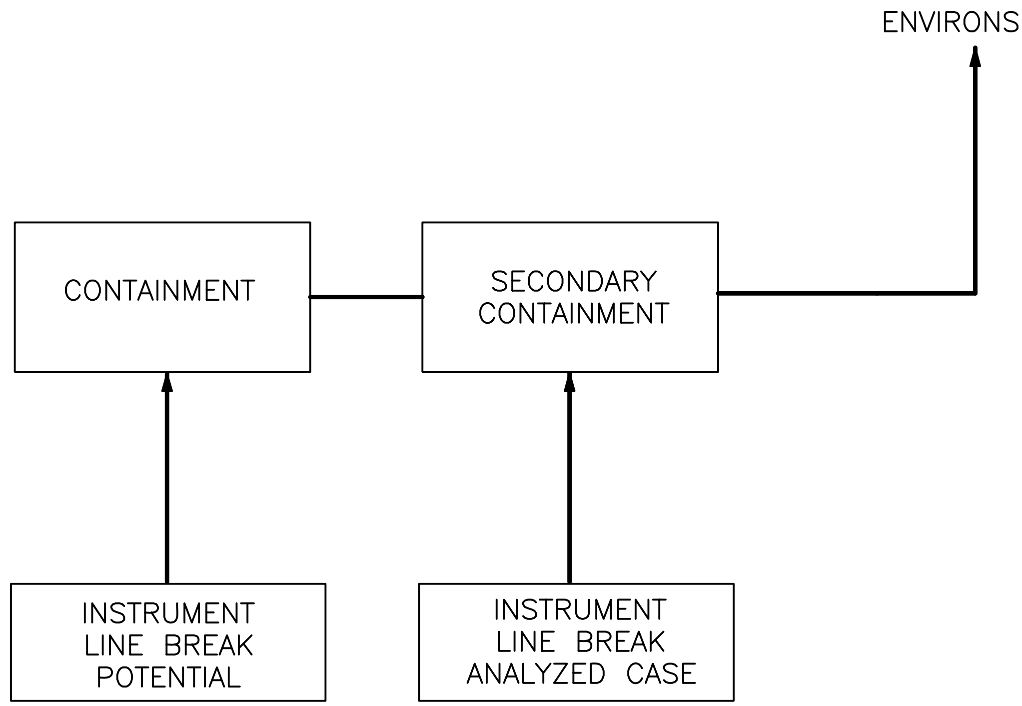
TABLE 15.6-26

FEEDWATER LINE BREAK OUTSIDE CONTAINMENT
RADIOLOGICAL DOSES (REM TEDE)

Location/ Dose Type	Design Basis Analysis		Realistic Analysis
	Case 1[1]	Case 2[2]	
Site Boundary (2 Hr)	3.40E-03	6.77E-02	1.44E-04
Low Population Zone (Duration)	2.01E-04	4.00E-03	5.32E-06
CRHE (Duration)	9.52E-04	1.90E-02	2.58E-04

Notes:

1. Case 1: iodine concentration in coolant = 0.2 μCi/gm dose equivalent I-131.
2. Case 2: iodine concentration in coolant = 4.0 μCi/gm dose equivalent I-131.



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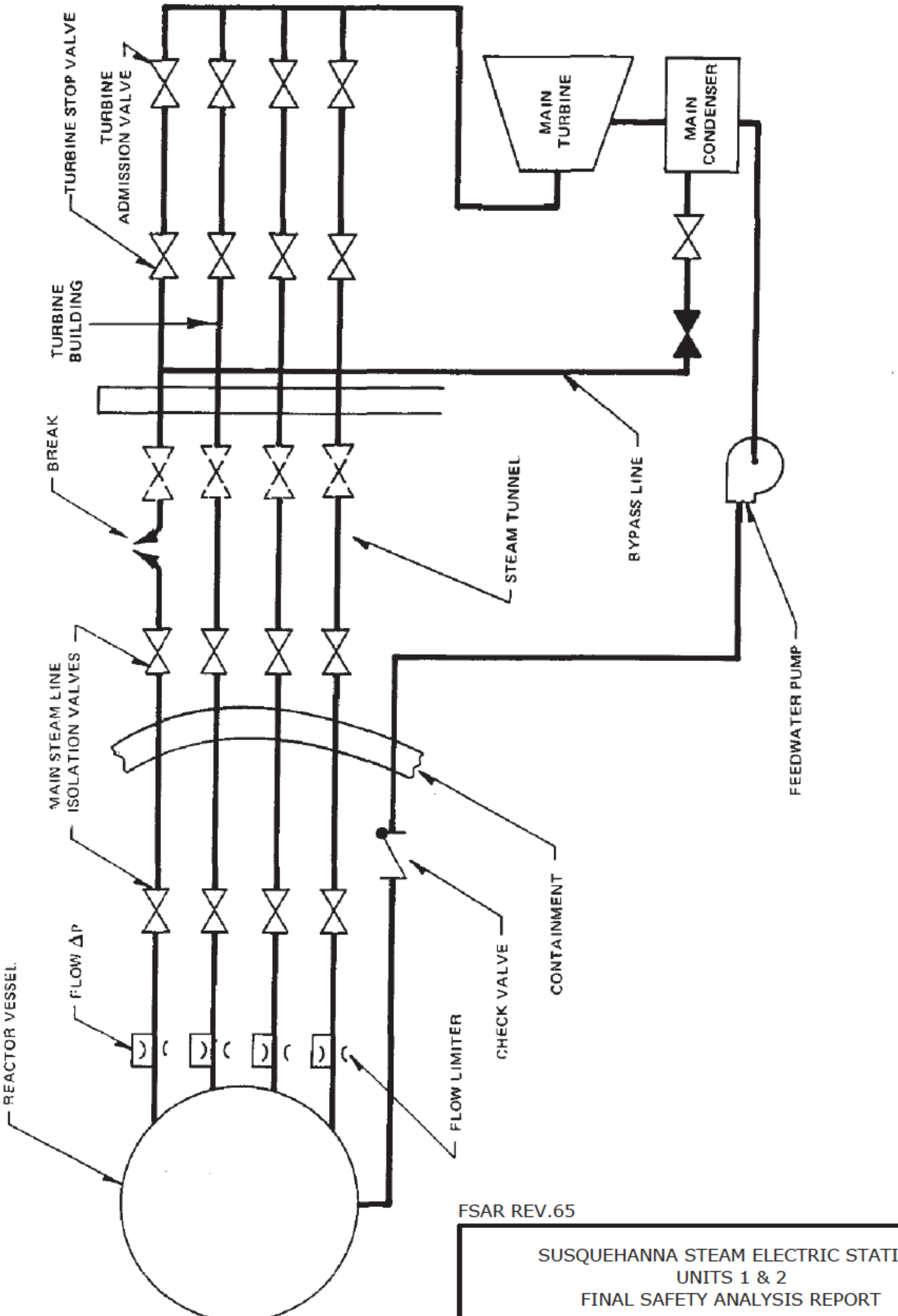
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LEAKAGE PATH
FOR
INSTRUMENT LINE BREAK

FIGURE 15.6-1, Rev 51

AutoCAD: Figure Fsar 15_6_1.dwg



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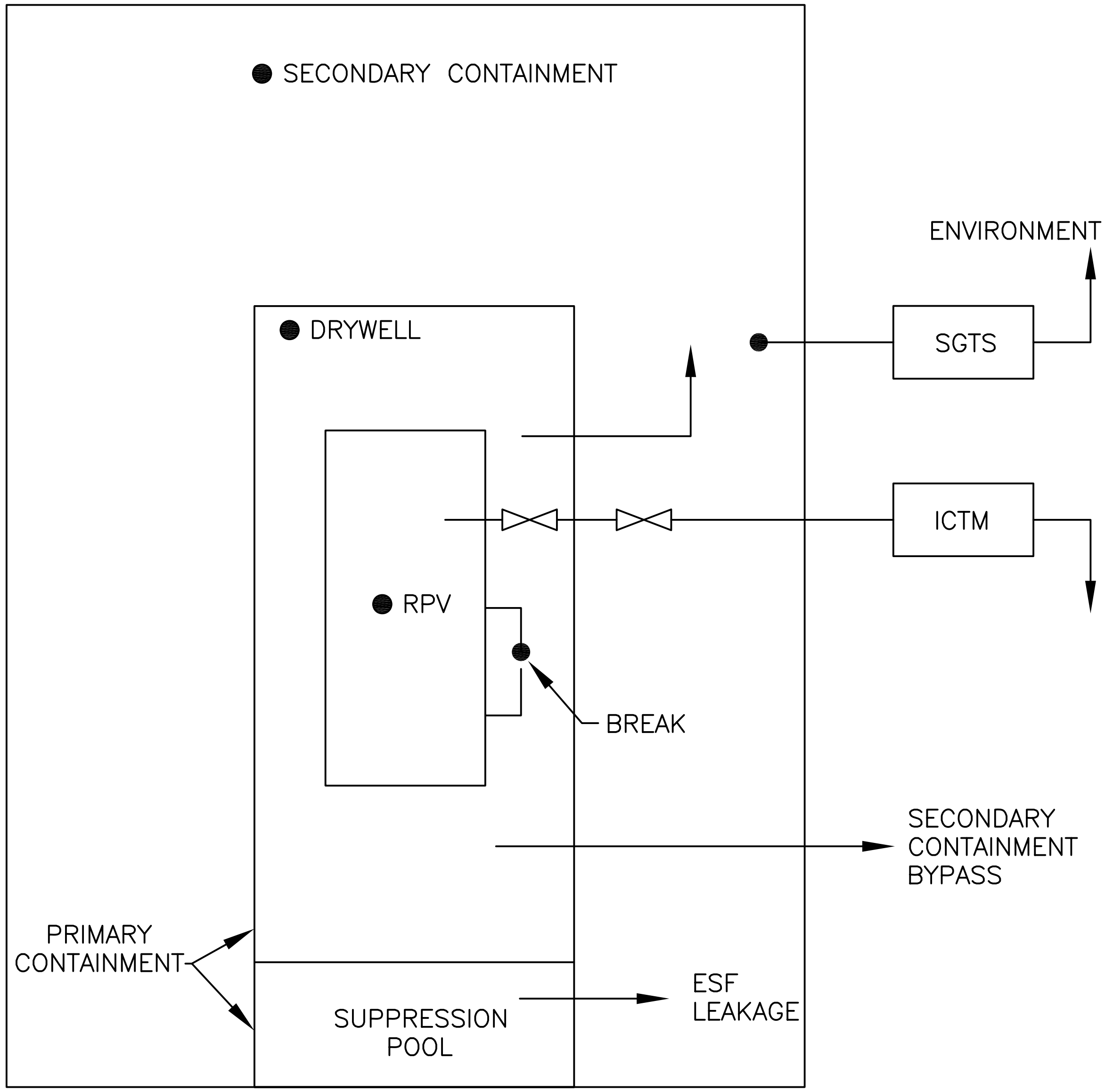
STEAM FLOW SCHEMATIC
FOR STEAM BREAK
OUTSIDE CONTAINMENT

FIGURE 15.6-2, Rev 49

AutoCAD: Figure Fsar 15_6_2.dwg



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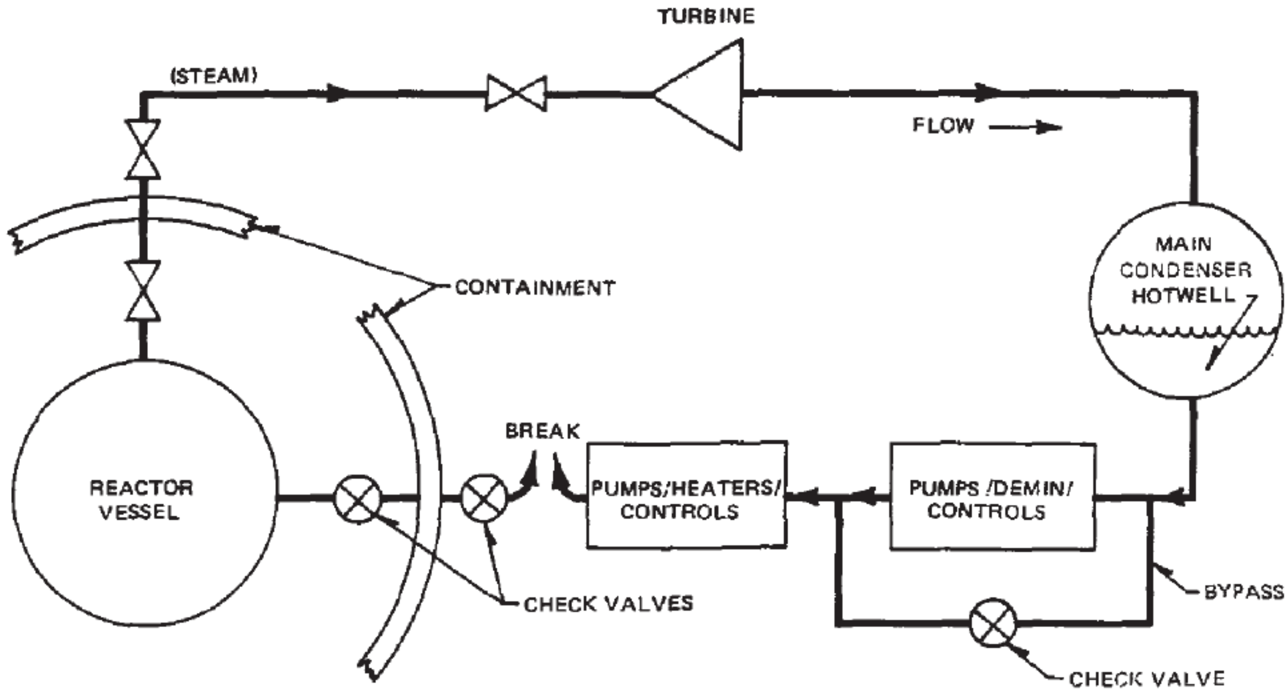
LEAKAGE FLOW FOR LOCA

FIGURE 15.6-3, Rev 53

AutoCAD: Figure Fsar 15_6_3.dwg



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UNITS 1 & 2
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LEAKAGE PATH FOR
FEEDWATER LINE BREAK
OUTSIDE CONTAINMENT

FIGURE 15.6-4, Rev 49

AutoCAD: Figure Fsar 15_6_4.dwg

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 GASEOUS RADWASTE SYSTEM LEAK OR FAILURE

The following gaseous radwaste system components are examined under severe failure mode conditions for effects on the plant safety profile:

(1) Main condenser offgas treatment system failure

(2) Malfunction of main turbine gland sealing system

(3) Failure of Air Ejector Lines.

15.7.1.1 Ambient Charcoal Offgas Treatment System Failure

15.7.1.1.1 Identification of Causes and Frequency Classification

15.7.1.1.1.1 Identification of Causes

Those potential events which could cause a gross failure in the offgas treatment system are:

(1) A seismic occurrence - greater than design basis

The seismic event is considered to be the most probable and most severe which the system is designed to prevent or accommodate. The seismic failure is the only conceivable event which could cause significant system damage.

(2) A hydrogen explosion in housing unit

The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a failure mode which would cause significant system damage.

(3) A fire in the filter assemblies, and

The decay heat on the filters is easily handled inherently by the system and certainly by the available air flows.

(4) Failure of spatially related equipment.

The system is reasonably isolated from other systems or components which could cause any serious interaction or failure.

The only credible event which could result in the release of significant activity to the environment is an earthquake.

Even though the offgas system is designed to uniform building code seismic requirements, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the offgas system.

The design basis, description, and performance evaluation of the subject system is given in Section 11.3.

15.7.1.1.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.1.2 Sequence of Events and System Operation

15.7.1.1.2.1 Sequence of Events

The offgas treatment system is assumed to fail, resulting in releases from the offgas system charcoal adsorption beds, delay line, and the SJAE. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

Detection and reporting of a failure in the offgas treatment system will be performed in accordance with plant operating procedures.

The sequence of events following this failure is shown in Table 15.7-1.

15.7.1.1.2.2 Identification of Operator Actions

Upon verification of a failure, the operator may initiate a normal shutdown of the reactor to reduce the gaseous activity being discharged.

Gross failure of this system may require manual isolation of this system from the main condenser. This isolation results in high condenser pressure and a reactor scram. The operator will monitor the turbine generator auxiliaries and break vacuum as soon as possible. The operator must notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for reentry.

15.7.1.1.2.3 Systems Operation

In analyzing the postulated Offgas System failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for plant operating procedures and the functioning of normally operating plant instruments and controls and other systems only in assuming the following:

(1) Capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by Area Radiation Monitoring System, loss of flow in the Offgas System, and/or in an alarmed increase in activity at the vent release.

(2) Capability to isolate the system and shutdown the reactor.

(3) Operational indicator and annunciators in the main control room.

15.7.1.1.2.4 The Effect of Single Failures and Operator Errors

After the initial system gross failure, the inability of the operator to actuate a system isolation could affect the analysis.

However, the seismic event which is assumed to occur beyond the present plant design basis for non-safety equipment will undoubtedly cause the tripping of turbine or will lead to a load rejection. This will initiate a scram and negate a need for the operator to initiate a reactor shutdown via system isolation.

See Appendix 15A for a further discussion.

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Subsection 15.2.5.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the Offgas System pressure boundary. No credit is taken for performance of secondary barriers.

15.7.1.1.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions, for the purpose of determining adequacy of the plant design to limit the offsite doses to levels that are well within 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic yet conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis.”

The radiological analyses assume that as a result of the offgas system failure, activity is released to the environment from the following sources:

- The total radioactive content of the offgas system charcoal adsorption beds.

- The total radioactive content of the offgas delay line.

- The release from the steam jet air ejector is assumed to occur from a break in the delay line just downstream of the SJAE. The SJAE is assumed to operate for a period of 1 hour after the accident. The release from the SJAE is assumed to be at ground level and a delay of 5 minutes is assumed to account for transit from the SJAE to the break in the delay line.

15.7.1.1.5.1 Design Basis Analysis

The gross failure of the offgas treatment system is assumed to release 100 percent of the noble gas inventory stored in the system with the continued release of offgas from the SJAE for a period of one hour. The radioactive content of the offgas during this period is conservatively based on an assumed 403,200 μCi/sec noble gas release rate at 30 minutes decay. The total activity released to the environment is given in Table 15.7-2. Specific parametric values used in this evaluation are presented in Table 15.7-5.

15.7.1.1.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The radiation source terms are based on a noble gas release rate from the SJAE equal to the design basis offgas release rate of 100,000 μCi/sec after 30 minutes delay. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-5. The total activity released to the environment is given in Table 15.7-3.

15.7.1.1.5.3 Results

The calculated exposures offsite and at the control room for the design basis and realistic analyses of the Offgas Treatment System failure are presented in Table 15.7-4. A detailed description of the control room model is shown in Appendix 15B. The radiation dose consequences are well within the guideline values given in 10CFR50.67.

15.7.1.2 Malfunction of Main Turbine Gland Sealing System

15.7.1.2.1 Identification of Causes and Frequency Classification

15.7.1.2.1.1 Identification of Causes

Possible causes of the malfunction of the Turbine Gland Sealing System include the failure of the gland steam evaporator and its backup steam supply, failure of the gland steam condenser exhausters, and excessive pressure in the steam seal header.

15.7.1.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.2.2 Sequence of Events and System Operation

15.7.1.2.2.1 Sequence of Events

15.7.1.2.2.2 Identification of Operator Actions

It is assumed that the system fails near the condenser. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

The operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result in a turbine trip and reactor shutdown.

15.7.1.2.2.3 System Operation

Failure of the gland steam evaporator and its backup steam supply would result in the discharge of a small amount of contaminated steam from the HP and LP shaft seals to the gland steam condenser exhauster.

Failure of both of the gland steam condenser exhausters results in the escape of clean steam from the HP and LP shaft seals.

Excessive pressure in the steam seal header as a result of a malfunction of the gland steam evaporator or the backup steam supply valve is prevented by a relief valve so that there is no detrimental effect on the operation of the seals.

15.7.1.2.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply systems (NSSS). It will, of course, lead to decoupling of the NSSS with the power conversion system.

The tripping of the main turbine via main condenser signals will result in an anticipated operational transient examined earlier in Chapter 15.

This failure has no effect on the core or the NSSS safety performance.

15.7.1.2.4 Barrier Analysis

This release occurs outside the containment hence does not involve any barrier integrity aspects.

15.7.1.2.5 Radiological Consequences

Each of the assumed malfunctions results in negligible releases of activity. Therefore, the doses which result from these failures are inconsequential.

15.7.1.3 Failure of Air Ejector Lines

15.7.1.3.1 Identification of Causes and Frequency Classification

An evaluation of events that could cause a failure of the air ejector line indicates that a seismic event more serious than the system is designed to withstand is the only event that could rupture the lines. The lines are designed to withstand the effects of a hydrogen explosion.

This event is categorized as a limiting fault.

15.7.1.3.2 Sequence of Events and Systems Operation

The sequence of the events following this failure is shown in Table 15.7-18.

It is assumed that the line leading from the steam jet air ejector to the offgas treatment system fails. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

Upon verification of a failure in the SJAE lines, the operator will initiate a normal shutdown of the reactor to reduce the gaseous activity being discharged. The operator will isolate the main condenser, which results in high condenser pressure and a reactor scram. The operator will notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for re-entry.

15.7.1.3.3 Core and System Performance

This auxiliary system does not directly affect the reactor core or the power cycle systems but is coupled only through operator alarms in the control room.

15.7.1.3.4 Barrier Analysis

This release occurs outside the containment; therefore, barrier integrity is not involved.

15.7.1.3.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions, similar to the offgas system failure, for the purpose of determining adequacy of the plant design to limit the offsite doses to levels that are well within 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis.”

15.7.1.3.5.1 Design Basis Analysis

For the design basis analysis, it is assumed that the reactor is operating at a steam flow of $1.69X10^7$ lb/hr, with the reactor coolant activity at design basis concentrations. The reactor steam concentrations of noble gases are based on an offgas release rate of 403,200 μCi/sec at 30 minutes decay. The iodine activity per pound of steam is assumed to be 8 percent of the iodine activity per pound of reactor coolant and an iodine partition factor of 100 exists between the condenser water and the offgas. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-8.

15.7.1.3.5.2 Realistic Analysis

For the realistic analysis, it is assumed that the reactor is operating at a steam flow of $1.69X10^7$ lb/hr, with the reactor coolant activity at design basis concentrations. The reactor steam concentrations of noble gases are based on an offgas release rate of 100,000 μCi/sec at 30 minutes decay. The iodine activity per pound of steam is assumed to be 8 percent of the iodine activity per pound of reactor coolant and an iodine partition factor of 140 exists between the condenser water and the offgas. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-8.

15.7.1.3.5.3 Fission Product Release

The hypothetical failure of the air ejector lines is postulated to occur downstream of the hydrogen recombiner system. Automatic isolation of the system is not provided and it is conservatively assumed that the SJAE is isolated within 24 hours.

15.7.1.3.5.4 Fission Product Transport to the Environment

It is conservatively assumed that all the activity released from the SJAE line break is released to the environment, with no credit taken for plateout in the Turbine Building, prior to release to the environs.

These analyses assume no decay during transport and that the uncontrolled release period is 24 hours before the release is terminated. The total activity release rates for the design basis and realistic accidents are given in Table 15.7-6.

15.7.1.3.5.5 Results

The calculated exposures offsite and at the control room for the design basis and realistic analyses of the Steam Jet Air Ejector failure are presented in Table 15.7-7. A detailed description of the control room model is provided in Appendix 15B. The radiation dose consequences are well within the guideline values given in 10CFR50.67.

15.7.2 LIQUID RADWASTE SYSTEM FAILURE

15.7.2.1 Miscellaneous Small Releases Outside Containment

Releases that could occur from piping failures outside the containment include the feedwater system piping break (Subsection 15.6.6) and the main steam line break (Subsection 15.6.4) accidents. The analysis of these events provides doses that might occur for such a classification of piping failure events.
Releases to the environment that could occur from radwaste system component failures outside containment are addressed in Section 15.7.3 for radioactive gaseous releases to the atmosphere and in Sections 2.4.12 and 2.4.13 for radioactive liquid releases via the surface and groundwater pathways.

Other releases that could occur outside the containment include small spills and leaks of radioactive materials inside structures housing process equipment. Conservative values for leakage have been assumed and evaluated in Sections 11.2 and 11.3 under routine plant releases. The offsite dose that results from any small spill, which could occur outside the containment, will be negligible in comparison to the dose resulting from the postulated leakages.

Because the above references to other FSAR sections provide bounding uses for all conceivable small releases outside the containment, no further descriptions of core and system performance, barrier performance and radiological consequences are provided here.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID RADWASTE TANK FAILURE

15.7.3.1 Identification of Causes and Frequency Classification

15.7.3.1.1 Identification of Causes

An unspecified event causes complete release of the radioactive inventory in a liquid containing waste tank with the largest quantity of volatile radionuclides in the Radioactive Waste Management Systems. This component is the RWCU phase separator tank located in the radwaste enclosure. The airborne radioactivity released during this event is assumed to pass directly to the environment via the turbine building exhaust vent.

Postulated events that could cause release of the radioactive inventory of the RWCU phase separator are cracks in the vessels and operator error. The possibility of small cracks and consequently low-level release rates receives primary consideration in system and component design. The RWCU phase separator is designed to operate at atmospheric pressure and at a maximum temperature of 200°F so that the possibility of failure is considered small. A radioactive release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions. Should a liquid radioactive release occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid.

15.7.3.1.2 Frequency Classification

The complete rupture of the RWCU phase separator tank is considered a remote possibility. Although not analyzed for the requirements of Seismic Category I equipment, the Radioactive Waste Management System components are constructed in accordance with sound engineering principles. Therefore, simultaneous failure of all the tanks is not considered a credible event. Accordingly, this accident is expected to occur with the frequency of a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

The sequence of events expected to occur is as follows:

Sequence of Events

1. Event begins - failure occurs
2. Sump high level and or radiation alarms alert plant personnel
3. Operator actions begin

The rupture of the RWCU Phase Separator would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available, however isolation of the radwaste area would minimize personnel exposure. A high water level alarm in the radwaste building sump and radiation alarms in the turbine building exhaust vent and in the radwaste building area are available to alert the operator to the failure and followup actions would be taken to isolate the radwaste area ventilation system and proceed with cleanup operations. However, no credit for any operator action or for ventilation system isolation is assumed in evaluating the radiological consequences of this event.

15.7.3.3 Core and System Performance

The failure of the RWCU phase separator tank does not affect the Nuclear Steam Supply System (NSSS).

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.3.4 Barrier Performance

This release occurs outside containment, therefore the event does not involve the primary containment barrier integrity.

15.7.3.5 Radiological Consequences

15.7.3.5.1 Design Basis Analysis

It is assumed that the RWCU phase separator tank contains the design basis inventory of radioactive material as presented in Table 15.7-9. The RWCU phase separator tank, which contains the largest amount of radioactive materials that could be released, is assumed to fail. The failure releases the entire contents of this tank to the radwaste enclosure. The radioactive materials in the RWCU phase separator tank are attached to powdered resin which is at ambient room temperature and are not expected to readily become airborne, if spilled. Consequently, the failure of the RWCU phase separator is not expected to result in any significant release of radioactive materials to the building atmosphere.

Nevertheless, a hypothetical event resulting in the release of radioactive iodine is evaluated. An iodine partition factor of 0.002 is assumed for the spilled liquid. This airborne iodine activity is vented through the radwaste ventilation system and exhausted instantaneously via the turbine building exhaust vent. No credit is assumed for iodine removal by the charcoal filters in the turbine building exhaust. Specific parametric values used in this evaluation are presented in Table 15.7-11. Table 15.7-9A lists the iodine activity assumed to be released to the environment. The offsite radiological doses for the RWCU phase separator rupture accident are given in Table 15.7-10.

15.7.3.5.2 Realistic Analysis

It is assumed that the inventory in the RWCU phase separator tank corresponds to an expected 0.05 Ci/sec offgas release rate at 30 minutes delay under normal operation as given in Table 15.7-9. Other parameters and assumptions are the same as those of the design basis analysis.

Activities released to the environment and the offsite doses are presented in Tables 15.7-9A and 15.7-10, respectively.

15.7.4 FUEL AND EQUIPMENT HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of the failure of the fuel assembly lifting mechanism resulting in the drop of a channeled fuel assembly, grapple, and mast onto other fuel bundles.

An equipment handling accident is assumed to occur as a consequence of the failure of the upper crane resulting in the drop of an object onto other fuel bundles. The total weight of the dropped object is 1100 lbs or less. Movement of objects in excess of 1000 lbs. are controlled by the Susquehanna Heavy Loads program.

A variety of events which qualify for the class of accidents termed "fuel and equipment handling accidents" have been investigated. The accidents which produce the most severe radiological consequences are the drop of a discharged channeled fuel assembly, grapple, and mast; or piece of equipment into the reactor core when the reactor vessel head is off or into the spent fuel pool.

Because the severity of the accident depends on a number of factors such as height of drop over impact site, depth of water over fuel (affects filtering of iodines), and recent irradiation/power history of fuel involved in impact (affects isotopic inventory or source term), a set of conservative assumptions is utilized to cover all possible scenarios. As such the fuel and equipment handling accident is analyzed to bound any credible event occurring over the core or over the spent fuel pool.

15.7.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

A typical sequence of events is as follows:

Event		Approximate Elapsed Time
(1)	A channeled fuel assembly is being handled over the core or spent fuel pool by the refueling equipment; or an object is being handled on the overhead crane over the core or spent fuel pool. The fuel assembly, grapple, and mast; or the object being handled on the overhead crane drops.	0
(2)	Fuel rods in the dropped fuel assembly and/or reactor core are damaged resulting in the release of gaseous fission products to the reactor coolant or spent fuel pool and eventually to the reactor building atmosphere.	0
(3)	The reactor building ventilation radiation monitoring system alarms to alert plant personnel, isolates the ventilation system, and starts operation of the Reactor Building Recirculation system and the SGTS.	1 min
(4)	Operator actions begin.	5 min

15.7.4.2.2 Identification of Operator Actions

(1) The operator will immediately initiate the evacuation of the refuel floor. If radiological conditions warrant, evacuation of the reactor building and the locking of the reactor building doors may also be initiated.

(2) The Refueling Floor Supervisor will instruct personnel to go immediately to the radiation protection personnel decontamination area.

(3) The Refueling Floor Supervisor will make the shift manager aware of the accident.

(4) The shift manager will immediately determine if the normal ventilation system has isolated, and the Reactor Building Recirculation system and the standby gas treatment system (SGTS) are in operation.

(5) The shift manager will initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the reactor building.

(6) The shift manager or his delegate will determine if the standby gas treatment system is performing as designed.

(7) The shift manager will post the appropriate radiological control signs at the entrance of the reactor building.

(8) Before entry to the reactor building is made, a careful study of conditions, radiation levels, etc., will be performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function although credit is taken only for the isolation of the normal ventilation system and the operation of the Reactor Building Recirculation System and the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

15.7.4.2.4 The Effects of Single Failures and Operator Errors

The automatic ventilation isolation system, which includes: a) the radiation monitoring detectors, b) isolation valves, and c) the Reactor Building Recirculation system and the SGTS are designed to single failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4 and to Appendix 15A for further details.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the mechanical consequences of the fuel and equipment handling accidents provide a conservative assessment of the number of fuel rods expected to fail.

For both the fuel handling and equipment handling accidents, a simple kinetic energy approach is used to determine fuel rod failures in the struck assemblies.

The fuel handling accident considers two impacts. First, an initial impact where the entire amount of kinetic energy is dispersed. After the initial impact, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally. The energy associated with the second impact is calculated by assuming a linear weight distribution over the length of the assembly and a point load at the top of the assembly representing the grapple and mast.

Half of the total kinetic energy available in each impact is assumed to be absorbed by the dropped fuel assembly, grapple, and mast. The dropped assembly is considered to impact at a small angle subjecting all the fuel rods in the dropped assembly to bending moments which result in the failure of all the fuel rods in the dropped assembly.

The other half of the total kinetic energy available in each impact is assumed to be absorbed by the non-fuel components of the struck fuel assemblies. This energy is then multiplied by the cladding weight fraction of the non-fuel components (weight of cladding to weight of total non-fuel components) of the struck fuel assemblies to determine the total amount of energy absorbed by the struck fuel rods.

The equipment handling accident considers one impact where the entire amount of available kinetic energy is absorbed by the struck fuel assemblies. This energy is then multiplied by the cladding weight fraction of the non-fuel components of the struck fuel assemblies to determine the total amount of energy absorbed by the struck fuel rods.

The total amount of energy absorbed by the struck fuel rods is divided by the fuel rod cladding failure energy to yield the expected number of failed fuel rods associated with each impact.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are listed below:

(1) For the fuel handling accident a load of 1500 lbs. which conservatively represents the channeled fuel assembly, grapple, and mast is assumed to drop 32.95 ft (the maximum height that an irradiated fuel assembly can be carried) and impact other assemblies in the core or spent fuel pool.

For the equipment handling accident, a conservative load of 1100 lbs. is assumed to drop 150 ft (the maximum height that the overhead crane can carry an object) and impact onto fuel assemblies in the core or spent fuel pool.

(2) All of the fuel rods in the dropped fuel assembly (fuel handling accident) are conservatively assumed to fail as a result of the dropped fuel assembly being considered to impact at a small angle and being subjected to bending moments. Bending moments require significantly less energy (on the order of 1 ft-lb) to fail a fuel rod.

(3) It is assumed that no energy is absorbed by the uranium fuel material (UO_2) in the struck assemblies.

(4) It is assumed that no kinetic energy is dissipated in the water above the core or spent fuel pool.

(5) For the ATRIUM^{TM}-10 fuel design the cladding weight fraction of the non-fuel components is approximately 0.488.

(6) The energy required to produce cladding failure due to compression for an FANP ATRIUM^{TM}-10 fuel rod is approximately 216 ft-lbs. This is based upon a 1% plastic hoop strain in the rod.

15.7.4.3.3 Results

The results for fuel handling and equipment handling accidents involving freshly discharged ATRIUM^{TM}-10 fuel are the most limiting of all the fuel types used in Susquehanna Units 1 and 2. The Atrium^{TM}-10 results also conservatively bound all fuel designs in spent fuel pool, including the PANP 8x8, GE 8x8, FANP 9x9-2, GE 12, and SVEA-96+ designs.

For each fuel type(s) specified below, the basis for why ATRIUM^{TM}-10 is more limiting is provided:

GE 8x8 and FANP 8X8 fuel

Due to the extended decay time these fuel bundles have experienced since discharge from the reactor, the ATRIUM^{TM}-10 source term will be larger.

GE12 and ABB SVEA-96+ LUAs

See section 15.7.4.3.3.3.3 for fuel handling accident results. The number of LUA fuel assembly failures, as calculated by the respective fuel vendors, is less than ATRIUM™-10. The same conclusion can be extended to the equipment handling accident.

FANP 9x9-2 fuel

FANP has reported and documented that the ATRIUM™-10 fuel and equipment handling accident is bounding over the 9x9-2. This is reasonable because the threshold to fail one ATRIUM™-10 fuel assembly is less than that to fail one 9x9-2 fuel assembly. Since the source term is about the same for an ATRIUM™-10 and 9x9-2 fuel assembly, the ATRIUM™-10 fuel and equipment handling accidents would result in more assembly failures and a higher radiological release.

15.7.4.3.3.1 Energy Available

For the initial impact of the fuel handling accident, a load of 1500 lbs. representing the channeled fuel assembly, grapple, and mast is assumed to drop 32.95 ft and impact onto other fuel assemblies with a maximum kinetic energy of 49,425 ft-lbs. Following the initial impact, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

For the equipment handling accident, a load of 1100 lbs. is assumed to drop 150 ft and impact onto other fuel assemblies with a maximum kinetic energy of 165,000 ft-lbs.

15.7.4.3.3.2 Energy Loss Per Impact

Each impact is conservatively assumed to dissipate the total amount of kinetic energy available, with no credit taken for partial energy dissipation.

15.7.4.3.3.3 Fuel Rod Failures

For the purpose of determining the radiological consequences due to the postulated fuel handling accident case, the estimated number of failed fuel rods is 254.8 rods for the Atrium 10 fuel assemblies. For the equipment handling accident case, the number of failed rods is estimated as 460.8 rods for the Atrium 10 fuel assemblies. To conservatively address the issue of lead fuel assemblies (whether in the reactor or the in the spent fuel pool), these estimates conservatively assume the number of fuel rods calculated to be damaged by the drop plus an additional Atrium 10 assembly which represents a complete failure of an additional lead use assembly (LUA).

15.7.4.3.3.3.1 First Impact Failures

For the Fuel Handling Accident, the fuel rod failure calculations assume that the dropped fuel assembly and the struck fuel assemblies are FANP ATRIUM™-10 assemblies. As noted in Section 15.7.4.3.3 this provides the most limiting fuel handling and equipment handling accident results.

For the initial impact of the fuel handling accident, a total kinetic energy of 49,425 ft-lbs. is dissipated.

Half of the energy is assumed to be absorbed by the dropped fuel assembly, grapple, and mast. However, all of the fuel rods in the dropped assembly are conservatively assumed to fail as a result of impacting at a small angle and being subjected to bending moments.

The other half of the energy is assumed to be absorbed by the non-fuel components of the struck fuel assemblies. The cladding weight fraction of the ATRIUM™-10's non-fuel components is 0.488. Therefore the total amount of energy absorbed by the struck fuel rods is approximately 12,060 ft-lbs. Dividing this by the cladding failure threshold of 216 ft-lbs yields approximately 56 failed rods in the struck fuel assemblies. Thus, the first impact of the fuel handling accident yields the following fuel failures:

Dropped Assembly	91	rods (all rods assumed to fail)
Struck Assemblies	56	rods (1st impact)
	147	rods

For the impact of the equipment handling accident, a total kinetic energy of 165,000 ft-lbs is dissipated. The total kinetic energy is assumed to be absorbed by the non-fuel components of the struck assemblies. The ATRIUM™-10 assembly non-fuel components cladding weight fraction is 0.488. Therefore the total amount of energy absorbed by the struck fuel rods is approximately 80,520 ft-lbs. Dividing this by the cladding failure threshold of 216 ft-lbs yields approximately 373 failed rods in the struck fuel assemblies. Thus, the impact of the equipment handling accident yields the following fuel failures:

Struck Assemblies	373	rods

15.7.4.3.3.3.2 Second Impact Failures

Following the initial impact in the fuel handling accident, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

Half of that energy is assumed to be absorbed by the non-fuel components of the struck assemblies. The ATRIUM™-10 assembly non-fuel components cladding weight fraction is 0.488. Therefore the total amount of energy absorbed by the struck fuel rods is approximately 4214 ft-lbs. Dividing this by the cladding failure threshold of 216 ft-lbs yields approximately 20 failed rods in the struck fuel assemblies. Thus, the second impact of the fuel handling accident yields the following fuel failures:

Struck Assemblies	20	rods (2nd Impact)

15.7.4.3.3.3.3 Total Failures

The total number of failed rods resulting from the fuel handling accident is as follows:

First impact	147	rods
Second impact	20	rods
	167	total failed rods (1.90 assemblies)*

The total number of failed rods resulting from the equipment handling accident is as follows:

First impact 373 total failed rods (4.25 assemblies)*

The four ABB SVEA-96+ lead fuel assemblies and four GE12 lead fuel assemblies have been discharged from the Susquehanna Units 1 and 2. For the GE12 fuel type, GE determined that 151 fuel rods (1.64 assemblies) would fail as a result of the fuel handling accident (Reference 15.7-6). For the SVEA-96+ fuel type, ABB determined that 124 fuel rods (1.29 assemblies) would fail as a result of the fuel handling accident (Reference 15.7-7). The results for both LUA designs are bounded by the ATRIUM-10 fuel handling accident results (1.90 failed assemblies).

However, to conservatively address the issue of lead fuel assemblies (whether in the reactor or in the spent fuel pool), radiological dose results assume that another ATRIUM™-10 assembly (representing a lead assembly) completely fails in addition to the previous results for the fuel handling and equipment handling accidents. No credit is taken for the energy absorption by the additional assembly.

*It is important to consider the total number of assemblies that fail because fuel designs have different numbers of rods. Consideration on a failed assembly basis provides a better measure of the relative severity of fuel and equipment handling accidents involving different mechanical designs because core average source terms do not account for assembly mechanical differences, and on an assembly basis are approximately the same.

15.7.4.4 Barrier Performance

The reactor coolant pressure boundary and primary containment are assumed to be open. The transport of fission products from the reactor building is discussed in Subsections 15.7.4.5.2.1 and 15.7.4.5.2.2 below.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for each refueling accident scenario:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "Design Basis Analysis."

(2) The second analysis is based on assumptions considered to provide a realistic but still conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis."

For the Design Basis and Realistic analyses, the fission product inventory in the fuel rods assumed to be damaged is based on an average assembly burnup of 39,000 MWd/MTU resulting from continuous operation at 4032 MW(t).

A 24-hour period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown. Figure 15.7-1 indicates the leakage flow path for these accidents.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based on NRC Regulatory Guide 1.183. The RADTRAD computer code is used to evaluate the radiological consequences (Reference 15.7-2). Specific values of parameters used in the evaluation for the fuel handling and equipment handling accidents are presented in Table 15.7-17.

Regulatory Guide 1.183 provides guidance on the use of pool decontamination factors for iodine for water depths of 23 feet or greater. For Susquehanna, the minimum water depth occurs over the spent fuel pool and is approximately 22 feet (for analysis purposes 21 feet is assumed, which is conservative). Regulatory Guide 1.183 states that if the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method. Regulatory Guide 1.183 indicates an acceptable method is provided in Staff Technical Paper, Evaluation of Fission Product Release and Transport, G. Burley (Reference 15.7-5). The overall decontamination factor for a pool depth of 21 feet determined using this methodology is a factor of 138.

15.7.4.5.1.1 Fission Product Release from Fuel

The fission product inventory of a core average rod (for radiological source term purposes, a core average rod for the ATRIUM™-10 assembly is conservatively based upon 87.8 equivalent full-length fuel rods per assembly) is adjusted by a peaking factor of 1.6 to establish the inventory of each damaged rod. The activity in the fuel rod gap available for release from the damaged rods is defined in accordance with Regulatory Guide 1.183 as eight percent of the I-131 inventory, 10 percent of the Kr-85 inventory, five percent of the noble gases and halogens and twelve percent of the alkali metals. These release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. A pool decontamination factor of 138 for iodine and 0 for noble gases is assumed. The activity airborne in the secondary containment is presented in Table 15.7-12 for both the fuel handling and equipment handling accident scenarios.

15.7.4.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of mixing in the fuel pool, migration from the pool to the secondary containment atmosphere and release to the environment through the SGTS (Standby Gas Treatment System).

After filtration by the SGTS (99% removal efficiency for iodine, 0% for noble gases, no filtration is assumed during the 10 minute drawdown) the airborne activity is assumed to be released to the environment over a 2 hour period. The dose over a 2-hour period from the start of the release is calculated at the exclusion area boundary. In addition, the dose over the 30-day period is calculated for the low population zone and the control room. No credit is taken for isotopic decay during the release.

The release of activity to the environment is presented in Table 15.7-13 for both the equipment and fuel handling accident scenarios.

15.7.4.5.1.3 Results

OFFSITE DOSES

The calculated radiological doses at the exclusion area boundary and low population zone for the design basis analyses are presented in Table 15.7-16. All doses are well within the 10CFR50.67 dose limits and the Regulatory Guide 1.183 acceptance criteria.

CONTROL ROOM DOSES

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.7-17. The radiological exposure to the control room personnel for the design basis case is given in Table 15.7-16. The calculated dose meets the 10CFR50.67 control room dose acception criterion.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD computer code (Reference 15.7-2) is used to evaluate the radiological consequences of the realistic analyses. Specific values of parameters used in the evaluation for the equipment handling and fuel handling accidents are presented in Table 15.7-17.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission product release estimates for the refueling accidents are based on the following assumptions:

(1) The reactor fuel has an average irradiation 39 GWD/MTU of up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hr decay period allows time to shut down the reactor, depressurize the nuclear system, remove the reactor vessel head, and remove the reactor vessel upper internals. It is not expected that these operations could be accomplished in less than 24 hr and probably will require at least 48 hrs.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments (Reference 15.7-3).

(3) Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

(4) It is conservatively assumed that the same number of fuel rods fail for the realistic equipment handling and fuel handling accidents as used in the design basis analysis. This is considered to be conservative because it is expected that many fewer rods would be damaged for these accident scenarios.

15.7.4.5.2.2 Fission Product Transport to the Environment

The following assumptions and conditions are used in calculating the release of activity to the environment for the equipment and fuel handling accidents.

(1) All of the noble gases released to the fuel pool become airborne in the secondary containment (reactor building).

(2) A pool decontamination factor of 138 is used for iodine activity released from the fuel.

(3) All of the activity is released from the secondary containment to the environment through the SGTS in two (2) hours (99% removal efficiency for iodine assumed after a 10 minute drawdown, no filtration is assumed during drawdown).

Based on these assumptions, the activity airborne in the reactor building for each refueling accident scenario is shown in Table 15.7-14.

The release rate of activity under normal ventilation conditions is sufficient to cause a trip of the Secondary Containment Discharge Plenum radiation monitors which results in secondary containment isolation and SGTS startup.

The cumulative release to the environment is presented in Table 15.7-15.

15.7.4.5.2.3 Results

OFFSITE DOSES

The calculated exposures for the realistic analyses are presented in Table 15.7-16 for both refueling accidents and demonstrate the margin of conservatism in the design basis analysis.

CONTROL ROOM DOSES

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.7-17. The radiological exposure to the control room personnel for the realistic basis case is given in Table 15.7-16.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

The spent fuel cask will be equipped with redundant sets of lifting lugs and yokes compatible with the reactor building crane main hook, thus preventing a cask drop due to a single failure. Therefore, an analysis of the spent fuel cask drop is not required. The On-Site Transfer Cask (described in Section 11.7.6.1) and yoke used to transfer spent fuel to the Independent Spent Fuel Storage Installation (ISFSI) is single failure proof and compatible with the Unit 1 Reactor Building Crane main hook, thus preventing an On-Site Transfer Cask drop due to single failure. Therefore, an analysis of the On-Site Transfer Cask drop is not required. Refer to Subsection 9.1.5 for a description of the reactor building crane and the interlocks which prevent moving the spent fuel cask over the fuel pool.

15.7.6 REFERENCES

15.7-1 ORNL-4628, ORIGEN - The ORNL Isotope Generation and Depletion Code, March 1974.

15.7-2 NUREG/CR-6604 and Supplements, RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, June 1999.

15.7-3 N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," APED 5756, March 1969.

15.7-4 "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application," NEDO-10958 and NEDE-10958, (November 1973).

15.7-5 Staff Technical Paper, Evaluation of Fission Product Release and Transport, G. Burley, 1971 (NRC Accession Number 8402080322)

15.7-6 PL-NF-97-003, Rev. 1, “Susquehanna SES Unit 2 Cycle 9 Reload Summary Report,” September 1997.

15.7-7 PL-NF-96-005, Rev. 2, “Susquehanna SES Unit 1 Cycle 10 Reload Summary Report,” July 1997.

TABLE 15.7-1

SEQUENCE OF EVENTS FOR MAIN CONDENSER OFFGAS TREATMENT SYSTEM FAILURE

Elapsed Time	Events
0 seconds	Event begins with system failure and the release of radioactive gases to the building.
0 to 1 hour	Event detection and termination of release: • Event detection is based on an alarmed increase in activity release via the building vent monitor. • Area radiation monitor alarms. • Loss of flow indication in the offgas system. Operator actions include: • Verification and assessment of the accident. • Notifications of plant personnel for area evacuation and radiation protection/area survey. • Initiate appropriate system isolations. • Manual scram actuation. • Assurance of reactor shutdown cooling.

TABLE 15.7-2

ACTIVITY INVENTORY STORED IN OFFGAS TREATMENT SYSTEM[1]
ACTIVITY RELEASED TO THE ENVIRONS (curies)
(DESIGN BASIS ANALYSIS)

Isotope	Activity
I-131	1.31E-01
I-132	1.45E+00
I-133	9.24E-01
I-134	3.42E+00
I-135	1.41E+00
Kr-83m	1.84E+02
Kr-85m	6.71E+02
Kr-85	1.51E+02
Kr-87	8.40E+02
Kr-88	1.51E+03
Kr-89	6.79E+02
Xe-131m	6.12E+01
Xe-133m	3.24E+02
Xe-133	2.19E+04
Xe-135m	4.50E+02
Xe-135	4.55E+03
Xe-137	9.73E+02
Xe-138	1.48E+03

1. SJAE, delay pipe, and offgas system delay beds.

TABLE 15.7-3

ACTIVITY INVENTORY STORED IN OFFGAS TREATMENT SYSTEM(1)
ACTIVITY RELEASED TO THE ENVIRONS (curies)
(REALISTIC ANALYSIS)

Isotope	Activity
I-131	2.33E-02
I-132	2.58E-01
I-133	1.64E-01
I-134	6.06E-01
I-135	2.50E-01
Kr-83m	4.01E+01
Kr-85m	1.53E+02
Kr-85	4.67E+00
Kr-87	1.73E+02
Kr-88	3.39E+02
Kr-89	1.61E+02
Xe-131m	1.99E+01
Xe-133m	7.75E+01
Xe-133	5.36E+03
Xe-135m	9.16E+01
Xe-135	1.12E+03
Xe-137	2.27E+02
Xe-138	3.03E+02

1. SJAE, delay pipe, and offgas system delay beds.

TABLE 15.7-4

MAIN CONDENSER OFFGAS TREATMENT SYSTEM FAILURE RADIOLOGICAL EFFECTS

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	4.38E-02	1.62E-03	1.68E-02
Design Basis Analysis	1.19E+00	7.02E-02	7.19E-02

TABLE 15.7-5

**OFFGAS TREATMENT SYSTEM FAILURE
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS**

I.	**Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents**	**Design Basis Analysis**	**Realistic Analysis**
A.	Reactor power (MWt)	4032	4032
B.	Fuel damage	None	None
C.	Reactor coolant activity before the accident		
	1. Iodine	-	Table 11.1-2
	2. Noble gas	Table 11.1-1	Table 11.1-1
D.	Reactor steam offgas release rate at 30 minutes decay (μCi/sec)	403,200	100,000
E.	Iodine carry over fraction reactor water to steam (percent)	8	8
II.	**Data and Assumptions Used to Estimate Activity Released to the Environment**		
A.	Total mass of charcoal in absorbers (lbs)	148,000	148,000
B.	Offgas system delay bed release to environs duration (hrs)	2	2
C.	Offgas system delay line release to environs duration (hrs)	2	2
D.	SJAE release to environs duration (hrs)	1	1
E.	Condenser air in-leakage / Common offgas recombiner low flow purge air (scfm)	6 1.46E+07	21.76 1.46E+07
F.	Reactor steam flow (lbm/hr)	65 – Kr	36 – Kr
G.	Dynamic absorption coefficients for the charcoal beds (cm^3/gm)	1000 – Xe	516 – Xe
III.	**Data and Assumptions Used to Evaluate Control Room Doses**		
A.	Control structure habitability envelope free volume(ft^3)	518,000	549/4827
B.	Control room free volume (ft^3)	110,000	110,000
C.	Control structure filtered air intake flow (cfm)	5229 – 6391	5229 - 6391
D.	Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10 500	10 500
E.	Control structure unidentified unfiltered outside air infiltration rate (cfm)	99	99
F.	Control structure filter efficiency for iodine (percent)		

TABLE 15.7-5

**OFFGAS TREATMENT SYSTEM FAILURE
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS**

IV.	**Dispersion Data**		
A.	EAB and LPZ distance (meters)	549/4829	549/4829
B.	EAB/XQ	Table 2.3-92 (0.5percentile)	Table 2.3-92 (0.5percentile)
C.	LPZ/QZ	Table 2.3-105 (0.5percentile)	Table 2.3-105 (0.5percentile)
D.	CRHE X/Q	Appendix 15B	Appendix 15B
V.	**Dose Date**		
A.	Method of dose calculations	Appendix 15B	Appendix 15B
B.	Dose conversion assumptions	Appendix 15B	Appendix 15B
C.	Doses	Table 15.7-4	Table 15.7-4

1. Times earlier than 30 days before the postulated accident is 100,000 μCi/sec at 30 minutes decay.

TABLE 15.7-6

FAILURE OF AIR EJECTOR LINES
ACTIVITY RELEASED TO THE ENVIRONMENT (curies/sec)

Isotope	Realistic Analysis	Design Basis Analysis
I-131	5.17E-06	2.92E-05
I-132	5.93E-05	3.35E-04
I-133	3.65E-05	2.06E-04
I-134	1.48E-04	8.33E-04
I-135	5.63E-05	3.18E-04
Kr-83m	3.40E-03	1.37E-02
Kr-85m	6.10E-03	2.46E-02
Kr-85	2.00E-05	8.06E-05
Kr-87	2.00E-02	8.06E-02
Kr-88	2.00E-02	8.06E-02
Kr-89	1.30E-01	5.24E-01
Xe-131m	1.50E-05	6.05E-05
Xe-133m	2.90E-04	1.17E-03
Xe-133	8.20E-03	3.31E-02
Xe-135m	2.60E-02	1.05E-01
Xe-135	2.20E-02	8.87E-02
Xe-137	1.50E-01	6.05E-01
Xe-138	8.90E-02	3.59E-01

TABLE 15.7-7

FAILURE OF STEAM JET AIR EJECTOR LINES
RADIOLOGICAL CONSEQUENCES

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	2.86E-02	1.09E-02	6.79E-02
Design Basis Analysis	2.06E+00	1.18E+00	7.23E-01

TABLE 15.7-8

FAILURE OF AIR EJECTOR LINES
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

			Design Basis Analysis	Realistic Analysis
I.	**Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents**			
	A.	Reactor power (MWt)	4302	4302
	B.	Fuel damage	None	None
	C.	Reactor coolant activity before the accident		
		1. Iodine	Table 11.1-2	Table 11.1-2
		2. Noble gas	Table 11.1-1	Table 11.1-1
	D.	Reactor steam offgas release rate at 30 minutes decay (µCi/sec)	403,200	100,000
	E.	Iodine carry over fraction reactor water to steam (percent)	8	8
II.	**Data and Assumptions Used to Estimate Activity Released to the Environment**			
	A.	Condenser partition coefficients for iodines:	100	140
	B.	Time for system isolation (hrs)	24	24
	C.	Reactor steam flow (lbm/hr)	169E+07	169E+07
III.	**Data and Assumptions Used to Evaluate Control Room Doses**			
	A.	Control structure habitability envelope free volume(ft^3)	518,000	518,000
	B.	Control room free volume(ft^3)	110,000	110,000
	C.	Control structure filtered air intake flow (cfm)	5229 - 6391	5229 - 6391
	D.	Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
			500	500
	E.	Control structure unidentified unfiltered outside air infiltration rate (cfm)	99	99
	F.	Control structure filter efficiency for iodine (percent)		
IV.	**Dispersion Data**			
	A.	EAB and LPZ distance (meters)	549/4827	549/4827
	B.	EAB X/Q	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (0.5 percentile)
	C.	LPZ/ X/Q	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (0.5 percentile)
	D.	CRHE X/Q	Appendix 15B	Appendix 15B
V.	**Dose Data**			
	A.	Method of dose calculations	Appendix 15B	Appendix 15B
	B.	Dose conversion assumptions	Appendix 15B	Appendix 15B
	C.	Doses	Table 15.7-7	Table 15.7-7

TABLE 15.7-9

RWCU PHASE SEPARATOR TANK FAILURE – INITIAL ACTIVITY

Design Basis Analysis					
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
Ba-139	4.60E+01	I-134	4.32E+01	Sr-89	7.88E+02
Ba-140	5.83E+02	I-135	1.81E+02	Sr-90	3.09E+02
Ce-141	1.74E+01	La-140	5.84E+02	Sr-91	1.38E+02
Ce-143	2.44E-01	La-141	1.09E+01	Sr-92	6.25E+01
Ce-144	3.08E+01	La-142	6.41E+00	Tc-99m	6.25E+02
Co-58	1.75E+03	Mo-99	3.07E+02	Te-129	4.27E+00
Co-60	6.36E+02	Nb-95	2.00E+01	Te-129m	6.80E+00
Cs-134	1.83E+02	Nd-147	7.79E-01	Te-132	8.12E+02
Cs-136	7.35E+00	Np-239	2.86E+03	Y-91	1.37E+02
Cs-137	3.23E+02	Pr-143	2.86E+00	Y-92	6.29E+01
I-131	5.32E+02	Pu-239	8.67E-02	Zr-95	1.28E+01
I-132	8.68E+02	Ru-103	3.78E+00	Zr-97	1.14E-01
I-133	3.91E+02	Ru-106	2.51E+00	-	-

Realistic Analysis					
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
Ba-139	2.02E+00	I-135	2.41E+01	Sr-90	5.68E+00
Ba-140	1.62E+01	La-140	1.62E+01	Sr-91	5.17E+00
Ce-141	3.05E+00	La-142	1.15E+00	Sr-92	3.85E+00
Ce-143	1.32E-01	Mo-99	1.75E+01	Tc-99m	3.17E+01
Ce-144	1.60E+00	Nb-95	2.40E+00	Te-129	1.27E+00
Co-58	4.31E+01	Nd-147	1.04E-01	Te-129m	4.20E+00
Co-60	3.07E+02	Np-239	5.25E+01	Te-131m	8.01E-01
Cs-134	2.07E+01	Pr-143	1.85E+00	Te-132	1.04E-01
Cs-136	3.34E+00	Pu-239	1.54E-03	Y-91	1.22E+01
Cs-137	1.62E+01	Rh-105	1.25E+00	Y-92	6.90E+00
I-131	4.60E+01	Ru-103	2.46E+00	Y-93	5.51E+00
I-132	1.20E+01	Ru-105	1.24E+00	Zr-95	1.57E+00
I-133	6.94E+01	Ru-106	1.75E+00	-	-
I-134	9.06E+00	Sr-89	1.56E+01	-	-

TABLE 15.7-10

RWCU PHASE SEPARATOR TANK FAILURE RADIOLOGICAL EFFECTS

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	7.53E-03	2.78E-04	6.28E-02
Design Basis Analysis	4.09E-01	2.42E-02	5.34E-01

TABLE 15.7-11

RWCU PHASE SEPARATOR TANK FAILURE

	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt) B. Fuel damaged C. Reactor coolant activity before the accident Realistic Conservative case	4032 None N/A Table 15.7-9A	4032 None Table 15.7-9A N/A
II. Data and Assumption Used to Estimate Activity Released		
A. Holdup in Radwaste Building B. Particulate/Iodine Partition Coefficient	No 0.002	No 0.002
III. Data and Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume(ft^3) B. Control room free volume(ft^3) C. Control structure filtered air intake flow(cfm) D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm) E. Control structure unidentified unfiltered outside air infiltration rate (cfm) F. Control structure filter efficiency for iodine (percent)	518,000 110,000 5229 - 6391 10 500 99	518,000 110,000 5229 - 6391 10 500 99
IV. Disposition Data		
A. Boundary for SB/LPZ distance (meters) B. X/Q's for Site Boundary C. X/Q's for LPZ D. X/Q's for CRHE	549/4827 Table 2.3-92 (0.5 percentile) Table 2.3-105 (0.5 percentile) Appendix 15B	549/4827 Table 2.3-92 (50 percentile) Table 2.3-105 (50 percentile) Appendix 15B
V. Dose Data		
A. Method of dose calculation B. Dose conversion assumptions C. Doses	Appendix 15B Appendix 15B Table 15.7-10	Appendix 15B Appendix 15B Table 15.7-10

TABLE 15.7-12

REFUELING ACCIDENTS
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity (curies)	
	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
I-131	6.37E+02	3.52E+02
I-132	5.10E+02	2.82E+02
I-133	4.06E+02	2.25E+02
I-134	2.27E-05	1.26E-05
I-135	6.69E+01	3.70E+01
Kr-83m	2.90E+01	1.60E+01
Kr-85m	3.64E+02	2.01E+02
Kr-85	1.63E+03	8.99E+02
Kr-87	6.21E-02	3.43E-02
Kr-88	1.17E+02	6.44E+01
Xe-131m	7.97E+02	4.41E+02
Xe-133m	3.46E+03	1.91E+03
Xe-133	1.13E+05	6.26E+04
Xe-135m	1.51E+03	8.33E+02
Xe-135	3.09E+04	1.71E+04

TABLE 15.7-13

REFUELING ACCIDENTS
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

Isotope	Activity Released to Environment (curies)	
	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
I-131	5.920E+01	3.273E+01
I-132	4.452E+01	2.462E+01
I-133	3.748E+01	2.072E+01
I-134	1.815E-06	1.004E-06
I-135	6.080E+00	3.362E+00
Kr-83m	2.004E+01	1.108E+01
Kr-85m	3.119E+02	1.724E+02
Kr-85	1.636E+03	9.046E+02
Kr-87	3.696E-02	2.044E-02
Kr-88	9.182E+01	5.077E+01
Xe-131m	7.979E+02	4.412E+02
Xe-133m	3.405E+03	1.883E+03
Xe-133	1.128E+05	6.236E+04
Xe-135m	2.431E+02	1.344E+02
Xe-135	2.866E+04	1.585E+04

TABLE 15.7-14

REFUELING ACCIDENTS
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Realistic Accident)

Isotope	Reactor Building Airborne Activity (curies)	
	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
I-131	2.55E+01	1.41E+01
I-132	3.26E+01	1.80E+01
I-133	2.60E+01	1.44E+01
I-134	1.46E-06	8.05E-07
I-135	4.28E+00	2.37E+00
Kr-83m	1.04E+01	5.77E+00
Kr-85m	1.31E+02	7.24E+01
Kr-85	2.93E+02	1.62E+02
Kr-87	2.24E-02	1.24E-02
Kr-88	4.19E+01	2.32E+01
Xe-131m	2.87E+02	1.59E+02
Xe-133m	1.24E+03	6.88E+02
Xe-133	4.08E+04	2.25E+04
Xe-135m	5.42E+02	3.00E+02
Xe-135	1.11E+04	6.15E+03

TABLE 15.7-15

REFUELING ACCIDENTS
AIRBORNE RELEASED TO ENVIRONMENT (curies)
(Realistic Accident)

Isotope	Activity Released to Environment (curies)	
	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
I-131	2.370E+00	1.310E+00
I-132	2.846E+00	1.574E+00
I-133	2.400E+00	1.327E+00
I-134	1.167E-07	6.455E-08
I-135	3.890E-01	2.151E-01
Kr-83m	7.185E+00	3.973E+00
Kr-85m	1.122E+02	6.206E+01
Kr-85	2.941E+02	1.626E+02
Kr-87	1.333E-02	7.372E-03
Kr-88	3.288E+01	1.818E+01
Xe-131m	2.873E+02	1.589E+02
Xe-133m	1.228E+03	6.787E+02
Xe-133	4.072E+04	2.251E+04
Xe-135m	8.727E+01	4.825E+01
Xe-135	1.030E+04	5.693E+03

TABLE 15.7-16 REFUELING ACCIDENTS – RADIOLOGICAL EFFECTS		
	Doses (REM TEDE)	
Design Basis Analysis	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
Acceptance Criterion - Offsite	6.30	6.30
2 HR Exclusion Area Boundary	2.33	1.29
Low Population Zone (30 day)	0.137	0.076
Acceptance Criterion - CRHE	5.00	5.00
CRHE	0.1781	0.0985
Realistic Analysis	Equipment Handling Accident (460.8 Failed Rods)	Fuel Handling Accident (254.8 Failed Rods)
Acceptance Criterion - Offsite	6.30	6.30
2 HR Exclusion Area Boundary	0.098	0.054
Low Population Zone (30 day)	0.0036	0.0020
Acceptance Criterion - CRHE	5.00	5.00
CRHE	0.0366	0.0202

CRHE - Control Room Habitability Envelope

TABLE 15.7-17

REFUELING ACCIDENTS - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Assumptions
I. Data and assumptions used to estimate radioactive sources from postulated accidents		
A. Reactor power (MWt)	4032	4032
B. Radial peaking factor	1.6	1.6
C. Fuel damaged EHA FHA	 460.8 254.8	 460.8 254.8
D. Release of activity by nuclide	5 percent of noble gases and halogens 10 percent of Kr-85 8 percent of I-131 12 percent of alkali metals	1.8 percent of noble gases 0.32 percent of iodines
II. Data and assumptions used to estimate activity released		
A. Secondary containment leak rate	All activity released to environment over 2 hour period	All activity released to environment over 2 hour period
B. SGTS filtration efficiencies (percent)		
iodines	0% all iodine species for the first 10 minutes, then 99% thereafter for the duration of the event	0% all iodine species for the first 10 minutes, then 99% thereafter for the duration of the event
noble gases	0	0
C. Fuel pool noble gase decontamination factor	0	0
D. Fuel pool iodine decontamination factor	138	138
E. Decay time prior to accident, hr.	24	24
F. Time delay in SGTS filtration (min)	10	10
III. Data And Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume (ft^3)	518,000	518,000
B. Control room free volume (ft^3)	110,000	110,000
C. Control structure filtered air intake flow(cfm)	5229 – 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency for iodine (percent)	99	99

IV Dispersion data		
A. Boundary and LPZ distance (meters)	549/4827	549/4827
B. X/Q's for EAB C. X/Q's for LPZ D. X/Q's for CRHE	Table 2.3-92 (0.5 percentile) Table 2.3-105(0.5 percentile) Appendix 15B	Table 2.3-92 (50 percentile) Table 2.3-105 (50 percentile) Appendix 15B
V. Dose data		
A. Method of dose calculation	Reg. Guide 1.183 Appendix 15B	Appendix 15B
B. Dose conversion	Appendix 15B	Appendix 15B
C. Activity in secondary containment	Table 15.7-12	Table 15.7-14
D. Activity released to environment	Table 15.7-13	Table 15.7-15
E. Doses	Table 15.7-16	Table 15.7-16

TABLE 15.7-18

SEQUENCE OF EVENTS FOR SJAE FAILURE

Approximate Elapsed Time	Events
0 sec.	Event begins with system failure and the release of radioactive noble gases and iodines to the building.
0 to 24 hours	Event detection and termination of release: Event detection is based on a loss of offgas system flow. Notification of control room of loss of offgas system flow. Verification of loss of offgas system flow by one or more of the following: 1. Verification of an activity release into the Turbine and/or Radwaste Building or to the environment by evaluation of appropriate ARMs, CAMs, Turbine Building Vent SPINGs, and/or airborne radiation surveys. 2. Verification of loss of offgas system flow by evaluating the operability of the offgas system flow instrumentation and monitoring of other offgas system parameters. Operator actions begin with: 1. Initiation of appropriate system isolations. 2. Manual scram actuation. 3. Assurance of reactor shutdown cooling.

TABLE 15.7-9A

RWCU PHASE SEPARATOR TANK FAILURE ACTIVITY RELEASE TO ENVIRONMENT

Realistic Source		Design Basis Source	
Isotope	**Curies**	**Isotope**	**Curies**
Ba-139	4.04E-03	Ba-139	9.20E-02
Ba-140	3.24E-02	Ba-140	1.17E+00
Ce-141	6.10E-03	Ce-141	3.48E-02
Ce-143	2.64E-04	Ce-143	4.88E-04
Ce-144	3.20E-03	Ce-144	6.16E-02
Co-58	8.62E-02	Co-58	3.50E+00
Co-60	6.14E-01	Co-60	1.27E+00
Cs-134	4.14E-02	Cs-134	3.66E-01
Cs-136	6.68E-03	Cs-136	1.47E-02
Cs-137	3.24E-02	Cs-137	6.46E-01
I-131	9.20E-02	I-131	1.06E+00
I-132	2.40E-02	I-132	1.74E+00
I-133	1.39E-01	I-133	7.82E-01
I-134	1.81E-02	I-134	8.64E-02
I-135	4.82E-02	I-135	3.62E-01
La-140	3.24E-02	La-140	1.17E+00
La-142	2.30E-03	La-141	2.18E-02
Mo-99	3.50E-02	La-142	1.28E-02
Nb-95	4.80E-03	Mo-99	6.14E-01
Nd-147	2.08E-04	Nb-95	4.00E-02
Np-239	1.05E-01	Nd-147	1.56E-03
Pr-143	3.70E-03	Np-239	5.72E+00
Pu-239	3.08E-06	Pr-143	5.72E-03
Rh-105	2.50E-03	Pu-239	1.73E-04
Ru-103	4.92E-03	Ru-103	7.56E-03
Ru-105	2.48E-03	Ru-106	5.02E-03
Ru-106	3.50E-03	Sr-89	1.58E+00
Sr-89	3.12E-02	Sr-90	6.18E-01
Sr-90	1.14E-02	Sr-91	2.76E-01
Sr-91	1.03E-02	Sr-92	1.25E-01
Sr-92	7.70E-03	Tc-99m	1.25E+00
Tc-99m	6.34E-02	Te-129	8.54E-03
Te-129	2.54E-03	Te-129m	1.36E-02
Te-129m	8.40E-03	Te-132	1.62E+00
Te-131m	1.60E-03	Y-91	2.74E-01
Te-132	2.08E-04	Y-92	1.26E-01
Y-91	2.44E-02	Zr-95	2.56E-02
Y-92	1.38E-02	Zr-97	2.28E-04
Y-93	1.10E-02		
Zr-95	3.14E-03		

Table 15.7-16A

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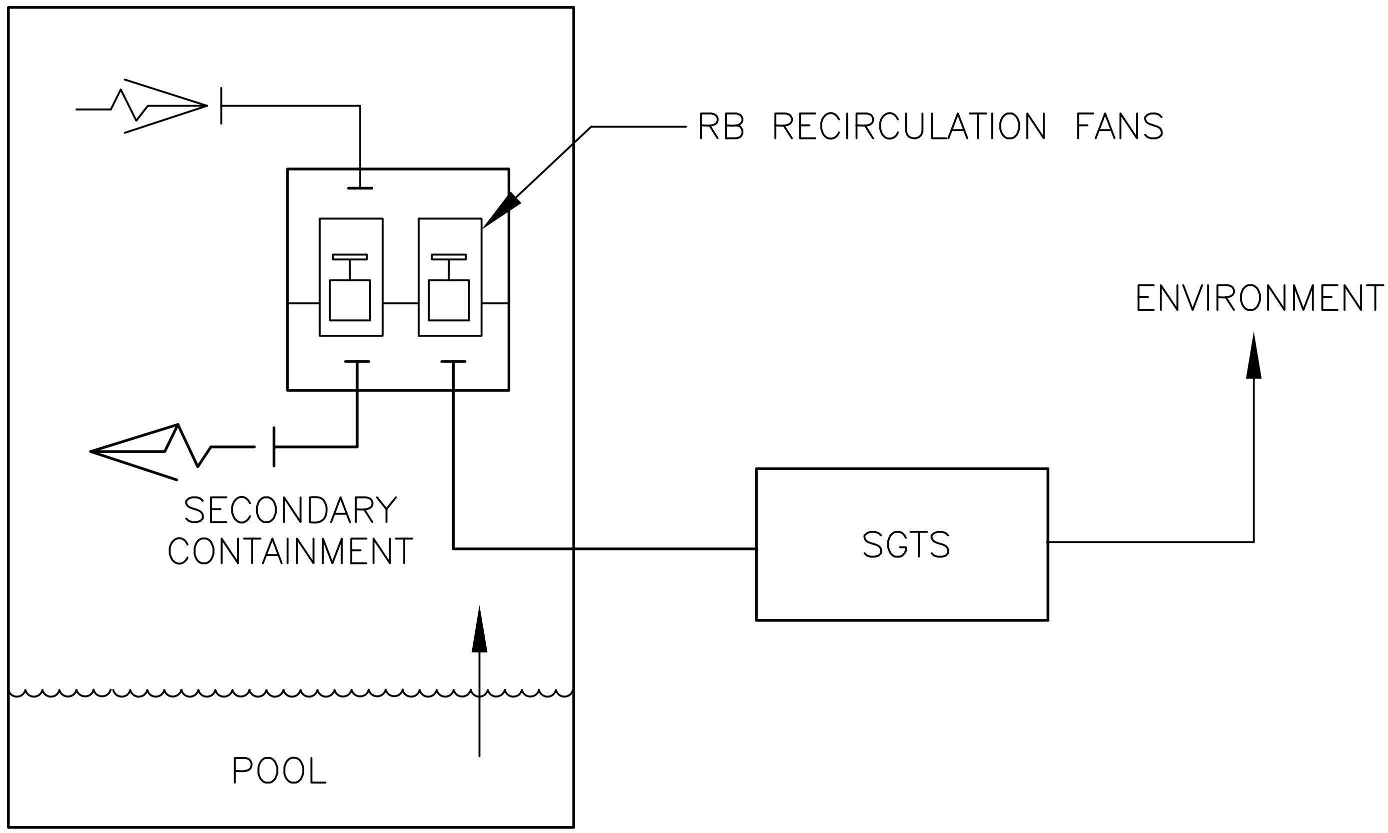

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LEAKAGE PATH
FOR
REFUELING ACCIDENTS

FIGURE 15.7-1, Rev 54

AutoCAD: Figure Fsar 15_7_1.dwg

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

15.8.1 Causes, Frequency Classification, Initiating Events, Acceptance Criteria, Mathematical Models, Input Parameters, and Initial Conditions

An Anticipated Transient Without Scram (ATWS) event was not part of the SSES design basis at the time SSES was designed and built. Codification of the ATWS rule in 10CFR50.62 made ATWS an event for which mitigation capability is required. Thus, ATWS is an event for which SSES structures, systems and components (SSC's) are required to function. The specific functions, ranges of values, etc. of SSC's that are required to mitigate an ATWS are considered to be within Susquehanna's "design basis". These functions, ranges of values etc., are within the Susquehanna "design basis" because they are required to comply with 10CFR50.62. Note, however, that the ATWS event it not considered to be a Design Basis Accident, but is considered to be an "other event" specifically addressed in regulation.

The specific functions, ranges of value etc. of SSC's that are required to mitigate an ATWS, however, do not necessarily relate to SSC's operability requirements as specified in SSES Technical Specifications. Operability of SSC's required to mitigate an ATWS is determined based on the requirements provided in SSES Technical Specifications.

15.8.1.1 Identification of Causes

A failure to scram event may be caused by electrical or mechanical problems. An electrical ATWS is characterized by a failure to vent the compressed gas from the scram exhaust and scram inlet valve operators so that the exhaust valve and the inlet valve do not open on demand as required for a reactor trip. The smallest number of components which must fail in order to cause an "electrical" failure to scram is four RPS relays. The RPS logic is divided so that the control rods are segregated into four electrical groups. The design of the logic system is such that there is no reasonable combination of failures, other than relays, which is expected to cause less than all four rod groups to insert. For this reason, the most probable cause of an "electrical" ATWS appears to be the simultaneous failure of four RPS relays, and the result is a full ATWS. In addition to these failures, the occurrence of an electrical ATWS requires failure of the ARI (Alternate Rod Insertion) system which is a redundant and independent set of components to vent the scram air header.

As discussed above, electrical ATWS events result in failure of the rods to insert because of excessive back pressure on the piston due to failure of the scram exhaust valve to open. In the case of the "mechanical" failure to scram, the excessive back pressure is most likely caused by inadequate exhaust volume in the SDV (Scram Discharge Volume). In order for a mechanical ATWS to occur, a source of water to the SDV must be present. This could result from valves leaking or inadequate draining from a previous scram. In addition, the SDV drain valve must fail to open or the drain line must be blocked to prevent draining of the water source. And finally, the level instrumentation and SDV level logic must fail to provide an alarm and scram signal from the high water level.

The probability of an ATWS event is very low since multiple failures are required to result in insufficient SDV capacity or failure to vent the scram air header.

15.8.1.2 Frequency Classification

The occurrence of an ATWS event is not expected in the life of the plant. The initiating event for the failure-to-scram event is an incident of moderate frequency (anticipated operational transient).

15.8.1.3 Initiating Events

In accordance with Regulatory Guide 1.70, Rev. 2, the following seven initiating events are considered:

- Inadvertent Control Rod Withdrawal,
- Loss of Feedwater,
- Loss of Normal A.C. Power,
- Loss of Electrical Load,
- Loss of Condenser Vacuum,
- Turbine Trip, and
- Closure of Main Steam Line Isolation Valves.

In addition to the seven initiating events required by Reg. Guide 1.70, the following three initiating events are included:

- Pressure Regulator Failure Open,
- Feedwater Controller Failure Open, and
- Inadvertent Opening of a S/R Valve.

These three initiating events also were examined as part of the Susquehanna ATWS evaluation for Power Uprate. Ref. 15.8-1A

15.8.1.4 Acceptance Criteria

15.8.1.4.1 Peak Clad Temperature and Vessel Pressure Associated with Initial Pressurization Transient

The most severe ATWS events are initiated by a pressurization transient (MSIV Closure or turbine trip) or by an equipment failure which leads to a pressurization transient (e.g., pressure regulator failure; loss of condenser vacuum). With scram failure, a pressurization transient can result in a large power spike which may be several hundred percent of rated power. The large increase in power exacerbates vessel pressurization. Acceptance criteria are specified in [EC-PUPC – 10902 (Ref. 15.8-1A) to ensure that the initial power and pressure transients do not threaten fuel and vessel integrity. Specifically these criteria consist of

- Reactor Pressure Vessel (RPV) Integrity. The peak RPV pressure must be less than 1500 psig (Service Level C).

- Fuel Integrity. The maximum fuel cladding temperature cannot exceed 2200°F and the local cladding oxidation must be less than 17%.

15.8.1.4.2 Peak Suppression Pool Temperature and Containment Pressure

In an isolation ATWS, suppression pool temperature increases rapidly as steam generated by fission power is condensed within the pool. The increasing vapor pressure associated with rising pool temperature drives steam and nitrogen through the vacuum breakers to the drywell which causes an increase in drywell pressure. Suppression pool temperature and containment pressure limits are specified in EC-PUPC-10902 to ensure the effectiveness of mitigating actions (recirculation pump trip, boron injection, and reactor water level reduction) on containment thermal loading. The acceptance criteria consist of

- Suppression Pool Temperature. The peak Suppression Pool bulk temperature must remain less than 220°F.

- Containment Pressure. The peak Containment pressure must remain below the design pressure of 53 psig.

Note that if the suppression pool temperature is below 190°F, it is not necessary to explicitly evaluate the containment pressure as it will remain well below the design limit of 53 psig.

15.8.1.4.3 Fuel Integrity Under Unstable Operation

An instability event at LaSalle County Nuclear Station Unit 2 in March 1988 led to an NRC and BWR Owners' Group investigation into the impact of unstable operation on fuel integrity in ATWS events (Ref. 15.8-2). The LaSalle event was initiated by an inadvertent trip of both recirculation pumps. The reduction in power and turbine steam flow caused by the pump trip led to automatic isolation of some feedwater preheaters. The decrease in core flow accompanied by a decrease in feedwater temperature led to diverging power oscillations which were terminated by the high neutron flux trip.

Because of the similarity between the LaSalle instability event and the turbine trip ATWS, which involves a trip of recirculation pumps and a complete loss of feedwater heating, calculations were performed by General Electric to investigate instabilities under ATWS conditions (Ref. 15.8-2). Results for a bounding turbine trip ATWS showed the development of severe power/flow instabilities with potential for localized cladding damage and centerline fuel melting. Owing to the localized nature of the fuel/cladding damage, the NRC has concluded that significant distortion of the fuel to impede core cooling or prevent safe shutdown is unlikely (Ref. 15.8-3). The NRC also concluded that instabilities should not change qualitatively the containment response. Therefore, the radiological conditions should be within 10CFR50.67 guidelines (Ref. 15.8-4).

Since the impact of unstable operation on fuel integrity has already been evaluated on a generic basis by considering a bounding turbine trip ATWS scenario, it is not necessary to reevaluate Susquehanna for ATWS instability on a cycle-specific basis.

In order to minimize the consequences of large-amplitude power instabilities, the NRC has proposed modifications to the Emergency Procedure Guidelines. These modifications consist of: (1) reduction of water level below the feedwater sparger immediately upon confirmation of ATWS, and (2) initiation of boron injection upon detection of oscillations during an ATWS event regardless of suppression pool conditions (Ref. 15.8-3).

Lowering RPV water level below the feedwater sparger is very effective in mitigating unstable operation because subcooled feedwater is preheated by mixing with saturated steam before it enters the core. Early boron injection also helps mitigate instabilities, but its effect is considerably slower than that of level reduction. The Susquehanna EOPs comply with the water level guidance and exceed the boron injection requirement put forth by the NRC (the ATWS EOP requires boron injection for any ATWS event where power is greater than 5%, or cannot be determined, rather than waiting for the development of power oscillations.)

Since the NRC has concluded, on a generic basis, that

- unstable operation will not significantly distort the fuel to impede core cooling or prevent safe shutdown,
- instabilities should not change qualitatively the containment response, and
- radiological conditions should be within 10CFR50.67 guidelines,

and since the Susquehanna EOPs satisfy the NRC requirements for mitigation of instabilities in ATWS events, no further analysis on ATWS instability is required as long as the assumptions used in Ref. 15.8-2 remain bounding.

15.8.1.4.4 Radiological Consequences and Long-Term Shutdown and Cooling Capability

Acceptance criteria with regard to radiological consequences and long-term shutdown and cooling capability are specified in NUREG 0460 (Ref. 15.8-5). As discussed in Section 15.8.1.4.3, unstable operation under ATWS conditions does not lead to violation of these criteria. Given this fact, it can be concluded that the radiological and long-term-shutdown and cooling-capability requirements are met as long as the criteria in Sections 15.8.1.4.1 and 15.8.1.4.2 are satisfied (Ref. 15.8-6). Therefore, only the criteria in Sections 15.8.1.4.1 and 15.8.1.4.2 need be considered in evaluating the performance of Susquehanna for ATWS events.

15.8.1.5 Mathematical Models

The ATWS analysis was performed by General Electric, and the analysis methods are described..

15.8.1.6 Input Parameters and Initial Conditions

Input parameters for the ATWS analysis are listed in Table 15.8-1. The Hot Shutdown Boron Concentration is based on Hot Full Power Xenon concentration.

The ATWS simulations are initiated at a core power of 100% of rated or greater, and 99 MLbm/hr total core flow. This power/flow condition corresponds to the Maximum extended load line limit (MELLL). The cycle exposure corresponds to end of full power (all rods out). Table 15.8-3 list the initial conditions.

15.8.2 Inadvertent Control Rod Withdrawal

In Section 3.1.16 of NEDE-24222 (Ref. 15.8-9), General Electric presents a detailed discussion of the consequences of a rod withdrawal error at full power and within the startup range. GE has concluded that the consequences of the control rod withdrawal error are such that analysis of this event is not necessary.

15.8.3 Loss of Feedwater (LOFW)

The LOFW event is initiated by an assumed loss of all feedwater. Reactor water level drops to L2 (-38") in about 15 seconds. At this time, an RPT occurs (with 10 second delay), and HPCI and RCIC initiate. The operator is assumed to initiate boron injection 90 seconds after Level 2 is reached.

Since the condenser remains available, no SRVs lift, and there is no significant increase in suppression pool temperature (pool temperature increases a small amount because of steam exhausted from HPCI/RCIC turbines). After the LOFW, the reactor pressure and neutron flux begin to fall. Thus, the peak values for these parameters occur at the beginning of the event.

This event was analyzed for Susquehanna by GE in GENE-637-024-0893, and was found to be less severe than the MSIV Closure ATWS; therefore, it does not need to be reanalyzed.

15.8.4 Loss of Offsite Power (LOOP)

An analysis of this event was performed for Susquehanna in EC-PUPC-10902. Initially, there is a loss of power to the recirculation pumps and condensate pumps. Pressure, power, and water level begin to decline due to the loss of feedwater and the recirculation pump trip. At two seconds, the MSIVs are assumed to begin closing due to the loss of A.C. power, and this results in a rapid rise in pressure and neutron flux. HPCI and RCIC initiate on Level 2, and the operator is assumed to initiate boron injection 90 seconds after the ATWS high pressure setpoint is reached.

Peak vessel pressure and PCT for the LOOP event are bounded by the MSIV closure ATWS. Power and pressure responses are less severe because the MSIVs do not start to close until after the recirculation pumps are tripped in the LOOP event. Peak suppression pool temperature is also bounded by the MSIV closure event. The loss of feedwater at the beginning of the LOOP event leads to a substantial reduction in water level and power following the MSIV closure. In contrast, feedwater is available for 1 to 2 minutes following containment isolation in the MSIV closure ATWS. The availability of feedwater in the MSIV Closure ATWS results in higher water level and power. The higher power level, with the MSIVs closed, results in more energy deposited in the suppression pool for the MSIV closure ATWS than for the LOOP event. However, the LOOP event is the most limiting event with regard to operation of the SLC system. LOOP results in a loss of Containment Instrument Gas to the SRVs terminating the relief mode function. Therefore, the SRVs will lift at the higher safety mode setpoints, which could affect SLCS performance. SLCS is designed to inject against a reactor steam dome pressure of 1500 psig (Section 3.9.3.1.12). Therefore, SLCS design is sufficient to ensure performance in this event.

15.8.5 Loss of Electrical Load

In the loss of electrical load event, the turbine-generator lock out relays trip to initiate turbine control valve fast closure. The fast control valve closure initiates a recirculation pump trip.

This event is essentially the same as the turbine trip ATWS event which was analyzed in GENE-637-024-0893. The turbine trip initiates closure of the main stop valves which in turn initiates a recirculation pump trip. Since the events are practically the same, only the turbine trip ATWS event (Section 15.8.7) has been analyzed for Susquehanna.

15.8.6 Loss of Condenser Vacuum

The loss of condenser vacuum ATWS event was analyzed by General Electric, on a generic basis, in NEDE-24222. The transient starts with closure of all turbine stop valves when an unexpected decline in condenser vacuum reaches the turbine trip setpoint. Thus, the beginning of this event is the same as the turbine trip ATWS. Feedwater turbines also isolate on low condenser vacuum early in the event.

As condenser vacuum decays further, the MSIVs and turbine bypass valves also close. Since the recirculation pumps and feedwater turbines are already tripped at this point, the pressurization and neutron flux transients due to the MSIV closure are much less severe than those generated by the initial turbine trip.

Since feedwater is not available following MSIV closure in the loss of condenser vacuum ATWS, the suppression pool temperature response will be bounded by the MSIV closure ATWS event. In the MSIV closure event, feedwater remains operable for 1 to 2 minutes into the event. When feedwater injection is available, reactor power is much higher than it is when level is maintained with the lower-capacity HPCI/RCIC systems.

Since the beginning part of the loss-of-condenser-vacuum event is the same as the turbine trip ATWS (Section 15.8.7), and the pool heat up is bounded by the MSIV closure ATWS (Section 15.8.8), a plant specific analysis is not performed.

15.8.7 Turbine Trip

This transient was evaluated for Susquehanna in GENE-637-024-0893. The Turbine Trip event begins with rapid closure of the turbine stop valves and the resultant opening of the turbine bypass valves.

After the stop valves close, the pressure immediately begins to rise which results in a reduction of the core void fraction and a rapid increase in power. The pressure and power rise are mitigated by the RPT which is initiated directly from the turbine stop valve closure. Pressure continues to rise until it is halted by the opening of relief valves. In GENE-637-024-0893 it is conservatively assumed that the operator initiates boron injection 2 minutes after suppression pool temperature reaches the BIIT (Boron Injection Initiation Temperature) which is 110°F. It is also assumed that the operator begins to lower RPV water level at this time.

Simulation results for suppression pool temperature and peak vessel pressure in GENE-637-024-0893 show that the Turbine Trip ATWS is not limiting for Susquehanna. Vessel pressure and pool temperature are bounded by the MSIV closure event. The PCT due to the initial power spike was not calculated for the Turbine Trip ATWS in GENE-637-024-0893 because based on the core power response, this transient was not considered limiting.

Since this event is less severe than the MSIV closure ATWS, it does not need to be reanalyzed for any changes in plant conditions.

15.8.8 Closure of Main Steam Line Isolation Valves

The Susquehanna ATWS Evaluation, shows that the MSIV Closure ATWS is one of the two limiting ATWS events (the other is the Pressure Regulator Failure - Open event). The evaluation was performed by GE using the computer models described in Section 15.8.1.5.2.

15.8.8.1 Sequence of Events, Systems Operation, and Operator Actions

The sequence of events for the MSIV Closure ATWS are listed in Table 15.8-5. Credit is taken for HPCI, RCIC, and CRD systems for coolant makeup to the vessel. Feedwater injects until main steamline pressure decays to the point where it is insufficient to run the feedwater turbines.

Operator actions assumed for mitigation of the ATWS event are consistent with the EOPs (Emergency Operating Procedures). These actions consist of:

- Initiate SLCS,
- Lower RPV level to within the target band specified by the EOPs,
- Maintain HPCI suction on the CST if sufficient time available for operator action,
- Inhibit ADS,
- Initiate suppression pool cooling, and
- Raise RPV water level to normal range when the HSBW (Hot Shutdown Boron Weight) has been injected.

Although the operator will bypass the RWM (Rod Worth Minimizer) using a control room keylock bypass switch and initiate MRI (manual control rod insertion) to accelerate reactor shutdown, no credit is taken for MRI in order to add conservatism to the suppression pool temperature results.

15.8.8.2 Results

Calculation results for the MSIV closure ATWS scenarios are presented in Table 15.8-7. The peak vessel pressure, PCT, and peak suppression pool temperature remain below the applicable limits.

As discussed in Section 15.8.1.4.2, there is a large margin to the primary containment design pressure limit of 53 psig if suppression pool temperature is less than the 220°F limit. For ATWS, the increase in containment pressure is primarily driven by the vapor pressure of the suppression pool which is relatively small when pool temperatures are within the acceptance criterion.

Comparing PCTs for the MSIV Closure ATWS (Table 15.8-7) against values computed in the generic ATWS study performed by General Electric (NEDE-24222), indicates that cladding oxidation will be significantly below the accepted maximum value of 17% of cladding volume (Ref. 15.8-9, Section 4.4.2).

15.8.9 Pressure Regulator Failure - Open

The ATWS event initiated by failure of the pressure regulator to maximum demand has been found to be a limiting event (along with the MSIV Closure ATWS) for Susquehanna. The evaluation for the PREGO-initiated ATWS was performed by GE using the computer models described in Section 15.8.1.5.2.

15.8.9.1 Sequence of Events, Systems Operation, and Operator Actions

The sequence of events for the PREGO ATWS are listed in Table 15.8-9. Credit is taken for HPCI, RCIC and CRD systems for coolant makeup to the vessel. Feedwater injects until main steamline pressure decays to the point where it is insufficient to run the feedwater turbines.

Assumed operator actions are consistent with the EOPs, and these actions consist of:

- Initiate SLCS,
- Lower RPV level to within the target band specified in the EOPs,
- Maintain HPCI suction on the CST if sufficient time available for operator action,
- Inhibit ADS,
- Initiate suppression pool cooling, and
- Raise RPV water level to normal range when the HSBW has been injected.

As in the case of the MSIV Closure ATWS (Section 15.8.8), conservatism is added to the peak suppression pool temperature result by not taking credit for manual insertion of control rods.

15.8.9.2 Results

Table 15.8-11 list the calculation results for the PREGO ATWS event. The peak vessel pressure, PCT, and peak suppression pool temperature remain below the applicable limits.

There is a large margin to the primary containment design pressure with pool temperature less than the 220 °F Limit. Based on the PCTs for the PREGO event (Tables 15.8-11 and 15.8-12) and the generic results reported by General Electric in Section 4.4.2 of NEDE-24222, cladding oxidation will be significantly below the accepted maximum value of 17% of cladding volume.

15.8.10 Feedwater Controller Failure—Open (FWCFO)

The initiating event for this transient is failure of the feedwater controller to the maximum demand position. As soon as the feedwater controller is assumed to fail, the reactor water level, pressure, and power begin to rise slowly as the higher subcooling due to increasing feedwater flow reduces the core void fraction. Water level continues to rise until the turbine and feedwater pumps trip on Level 8. RPT is initiated by turbine stop valve closure. Following the turbine trip, reactor pressure begins to rise more rapidly and the ATWS high pressure setpoint is reached. Boron is assumed to be initiated manually 90 seconds after the ATWS high pressure setpoint is reached. There is little increase in suppression pool temperature for this event because the main condenser remains available. The availability of the turbine bypass valves and the larger steam volume (more than the volume from the vessel to the MSIVs) should keep the peak vessel pressure and peak clad temperature less severe than for the MSIV closure ATWS event.

The Susquehanna ATWS evaluation performed by General Electric (Ref. 15.8-1) shows that the effects of the FWCFO event are bounded by the MSIV Closure ATWS and the PREGO ATWS. Therefore, the event does not need to be reanalyzed for any changes in plant conditions.

Since the time of the ATWS evaluation performed by General Electric, Susquehanna has installed a digital Integrated Control System encompassing the control of reactor feedwater level. Common mode failure of the reactor feedwater level control system to the maximum demand position will result in the described initiating event and transient.

15.8.11 Inadvertent Opening of a S/R Valve

The inadvertent opening of a relief valve (IORV) transient is initiated by an assumed failure of a relief valve in the open position. This event involves no rapid increase in reactor pressure and power, but is merely a long-term suppression pool heatup and vessel depressurization. GENE-637-024-0893 shows that peak pool temperature is substantially below the values predicted for the MSIV Closure and PREGO events. Consequently, this event does not need to be reanalyzed for any changes in plant conditions.

15.8.12 References

15.8-1 Claassen, L. B., "Evaluation of Susquehanna ATWS Performance for Power Uprate Conditions," GENE-637-024-0893, September 1993

15.8-2 NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," General Electric Company, June 1995.

15.8-3 "Safety Evaluation Report ATWS Rule Issues and Mitigative Actions in Response to ATWS with Large Power Oscillations NEDO-32047 and NEDO-32164, Revision 0," Attachment to Letter from Ashok C. Thadani (NRC) to Les England (BWR Owner's Group), TAC No. M79766, February 5, 1994.

15.8-4 Technical Evaluation Report on Review of the Consequences of Thermohydraulic Instability during ATWS and the Effect of Mitigative Actions, ORNL/NRC/LTR-93-04, prepared by Oak Ridge National Laboratory, April, 1993.

15.8-5 NUREG 0460, "Anticipated Transients Without Scram for Light Water Reactors," Volume 1, Section 7.1, April 1978.

15.8-6 Letter from M. M. Urioste (General Electric Nuclear Services Manager) to R. J. Poshefko (PP&L), "Susquehanna Steam Electric Station Proposed Change to Power Uprate Contract to Provide ATWS Analysis Proposal #295 - 1DKM4-KRO," GKR-92-128 Dated November 24, 1992.

15.8-7 PLA-4480, "Susquehanna Steam Electric Station Unit 2 Cycle 9 ATWS Evaluation," File R41-2, Docket No. 50-388, Dated July 23, 1996.

15.8-8 NRC Safety Evaluation: Modifications to the Boiling Water Reactor (BWR) Emergency Procedure Guidelines to Address Reactor Core Instabilities, June 6, 1996.

15.8-9 "Assessment of BWR Mitigation of ATWS, Volume II (NUREG 0460 Alternate No. 3)," NEDE-24222, December, 1979.

TABLE 15.8-1

INPUT PARAMETERS FOR UNIT 1 ATWS ANALYSIS

Closure Time of MSIV (sec)	4
ATWS High Pressure RPT Setpoint, UAL (psig)	1170
Setpoint for Low Water Level Closure of MSIV	L1(-129”)
Setpoint for Low Steam Line Pressure Closure of MSIV (psig)	861
Relief Valve Setpoints	†
HPCI Flow Rate (gpm)	5000
HPCI Start/Stop Levels	L2(-38”) / L8(+54”)
RCIC Flow Rate (gpm)	600
RCIC Start/Stop Levels	L2(-38”) / L8(+54”)
Hot Shutdown Boron Weight (ppm)	494
SLCS Boron Injection Rate Per Pump (GPM)	40
Boron Transport Time from SLCS Pumps to Vessel (sec)	40
Condensate Storage Tank Water Temperature (^{o}F)	140
ATWS Low Water Level RPT Setpoint	-38”
RHR Pool Cooling Capacity (1st / 2nd Loop) (Btu/sec ^{o}F)	322.4/324.1
Service Water Temperature (^{o}F)	88
Number of Operating SLCS Pumps	1

† SRV set points are taken from Ref. 15.8-1A.

Table 15.8-2

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TABLE 15.8-3

INITIAL OPERATING CONDITIONS FOR UNIT 1 ATWS ANALYSIS

Dome Pressure (psia)	1050
Total Core Flow (Mlbm/hr)	99.0
Core Thermal Power (Mwth)	3952
Narrow Range Water Level (INCHES AVZ)	562.5
Suppression Pool Liquid Volume (ft^3)	122,410
Suppression Pool Temperature (^{o}F)	90

Table 15.8-4

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TABLE 15.8-5

SEQUENCE OF EVENTS FOR MSIV CLOSURE ATWS

Event	Time (sec)
MSIV Isolation Initiates	0
MSIV Fully Closed	4.0
Peak Neutron Flux	4.02
High Pressure ATWS Setpoint	4.17
Opening of the First Relief Valve	4.34
Recirculation Pumps Tripped	4.7
Peak Heat Flux Occurs	4.81
Peak Vessel Pressure	6.84
Feedwater Reduction Initiated (feedwater stopped completely)	104
SLCS Pump Starts	124
RHR Cooling Initiated (first train/second train)	1100/1600
Peak Suppression Pool Temperature	1508
Hot Shutdown Achieved (Neutron flux remains <0.1%)	1618

Table 15.8-6

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TABLE 15.8-7

RESULTS FOR MSIV CLOSURE ATWS EVENT

Parameter	Result	Limit
Peak Vessel Pressure (psig)	1333	1500
Peak Clad Temperature (^{o}F)	1247	2200
Peak Suppression Pool Temperature (^{o}F)	206	220

Table 15.8-8

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TABLE 15.8-9

SEQUENCE OF EVENTS FOR PREGO ATWS

Event	Time (sec)
Turbine Control and Bypass Valve Start Open	0.11
MSIV Closure Initiated by Low Steamline Pressure	12.6
Peak Neutron Flux	16.6
MSIVs Fully Closed	16.6
High Pressure ATWS Setpoint	18.7
Opening of the First Relief Valve	18.9
Peak Heat Flux Occurs	19.2
Recirculation Pumps Tripped	19.2
Peak Vessel Pressure	21.3
Feedwater Reduction Initiated (feedwater stopped completely)	118
SLCS Pumps Start	139
RHR Cooling Initiated (first train/second train)	1100/1600
Peak Suppression Pool Temperature	1959
Hot Shutdown Achieved (Neutron flux remains <0.1%)	1656

Table 15.8-10

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TABLE 15.8-11

RESULTS FOR PREGO ATWS EVENT

Parameter	Result	Limit
Peak Vessel Pressure (psig)	1336	1500
Peak Clad Temperature (^{o}F)	1434	2200
Peak Suppression Pool Temperature (^{o}F)	206	220

Table 15.8-12

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15.9 STATION BLACKOUT (SBO)

15.9.0 COPING ASSESSMENT FOR THE SUSQUEHANNA STEAM ELECTRIC STATION DURING A STATION BLACKOUT

10CFR50.63, "Loss of All Alternating Current Power," requires all licensees to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during a station blackout (SBO) and to have procedures to cope with such an event. In order to comply with this Nuclear Regulatory Commission rule, a detailed coping assessment for the Susquehanna Steam Electric Station (SSES) was undertaken, based on Regulatory Guide 1.155 (Reference 15.9-1), utilizing the methodology provided by the Nuclear Management and Resources Council (NUMARC) in NUMARC 87-00, Revision 1 (Reference 15.9-2). This assessment concluded that the required SSES SBO coping time was four (4) hours and this coping assessment received NRC review and approval (Reference 15.9-3). Plant specific analysis was used in areas where necessary to more accurately represent SSES and a detailed evaluation of SSES response to an SBO event was performed. The results demonstrate that SSES can successfully cope with an SBO event, using current plant procedures, for the required four (4) hour period.

The preferred method of coping with SBO at SSES is based upon the following criteria as identified by PPL's approach to accident management:

- Extending the time to Reactor Vessel and Primary Containment challenge; and
- Maximizing the availability of plant equipment necessary to cope with and recover from SBO.

Using these criteria will assure that the required four (4) hour SBO coping time (before fuel integrity, or any other acceptance criterion in either 10CFR50.63 or NUMARC 87-00, is challenged) continues to be met.

All Plant equipment (i.e., systems and instrumentation) necessary to cope with SBO, recover from SBO, and ensure Primary Containment isolations were identified and investigated to assure that all items necessary for the equipment to function would be available for at least 4 hours. Instrumentation required for the 4 hour station blackout commitment is listed in Table 15.9-1 for Unit 1 and Table 15.9-2 for Unit 2. In addition, coping beyond 4 hours was analyzed with respect to equipment necessary to cope with and recover from SBO.

15.9.1 REFERENCES

15.9-1 U. S. Nuclear Regulatory Commission, Regulatory Guide 1.155, ‘Station Blackout, ’August 1988.

15.9-2 NUMARC 87-00, Revision 1, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout in Light Water Reactors, Nuclear Management and Resource Council, August, 1991.

15.9-3 Letter, George F. Maxwell (NRC) to Harold W. Keiser, “Supplement Safety Evaluation (SSE) on the Station Blackout Rule for Susquehanna Steam Electric Station, Units 1 and 2 (TAC Nos. M68613 and M68614), June 16, 1992.

Table 15.9-1 – Unit 1 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
LI-14201A	LI-14201A	Reactor Vessel Level	1C601	12/729	CR	1Y11501	Y	I	J-803-4
	LT-14201A	(Wide Range)	1C225	27/749	I-513	1Y11505	Y	I	M-142-1
	LT-14203A	(Extended Range)	1C224	27/749	I-513	1Y11505	Y	I	J-442-1
LI-14201A1	LI-14201A1	(Extended Range)	1C651	12/729	CR	1Y11501	Y	I	J-803-4
	LT-14201A	(Extended Range)	1C225	27/749	I-513	1Y11505	Y	I	M-142-1
	LT-14203A	(Extended Range)	1C224	27/749	I-513	1Y11505	Y	I	J-442-1
LI-14203A	LI-14203A	(Extended Range)	1C601	12/729	CR	1Y11501	Y	I	J-802-6
	LT-14203A	(Extended Range)	1C224	27/749	I-513	1Y11505	Y	I	M-142-1
	LT-14201A	(Extended Range)	1C225	27/749	I-513	1Y11505	Y	I	J-442-1
UR-14201A	UR-14201A	Reactor Vessel Level/Press.	1C601	12/729	CR	1Y11501	Y	I	J-802-4
	LT-14202A	(Fuel Zone Range)	Local	29/719	I-401	1Y11505	Y	I	M-142-1
	LT-14201A	(Wide Range)	1C225	27/749	I-513	1Y11505	Y	I	J-442-3
	PT-14201A	(Wide Range)	1C225	27/749	I-513	1Y11505	Y	I	J-442-3
LI-14201B	LI-14201B	Reactor Vessel Level	1C601	12/729	CR	1Y12501	Y	II	J-802-3
	LT-14201B	(Wide Range)	1C225	27/749	I-513	1Y12505	Y	II	M-142-1
	LT-14203B	(Extended Range)	1C224	27/749	I-513	1Y12505	Y	II	
LI-14201B1	LI-14201B1	(Extended Range)	1C651	12/729	CR	1Y12501	Y	II	J-803-4
	LT-14201B	(Extended Range)	1C225	27/749	I-513	1Y12505	Y	II	M-142-1
	LT-14203B	(Extended Range)	1C224	27/749	I-513	1Y12505	Y	II	
LI-14203B	LI-14203B	(Extended Range)	1C601	12/729	CR	1Y12501	Y	II	J-802-6
	LT-14203B	(Extended Range)	1C224	27/749	I-513	1Y12505	Y	II	M-142-1
	LT-14201B	(Extended Range)	1C225	27/749	I-513	1Y12505	Y	II	
UR-14201B	UR-14201B	Reactor Vessel Level/Press.	1C601	12/729	CR	1Y12501	Y	II	J-802-7
	LT-14202B	(Fuel Zone Range)	Local	29/719	I-401	1Y12505	Y	II	M-142-1
	LT-14201B	(Wide Range)	1C225	27/749	I-513	1Y12505	Y	II	M-142-1
	PT-14201B	(Wide Range)	1C225	27/749	I-513	1Y12505	Y	II	M-142-1
PI-14202A	PI-14202A	Reactor Vessel Pressure	1C601	12/729	CR	1Y11501	Y	I	J-802-3
	PT-14201A	(Wide Range)	1C225	27/749	I-513	1Y11505	Y	I	J-442-3
PI-14202A1	PI-14202A1	Reactor Vessel Pressure	1C651	12/729	CR	1Y11501	Y	I	J-803-4
	PT-14201A	(Wide Range)	1C225	27/749	I-513	1Y11505	Y	I	J-442-3
PI-14204A	PI-14204A	Reactor Vessel Pressure	1C601	12/729	CR	1Y11501	Y	I	J-806-6
	PT-14201A	(Wide Range)	1C225	27/729	I-513	1Y11505	Y	I	J-442-3
PI-14202B	PI-14202B	Reactor Vessel Pressure	1C601	12/729	CR	1Y12501	Y	II	J-802-3
	PT-14201B	(Wide Range)	1C224	27/749	I-513	1Y12505	Y	II	M-142-1
	PT-14203B		1C225	27/749	I-513	1Y12505	Y	II	

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Table 15.9-1 – Unit 1 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
PI-14202B1	PI-14202B1	Reactor Vessel Pressure	1C651	12/729	CR	1Y12501	Y	II	J-803-4
	PT-14201B	(Wide Range)	1C224	27/749	I-513	1Y12505	Y	II	M-142-1
	PT-14203B		1C225	27/740	I-513	1Y12505	Y	II	
PI-14204B	PI-14204B	Reactor Vessel Pressure	1C601	12/729	CR	1Y12501	Y	II	J-802-6
	PI-14201B	(Wide Range)	1C224	27/749	I-513	1Y12505	Y	II	M-142-1
	PI-14203B		1C225	27/749	I-513	1Y12505	Y	II	
PI-E51-1R602	PI-E51-1R602	Reactor Pressure	1C601	12/729	CR	1D61407	Y	II	J-449
	PT-E51-1N007	(RCIC Turbine Steam Supply)	1C017	28/645	I-012	1D64107	Y	II	M-149
PI-E41-1R602	PI-E41-1R602	Reactor Pressure	1C601	12/729	CR	1D62406	Y	II	
	PT-E41-1N013	(HPCI Turbine Steam Supply)	1C014	25/645	I-010	1D62406	Y	II	M-155
TIAH-15751	TIAH-15751	Supp. Pool Temp. Monitoring	1C601	12/729	CR	1Y11501	Y	I	J-802-5
	TX-15751	Supp. Pool Temp. Monitoring	1C690A	12/729	CR	1Y11502	Y	I	J-457-9
	TE-15753	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	E-64-15
	TE-15755	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	E-25-1
	TE-15757	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	M-157-3
	TE-15759	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	
	TE-15763	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	
	TE-15765	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	
	TE-15767	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	
	TE-15769	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y11502	Y	I	
	TE-15751	Supp. Pool Temp. Monitoring	Local	26/683	I-206	1Y11502	Y	I	
	TE-15756	Supp. Pool Temp. Monitoring	Local	26/683	I-206	1Y11502	Y	I	
	TE-15761	Supp. Pool Temp. Monitoring	Local	26/683	I-206	1Y11502	Y	I	
	TE-15764	Supp. Pool Temp. Monitoring	Local	26/683	I-206	1Y11502	Y	I	
TIAH-15752	TIAH-15752	Supp. Pool Temp. Monitoring	1C601	12/729	CR	1Y12501	Y	II	J-802-5
	TX-15752	Supp. Pool Temp. Monitoring	1C609B	12/729	CR	1Y12502	Y	II	M-157-3
	TE-15752	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	J-457-10
	TE-15754	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	E-64-15
	TE-15758	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	E-25-1
	TE-15760	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	
	TE-15762	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	
	TE-15766	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	
	TE-15768	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	
	TE-15770	Supp. Pool Temp. Monitoring	Local	26/692	I-206	1Y12502	Y	II	
UR-15776A	UR-15776A	Suppression Pool Level	1C601	12/729	CR	1Y11501	Y	I	J-802-5
	LT-15766A	(Wide Range)	Local	27/645	I-17	1Y11505	Y	I	M-157-3

Table 15.9-1 – Unit 1 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
	LT-15775A	(Narrow Range)	Local	27/645	I-17	M	Y	I	J-457-2
LI-15775A	LI-15775A	Suppression Pool Level	1C601	12/729	CR	1Y11505	Y	I	J-802-2
	LT-15775A	(Narrow Range)	Local	27/645	I-17	1Y11505	Y	I	M-157-3
									J-457-2
UR-15776B	UR-15776B	Suppression Pool Level	1C601	12/729	CR	1Y21501	Y	II	J-802-6
	LT-15766B	(Wide Range)	Local	25/645	I-10	1Y21505	Y	II	M-157-3
	LT-15775B	(Narrow Range)	Local	25/645	I-10	1Y21505	Y	II	J-457-2A
	LY-15776B		1C661	12/698	C-203	1Y12505	Y	I	J-457-2A
LI-15775B	LI-15775B	Suppression Pool Level	1C601	12/729	CR	1Y12505	Y	II	J-802-8
	LT-15775B	(Narrow Range)	Local	25/645	I-10	1Y12505	Y	II	M-157-3
LI-15776A2	LI-15776A2	Suppression Pool Level	Local	27/645	I-17	N/R	-	-	M-157
UR-15701A	UR-15701A	Drywell Pressure	1C601	12/729	CR	1Y11505	Y	I	J-802-5
	PT-15710A	(LOCA Range)	Local	27/719	I-401	1Y11505	Y	I	M-157-3
	PT-15709A	(Hi Accident Range)	Local	27/719	I-401	1Y11505	Y	I	J-457-3
UR-15701B	UR-15701B	Drywell Pressure	1C601	12/729	CR	1Y12505	Y	II	J-802-6
	PT-15710B	(LOCA Range)	Local	28/719	I-401	1Y12505	Y	II	M-157-3
	PT-15709B	(Hi Accident Range)	Local	28/719	I-401	1Y12505	Y	II	J-457-3A
PI-15702	PI-15702	Drywell Pressure	1C601	12/729	CR	1Y62925	Y	N	M-157-3
	PT-15728A	(Containment/Supp Chamber)	Local	27/683	I-203	1Y62925	Y	N	J-457-3
UR-15701A	UR-15701A	Drywell Temperature	1C601	12/729	CR	1Y11505	Y	I	J-802-5
	TT-15790A	Drywell Temperature	1C661A	12/754	URR	1Y11505	Y	I	M-157-1
	TE-15790A	Drywell Temperature	Local	26/752	I-516	1Y11505	Y	I	J-457-7
UR-15701B	UR-15701B	Drywell Temperature	1C601	12/729	CR	1Y12505	Y	II	J-802-6
	TT-15790B	Drywell Temperature	1C661B	12/698	LRR	1Y12505	Y	II	M-157-3
	TE-15790B	Drywell Temperature	Local	26/752	I-516	1Y12505	Y	II	J-457-7A
TI-15727A	TI-15727A	RHR Heat Ex. Disch. Temp.	1C601	12/729	CR	1Y11501	Y	I	M-151-1
	TT-15727A	RHR Heat Ex. Disch. Temp.	1C661A	12/754	URR	1Y11505	Y	I	J-451-15
	TE-E11-1N027A	RHR Heat Ex. Disch. Temp.	Pipe	29/683	I-201	1Y11505	Y	I	
TI-15727B	TI-15727B	RHR Heat Ex. Disch. Temp.	1C601	12/729	CR	1Y12501	Y	II	J-802-7
	TT-15727B	RHR Heat Ex. Disch. Temp.	1C661B	12/683	LRR	1Y12505	Y	II	M-151-3
	TE-E11-1N027B	RHR Heat Ex. Disch. Temp.	Pipe	29/683	I-201	1Y12505	Y	II	M-151-2
FI-15120A	FI-15120A	Drywell & Wetwell Spray Flow	1C601	12/729	CR	1Y11501	Y	I	J-802-4
	FY-15120A	Drywell Spray	1C661A	12/754	URR	1Y11505	Y	I	M-151-3
	FT-15120A	Drywell Spray	Local	29/749	I-513	1Y11505	Y	I	J-470-2
	FY-15121A	Suppression Pool Spray	1C661A	12/754	URR	1Y11505	Y	I	E-64-15
	FT-15121A	Suppression Pool Spray	Local	27/683	I-203	1Y11505	Y	I	

Table 15.9-1 – Unit 1 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
FI-15120B	FI-15120B	Drywell & Wetwell Spray Flow	1C601	12/729	CR	1Y12501	Y	II	J-802-7
	FY-15120B	Drywell Spray	1C661B	12/698	LRR	1Y12505	Y	II	M-151-1
	FT-15120B	Drywell Spray	Local	25/749	I-509	1Y12505	Y	II	J-470-2
	FY-15121B	Suppression Pool Spray	1C661B	12/698	LRR	1Y12505	Y	II	E-64-15
	FT-15121B	Suppression Pool Spray	Local	25/683	I-200	1Y12505	Y	II	
PI-15702	PI-15702	Suppression Chamber Press.	1C601	12/729	CR	1Y62925	Y	N	M-157-3
	PT-15702		Local	27/683	I-203	1Y62925	Y	N	J-457-3
PI-E41-1R606	PI-E41-1R606	HPCI Pump Suction Pressure	1C601	12/729	CR	1D62406	Y	II	J-456-3
	PT-E41-1N019		1C014	25/645	I-010	1D62406	Y	II	M-156-1
PI-E41-1R602	PI-E41-1R602	HPCI Steam Supply Pressure	1C601	12/729	CR	1D62406	Y	II	J-455-1
	PT-E41-1N013		1C014	25/645	I-010	1D62406	Y	II	M-155
PI-E41-1R603	PI-E41-1R603	HPCI Turbine Exhaust Press.	1C601	12/729	CR	1D62406	Y	II	J-456-3
	PT-E41-1N016		1C014	25/645	I-010	1D62406	Y	II	M-156-1
PI-E41-1R601	PI-E41-1R601	HPCI Pump Discharge Press.	1C601	12/729	CR	1D62406	Y	II	J-455-1
	PT-E41-1N009		1C014	25/645	I-010	1D62406	Y	II	M-155
FI-E41-1R600-1	FI-E41-1R600-1	HPCI Flow	1C601	12/729	CR	1D62406	Y	II	J-455-2
	FY-E41-1K601		1C601	12/729	CR	1D62406	Y	II	M-155
	FT-E41-1N008		1C014	25/645	I-010	1D62406	Y	II	
SI-E41-1R604	SI-E41-1R604	HPCI Turbine Speed	1C601	12/729	CR	1D62406	Y	II	J-455-2
	SY-15683		TB0078	28/645	I-011	1D62406	Y	II	
	SE-15661		15211	28/645	I-011	1D62406	Y	II	
LI-00812A	LI-00812A	CST ‘A’ Level	OCB518A	36/670	I-130	1Y62927	Y	N	J-408-2
	LT-00812A		OCB518A	36/670	I-130	1Y62927	Y	N	M-108
LI-00812B	LI-00812B	CST ‘B’ Level	OCB518B	45/670	-	1Y62927	Y	N	J-408-2
	LT-00812B		OCB518B	45/670	-	1Y62927	Y	N	M-108
LI-00802	LI-00802	RWST Level	OCB517	35/670	I-130	1Y62927	Y	N	J-408-2
	LT-00802		OCB517	35/670	I-130	1Y62927	Y	N	M-108
PI-12649	PI-12649	Containment Instrument Gas	1C601	12/729	CR	1Y12501	Y	I	J-426-3
	PT-12649	Bottle Pressure	Local	25/719	I-408	1Y12501	Y	I	
PI-E51-1R603	PI-E51-1R603	RCIC Turbine Exhaust Press.	1C601	12/729	CR	1D61407	Y	I	J-450-1
	PT-E51-1N008		1C017	28/645	I-012	1D61407	Y	I	M-150
PI-E51-1R602	PI-E51-1R602	RCIC Turbine Inlet Pressure	1C601	12/729	CR	1D61407	Y	I	J-449-1
	PT-E51-1N007		1C017	28/645	I-012	1D61407	Y	I	M-149
PI-E51-1R604	PI-E51-1R604	RCIC Pump Suction Pressure	1C601	12/729	CR	1D61407	Y	I	J-450-1
	PT-E51-1N005		1C017	28/645	I-012	1D61407	Y	I	M-150
PI-E51-1R601	PI-E51-1R601	RCIC Pump Discharge Press.	1C601	12/729	CR	1D61407	Y	I	J-449-1

Table 15.9-1 – Unit 1 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
	PT-E51-1N004		1C017	28/645	I-012	1D61407	Y	I	M-149
FI-E51-1R600-1	FI-E51-1R600-1	RCIC Flow	1C601	12/729	CR	1D61407	Y	I	J-449-2
	FY-E51-1K601		1C601	12/729	CR	1D61407	Y	I	M-149
	FT-E51-1N003		1C017	28/645	I-012	1D61407	Y	I	
SI-15001A	SI-15001A	RCIC Turbine Speed	1C601	12/729	CR	1D61407	Y	I	J-449-2
	EGM		TB0145	28/645	I-012	1D61407	Y	I	M-149
	SE(Not Shown)		1C212	28/645	I-012	1D61407	Y	I	
TI-14185	TI-14185	RCIC Area Temperature	1C614	12/729	CR	1D63401	Y	I	J-450-2
	TY-14185		1C614	12/729	CR	1D63401	Y	I	
	TSH-E51-1N600A	Ambient Temperature	1C614	12/729	CR	1D63401	Y	I	
	TE-E51-1N011A	Ambient Temperature	Local	28/645	I-012	1D63401	Y	I	
	TSH-E51-1N602A	Emerg. Area Clr. Amb. Temp.	1C614	12/729	CR	1D63401	Y	I	
	TSH-E51-1N023A	Emerg. Area Clr. Amb. Temp.	Duct	28/645	I-012	1D63401	Y	I	

Table 15.9-2 – Unit 2 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
LI-24201A	LI-24201A	Reactor Vessel Level	2C601	21/729	CR	2Y11501	Y	I	J-2802-3
	LT-24201A	(Wide Range)	2C225	30/749	II-513	2Y11505	Y	I	M-2142-1
	LT-24203A	(Extended Range)	2C224	33/749	II-513	2Y11505	Y	I	J-2802-3
LI-24201A1	LI-24201A1	(Extended Range)	2C652	21/729	CR	2Y11501	Y	I	J-2803-4
	LT-24201A	(Extended Range)	2C225	30/749	II-513	2Y11505	Y	I	M-2142-1
	LT-24203A	(Extended Range)	2C224	33/749	II-513	2Y11505	Y	I	J-2442-2
LI-24203A	LI-24203A	(Extended Range)	2C601	21/729	CR	2Y11501	Y	I	J-2802-6
	LT-24203A	(Extended Range)	2C224	33/749	II-513	2Y11505	Y	I	M-2142-1
	LT-24201A	(Extended Range)	2C225	30/749	II-513	2Y11505	Y	I	J-2802-6
UR-24201A	UR-24201A	Reactor Vessel Level/Press.	2C601	21/729	CR	2Y11501	Y	I	J-2802-4
	LT-24202A	(Fuel Zone Range)	Local	34/719	II-401	2Y11505	Y	I	M-2142-1
	LT-24201A	(Wide Range)	2C225	30/749	II-513	2Y11505	Y	I	M-2142
	PT-24201A	(Wide Range)	2C225	30/749	II-513	2Y11505	Y	I	M-2142
LI-24201B	LI-24201B	Reactor Vessel Level	2C601	21/729	CR	2Y12501	Y	II	J-2802-3
	LT-24201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142
	LT-24203B	(Extended Range)	2C225	30/749	II-513	2Y12505	Y	II	
LI-24201B1	LI-24201B1	(Extended Range)	2C651	21/729	CR	2Y12501	Y	II	J-2803-4
	LT-24201B	(Extended Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142
	LT-24203B	(Extended Range)	2C225	33/749	II-513	2Y12505	Y	II	
LI-24203B	LI-24203B	(Extended Range)	2C601	21/729	CR	2Y12501	Y	II	J-2802-6
	LT-24203B	(Extended Range)	2C225	30/749	II-513	2Y12505	Y	II	M-2142
	LT-24201B	(Extended Range)	2C224	33/749	II-513	2Y12505	Y	II	
UR-24201B	UR-24201B	Reactor Vessel Level/Press.	2C601	21/729	CR	2Y12501	Y	II	J-2802-7
	LT-24202B	(Fuel Zone Range)	Local	30/719	II-401	2Y12505	Y	II	M-2142
	LT-14201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142
	PT-14201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	J-2802-4
PI-24202A	PI-24202A	Reactor Vessel Pressure	2C601	21/729	CR	2Y11501	Y	II	J-2802-3
	PT-24201A	(Wide Range)	2C225	30/749	II-513	2Y11505	Y	II	M-2142
PI-24202A1	PI-24202A1	Reactor Vessel Pressure	2C651	21/729	CR	2Y11501	Y	I	J-2803-4
	PT-24201A	(Wide Range)	2C225	30/749	II-513	2Y11505	Y	I	M-2142
PI-24204A	PI-24204A	Reactor Vessel Pressure	2C601	21/729	CR	2Y11501	Y	I	J-2802-6
	PT-24201A	(Wide Range)	2C225	30/729	II-513	2Y11505	Y	I	M-2142
PI-24202B	PI-24202B	Reactor Vessel Pressure	2C601	21/729	CR	2Y12501	Y	II	J-2802-3
	PT-24201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142

Table 15.9-2 – Unit 2 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
	PT-24203B		2C225	30/749	II-513	2Y12505	Y	II	J-2802-4
PI-24202B1	PI-24202B1	Reactor Vessel Pressure	2C651	21/729	CR	2Y12501	Y	II	
	PT-24201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142
	PT-24203B		2C225	30/740	II-513	2Y12505	Y	II	
PI-24204B	PI-24204B	Reactor Vessel Pressure	2C601	21/729	CR	2Y12501	Y	II	J-2802-6
	PI-24201B	(Wide Range)	2C224	33/749	II-513	2Y12505	Y	II	M-2142
	PI-24203B		2C225	30/749	II-513	2Y12505	Y	II	
PI-E51-2R602	PI-E51-2R602	Reactor Pressure	2C601	21/729	CR	2D61407	Y	I	J-2449-1
	PT-E51-2N007	(RCIC Turbine Steam Supply)	2C017	33/645	II-012	2D64107	Y	I	M-2149
PI-E41-2R602	PI-E41-2R602	Reactor Pressure	2C601	21/729	CR	2D62406	Y	II	
	PT-E41-2N013	(HPCI Turbine Steam Supply)	2C014	33/645	II-010	2D62406	Y	II	M-2155
TIAH-25751	TIAH-25751	Supp. Pool Temp. Monitoring	2C601	21/729	CR	2Y11501	Y	I	J-2802-5
	TX-25751	Supp. Pool Temp. Monitoring	2C690A	21/729	CR	2Y11502	Y	I	M-2157
	TE-25753	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	J-2457-9
	TE-25755	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25757	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25759	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25763	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25765	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25767	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25769	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y11502	Y	I	
	TE-25751	Supp. Pool Temp. Monitoring	Local	31/683	II-206	2Y11502	Y	I	
	TE-25756	Supp. Pool Temp. Monitoring	Local	31/683	II-206	2Y11502	Y	I	
	TE-25761	Supp. Pool Temp. Monitoring	Local	31/683	II-206	2Y11502	Y	I	
	TE-25764	Supp. Pool Temp. Monitoring	Local	31/683	II-206	2Y11502	Y	I	
TIAH-25752	TIAH-25752	Supp. Pool Temp. Monitoring	2C601	21/729	CR	2Y12501	Y	II	J-2802-5
	TX-25752	Supp. Pool Temp. Monitoring	2C609B	21/729	CR	2Y12502	Y	II	M-2157
	TE-25752	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	J-2457-10
	TE-25754	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25758	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25760	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25762	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25766	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25768	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	
	TE-25770	Supp. Pool Temp. Monitoring	Local	31/692	II-206	2Y12502	Y	II	

Table 15.9-2 – Unit 2 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
LR-25776A	LR-25776A	Suppression Pool Level	2C601	21/729	CR	2Y11501	Y	I	J-2802-5
	LT-25766A	(Wide Range)	Local	32/645	II-017	2Y11505	Y	I	M-2157-3
	LT-25775A	(Narrow Range)	Local	32/645	II-017	2Y11505	Y	I	J-2457-2
	LY-25776A		2C661	21/754	C-502	2Y11505	Y	I	J-2457-2
LI-25775A	LI-25775A	Suppression Pool Level	2C601	21/729	CR	2Y11505	Y	I	J-2802-3
	LT-25775A	(Narrow Range)	Local	32/645	II-017	2Y11505	Y	I	M-2157-3
LR-25776B	LR-25776B	Suppression Pool Level	2C601	21/729	CR	2Y21505	Y	II	J-2802-6
	LT-25775B	(Marrow Range)	Local	30/645	II-10	2Y21505	Y	II	M-2157-3
	LT-25776B	(Wide Range)	Local	30/645	II-10	2Y21505	Y	II	J-2457-2
LI-25775B	LI-25775B	Suppression Pool Level	2C601	21/729	CR	2Y12505	Y	II	J-2802-8
	LT-25775B	(Narrow Range)	Local	30/645	II-10	2Y12505	Y	II	M-2157-3
LI-25776A2	LI-25776A2	Suppression Pool Level	Local	30/645	II-010	N/R	-	-	M-2157-3
UR-25701A	UR-25701A	Drywell Pressure	2C601	21/729	CR	2Y11505	Y	I	J-2802-5
	PT-25710A	(LOCA Range)	Local	32/719	II-401	2Y11505	Y	I	M-2157
	PT-25709A	(Hi Accident Range)	Local	32/719	II-401	2Y11505	Y	I	J-2457-3
UR-25701B	UR-25701B	Drywell Pressure	2C601	21/729	CR	2Y12505	Y	II	J-2802-6
	PT-25710B	(LOCA Range)	Local	33/719	II-413	2Y12505	Y	II	M-2157
	PT-25709B	(Hi Accident Range)	Local	33/719	II-413	2Y12505	Y	II	J-2457-3A
PI-25702	PI-25702	Drywell Pressure	2C601	21/729	CR	2Y62925	Y	N	M-2157
	PT-25728A1	(Containment/Supp Chamber)	Local	32/719	II-401	2Y62925	Y	N	J-2457-3
UR-25701A	UR-25701A	Drywell Temperature	2C601	21/729	CR	2Y11505	Y	I	J-2802-5
	TT-25790A	Drywell Temperature	2C661A	21/754	URR	2Y11505	Y	I	M-2157
	TE-25790A	Drywell Temperature	Local	31/752	II-516	2Y11505	Y	I	J-2457-7
UR-25701B	UR-25701B	Drywell Temperature	2C601	21/729	CR	2Y12505	Y	II	J-2802-6
	TT-25790B	Drywell Temperature	2C661B	21/698	LRR	2Y12505	Y	II	M-2157
	TE-25790B	Drywell Temperature	Local	31/752	II-516	2Y12505	Y	II	J-2457-7A
TI-25727A	TI-25727A	RHR Heat Ex. Disch. Temp.	2C601	21/729	CR	2Y11501	Y	I	M-2151
	TT-25727A	RHR Heat Ex. Disch. Temp.	2C661A	21/754	URR	2Y11505	Y	I	J-2802-4
TI-25727B	TI-25727B	RHR Heat Ex. Disch. Temp	2C601	21/729	CR	2Y12501	Y	II	J-2802-7
	TT-25727B	RHR Heat Ex. Disch. Temp.	2C661B	21/683	LRR	2Y12505	Y	II	M-2151
	TE-E11-2N027B	RHR Heat Ex. Disch. Temp.	Pipe	33/683	II-202	2Y12505	Y	II	
FI-25120A	FI-25120A	Drywell & Wetwell Spray Flow	2C601	21/729	CR	2Y11501	Y	I	J-2802-4
	FY-25120A	Drywell Spray	2C661A	21/754	URR	2Y11505	Y	I	M-2151
	FT-25120A	Drywell Spray	Local	33/749	II-513	2Y11505	Y	I	J-2452-2
	FY-25121A	Suppression Pool Spray	2C661A	21/754	URR	2Y11505	Y	I	

Table 15.9-2 – Unit 2 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
	FT-25121A	Suppression Pool Spray	Local	34/683	II-203	2Y11505	Y	I	
FI-25120B	FI-25120B	Drywell & Wetwell Spray Flow	2C601	21/729	CR	2Y12501	Y	II	J-2802-7
	FY-25120B	Drywell Spray	2C661B	21/698	LRR	2Y12505	Y	II	M-2151
	FT-25120B	Drywell Spray	Local	33/749	II-513	2Y12505	Y	II	J-2451-2
	FY-25121B	Suppression Pool Spray	2C661B	21/698	LRR	2Y12505	Y	II	
	FT-25121B	Suppression Pool Spray	Local	30/683	Ii-200	2Y12505	Y	II	
PI-22649	PI-22649	Containment Instrument Gas	1C601	21/729	CR	2Y121501	Y	II	J-2426-2
	PT-22649	Bottle Pressure	Local	30/719	II-408	2Y12505	Y	II	
PI-25702	PI-25702	Suppression Chamber Press.	2C601	21/729	CR	2Y62925	Y	N	M-2157
	PT-25702		Local	32/683	II-203	2Y62925	Y	N	J-2457-3
PI-E41-2R606	PI-E41-2R606	HPCI Pump Suction Pressure	2C601	21/729	CR	2D62406	Y	II	J-2456-3
	PT-E41-2N019		2C014	30/645	II-010	2D62406	Y	II	M-2156
PI-E41-2R602	PI-E41-2R602	HPCI Steam Supply Pressure	2C601	21/729	CR	2D62406	Y	II	J-2455-1
	PT-E41-2N013		2C014	30/645	II-010	2D62406	Y	II	M-2155
PI-E41-2R603	PI-E41-2R603	HPCI Turbine Exhaust Press.	2C601	21/729	CR	2D62406	Y	II	J-2456-3
	PT-E41-2N016		2C014	30/645	II-010	2D62406	Y	II	M-2156
PI-E41-2R601	PI-E41-2R601	HPCI Pump Discharge Press.	2C601	21/729	CR	2D62406	Y	II	J-2455-1
	PT-E41-2N009		2C014	30/645	II-010	2D62406	Y	II	M-2155
FI-E41-2R600-1	FI-E41-2R600-1	HPCI Flow	2C601	21/729	CR	2D62406	Y	II	J-2455-2
	FY-E41-2K601		2C601	21/729	CR	2D62406	Y	II	M-2155
	FT-E41-2N008		2C014	30/645	II-010	2D62406	Y	II	
SI-E41-2R604	SI-E41-2R604	HPCI Turbine Speed	2C601	21/729	CR	2D62406	Y	II	J-2455-2
	SY-25683		TB0195	33/645	II-011	2D62406	Y	II	
	SE-25661		25211	33/645	II-011	2D62406	Y	II	
LI-00812A	LI-00812A	CST 'A' Level	OCB518A	36/670	I-130	1Y62927	Y	N	J-408-2
	LT-00812A		OCB518A	36/670	I-130	1Y62927	Y	N	M-108
LI-00812B	LI-00812B	CST 'B' Level	OCB518B	45/670	-	1Y62927	Y	N	J-408-2
	LT-00812B		OCB518B	45/670	-	1Y62927	Y	N	M-108
LI-00802	LI-00802	RWST Level	OCB517	35/670	I-130	1Y62927	Y	N	J-408-2
	LT-00802		OCB517	35/670	I-130	1Y62927	Y	N	M-108
PI-E51-2R603	PI-E51-2R603	RCIC Turbine Exhaust Press.	2C601	21/729	CR	2D61407	Y	I	J-2450-1
	PT-E51-2N008		2C017	33/645	II-012	2D61407	Y	I	
PI-E51-2R602	PI-E51-2R602	RCIC Turbine Inlet Pressure	2C601	21/729	CR	2D61407	Y	I	J-2449-1
	PT-E51-2N007		2C017	33/645	II-012	2D61407	Y	I	
PI-E51-2R604	PI-E51-2R604	RCIC Pump Suction Pressure	2C601	21/729	CR	2D61407	Y	I	J-2450-1

Table Rev. 1

Table 15.9-2 – Unit 2 Station Blackout Instrumentation List

Instrument	Loop Devices	Parameter	Location	Area/Elev.	Room	Power Supply	Bat.	Div.	Drwg.
	PT-E51-2N005		2C017	33/645	II-012	2D61407	Y	I	
PI-E51-2R601	PI-E51-2R601	RCIC Pump Discharge Press.	2C601	21/729	CR	2D61407	Y	I	J-2449-1
	PT-E51-2N004		2C017	33/645	II-012	2D61407	Y	I	
FI-E51-2R600-1	FI-E51-2R600-1	RCIC Flow	2C601	21/729	CR	2D61407	Y	I	J-2449-2
	FY-E51-2K601		2C601	21/729	CR	2D61407	Y	I	
	FT-E51-2N003		2C017	33/645	II-012	2D61407	Y	I	
SI-25001A	SI-25001A	RCIC Turbine Speed	2C601	12/729	CR	2D61407	Y	I	J-2449-2
	EGM		TB0520	33/645	II-012	2D61407	Y	I	
	SE(Not Shown)		25212	33/645	II-012	2D61407	Y	I	
TI-24185	TI-24185	RCIC Area Temperature	2C614	21/729	CR	2D63401	Y	I	J-2450-2
	TY-24185		2C614	21/729	CR	2D63401	Y	I	
	TSH-E51-2N600A	Ambient Temperature	2C614	21/729	CR	2D63401	Y	I	
	TE-E51-2N011A	Ambient Temperature	Local	33/645	II-012	2D63401	Y	I	
	TSH-E51-2N602A	Emerg. Area Clr. Amb. Temp.	2C614	21/729	CR	2D63401	Y	I	
	TSH-E51-2N023A	Emerg. Area Clr. Amb. Temp.	Duct	33/645	II-012	2D63401	Y	I	

START HISTORICAL

APPENDIX 15A

NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) -
(A System-Level/Qualitative Type Plant FMEA)

15A.1 OBJECTIVES

15A.1.1 General Objectives

The general objectives of the Nuclear Safety Operational Analysis (NSOA) are cited below along with the mission of each objective.

(1) *Essential Protective Sequences - to identify and demonstrate that the essential protection sequences needed to accommodate the plant normal operations, anticipated and abnormal operational transients, and design basis accidents are available and adequate.*

(2) *Design, Basis Adequacy - to identify and demonstrate that the safety design basis of the various structures, systems or components, needed to satisfy the plant essential protection sequences are appropriate, available and adequate.*

(3) *System-Level/Qualitative Type FMEA - to provide a system level/qualitative-type Failure Modes and Effects Analysis (FMEA) of essential protective sequences to show compliance with the Single Active Component Failure (SACF) or Single Operator Error (SOE) criteria;*

(4) *NSOA Criteria Relative to Plant Safety Analysis - to identify the systems, equipment, or components' operational conditions or requirements essential to satisfy the nuclear safety operational criteria utilized in the Chapter 15 plant events; and*

(5) *Technical Specification Operational Basis - to establish limiting operating conditions, testing, and surveillance bases relative to plant technical specification operational requirements.*

15A.1.2 Specific Objectives

The specific objectives of the Nuclear Safety Operational Analysis (NSOA) are cited below:

(1) *Essential Protective Sequences - Each event considered in the plant safety analysis (Chapter 15) is further examined and analyzed. Essential protective sequences are identified. The appropriateness of each sequence is discussed for all operating modes. Each protective sequence path is evaluated for SACF.*

(2) *Design Basis Adequacy - Each essential protective sequence involves specific structures, systems or components performing safety or power generation functions. There are also interrelationships between primary systems and secondary or auxiliary equipment in providing these functions. The individual design bases (identified throughout the FSAR for each structure, system, or component) are brought together in this section. The entire plant safety analysis is evaluated here.*

(3) *System-Level/Qualitative Type FMEA - A system-level, qualitative-type FMEA is presented here. Each protective sequence entry is evaluated relative to SACF or SOE criteria. Safety classification aspects and interrelationships between systems are also considered. The system-level SACF or SOE is a conservative "worst-case" envelope evaluation. Discounting any less severe evaluations than SACF or SOE such as by quantitative analysis is not claimed in this section although certainly it would assure less limiting results than shown.*

(4) *NSOA Criteria Relative to Plant Safety Analysis - The safety analysis performed in Chapter 15 is further examined relative to the systematic classification of plant events by frequency of occurrence, radiological impact, unacceptable results, and allowable limits of the safety criteria for the various event classifications; normal (planned) operation, anticipated (expected) and abnormal (unexpected) operational transients, and design basis accidents are described.*

(5) *Technical Specifications Operational Basis - Evaluations presented in this section provide the basis for justifications of more realistic, engineered technical specifications including system or equipment surveillance requirements, allowable down times, etc.*

15A.2 APPROACH TO OPERATIONAL NUCLEAR SAFETY

15A.2.1 General Philosophy

The objective of this appendix is to derive nuclear safety operational requirements and analyses for the plant that are based on specified measures of nuclear safety.

The specified measures of safety used in this analysis are referred to as "unacceptable results-oriented." They are analytically determinable limits on the consequences of different classifications of plant events. The nuclear safety operational analysis is thus an "event-consequence-oriented" evaluation.

15A.2.2 Specific Philosophy

In this appendix the following guidelines are utilized to develop the NSOA.

1) *Scope and Classification of Plant Events*

The scope and classification of the situations analyzed will include:

a) Normal (Planned) Operations
b) Anticipated (Expected) Operational Transients
c) Abnormal (Unexpected) Operational Transients
d) Design Basis (Postulated) Accidents
e) Special (Hypothetical) Events

Refer to Tables 15A.2-1 through 15A.2-6 for specific event/classifications.

The events referenced and classified above represent the plant situations considered applicable to safety evaluation.

2) Safety and Power Generation Aspects

Safety considerations directly involve the health and safety of the off-site public. Matters identified with "safety" classification are governed by regulatory requirements. Safety functions include:

a) The accommodation of abnormal operational transients and postulated design basis accidents.
b) The maintenance of containment integrity, when necessary.
c) The assurance of ECCS, when necessary, and
d) The continuance of RCPB integrity, when necessary.

Safety is related to 1OCFR100 dose limits, infrequent and low probability occurrences, SACF criteria, worst case operating conditions and initial assumptions, automatic (10 minute) corrective action, significant unacceptable dose and environmental effects, and the involvement of other coincident (mechanistic or non-mechanistic) plant and environmental situations.

Power generation considerations are directly related to continued plant power generation operation, equipment operational matters, component availability aspects and indirectly related to long term off-site public effects.

Matters identified with "power generation" classification are also covered by regulatory guidelines. Power generation functions include:

a) the accommodation of planned operations and anticipated operational transients,
b) the minimization of radiological releases to appropriate levels,
c) the assurance of safe and orderly reactor shutdown, when necessary, and/or return to power generation operation, and
d) the continuance of plant equipment design conditions to ensure long term reliable operation.

Power generation is related to 10CFR20 and 10CFR50, Appendix I dose limits, moderate and high probability occurrences nominal operating conditions and initial assumptions, allowable immediate operator manual actions, and insignificant unacceptable dose and environmental effects.

3) Frequency of Events

Consideration of the frequency of the initial (or initiating) event is reasonably straight-forward. Added considerations of further component failures or operator errors complicates the classification grouping and the related limits or acceptable consequences. The events in this appendix are initially grouped by initiating frequency occurrence. The imposition of further failures will necessitate further reclassification. This reclassification will result in the event being listed in a less restrictive category.

The introduction of SACF or SCF or SOE in planned operation/anticipated and abnormal operational transient evaluations has not been previously considered a design basis or evaluation prerequisite. It is entertained here for plant capability demonstration purposes.

4) Conservative Analysis - Margins

The unacceptable results established in this appendix relative to the public health and safety aspects are in themselves in conformance to regulatory requirements. They are also in conformance with regulations by large margins even though the events, their assumptions, conditions of evaluation, coincident situations, the limits, etc., are equally conservative in themselves by large margins. Further introduction of large margin operational requirements is not reasonable or justifiable. The results of this NSOA should directly lead to envelope technical specifications.

The utilization of a margin allowance to introduce further limiting restrictions is not safety oriented.

5) Safety Function Definition

Consideration of the frequency of the need for a safety function should be very carefully weighed and examined in order to truly assess real design basis, operational and availability requirements.

First, the essential protective sequences shown for an event in this appendix are the minimum required to be available to satisfy the SACF or SOE evaluation aspects of the event and yet meet all safety functional objectives. Many more protective "success paths" exist with the event than are shown.

Second, not all the events involve the same natural, environmental or plant conditional assumptions. For example, LOCA and SSE are associated with Event 44. In Event 41, CRDA is not assumed to be associated with any SSE or OBE occurrence, therefore, seismic safety function requirements are inappropriate for Event 41, although most safety function equipment associated with the protective sequence are capable of more limiting events, such as Event 44. The probability of Event 41 is far less than Event 44 occurrence-wise and certainly evaluation-assumption-wise.

Third, containment may be a safety function for some events (when uncontained radiological effects would be unacceptable) but for other events it may not be applicable (e.g. during refueling). The requirement to maintain the containment during post-accident recovery is only needed to limit doses to less than 10CFR100. After radiological sources are depleted with time, further containment is unnecessary. Thus the time domain and need for a function is taken into account and considered when evaluating the events in this appendix.

Fourth, the use of low frequency, high priority ESF equipment, limiting unacceptable result events for high probability, minor unacceptable result events should not be misunderstood to require similar pedigree equipment requirements on other supplement motor-safety components.

The interpretation of the use of ESF-SACF capable systems for anticipated operational transient protective sequences should not lead to the assumption that these equipment requirements (seismic, redundancy, diversity, testable, IEEE, etc.) are required for this event or associated with the event.

6) *Envelope and Actual Event Analyses*

The event analyses presented in Chapter 15, when examined from the frequency standpoint, would lead to the conclusion that each year a spectrum of the events occur as postulated. Study of the operating and plant occurrences verifies that the protective sequences cited in Chapter 15 are conservative, and in most cases never needed. Experience, of course, has been confined to planned operation, anticipated operational transients, and a very small number of abnormal operational transients situations. Operator action is valuable and repeatedly demonstrated yet ignored as a protective sequence. Consideration of and credit for this success path should be allowed for operational transients.

15A.2.2.1 Consistency of the Analysis

An objective of this analysis is consistency. Therefore, it is worthwhile to investigate possible inconsistencies in the selection of nuclear safety operational requirements (and technical specifications); then it will be seen in the presented NSOA that such inconsistencies are avoided.

Figure 15A.2-1 illustrates three inconsistencies. Panel A shows the possible inconsistency resulting from operational requirements being placed on separated levels of protection for one event. If the second and sixth levels of protection are important enough to warrant operational requirements, then so are the third, fourth, and fifth levels. Panel B shows the possible inconsistency resulting from operational requirements being arbitrarily placed on some action thought to be important to safety. In the case shown, scram represents different protection levels for two similar events in one category; if the fourth level of protection for Event B is important enough to warrant an operational requirement, then so is the fourth level for Event A. Thus, to simply place operational requirements on all equipment needed for some action (scram, isolation, etc.) could be inconsistent and unreasonable if different protection levels are represented. Panel C shows the possible inconsistency resulting from operational requirements being placed on some arbitrary level of protection for any and all postulated events. Here the *inconsistency is not recognizing and accounting for different event categories based on cause or expected frequency of occurrence.*

Inconsistencies of the types illustrated in Figure 15A.2-1 are avoided in the NSOA by directing the analysis to "event-consequences-oriented" aspects. Analytical inconsistencies are avoided by treating all the events of categories under the same set of functional rules. Thus, it is valid to compare the results of the analyses of the events in any one category and invalid to compare events of different category to each other.

15A.2.3 Comprehensiveness of the Analysis

The analysis must be sufficiently comprehensive in method that (1) all plant hardware is considered; and, (2) that the full range of plant operating conditions are considered. The tendency to be preoccupied with "worst cases" (those that appear to give the most severe consequences) is recognized; however, the protection sequences essential to lesser cases may be different (more or less restrictive) from the "worst-case" sequence. To assure that operational and design basis requirements are defined and appropriate for all equipment essential to attaining acceptable consequences, all essential protection sequences must be identified for each of the plant safety events examinations.

Only in this way is a comprehensive level of safety attained. Thus, the NSOA is also "protection sequence-oriented" to achieve comprehensiveness.

15A.2.4 Systematic Approach of the Analysis

In summary, the systematic method utilized in this analysis contributes to both the consistency and comprehensiveness of the analysis mentioned above. The desired characteristics representative of a systematic approach to selecting BWR operational requirements are listed as follows:

- *(1) Specified measures of safety-unacceptable results*
- *(2) Consideration of all planned operations*
- *(3) Systematic event selection*
- *(4) Common treatment analysis (FMEA, SACF, SOE) of all events of any one type*
- *(5) Systematic identification of plant actions and systems essential to avoiding unacceptable results*
- *(6) Emergence of operational requirements and limits from system analysis*

Figure 15A.2-2 illustrates the systematic process by which the operational and design basis nuclear safety requirements and technical specifications are derived. The process involves the evaluation of carefully selected plant events relative to the unacceptable results (specified measures of safety). Those limits, actions, systems, and components found to be essential to achieving acceptable consequences are the subjects of operational requirements.

15A.2.5 Relationship of Nuclear Safety Operational Analysis to Safety Analyses of Chapter 15

One of the main objectives of the operational analysis is to identify all essential protection sequences and to establish the detailed equipment conditions essential to satisfying the nuclear safety operational criteria. The spectrum of events examined in Chapter 15 represent a complete set of plant safety considerations. The main objective of the earlier analyses of Chapter 15, is, of course, to provide detailed "worst-case" (limiting or envelope) analysis of the plant events. The "worst cases" are correspondingly analyzed and treated likewise in this appendix, but in light of frequency at occurrence, unacceptable results, assumption categorization, etc.

The detailed discussion relative to each of the events covered in Chapter 15 will not be repeated in this appendix. Tables 15A.2-1 through 15A.2-5 provide cross-correlation between the NSOA event, its protection sequence diagram, and its safety evaluation in Chapter 15.

15A.2.6 Relationship Between NSOA and Operational Requirements, Technical Specifications, Design Basis, and SACF Aspects

By definition, "an operational requirement" is a requirement or restriction (limit) on either the value of a plant variable or the operability condition associated with a plant system. Such requirements must be observed during all modes of plant operation (not just at full power) to assure that the plant is operated safely. There are two kinds of operational requirements for plant hardware;

(1) Limiting condition for operation: the required condition for a system while the reactor is operating in a specified state.
(2) Surveillance requirements: the nature and frequency of tests required to assure that the system is capable of performing its essential functions.

Operational requirements are systematically selected for one of two basic reasons:

(1) To assure that unacceptable results are avoided or mitigated following specified plant events by examining and challenging the system, component, and equipment design basis.
(2) To assure the existence of a single failure proof success path to acceptable consequences should a transient or accident occur by confirming SACF or SOE criteria conformance.

The operational requirements that emerge from the NSOA are frequently complex hardware requirements applicable only under certain carefully specified plant conditions. Although these complex operational requirements are the true safety requirements, they frequently are too complicated for direct use as a technical specification. As shown in Figure 15A.2-2, the complex operational requirements are conservatively simplified as a final step in the process so that a practical set of technical specifications and operating procedures may be obtained.

The individual structures, systems, components which perform a safety function are required to do so under design basis conditions including environmental consideration and under single *active component failure assumptions. The NSOA confirms the previous examination of the individual equipment (See "Evaluations" subsection) requirement conformance analyses.*

15A.2.7 Unacceptable Results Criteria

Tables 15A.2-6 through 15A.2-10 identify the unacceptable results associated with different event categories. In order to prevent or mitigate them, they are recognized as the major bases for identifying system operational requirements as well as the bases for all other safety analyses vs. criteria throughout the FSAR.

15.A.2.8 General Nuclear Safety Operational Criteria

The following general nuclear safety operational criteria are used to select operational requirements:

Applicability	***Nuclear Safety Operational Criteria***
Planned operation, anticipated abnormal operational transients, design basis accidents, and additional separate plant capability events	*The plant shall be operated so as to avoid unacceptable results.*
Anticipated and abnormal operational transients and design basis accidents	*The plant shall be operated in such a way that no Single Active Component Failure (SACF) can prevent the safety actions essential to avoiding the unacceptable results associated with anticipated or abnormal operational transients or design basis accidents. However, this requirement is not applicable during structure, system, or component repair if the availability of the safety action is maintained either by restricting the allowable repair time or by more frequently testing a redundant structure, system, or component.*

The unacceptable results associated with the different categories of plant operation and events are dictated by:

- *a) frequency of occurrence (probability),*
- *b) allowable limits (per the probability) - related to radiological, structural, environmental, etc., aspects,*
- *c) coincidence of other related or unrelated disturbances, and*
- *d) time domain of event and consequences consideration.*

15A.3 METHOD OF ANALYSIS

15A.3.1 General Approach

The NSOA is performed assuming that the plant design has been established. The end products of the analysis are the nuclear safety operational requirements and the restrictions on plant hardware and its operation that must be observed (1) to satisfy the nuclear safety operational criteria, and (2) to show compliance of the plant safety and power generation systems with plant wide requirements. Figure 15A.2-2 shows the process used in the analysis. The following inputs are required for the analysis of specific plant events:

(1) *Applicable unacceptable results (Subsection 15A.2.7)*
(2) *Applicable nuclear safety operational criteria (Subsection 15A.2.8)*
(3) *Definition of BWR operating states (Subsection 15A.3.2)*
(4) *Event selection criteria (Subsection 15A.3.3)*
(5) *Rules for event analysis (Subsection 15A.3.5)*

With this information, each selected event can be evaluated to determine systematically, the actions, the systems, and the limits essential to avoiding the defined unacceptable results. The essential plant components and limits so identified are then considered to be in agreement with and subject to nuclear operational, design basis requirements and technical specification restrictions.

15A.3.2 BWR Operating States

Four BWR operating states in which the reactor can exist are defined in Table 15A.3-1. The main objective in selecting operating states is to divide the BWR operating spectrum into sets of initial conditions to facilitate consideration of various events in each state.

Each operating state includes a wide spectrum of values for important plant parameters. Within each state, these parameters are considered over their entire range to determine the limits on their values necessary to satisfy the nuclear safety operational criteria. Such limitations are presented in the subsections of the FSAR that describe the systems associated with the parameter limit. The plant parameters to be considered in this manner include the following:

Reactor coolant temperature
Reactor vessel water level
Reactor vessel pressure
Reactor vessel water quality
Reactor coolant forced circulation flow rate
Reactor power level (thermal and neutron flux)
Core neutron flux distribution
Feedwater temperature
Containment temperature and pressure
Suppression pool water temperature and level
Spent fuel pool water temperature and level

15A.3.3 Selection of Events for Analysis

15A.3.3.1 Planned Operations

"Planned operation" refers to normal plant operation under predetermined conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or additional plant capability event) are not considered planned operations until the actions taken or equipment used in the plant are identical to those that would be used had the incident not occurred. As defined, the planned operations can be considered as a chronological sequence: refueling outage, achieving criticality, heatup, power operation, achieving shutdown, cooldown, and refueling outage.

The planned operations are defined below.

(1) *Refueling outage: includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is taken critical and returned to the shutdown condition. The following planned operations are included in refueling outage:*

   a. *Planned, physical movement of core components (fuel, control rods, etc.)*
   b. *Refueling test operations (except criticality and shutdown margin tests)*
   c. *Planned maintenance*
   d. *Required inspection*

(2) *Achieving criticality: Includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.*

(3) *Heatup: Begins when achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine-generator.*

(4) *Power operation: Begins when heatup ends and includes continued plant operation at power levels in excess of heatup power.*

(5) *Achieving Shutdown: Begins when the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation.*

(6) *Cooldown: Begins when achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure.*

The exact point at which some of the planned operations end and others begins cannot be precisely determined. It will be shown later that such precision is not required, for the protection requirements are adequately defined in passing from one state to the next. Dependence on several planned operations on the one rod subcritical condition provides an exact point on either side of which protection (especially scram) requirements differ. Thus, where a precise boundary between planned operations is needed, the definitions provide the needed precision.

Together, the BWR operating states and the planned operations define the full spectrum of conditions from which transients, accidents, and special events are initiated. The BWR operating states define only the physical condition (pressure, temperature, etc.) of the reactor; the planned operations define what the plant is doing. The separation of physical conditions from the operation being performed is deliberate and facilitates careful consideration of all possible initial conditions from which incidents may occur.

15A.3.3.2 Anticipated (Expected) Operational Transients

To select anticipated operational transients, eight nuclear system parameter variations are considered as potential initiating causes of threats to the fuel and the reactor coolant pressure boundary. The parameter variations are as follows:

- *(1) Nuclear system pressure increase*
- *(2) Reactor vessel water (moderator) temperature decrease*
- *(3) Positive reactivity insertion*
- *(4) Reactor vessel coolant inventory decrease*
- *(5) Reactor core coolant flow decrease*
- *(6) Reactor core coolant flow increase*
- *(7) Core coolant temperature increase*
- *(8) Excess of coolant inventory*

These parameter variations, if uncontrolled, could result in damage to the reactor fuel or reactor coolant pressure boundary, or both. A nuclear system pressure increase threatens to rupture the reactor coolant pressure boundary from internal pressure. A pressure increase also collapses voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage as a result of overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in coolant flow through the core threaten to overheat the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, resulting in an increased fission rate. A core coolant temperature increase threatens the integrity of the fuel; such a variation could be the result of a heat exchanger malfunction during operation in the shutdown cooling mode. An excess of coolant inventory could be the result of malfunctioning water level control equipment; such a malfunction can result in a turbine trip, which causes an increase in nuclear system pressure and an increased fission rate.

The eight parameter variations listed above include all effects within the nuclear system caused by anticipated operational transients that threaten the integrity of the reactor fuel or reactor coolant pressure boundary. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Anticipated operational transients are defined as transients resulting from single equipment failures or single operator errors that can be reasonably expected (moderate probability of occurrence once per day to once in 20 years) during any mode of plant operation. Examples of single operational failures or operator errors in this range of probability are:

- *(1) Opening or closing any single valve (a check valve is not assumed to close against normal flow)*
- *(2) Starting or stopping any single component*
- *(3) Malfunction or maloperation of any single control device*
- *(4) Any single electrical failure*
- *(5) Any single operator error*

An operator error is defined as an active deviation from nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single reasonably expected erroneous decision. The set of actions is limited as follows:

- *(1) Those actions that could be performed by only one person.*
- *(2) Those actions that would have constituted a correct procedure had the initial decision been correct.*
- *(3) Those actions that are subsequent to the initial operator error and that affect the designed operation of the plant, but are not necessarily directly related to the operator error.*

Examples of single operator errors are as follows:

- *(1) An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.*
- *(2) The selection and complete withdrawal of a single control rod out of sequence.*
- *(3) An incorrect calibration of an average power range monitor.*
- *(4) Manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indication.*

The five types of a single operator error or a single equipment malfunction are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

15A.3.3.3 Abnormal (Unexpected) Operational Transients

To select abnormal operational transients, eight nuclear system parameter variations are considered as potential initiating causes of gross core-wide fuel failures and threats of the reactor coolant pressure boundary. The parameter variations are as follows:

- *(1) Nuclear system pressure increase*
- *(2) Reactor vessel water (moderator) temperature decrease*
- *(3) Positive reactivity insertion*
- *(4) Reactor vessel coolant inventory decrease*
- *(5) Reactor core coolant flow decrease*
- *(6) Reactor core coolant flow increase*

(7) Core coolant temperature increase
(8) Excess of coolant inventory

These parameter variations, if uncontrolled, could result in gross core-wide reactor fuel failure or damage to the reactor coolant pressure boundary, or both.

The eight parameter variations listed above include all effects within the nuclear system caused by abnormal operational transients that threaten gross core-wide reactor fuel integrity or seriously affect reactor coolant pressure boundary. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Abnormal operational transients are defined as incidents resulting from single or multiple equipment failures and/or single or multiple operator errors that are not reasonably expected (less than one event in 20 years to one in 100 years) during any mode of plant operation.

Examples of single or multiple operational failures and/or single or multiple operator errors are:

(1) Catastrophic failure of major power generation equipment components
(2) Multiple electrical failures
(3) Multiple operator errors
(4) Combinations of equipment failure and an operator error

Operator error is defined as an active deviation from nuclear plant standard operating practices. A multiple operator error is the set of actions that is a direct consequence of several unexpected erroneous decisions.

Examples of multiple operator errors are as follows:

(1) Inadvertent loading and operating a fuel assembly in an improper position.
(2) The movement of a control rod during refueling operations.

The various types of single errors and/or single malfunctions are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the various radioactive material barriers.

15A.3.3.4 Accidents

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and that are not expected during plant operations. These are plant events, equipment failures, combinations of initial conditions which are of extremely low probability (once in 100 years to once in 10,000 years). The postulated accident types considered are as follows:

(1) Mechanical failure of a single component leading to the release of radioactive material from one or more barriers. The components referred to here are not

those that act as radioactive material barriers. Example of mechanical failure is breakage of the coupling between a control rod drive and the control rod.

(2) Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor coolant pressure boundary. This kind of accident is considered only under conditions in which the nuclear system is pressurized.

For purposes of analysis, accidents are categorized as those events that result in releasing radioactive material:

(1) From the fuel with the reactor coolant pressure boundary and reactor building initially intact. (Event 40)
(2) Directly to the containment. (Event 42)
(3) Directly to the reactor or turbine buildings with the containment initially intact. (Events 40, 43, 44, 45, 50)
(4) Directly to the reactor building with the containment not intact. (Events 41, 50)
(5) Directly to the spent fuel containing facilities. (Events 41, 50)
(6) Directly to the turbine building (Events 46, 47)
(7) Directly to the environs (Events 48, 49)

The effects of various accident types are investigated, with consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than day other accident considered under the same general accident assumptions are designated design basis accidents.

15A.3.3.5 Additional Special Plant Capability Events

A number of additional events are evaluated to demonstrate plant capabilities relative to special arbitrary nuclear safety criteria. These special events involve extremely low- probability occurrence situations. As an example, the adequacy of the redundant reactivity control system is demonstrated by evaluating the special event: "reactor shutdown without control rods."

Another similar example, the capability to perform a safe shutdown from outside the main control room is demonstrated by evaluating the special event "reactor shutdown from outside the main control room."

15A.3.4 Applicability of Events to Operating States

The first step in performing an operational analysis for a given "incident" (transient, accident, or special event) is to determine in which operating states the incident can occur. An incident is considered applicable within an operating state if the incident can be initiated from the physical conditions that characterize the operating state. Applicability of the "planned operations" to the operating states follows from the definitions of planned operations. A planned operation is considered applicable within an operating state if the planned operation can be conducted when the reactor exists under the physical conditions defining the operating state.

15A.3.5 Rules for Event Analysis

The following functional rules are followed in performing SACF, operational and design basis analyses for the various plant events:

(1) An action, system, or limit shall be considered essential only if it is essential to avoiding an unacceptable result or satisfying the nuclear safety operational criteria.

(2) The full range of initial conditions (as defined in paragraph 15A.3.5.(3) shall be considered for each event analyzed so that all essential protection sequences are identified. Consideration is not limited to "worst cases" because lesser cases sometimes may require more restrictive actions or systems different from the "worst cases."

(3) The initial conditions of transients, accidents, and additional plant capability events shall be limited to conditions that would exist during planned operations in the applicable operating state.

(4) For planned operations, consideration shall be made only for actions, limits, and systems essential to avoiding the unacceptable results during operation in that state (as opposed to transients, accidents, and additional plant capability events, which are followed through to completion). Planned operations are treated differently from other events because the transfer from one state to another during planned operations is deliberate. For events other than planned operations, the transfer from one state to another may be unavoidable.

(5) Limits shall be derived only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be included in the requirement for the operability of the system. Limits on frequently monitored process parameters are called "envelope limits," and limits on parameters associated with the operability of a safety system are called "operability limits." Systems associated with the control of the envelope parameters are considered nonessential if it is possible to place the plant in a safe condition without using the system in question.

(6) For transients, accidents and additional plant capability events, consideration shall be made for the entire duration of the event and aftermath until some planned operation is resumed. Planned operation is considered resumed when the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations.

(7) Credit for operator action shall be taken on a case-by-case basis depending on the conditions that would exist at the time operator action would be required. Because transients, accidents, and additional plant capability events are considered through the entire duration of the event until planned operation is resumed, manual operation of certain systems is sometimes required following the more rapid or automatic portions of the event. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions.

(8) For transients, accidents, and additional plant capability events, only those actions, limits, and systems shall be considered essential for which there arises a unique requirement as a result of the event. For instance, if a system that was operating prior to the event (during planned operation) is to be employed in the same manner following the event and if the event did not affect the operation of

the system, then the system would not appear on the protection sequence diagram.

(9) The operational analyses shall identify all the support or auxiliary systems essential to the functioning of the front-line safety systems. Safety system auxiliaries whose failure results in safe failure of the front-line safety systems shall be considered nonessential.

(10) A system or action that plays a unique role in the response to a transient, accident, or additional plant capability event shall be considered essential unless the effects of the system or action are not included in the detailed analysis of the event.

15A.3.6 Steps in an Operational Analysis

All information needed to perform an operational analysis for each plant event has been presented (Figure 15A.2-2). The procedure followed in performing an operational analysis for a given event (selected according to the event selection criteria) is as follows:

(1) Determine the BWR operating states in which the event is applicable.

(2) Identify all the essential protection sequences (safety actions and front-line safety systems) for the event in each applicable operating state.

(3) Identify all the safety system auxiliaries essential to the functioning of the front-line safety systems.

The above three steps are performed in later sections of this appendix.

To derive the operational requirements and technical specifications for the individual components of a system included in any essential protection sequence, the following steps are taken:

(1) Identify all the essential actions within the system (intrasystem actions) necessary for the system to function to the degree necessary to avoid the unacceptable results.

(2) Identify the minimum hardware conditions necessary for the system to accomplish the minimum intra-system actions.

(3) If the single-failure criterion applies, identity the additional hardware conditions necessary to achieve the plant safety actions (scram, pressure relief, isolation, cooling, etc.) in spite of single failures. This step gives the nuclear safety operational requirements for the plant components so identified.

(4) Identify surveillance requirements and allowable repair times for the essential plant hardware (Subsection 15A.5.2).

(5) Simplify the operational requirements determined in steps (3) and (4) so that technical specifications may be obtained that encompass the true operational requirements and are easily used by plant operations and management personnel.

15A.4 DISPLAY OF OPERATIONAL ANALYSIS RESULTS

15A.4.1 General

To fully identify and establish the requirements, restrictions, and limitations that must be observed during plant operation, plant systems and components must be related to the needs for their actions in satisfying the nuclear safety operational criteria. This appendix displays these relationships in a series of block diagrams.

First, a table like Table 15A.3-1 will be supplied indicating in which operating states each event is applicable. Then, for each event, a block diagram is presented showing the, conditions and systems required to achieve each essential safety action. The block diagrams show only those systems necessary to provide the safety actions such that the nuclear safety operational and design basis criteria are satisfied. The total plant capability to provide a safety action is generally not shown, only the minimum capability essential to satisfying the operational criteria. It is very important to understand that only enough protective equipment is cited in the diagram to provide the necessary action. Many events can utilize many more paths to success then are shown. These operational analyses involve the minimum equipment needed to prevent or avert an unacceptable result. Thus, the diagrams depict essential protection sequences for each event with the least amount of protective equipment needed. Once all of these protection sequences are identified in block diagram form, system requirements are derived by considering all events in which the particular system is employed. The analysis considers the following conceptual aspects:

(1) The BWR operating state.
(2) Types of operations or events that are possible within the operating state.
(3) Relationships of certain safety actions to the unacceptable results and to specific types of operations and events.
(4) Relationships of certain systems to safety actions and to specific types of operations and events.
(5) Supporting or auxiliary systems essential to the operation of the front-line safety systems.
(6) Functional redundancy, the single-failure criterion applied at the safety action level. This is, in effect, a qualitative/ system level/FMEA-type analysis.

Each block in the sequence diagrams represents a finding of essentiality for the safety action, system, or limit under consideration. Essentiality in this context means that the safety action, system, or limit is needed to satisfy the nuclear safety operational criteria. Essentiality is determined through an analysis in which the safety action, system, or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system, or limit, then the safety action, system, or limit is not essential, and no operational nuclear safety requirement would be indicated. When disregarding a safety action, system, or limit results in violating one or more nuclear safety operational criteria, the safety action, system, or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable results.

15A.4.2 Protection Sequence and Safety System Auxiliary Diagrams

Block diagrams illustrate essential protection sequences for each event requiring unique safety actions. These protection sequence diagrams show only the required front-line safety systems. The format and conventions used for these diagrams are shown in Figure 15A.4-1.

The auxiliary systems essential to the correct functioning of front-line safety systems are shown on safety system auxiliary diagrams. The format used for these diagrams is shown in Figure 15A.4-2.

The diagram indicates that auxiliary systems A, B, and C are required for proper operation of front-line safety system X.

Total plant requirements for an auxiliary system or the relationships of a particular auxiliary system to all other safety systems (frontline and auxiliary) within an operating state are shown on the commonality of auxiliary diagrams. The format used for these diagrams is shown in Figure 15A.4-3.

The convention employed in Figure 15A.4-3 indicates that auxiliary system A is required:

(1) *to be single-failure proof relative to system q in State A-events X, Y; State B-events X, Y; State C-events X, Y, Z; State D-events X, Y, Z.*
(2) *to be single-failure proof relative to the parallel combination of systems a and b in State A-events U, V, W; State B-events V, W; State C-events U, V, W, X; State D-events U, V, W, X.*
(3) *to be single-failure proof relative to the parallel combination of system ! and ± system e in series with the parallel combination of systems u and C1 in State C-events Y, W; State D-events Y, W, Z. As noted, system e is part of the combination but does not require auxiliary system A for its proper operation.*
(4) *for system W in State B-events Q, R; State D- events Q, R, S.*

With these three types of diagrams, it is possible to determine for each system the detailed functional requirements and conditions to be observed regarding system hardware in each operating state. The detailed conditions to be observed regarding system hardware include such nuclear safety operational requirements as test frequencies and the number of components that must be operable.

15A.5 BASES FOR SELECTING SURVEILLANCE TEST FREQUENCIES

15A.5.1 Normal Surveillance Test Frequencies

After the essential nuclear safety systems and engineered safeguards have been identified by applying the nuclear safety operational criteria, surveillance requirements are selected for these systems. In this selection process, the various systems are considered in terms of relative availability, test capability, plant conditions necessary for testing, and engineering experience with the system type. The surveillance test frequency selected represents the application of engineering judgment integrating all of these considerations.

15A.5.2 Allowable Repair Times

Allowable repair times are selected by computation using availability analysis methods (Reference 15A-1) for redundant standby systems. The resulting maximum average allowable repair times assure that a system's long-term availability, including allowance for repair, is not reduced below the theoretical availability that would be achieved if repairs could be made in zero time.

15A.5.3 Repair Time Rule

A safety system can be repaired while the reactor is in operation if the repair time is equal to or less than the maximum allowable average repair time. If repair is not complete when the allowable repair time expires, the plant must be placed in its safest mode (with respect to the protection lost).

To maintain the validity of the assumptions used to establish the above repair time rule, the following restrictions must be observed:

(1) The allowable repair time should only be used as needed to restore failed equipment to operation, not for routine maintenance.

Using this time should be an event as rare as failure of the equipment itself. Routine maintenance should be scheduled when the equipment is not needed.

(2) When a failure is discovered by test, all the redundant components should be tested to establish that they are good at the beginning of the repair time for the failed component and do not suffer from the same failure mode discovered in the failed component. If there are multiple failures of the same mode, the repair time allowance does not apply and the plant must be placed in a condition in which the actions of the safety system are not essential to avoiding the unacceptable safety results.

(3) At the conclusion of the repair, the repaired component must be retested and placed in service. The redundant components must also be retested, not only to validate the assumptions, but to assure that the repair did not inadvertently invalidate a good component.

(4) Once the need for repair of a failed component is discovered, repairs should proceed as quickly as possible consistent with good craftsmanship.

Alternatively, if a system is expected to be out of repair for an extended time, the availability of the remaining systems can be maintained at the prefailure level by testing them more often. This technique is fully developed in Reference 15A-1.

15A.6 OPERATIONAL ANALYSES

Results of the operational analyses are discussed in the following subsections and displayed on Figures 15A.6-1, 15A.6-2, 15A.6-3, 15A.6-4 and 15A.6-5. Tables 15A.6-1 through 15A.6-5 indicate the BWR operating states in which each of the approximately 50 events is applicable.

15A.6.1 Safety System Auxiliaries

Figures 15A.6-1 and 15A.6-2 show the safety system auxiliaries essential to the functioning of each front-line safety system. Commonality of auxiliary diagrams are shown in Figures 15A.6-54, 15A.6-55, 15A.6-56, 15A.6-57, 15A.6-58 and 15A.6-59.

15A.6.2 Planned (Normal) Operations

15A.6.2.1 General

Requirements for the planned operations normally involve limits (L) on certain key process variables and restrictions (R) on certain plant equipment. The control block diagrams for each operating state (Figures 15A.6-3, 15A.6-4, 15A.6-5 and 15A.6-6) show only those controls necessary to avoid unacceptable safety results 1-1 through 1-4. Refer to Table 15A.2-6 for unacceptable results criteria.

Following is a description of the planned operations (Events 1 through 6), as they pertain to each of the four operating states. The description of each operating state contains a definition of that state, a list of the planned operations that apply to that state, and a list of the safety actions that are required to avoid the unacceptable safety results.

15A.6.2.2 Event Definitions

Event 1 - Refueling Outage

Refueling outage includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is made critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

(1) Planned, physical movement of core components (fuel, control rods, etc.)
(2) Refueling test operations (except criticality and shutdown margin tests)
(3) Planned maintenance
(4) Required inspection

Event 2 - Achieving Criticality

Achieving criticality includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

Event 3 - Reactor Heatup

Heatup begins where achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine generator.

<u>Event 4 - Power Operation - Electric generation</u>

Power operation begins where heatup ends and continued plant operation at power levels in excess of heatup power or steady state operation. It also includes plant maneuvers such as:

(1) Daily electrical load reduction and recoveries
(2) Electrical grid frequency control adjustment
(3) Control rod/reactor fuel/core management movements
(4) Power generation surveillance testing involving:

  a. Turbine stop valve closing
  b. Turbine control valve adjustments
  c. MSLIV exercising

<u>Event 5 - Achieving Reactor Shutdown</u>

Achieving shutdown begins where the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) after power operation.

<u>Event 6 - Reactor Cooldown</u>

Cooldown begins where achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure.

<u>15A.6.2.3 Required Safety Actions/Related Unacceptable Results</u>

The following paragraphs describe the safety actions for planned operations. Each description includes a selection of the operating states that apply to the safety action, the plant system affected by limits or restrictions, and the unacceptable result that is avoided. The four operating states are defined in Table 15A.3-1. The unacceptable results criteria are tabulated in Table 15A.2-6.

<u>15A.6.2.3.1 Radioactive Material Release Control</u>

Radioactive materials may be released to the environs in any operating state; therefore, radioactive material release control is required in all operating states. Because of the significance of preventing excessive release of radioactive materials to the environs, this is the only safety action for which monitoring systems are explicitly shown. The offgas vent radiation monitoring system provides indication for gaseous release through the main vent. Gaseous releases through other vents are monitored by the ventilation monitoring system. The process liquid radiation monitors are not required, because all liquid wastes are monitored by batch sampling before a controlled release. Limits are expressed on the offgas vent system, liquid radwaste system, and solid radwaste system so that the planned releases of radioactive materials comply with the limits given in 10CFR20, 10CFR50, and 10C.FR71 (related unacceptable safety result 1-1).

15A.6.2.3.2 Core Coolant Flow Rate Control

In State D, when above approximately 10% NB rated power, the core coolant flow rate must be maintained above certain minimums (i.e., limited) to maintain the integrity of the fuel cladding (1-2) and assure the validity of the plant safety analysis (1-4).

15A.6.2.3.3 Core Power Level Control

The plant safety analyses of accidental positive reactivity additions have assumed as an initial condition that the neutron source level is above a specified minimum. Because a significant positive reactivity addition can only occur when the reactor is less than one rod subcritical, the assumed minimum source level need be observed only in States B and D. The minimum source level assumed in the analyses has been related to the counts/sec readings on the source range monitors (SRM); thus, this minimum power level limit on the fuel is expressed as a required SRM count level. Observing the limit assures validity of the plant safety analysis (1-4). Maximum core power limits are also expressed for operating States B and D to maintain fuel integrity (1-2) and remain below the maximum power levels assumed in the plant safety analysis (1-4).

15A.6.2.3.4 Core Neutron Flux Distribution Control

Core neutron flux distribution must be limited in State D, otherwise core power peaking could result in fuel failure (1-2). Additional limits are expressed in this state, because the core neutron flux distribution must be maintained within the envelope of conditions considered by plant safety analysis (1-4).

15A.6.2.3.5 Reactor Vessel Water Level Control

In any operating state, the reactor vessel water level could, unless controlled, drop to a level that will not provide adequate core cooling; therefore, reactor vessel water level control applies to all operating states. Observation of the reactor vessel water level limits protects against fuel failure (1-2) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.6 Reactor Vessel Pressure Control

Reactor vessel pressure control is not needed in States A and B because vessel pressure cannot be increased above atmospheric pressure. In State C, a limit is expressed on the reactor vessel to assure that it is not hydrostatically tested until the temperature is above the NDT temperature plus 60oF; this prevents excessive stress (1-3). Also, in States C and D a limit is expressed on the residual heat removal system to assure that it is not operated in the shutdown cooling mode when the reactor vessel pressure is greater than approximately 150 psig; this prevents excessive stress (1-3). In States C and D, a limit on the reactor vessel pressure is necessitated by the plant safety analysis (1-4).

15A.6.2.3.7 Nuclear System Temperature Control

In operating States A, C, and D, a limit is expressed on the reactor vessel to prevent the reactor vessel head bolting studs from being in tension when the temperature is less than 70oF to avoid excessive stress (1-3) on the reactor vessel flange. This limit does not apply in States A and B because the head will not be bolted in place during criticality tests or during refueling. In all *operating states, a limit is expressed on the reactor vessel to prevent an excessive rate of change of the reactor vessel temperature to avoid excessive stress (1-3). In States C and D, where it is planned operation to use the feedwater system, a limit is placed on the reactor fuel so that the feedwater temperature is maintained within the envelope of conditions considered by the plant safety analysis (1-4). For State D, a limit is observed on the temperature difference between the recirculation system and the reactor vessel to prevent the starting of the recirculation pumps. This operating restriction and limit prevents excessive stress in the reactor vessel (1-3).*

15A.6.2.3.8 Nuclear System Water Quality Control

In all operating states, water of improper chemical quality could produce excessive stress as a result of chemical corrosion (1-3). Therefore, a limit is placed on reactor coolant chemical quality in all operating states. For all operating states where the nuclear system can be pressurized (States C and D), and additional limit on reactor coolant activity assures the validity of the analysis of the main steamline break accident (1-4).

15A.6.2.3.9 Nuclear System Leakage Control

Because excessive nuclear system leakage could occur only while the reactor vessel is pressurized, limits are applied only to the reactor vessel in States C and D. Observing these limits prevents vessel damage due to excessive stress (1-3) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.10 Core Reactivity Control

In State A during refueling outage, a limit on core loading (fuel) to assure that core reactivity is maintained within the envelope of conditions considered by the plant safety analysis (1-4). In all states, limits are imposed on the control rod drive system to assure adequate control of core reactivity so that core reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4).

15A.6.2.3.11 Control Rod Worth Control

Any time the reactor is not shut down and is generating less than 30% power (State D), a limit is imposed on the control rod pattern to assure that control rod worth is maintained within the envelope of conditions considered by the analysis of the control rod drop accident (1-4).

15A.6.2.3.12 Refueling Restriction

By definition, planned operation event 1 (refueling outage) applies only to State A. Observing the restrictions on the reactor fuel and on the operation of the control rod drive system within the specified limit maintains plant conditions within the envelope considered by the plant safety analysis (1-4).

15A.6.2.3.13 Containment & Reactor Building Pressure and Temperature Control

In States C and D, limits are imposed on the containment and the suppression pool storage to maintain temperature and pressure within the envelope considered by plant safety analysis (1-4). These limits assure an environment in which instruments and equipment can operate *correctly within the containment. Limits on the pressure suppression pool apply to the water temperature and water level to assure that it has the capability of absorbing the energy discharged during a safety/relief valve blowdown.*

15A.6.2.3.14 Stored Fuel Shielding, Cooling, and Reactivity Control

Because both new and spent fuel will be stored during all operating states, stored fuel shielding, cooling, and reactivity control apply to all operating states. Limits are imposed on the spent fuel pool storage positions, water level, fuel handling procedures, and water temperature. Observing the limits on fuel storage positions assures that spent fuel reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4). Observing the limits on water level assures shielding in order to maintain conditions within the envelope of conditions considered by the plant safety analysis (1-4) and provides the fuel cooling necessary to avoid fuel damage (1-2). Observing the limit on water temperature avoids excessive fuel pool stress (1-3). A limit is imposed on the new fuel storage arrangement to assure that the fuel storage geometry is maintained within the envelope of reactivity conditions considered by the plant safety analysis (1-4).

15A.6.2.4 Operational Safety Evaluations

State A

In State A the reactor is in a shutdown condition, the vessel head is off, and the vessel is at atmospheric pressure. The applicable events for planned operations are refueling outage, achieving criticality, and cooldown (Events 1, 2, and 6, respectively).

Figure 15A.6-3 shows the necessary safety actions for planned operations, the corresponding plant systems, and the event for which these actions are necessary. As indicated in the diagram the required safety actions are as follows:

Safety Action
Radioactive material release control
Reactor vessel water level control
Nuclear system temperature control
Nuclear system water quality control
Core reactivity control
Refueling restrictions
Stored fuel shielding, cooling, and reactivity control

State B

In State B the reactor vessel head is off, the reactor is not shutdown, and the vessel is at atmospheric pressure. Applicable planned operations are achieving criticality and achieving shutdown (Events 2 and 5, respectively).

Figure 15A.6-4 relates the necessary safety actions for planned operations, the plant systems, and the event for which the safety actions are necessary. The required safety actions for planned operation in State B are as follows:

Safety Action
Radioactive material release control *Core power level control* *Reactor vessel water level control* *Nuclear system temperature control* *Nuclear system water quality control* *Core reactivity control* *Rod worth control* *Stored fuel shielding, cooling, and reactivity control*

State C

In State C the reactor vessel head is on and the reactor is shutdown. Applicable planned operations are achieving criticality and cooldown (Events 2 and 6, respectively).

Sequence diagrams relating safety actions for planned operations, plant systems, and applicable events are shown in Figure 15A.6-5. The required safety actions for planned operation in State C are as follows:

Safety Action
Radioactive material release control *Reactor vessel water level control* *Reactor vessel pressure control* *Nuclear system temperature control* *Nuclear system water quality control* *Nuclear system leakage control* *Core reactivity control* *Reactor building pressure and temperature control* *Stored fuel shielding, cooling, and reactivity control*

State D

In State D the reactor vessel head is on and the reactor is not shutdown. Applicable planned operations are achieving criticality, heatup, power operation and achieving shutdown (Events 2, 3, 4, and 5, respectively).

Figure 15A.6-6 relates safety actions for planned operations, corresponding plant systems, and events for which the safety actions are necessary. The required safety actions for planned operation in State D are as follows:

Safety Action
Radioactive material release control *Core coolant flow rate control* *Core power level control* *Core neutron flux distribution control* *Reactor vessel water level control* *Reactor vessel pressure control* *Nuclear system temperature control* *Nuclear system water quality control* *Nuclear system leakage control* *Core reactivity control* *Rod worth control* *Containment and reactor/auxiliary building pressure and temperature control* *Stored fuel shielding, cooling, and reactivity control*

15A.6.3 Anticipated (Expected) Operational Transients

15A.6.3.1 General

The safety requirements and protection sequences for anticipated operational transients are described in the following paragraphs for Events 7 through 29. The protection sequence block diagrams show the sequence of front-line safety systems. (Refer to Figure 15A.6-7 through 15A.6-29.) The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-54, 15A.6-55, 15A.6-56, 15A.6-57, 15A.6-58 and 15A.6-59).

15A.6.3.2 Required Safety Actions/Related Unacceptable Result

The following list relates the safety actions for anticipated operational transients that mitigate or prevent the unacceptable safety results. Refer to Table 15A.2-7 for the unacceptable results criteria.

Safety Action	*Related Unacceptable Result Criteria*	*Reason Action Required*
Scram and/or RPT *Pressure relief*	*2-2* *2-3* *2-3*	*To prevent fuel damage and to limit nuclear system pressure rise.* *To prevent excessive nuclear system pressure rise.*
Core and Containment cooling	*2-2*	*To prevent fuel and containment damage in the event that normal cooling is interrupted.*

Reactor vessel isolation	*2-2*	*To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.*
Restore ac power	*2-2*	*To prevent fuel damage by restoring ac power to systems essential to other safety actions.*
Prohibit rod motion	*2-2*	*To prevent exceeding fuel limits during transients.*
Containment isolation	*2-4*	*To minimize radiological effects.*

15A.6.3.3 Event Definitions & Operational Safety Evaluations

Event 7 - Manual & Inadvertent SCRAM

The deliberate manual or inadvertent automatic SCRAM due to single operator error is an event which can occur under any operating conditions. Although assumed to occur here for examination purpose, multi-operator error or action is necessary to initiate such an event.

While all the safety criteria apply, no unique safety actions are required to control the planned operation-like event after effects of the subject initiation actions. In all operating states, the safety criteria are therefore met through the basis design of the plant systems. Figure 15A.6-7 identifies the protection sequences for this event.

Event 8 - Loss-of-Plant Instrument Air

Loss of all plant instrument air system requires a manual reactor shutdown and causes the closure of isolation valves. Although these actions occur, they are not a requirement to prevent unacceptable results in themselves. Multi-equipment failures would be necessary in order to cause the deterioration of the subject system to the point that the components supplied with instrument air would cease to operate "normally" and/or "fail-safe." The resulting actions are identical to the Event 14 described later.

Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating States C and D. In operating States A and B, the main steam lines are continuously isolated.

Isolation of all main steam lines is most severe and rapid in operating State D during power operation.

Figure 15A.6-8 shows how scram is accomplished by annual actuation or by main steam line isolation through the actions of the reactor protection system and the control rod drive system. The nuclear system pressure relief system provides pressure relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either high-pressure core cooling system supplies water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.

Adequate reserve air supplies are maintained exclusively for the continual operation of the safety/relief valves until reactor shutdown is accomplished.

Event 9 - Inadvertent HPCI Pump (or any NSSS Pump) Start (Moderator Temperature Decrease)

An inadvertent pump start (temperature decrease) is defined as an unintentional start of any nuclear system pump that adds sufficient cold water to the reactor coolant inventory to cause a measurable decrease in moderator temperature. This event is considered in all operating states because it can potentially occur under any operating condition. Since the HPCI pump operates over nearly the entire range of the operating states and delivers the greatest amount of cold water to the vessel, the following analysis will describe its inadvertent operation rather than other NSSS pumps (e.g., RCICS, RHRS, CSCS).

While all the safety criteria apply, no unique safety actions are required to control the adverse effects of such a pump start (i.e., pressure increase and temperature decrease in States A and C). In these operating states, the safety criteria are met through the basic design of the plant systems, and no safety action is specified. In States B and D, where the reactor is not shutdown, the operator or the plant normal control system can control any power changes in the normal manner of power control.

Figure 15A-6-9 illustrates the protection sequence for the subject event. Single failures to the normal plant control system pressure regulator or the feedwater controller systems will result in further protection sequences. These are shown in Events 22 and 23. The single failure (SF) aspects of their protection sequences will, of course, not be required.

Event 10 - Startup of Idle Recirculation Pump

The cold-loop startup of an idle recirculation pump can occur in any state and is most severe and rapid for those operating states in which the reactor may be critical (States B and D). When the transient occurs in the range of 10 to 60% power operation, no safety action response is required. Reactor power is normally limited to approximately 60% design power because of core flow limitations while operability with one recirculation loop working. Above about 60% power, a high neutron flux scram is initiated. Should the event occur when the reactor is in operating State D but not at power operation, but critical (5% < power < 10%), the resulting transient may produce a high level neutron flux scram of the intermediate range monitors (IRM). No safety actions are required in State B since the power would be less than 5%.

As shown in Figure 15A.6-10, the scram action is accomplished through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems. At power operation (10 to 60%) the high level IRM scram is not initiated, because the core flux monitoring has been shifted to the average power range monitors (APRM).

Event 11 - Recirculation Flow Control Failure (Increasing Flow)

A recirculation flow control failure causing increased flow is applicable in States C and D. In State D, the accompanying increase in power level is accommodated through a reactor scram. As shown in Figure 15A.6-11, the scram safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems.

<u>Event 12 - Recirculation Flow Control Failure (Decreasing Flow)</u>

This recirculation flow control malfunction causes a decrease in core coolant flow. This event is not applicable to States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use.

The number and type of flow controller failure modes determine the protection sequence for the event. For M/G set flow control systems, failures of one or the master flow controller will result in a transient equivalent to one or two recirculation pump trips, respectively it is shown on Figure 15A.6.-12.

<u>Event 13 - Trip of One or Both Recirculation Pumps</u>

The trip of one recirculation pump produces a milder transient than does the simultaneous trip of two recirculation pumps.

The transient resulting from this two-loop trip is not severe enough to require any unique safety action. The transient is compensated for by the inherent nuclear stability of the reactor. This event is not applicable in States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use. The trip could occur in States C and D; however, the reactor can accommodate the transient with no unique safety action requirement. Figure 15A.6-13 provides the protection sequence for the event for one or both pump trip actuations.

In fact, this event constitutes all acceptable operational technique to reduce or minimize the effects of other event conditions. To this end, an engineered recirculation pump trip capability is included in the plant operational design to reduce pressure and thermohydraulic transient effects. Operating States C and D are involved in this event.

Tripping a single recirculation pump requires no protection system operation.

A two pump trip results in a high water level trip of the main turbine which further causes a stop valve closure and its subsequent SCRAM actuation. Main steam line isolation soon occurs and is followed by RCIC/HPCI systems initiation on low water level. Relief valve actuation will follow.

<u>Event 14 - Isolation of One or All Main Steam Lines</u>

Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating States C and D. In operating States A and B, the main steam lines are continuously isolated.

Isolation of all main steam lines is most severe and rapid in operating State D during power operation.

Figure 15A.6-14a shows how scram is accomplished by main steam line isolation through the actions of the reactor protection system and the control rod drive system. The nuclear system pressure relief system provides relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either high-pressure core cooling system supplies *water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.*

Isolation of one main steam line causes a significant transient only in State D during high power operation. Scram is the only unique action required to avoid fuel damage and nuclear system overpressure. Because the feedwater system and main condenser remain in operation following the event, no unique requirement arises for core cooling.

As shown in Figure 15A.6-14b, the scram safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems.

Event 15 - Inadvertent Opening of the Safety/Relief Valve

The inadvertent opening of a safety/relief valve is possible in any operating state. The protection sequences are shown in Figure 16A.6-15. In States A, B, and C, the water level cannot be lowered far enough to threaten fuel damage; therefore, no safety actions are required.
In State D, there is a slight decrease in reactor pressure following the event. The pressure regulator closes the main turbine control valves enough to stabilize pressure at a level slightly below the initial value. There are no unique safety system requirements for this event.

If the event occurs when the feedwater system is not active in State D, a loss in the coolant inventory results in a reactor vessel isolation. The low water level signal initiates reactor vessel isolation. The nuclear system pressure relief system provides pressure relief.

Core cooling is accomplished by the RCIC/HPCI system which is automatically initiated by the incident detection circuitry (IDC). The automatic depressurization system (ADS) or the manual relief valve system remain as the backup depressurization system if needed. After the vessel has depressurized, long term core cooling is accomplished by the LPCI, or CSCS, which are initiated on low water level by the IDC system or are manually operated. Containment/ suppression pool cooling is manually initiated.

Event 16 - Control Rod Withdrawal Error (During Refueling & Startup Operation)

Because a control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition, it must be considered in all operating states. For this specific event situation, only State A and B apply.

Refueling

No unique safety action is required in operating State A for the withdrawal of one control rod because the core is more than one control rod subcritical. Withdrawal of more than one control rod is precluded by the protection sequence shown in Figure 15A.6-16.

During core alterations, the mode switch is normally in the REFUEL position, which allows the refueling equipment to be positioned over the core and also inhibits control rod withdrawal. This transient, therefore, applies only to operating State A. No safety action is required because the total worth (positive reactivity) of one fuel assembly or control rod is not adequate to cause criticality. Moreover, mechanical design of the control rod assembly prevents physical removal without removing the adjacent fuel assemblies.

Startup

During low power operation (State B), the neutron monitoring system via the RPS will initiate SCRAM if necessary. Refer to Figure 15A.6-16.

Event 17 - Control Rod Withdrawal Error (During Power Operation)

Because a control rod withdrawal occur resulting in an increase of positive reactivity can occur under any operating condition, it must be considered in all operating states. For this specific event situation, only States C and D apply.

During power operation (Power Range) (State D), a number of plant protective devices of various designs prohibit the control rod motion before critical levels are reached. Refer to Figure 15A.6-17. While in State C no protective action is needed.

Systems in the power range (0 to 100% NBR) prevent the selection of an out-of-sequenced rod movement by use of the RWM (Banked Position or Notch Group). In addition, the movement of the rod is monitored and limited within acceptable intervals either by neutronic effects or actual rod motion, (notch counting). The RBM provides movement surveillance. Of course, beyond these rod motion control limits are the fuel/core SCRAM protection systems. While in State C no protective action is needed.

Event 18 - Loss of Shutdown Cooling

The loss of RHRS-shutdown cooling can occur only during the low pressure portion of a normal reactor shutdown and cooldown.

As shown in Figure 15A.6-18, for most single failures that could result in primary loss of shutdown cooling capabilities, no unique safety actions are required; in these cases, shutdown cooling is simply reestablished using redundant shutdown cooling equipment. In the cases where the RHRS-shutdown cooling suction line becomes inoperative, a unique arrangement for cooling arises. In States A and B, in which the reactor vessel head is off, the LPCI can be used to maintain reactor vessel water level. In States C and D, in which the reactor vessel head is on and the system can be pressurized, the automatic depressurization system (ADS) or manual operation of relief valves in conjunction with any of the ECCS and the RHRS suppression pool cooling mode (both manually operated) can be used to maintain water level and remove decay heat. Containment/Suppression pool cooling is actuated. Core and containment decay heat are removed by the RHRS containment cooling system.

Event 19 - RHF Shutdown Cooling Malfunction (Moderator Temperature Decrease)

An RHR shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered in States C and D if nuclear system pressure is too high to permit operation of the shutdown cooling (RHRS). Refer to Figure 15A.6-19. No unique safety actions are required to avoid the unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers.

In States B and D, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

Event 20 - Loss of All Feedwater Flow

A loss of feedwater results in a net decrease in the coolant inventory available for core cooling. A loss of feedwater flow can occur in States C and D. Appropriate responses to this transient include a reactor scram on low water level and maintenance of reactor vessel water level.

As shown in Figure 15A.6-20, the reactor protection and control rod drive systems effect a scram on low water level. The containment and reactor vessel isolation control system and the main steam line isolation valves act to isolate the reactor vessel. After the main steam line isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. Initial core cooling is necessary to restore and maintain water level. Either the RCIC or HPCI system can maintain adequate water level. For long term shutdown and extended core coolings, containment/suppression pool cooling systems are manually initiated.

The requirements for operating State C is the same as for State D except that the scram action is not required in State C.

Event 21 - Loss of a Feedwater Heater

Loss of a feedwater heater must he considered with regard to the nuclear safety operational criteria only in operating State D because significant feedwater heating does not occur in any other operating state.

A loss of feedwater heating causes a transient that requires no protective actions when the reactor is initially on automatic recirculation flow control. It the reactor is on manual flow control, however, the neutron flux increase associated with this event will reach the scram setting. As shown in Figure 15A.6-21, the scram safety action is accomplished through actions of the neutron monitoring, reactor protection, and control rod drive systems. Water level will initiate a turbine trip and isolation will soon follow.

Event 22 - Feedwater Controller Failure - Maximum Demand

A feedwater controller failure, causing an excess of coolant inventory in the reactor vessel is possible in all operating states. Feedwater controller failures considered are those that would give failures of automatic flow control, manual flow control, or feedwater bypass valve control. In operating States A and B, no safety actions are required since the vessel head is removed and the moderator temperature is low. In operating State D, any adverse responses by the reactor caused by cooling of the moderator can be mitigated by a scram. As shown in Figure 15A.6-22, the accomplishment of the scram safety action is satisfied through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems. Pressure relief is required in States C and D and is achieved through the operation of the nuclear system pressure relief system. Initial restoration of the core water level is by the RCIC/HPCI systems. Prolonged isolation may require extended core cooling and containment/suppression pool cooling.

<u>Event 23 - Pressure Regulator Failure (Open Direction)</u>

A pressure regulator failure in the open direction, causing the opening of a turbine control or bypass valve, applies only in operating States C and D, because in other states the pressure regulator is not in operation. A pressure regulator failure is most severe and rapid in operating State D at low power.

The various protection sequences giving the safety actions are shown in Figure 15A.6-23. Depending on plant conditions existing prior to the event, scram will be initiated either on main steamline isolation, main turbine trip, reactor vessel high pressure, or reactor vessel low water level. The sequence resulting in reactor vessel isolation also depends on initial conditions. With the mode switch in "Run," isolation is initiated when main steamline pressure decreases to approximately 800 psig. Under other conditions, isolation is initiated by reactor vessel low water level. After isolation is completed, decay heat will cause reactor vessel pressure to increase until limited by the operation of the relief valves. Core cooling following isolation can be provided by either the RCICS or HPCI. Shortly after reactor vessel isolation, normal core cooling can be re-established via the main condenser and feedwater systems or if prolonged isolation is necessary, extended core and containment cooling will be manually actuated.

<u>Event 24 - Pressure Regulator Failure - Closed</u>

A pressure regulator failure in the closed direction (or downscale), causing the closing of turbine control valves, applies only in operating States C and D, because in other states the pressure regulator is not in operation.

A single pressure regulator failure downscale would result in little or no effect on the plant operation. The second pressure regulator would provide turbine-reactor control. If the second unit failed this would result in the worst situation, yet it is much less severe than Events 25, 27, 30 and 31. The dual pressure regulator failures are most severe and rapid in operating State D at high power.

The various protection sequences giving the safety actions are shown in Figure 15A.6-24. Upon failure of one pressure regulator downscale, normally a backup regulator will maintain the plant in the present status upon the initial regulator downscale failure. An additional single failure (SF) of the backup regulator will result in a high flux or pressure SCRAM, system isolation, and subsequent extended isolation core cooling system actuations.

<u>Event 25 - Main Turbine Trips (With By-Pass System Operation)</u>

A main turbine trip can occur only in operating State D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power because the initial power level is low (<30%), thus minimizing the effects of the. transient and enabling return to planned operations via the by-pass system operation. For a turbine trip above 30% power, a scram will occur via turbine stop valve closure as will a recirculation pump trip (RPT). Subsequent relief valve actuation will occur. Eventual main steam line isolation and RCIC/HPCI system initiation will result from low water level. Figure 15A.6-25 depicts the protection sequences required for main turbine trips. Main turbine trip and main generator trip are similar anticipated operational transients and, although main turbine trip is a more severe transient than main generator trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 26 - Loss of Main Condenser Vacuum (Turbine Trip)

A loss of vacuum in the main turbine condenser can occur any time steam pressure is available and the condenser is in use; it is applicable to operating States C and D. This nuclear system pressure increase transient is the most severe of the pressure increase transients. However, scram protection in State C is not needed since the reactor is not coupled to the turbine system.

For State D above 30% power, loss of condenser vacuum will initiate a turbine trip with its attendant stop valve closures (which leads to SCRAM) and a recirculation pump trip (RPT). Loss of condenser vacuum will also initiate isolation, pressure relief valve actuation, and RCIC/HPCI initial core cooling. A scram is initiated by MSIV closure to prevent fuel damage and is accomplished with the actions of the reactor protection system and control rod drive system. Below 30% power (State D) scram is initiated by a high neutron flux signal. Figure 15A.6-26 shows the protection sequences. Decay heat will necessitate extended core and containment cooling. When the nuclear system depressurizes sufficiently, the low pressure core cooling systems provide core cooling until a planned operation via RHRS shutdown cooling is achieved.

Event 27 - Main Generator Trip (With By-Pass System Operation)

A main generator trip with by-pass system operation can occur only in operating State D (during heatup or power operation). Fast closure of the main turbine fast control valves (TGV) is initiated whenever an electrical grid disturbance occurs which results in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the main turbine-generator rotor. Closure of the turbine control valves will cause a sudden reduction in steam flow which results in an increase in system pressure. Above 30% power, scram will occur as a result of fast control valve closure. Turbine tripping will actuate the Recirculation Pump Trip (RPT). Subsequently main steam line isolation will result, pressure relief and initial core cooling by RCIC/HPCI will take place. Prolonged shutdown of the turbine-generator unit will necessitate extended core and containment cooling. A generator trip during heatup (<30%) is not severe because the turbine by-pass system can accommodate the decoupling of the reactor and the turbine-generator unit, thus minimizing the effects of the transient and enabling return to planned operations. Figure 15A.6-27 depicts the protection sequences required for a main generator trip. Main generator trip and main turbine trip are similar anticipated operational transients. Although the main generator trip is a less severe transient than a turbine trip due to the rapid closure of the turbine stop valves, the required safety actions for both are the same sequence.

Event 28 - Loss of Normal Onsite Power - Auxiliary Transformer Failure

There is a variety of possible plant electrical component failures which could affect the reactor system. The total loss of onsite ac power is the most severe. The loss of auxiliary power transformer results in a sequence of events similar to that resulting from a loss of feedwater flow. The most severe situation occurs in State D during power operation. Figure 15A.6-28 shows the safety actions required to accommodate a loss of normal onsite power in the States A, B, C, and D.

The reactor protection and control rod drive systems effect a scram on main turbine trip or loss of reactor protection system power sources. The turbine trip will actuate a recirculation pump trip (RPT). The containment and reactor vessel isolation control system (PCRVICS/CRVICS) *and the main steamline isolation valves act to isolate the reactor vessel. After the main steamline isolation valves (MSIV) close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. With continued isolation decay heat may cause increase in nuclear system pressure, eventually lifting relief valves and allowing reactor vessel water level to decrease. The core/containment cooling sequences shown in Figure 15A.6-28 denote the short- and long-term actions for achieving adequate cooling.*

Event 29 - Loss of Offsite Power - Grid Loss

There is a variety of plant/grid electrical component failures which can affect reactor operation. The total loss of offsite ac power is the most severe. The loss of both onsite and offsite auxiliary power sources results in a sequence of events similar to that resulting from a loss of feedwater flow (see Event 20). The most severe case occurs in State D during power operation.

Figure 15A.6-29 shows the safety actions required for a total loss of offsite power in all States A, B, C, and D.

The reactor protection and control rod drive systems affect a scram from main turbine trip or loss of reactor protection system power sources. The turbine trip will initiate recirculation pump trip (RPT). The containment and reactor vessel isolation control system (PCRVICS/CRVICS) and the main steam line isolation valves (MSLIV) act to isolate the reactor vessel. After the main steamline isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. After the reactor is isolated and feedwater flow has been lost, decay heat will cause an increase in nuclear system pressure, eventually lifting relief valves and allowing reactor vessel water level to decrease. The core and containment cooling sequence shown in Figure 15A.6-29 shows the short- and long-term sequences for achieving adequate cooling.

15A.6.4 Abnormal (Unexpected) Operational Transients

15A.6.4.1 General

The safety requirements and protection sequences for abnormal operational transients are described in the following paragraphs for Events 30 through 39. The protection sequence block diagrams show the sequence of front-line safety systems (refer to Figure 15A.6-30 through 15A.6-39). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-54, 15A.6-55, 15A.6-56, 15A.6-57, 15A.6-58 and 15A.6-59).

15.A.6.4.2 Required Safety Actions/Related Unacceptable Results

The following list relates the safety actions for abnormal operational transients to mitigate or prevent the unacceptable safety results cited in Table 15A.2-8.

Safety Action	*Related Unacceptable Result*	*Reason Action Required*
Scram and/or RPT Pressure relief	*3-2 3-3 3-3*	*To limit gross core-wide fuel damage and to limit nuclear system pressure rise. To prevent excessive nuclear system pressure rise.*
Core and Containment cooling	*3-2 3-4*	*To limit further fuel and containment damage in the event that normal cooling is interrupted.*
Reactor vessel isolation	*3-2*	*To limit further fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.*
Restore ac power	*3-2*	*To limit initial fuel damage by restoring a-c power to system essential to other safety actions.*
Containment isolation	*3-4*	*To limit radiological effects.*

15A.6.4.3 Event Definition & Operational Safety Evaluation

Event 30 - Main Generator Trip (Without By-Pass System Operation)

A main generator trip without by-pass system operation can occur only in operating State D (during heatup or power operation). A generator trip during heatup without by-pass operation results in the same situation as the power operation case. Figure 15A.6-30 depicts the protection sequences required for a main generator trip. The event is basically the same as that described in Event 27 at power levels above 30%. A scram, RPT, isolation, relief valve, and RCIC/HPCI operation will immediately result in prolonged shutdown, which will follow the same pattern as Event 27.

The thermohydraulic and thermodynamic effects on the core, of course, are more severe. Since the event is of lower probability than Event 27, the unacceptable results are less limiting.

The load rejection and turbine trip are similar abnormal operational transients and, although main generator trip is a less severe transient than a turbine trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 31 - Main Turbine Trip (Without By-Pass System Operation)

A main turbine trip without by-pass can occur only in operating State D (during heatup or power operation). Figure 15A.6-31 depicts the protection sequences required for main turbine trips. Plant operation with by-pass system operation above or below 30% power, due to by-pass system failure, will result in the same transient effects: a scram, a RPT, an isolation, subsequent *relief valve actuation, and immediate RCIC/HPCI actuation. After prolonged shutdown, similar extended core and containment cooling will be required as noted previously in Event 25.*

Turbine trips without by-pass system operations results in very severe thermohydraulic impacts on the reactor core. The allowable limit or acceptable calculational techniques for this event are less demanding or strict due to the low probability of the stated event relative to turbine trip with a by-pass operation event.

Main turbine trip and load rejections are similar abnormal operational transients and, although main turbine trip is a more severe transient than main generator trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 32 - Inadvertent Loading and Operation with Fuel Assembly in Improper Position

Operation with a fuel assembly in the improper position can occur in all operating states. No protection sequences are necessary relative to this event. Results of worst fuel handle loading error will not cause fuel cladding integrity damage. It requires three independent equipment/operator errors to allow this situation to develop. See Figure 15A.6-32 for the event sequence.

Events 33 through 37 - Not Used

Event 38 - Recirculation Loop Pump Seizure

A recirculation loop pump seizure event considers the instantaneous stoppage of the pump motor shaft of one recirculation loop pump. The case involves operation at design power in State D.

A main turbine trip will occur as vessel level swell exceeds the turbine trip setpoint. This results in a trip scram and a RPT when the turbine stop valves close. Relief valve opening will occur to control pressure level and temperatures. RCIC or HPCI systems will maintain vessel water level. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

The protection sequence for this event is given in Figure 15A.6-38.

Event 39 - Recirculation Loop Pump Shaft Break

A recirculation loop pump shaft break event considers the degraded, delayed stoppage of the pump motor shaft of one recirculation loop pump. The case involves operation at design power in State D. A main turbine trip will occur as vessel level swell exceeds the turbine trip setpoint. This results in a trip scram and a RPT when the turbine stop valves close. Relief valve opening will occur to control pressure level and temperatures. RCIC or HPCI systems will maintain vessel water level. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

The protection sequence for this event is given in Figure 15A.6-39.

<u>15A.6.5 Design Basis (Postulated) Accidents</u>

<u>15A.6.5.1 General</u>

The safety requirements and protection sequences for accidents are described in the following paragraphs for Events 40 through 49. The protection sequence block diagrams show the safety actions and the sequence of front-line safety systems used for the accidents (refer to Figures 15A.6-54, 15A.6-55, 15A.6-56, 15A.6-57, 15A.6-58 and 15A.6-59).

The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-60 through 15A.6-65).

<u>15A.6.5.2 Required Safety Actions/Unacceptable Results</u>

The following list relates the safety actions for design basis accidents to mitigate or prevent the unacceptable results cited in Table 15A.2-9.

Safety Action	*Related Unacceptable Result*	*Reason Action Required*
Scram	*4-2* *4-3*	*To prevent fuel cladding failure and to prevent excessive nuclear system pressures.* *Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalphy (control rod drop accident).*
Pressure relief	*4-3*	*To prevent excessive nuclear system pressure.*
Core Cooling	*4-2*	*To prevent fuel cladding failure.*
Reactor vessel isolation	*4-1*	*To limit radiological effect to not exceed the guideline values of 10 CFR 100.*
Establish reactor containment	*4-1*	*To limit radiological effects to not exceed the guideline values of 10 CFR 100.*
Containment cooling	*4-4*	*To prevent excessive pressure in the containment when containment is required.*
Stop rod ejection	*4-2*	*To prevent fuel cladding failure.*
Restrict loss of reactor coolant (passive)	*4-2*	*To prevent fuel cladding failure.*
Main Control Room environmental control	*4-5*	*To prevent overexposure to radiation of plant personnel in the control room.*
Limit reactivity insertion rate (passive)	*4-2* *4-3*	*To prevent fuel cladding failure and to prevent excessive nuclear system pressure.*

15A.6.5.3 Event Definition and Operational Safety Evaluations

Event 40 - Control Rod Drop Accident (CRDA)

The control rod drop accident (CRDA) results from an assumed failure of the control rod-to-drive mechanism coupling after the control rod (very reactive rod) becomes stuck in its fully inserted position. It is assumed that the control rod drive is then fully withdrawn before the stuck rod falls out of the core. The control rod velocity limiter, an engineered safeguard, limits the control rod drop velocity. The resultant radioactive material release is maintained far below the guideline values of 10CFR1OO.

The control rod drop accident is applicable only in operating State D.

The control rod drop accident cannot occur in State B because rod coupling integrity is checked on each rod to be withdrawn if more than one rod is to be withdrawn. No safety actions are required in States A or C where the plant is shutdown by more than one rod prior to the accident.

Figure 15A.6-40 presents the different protection sequences for the control rod drop accident. As shown in Figure 15A.6-40, the reactor is automatically scrammed and isolated. For all design basis cases, the neutron monitoring, reactor protection, and control rod drive systems will provide a scram from high neutron flux. The main steam line radiation monitoring system will initiate the isolation of certain containment lines. Any high radiation in the containment areas will initiate closure of other possible pathways to atmosphere, as necessary.

After the reactor has been scrammed and isolated, the pressure relief system allows the steam (produced by decay heat) to be directed to the suppression pool. Initial core cooling is accomplished by either the RCICS or the HPCIS or the normal feedwater system.

With prolonged isolation, as indicated in Figure 15A.6-40, the reactor operator initiates the RHBS/suppression pool cooling mode and depressurizes the vessel with the automatic depressurization system (ADS) or via normal manual relief valve operation. The LPCI, CSCS or HPCI maintain the vessel water level and accomplish extended core cooling. Isolation of turbine-condenser fission product releases will also be maintained.

Event 41 - Fuel Handling Accident (FHA)

Because a fuel-handling accident can potentially occur any time fuel assemblies are being manipulated, either over the reactor core or in a spent fuel pool, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-41. Containment and/or reactor building isolation and standby gas treatment operation are automatically initiated by the respective building or ventilation radiation monitoring systems.

Figure 15A.6-41 describes the protection sequences for the event.

Event 42 - Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within RPCB Inside Containment (DBA-LOCA)

Pipe breaks inside the containment are considered only when the nuclear system is significantly pressurized (States C and D). The result is a release of steam and water into the containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks including larger liquid recirculation loop piping down to small steam instrument line breaks. The most severe cases are the circumferential break of the largest (liquid) recirculation system pipe and the circumferential break of the largest (steam) main steam line.

As shown in Figure 15A.6-42, in operating State C (reactor shut down, but pressurized), a pipe break accident up to the DBA can be accommodated within the nuclear safety operational criteria through the various operations of the main steamline isolation valves, emergency core cooling systems (HPCI, ADS, LPCI, CSCS), containment and reactor vessel isolation control system, containment, reactor building, standby gas treatment system, main control room *heating, cooling and ventilation system, MSIV Leakage Isolated Condenser Treatment Method, emergency service water systems, hydrogen control system, equipment cooling systems, and the incident detection circuitry. For small pipe breaks inside the containment, pressure relief is effected by the nuclear system pressure relief system, which transfers decay heat to the suppression pool. For large breaks, depressurization takes place though the break itself. In State D (reactor not shut down, but pressurized), the same equipment is required as in State C but, in addition, the reactor protection system and the control rod drive system must operate to scram the reactor. The limiting items, on which the operation of the above equipment is based, are the allowable fuel cladding temperature and the containment pressure capability. The control rod drive housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the bottom of the reactor pressure vessel following the postulated rupture of one control rod drive housing (a lesser case of the design basis loss-of-coolant accident and a related preventive of a postulated rod ejection accident).*

After completion of the automatic action of the above equipment, manual operation of the RHRS (suppression pool cooling mode) and ADS (controlled depressurization) is required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

Event - 43, 44, 45 - Large, Small, Steam and Liquid Pipe Breaks Outside Containment (SLBA)

Pipe break accidents outside the containment are assumed to occur any time the nuclear system is pressurized (States C and D). This accident is most severe during operation at high power (State D). In State C, this accident becomes a lesser case of the State D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown in Figure 15A.6-43. The sequences also show that for small breaks (breaks not requiring immediate action) the reactor operator can use a large number of process indications to identify the break and isolate it.

In operating State D (reactor not shut down, but pressurized), scram is accomplished through operation of the reactor protection system and the control rod drive system. Reactor vessel isolation is accomplished through operation of the main steamline isolation valves and the containment and reactor vessel isolation control system.

For a main steamline break, initial core cooling is accomplished by either the HPCI or the automatic depressurization system (ADS) or manual relief valve operation in conjunction with either the CSCS or LPCI. These systems provide three parallel paths to effect initial core cooling, thereby satisfying the single-failure criterion. Extended core cooling is accomplished by the single-failure proof, parallel combination of CSCS, HPCI and LPCI. The automatic depressurization system (ADS) or relief valve system operation and the RHRS suppression pool cooling mode (both manually operated) are required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

Event 46 - Gaseous Radwaste System Leak or Failure

It is assumed that the line leading to the steam jet air ejector fails near the main condenser. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the *environment. This failure results in a loss-of-flow signal to the offgas system. This event can be considered only under States C and D.*

The reactor operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result (timing depending on leak rate) in a main turbine trip and ultimately a reactor shutdown. Refer to Event 26 for reactor protection sequence (see Figure 15A.6-26).

The protective sequences for this event are provided in Figure 15A.6-46.

Event 47 - Ambient Charcoal Offgas Treatment System Failure

An evaluation of those events which could cause a gross failure in the offgas system has resulted in the identification of a postulated seismic event, more severe than the one for which the system is designed, as the only conceivable event which could cause significant damage.

The detected gross failure of this system will result in manual isolation of this system from the main condenser. The isolation results in high main condenser pressure and ultimately a reactor scram.

The undetected postulated failure soon results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Event 26 (see Figure 15A.6-26).

The protective sequences for this event are provided in Figure 15A.6-47.

Event 48 - Liquid Radwaste System Leak or Failure

Releases which could occur inside and outside of the containment, not covered by Events 40, 41, 42, 43, 44, 45, 47, and 48 will probably include small spills and equipment leaks of radioactive materials inside structures housing the subject process equipment. Conservative values for leakage have been assumed and evaluated in the plant under routine releases. The offsite dose that results from any small spill which could occur outside containment will be negligible in comparison to the dose resulting from the accountable (expected) plan leakages.

The protective sequences for this event are provided in Figure 15A.6-48.

Event 49 - Liquid Radwaste System - Storage Tank Failure

An unspecified event causes the complete release of the average radioactivity inventory in the subject tank containing the largest quantities of significant radionuclides from the liquid radwaste system. This is assumed to be the concentrates waste tank in the radwaste building. The airborne radioactivity released during the accident passes directly to the environment via the radwaste building vent.

The postulated events that could cause release of the radioactive inventory of the concentrates waste tank include cracks in the vessels and an operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The concentrates waste tank is designed to operate at atmospheric pressure and 200°F maximum temperature so the possibility of failure is considered small. A *liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is provided to prevent inadvertent opening of a drain valve. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid to a contained storage tank.*

The protective sequences for this event are provided in Figure 15A.6-49.

15A.6.6 Special Plant-Capability Events

15A.6.6.1 General

Additional special events are postulated to demonstrate that the plant is capable of accommodating off-design occurrences. (Refer to Events 50 through 53). As such, these events are beyond the safety requirements of the other event categories. The safety actions shown on the sequence diagrams (refer to Figure 15A.6-50 through 15A.6-53) for the additional special events follow directly from the requirements cited in the demonstration of the plant capability.

Auxiliary system support analyses are shown in Figures 15A.6-1, 15A.6-2 and 15A.6-54, 15A.6-55, 15A.6-56, 15A.6-57, 15A.6-58 and 15A.6-59.

15A.6.6.2 Required Safety Action/Unacceptable Results

The following list relates the safety actions for special events to prevent the unacceptable results cited in Table 15A.2-10.

Safety Action	*Related Unacceptable Result*	*Reason Action Required*
Manually initiate all shutdown controls from local panels	*5-1 5-2*	*Local panel control has been provided and is available outside main control room.*
Manually initiate SLCS	*5-3*	*Standby Liquid Control System to control reactivity to cold shutdown is available.*

15A.6.6.3 Event Definitions and Operational Safety Evaluation

Event 50 - Shipping Cask Drop

Due to the redundant nature of the plant crane, the cask drop accident is not believed to be a credible accident. However, the accident is hypothetically assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall.

It is assumed that a spent fuel shipping cask containing irradiated fuel assemblies is in the process of being moved with the cask suspended from the crane above the rail car. The fuel assemblies have been out of the reactor for at least 90 days.

Through some unspecified failure, the cask is released from the crane and falls between 30 to 100 feet onto the rail car. Some of the coolant in the outer cask structure may leak from the cask.

The reactor operator will ascertain the degree of cask damage and, if possible, make the necessary repairs and refill the cask coolant to its normal level if coolant has been lost.

It is assumed that if the coolant is lost from the external cask shield, the operator will establish forced cooling of the cask by introducing water exterior surface. Maintaining the cask in a cool condition will, therefore, ensure no fuel damage as a result of a temperature increase due to decay heat.

Since the cask is still within the reactor building volume, any activity postulated to be released can be accommodated by the SGTS.

The protective sequences for this event are provided in Figure 15A.6-50.

Event 51 - Reactor Shutdown - ATWS

Reactor shutdown from a plant transient occurrence (e.g., turbine trip) without the use of mechanical control rods is an event currently being evaluated to determine the capability of the plant to be safely shutdown. The event is applicable in any operating state. Figure 15A.6-51 shows the protection sequence for this extremely improbable and demanding event in each operating state. In State A, no sequence is shown because the reactor is already in the condition finally required by definition.

State D is the most limiting case. Upon initiation of the plant transient situation (turbine trip), a scram will be initiated but no control rods are assumed to move. The recirculation pumps will be tripped by the initial turbine trip signal. If the nuclear system becomes isolated from the main condenser, low power neutron heat can be transferred from the reactor to the suppression pool via the relief valves. The incident detection circuitry initiated operation of the HPCIS on low water level which maintains reactor vessel water level. The standby liquid control system will be manually initiated and the transition from low power neutron heat to decay heat will occur. The RHRS suppression pool spray cooling mode is used to remove the low power neutron and decay heat from the suppression pool as required. When reactor pressure falls to 100 to 200 psig level, the RHRS shutdown cooling mode is started and continued to cold shutdown. Various single failure analytical exercises can be examined to further show additional capabilities to accommodate further plant system degradations.

Event 52 - Reactor Shutdown From Outside Main Control Room

Reactor shutdown from outside main control room is an event investigated to evaluate the capability of the plant to be safely shutdown and cooled to the cold shutdown state from outside the main control room. The event is applicable in any operating States A, B, C and D.

Figure 15A.6-52 shows the protection sequences for this event in each operating state. In State A, no sequence is shown because the reactor is already in the condition finally required for the event. In State C, only cooldown is required since the reactor is already shutdown.

A scram from outside the main control room can be achieved by opening the ac supply breakers for the reactor protection system. If the nuclear system becomes isolated from the main condenser, decay heat is transferred from the reactor to the suppression pool via the relief valves. The incident detection circuitry initiates operation of the RCIC/HPCI systems on low water level which maintains reactor vessel water level, and the RHRS suppression pool cooling mode is used to remove the decay heat from the suppression pool if required. When reactor pressure falls to 100 to 200 psig level, the RHRS shutdown cooling mode is started.

Event 53 - Reactor Shutdown Without Control Rods

Reactor shutdown without control rods is an event requiring an alternate method of reactivity control (the standby liquid control system). By definition, this event can occur only when the reactor is not already shutdown. Therefore, this event is considered only in operating States B and D.

The standby liquid control system must operate to avoid unacceptable result criteria 5-3. The design bases for the standby liquid control system result from these operating criteria when applied under the most severe conditions (State D at rated power). As indicated in Figure 15A.6-53, the standby liquid control system is manually initiated and controlled in States B and D.

15A.7 REMAINDER OF NSQA

With the information presented in the protection sequence block diagrams, the auxiliary diagrams, and the commonality of auxiliary diagrams, it is possible to determine the exact functional and hardware requirements for each system. This is done by considering each event in which the system is employed and deriving a limiting set of operational requirements. This limiting set of operational requirements established the lowest acceptable level of performance for a system or component, or the minimum number of components or portions of a system that must be operable in order that plant operation may continue.

The operational requirements derived using the above process may be complicated functions of operating states, parameter ranges, and hardware conditions. The final step is to simplify these complex requirements into technical specifications that encompass the operational requirements but are easily used by plant operations and management personnel.

15A.8 CONCLUSIONS

It is concluded that the nuclear safety operational and plant design basis criteria are satisfied when the plant is operated in accordance with the nuclear safety operational requirements determined by the method presented in this appendix.

15A.9 REFERENCES

15A-1 *Hirsch, M.M. "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems," January 1973 (NEDO-10739).*

END HISTORICAL

TABLE 15A.2-1

PLANNED (NORMAL) OPERATION

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
1	Refueling - Initial - Reload	15A.6-3,4,5,6	-
2	Achieving Criticality	15A.6-3,4,5,6	-
3	Heat-Up	15A.6-3,4,5,6	-
4	Power Operation – Generation - Steady State - Daily Load Reduction & Recovery - Grid Frequency Control Response - Control Rod Sequence Exchanges - Power Generation Surveillance Testing ° Turbine Stop Valve Surveillance Tests ° Turbine Control Valve Surveillance Tests ° MSLIV Surveillance Tests	15A.6-3,4,5,6	-
5	Achieving Shutdown	15A.6-3,4,5,6	-
6	Cooldown	15A.6-3,4,5,6	-

TABLE 15A.3-1

BWR OPERATING STATES

Conditions	States			
	A	B	C	D
Reactor vessel head off	X*	X*		
Reactor vessel head on			X	X
Shutdown	X		X	
Not shutdown		X		X

Definition

Shutdown: K_{eff} sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentrations.

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*Because the reactor vessel head is off in States A and B, pressure is atmospheric.

TABLE 15A.6-1

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

PLANNED (NORMAL) OPERATION

	BWR Operating States			
Types of Operation and Events	A	B	C	D
1. Refueling outage	X			
2. Achieving Criticality	X	X	X	X
3. Heatup				X
4. Power operation				X
5. Achieving Shutdown		X		X
6. Cooldown	X		X	

TABLE 15A.2-2

ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
7	Manual or Inadvertant SCRAM	15A.6-7	7.2
8	Loss of Plant Instrument Service Air Systems	15A.6-8	9.3.1
9	Inadvertant Start-Up of HPCI Pump	15A.6-9	15.5.1
10	Inadvertant Start-Up of Idle Recirculation Loop Pump	15A.6-10	15.4.4
11	Recirculation Loop Flow Control Failure with Increasing Flow	15A.6-11	15.4.5
12	Recirculation Loop Flow Control Failure with Decreasing Flow	15A.6-12	15.3.2
13	Recirculation Loop Pump Trip - With One Pump - With Two Pumps	15A.6-13	15.3.1
14	Inadvertant MSLIV Closure - With One Valve - With Four Valves	15A.6-14a 15A.6-14b	15.2.4

HISTORICAL INFORMATION

TABLE 15A.2-2

ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
15	Inadvertant Operation of One Safety/Relief Valve - Opening/Closing - Struck Open	15A.6-15	15.1.4
16	Continuous Control Rod Withdrawal Error - During Start-Up - During Refueling	15A.6-16	15.4.1
17	Continuous Control Rod Withdrawal Rod Error at Power	15A.6-17	15.4.2
18	RHRS – Shutdown Cooling Failure Loss of Cooling	15A.6-18	15.2.9
19	RHRS – Shutdown Cooling Failure Increased Cooling	15A.6-19	15.1.6
20	Loss of All Feedwater Flow	15A.6-20	15.2.7
21	Loss of Feedwater Heater	15A.6-21	15.1.1
22	Feedwater Controller Failure Maximum Demand	15A.6-22	15.1.2
23	Pressure Regulator Failure - Open	15A.6-23	15.1.3

HISTORICAL INFORMATION

TABLE 15A.2-2

ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
24	Pressure Regulator Failure - Closed	15A.6-24	15.2.1
25	Main Turbine Trip With Bypass System Operational	15A.6-25	15.2.3
26	Loss of Main Condenser Vacuum	15A.6-26	15.2.5
27	Main Generator Trip (Load Rejection) With Bypass System Operational	15A.6-27	15.2.2
28	Loss of Plant Normal On-Site AC Power – Auxiliary Transformer Failure	15A.6-28	15.2.6
29	Loss of Plant Normal Off-Site AC Power – Grid Connection Failure	15A.6-29	15.2.6

HISTORICAL INFORMATION

TABLE 15A.6-2

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

		BWR Operating States			
	Types of Operation and Events	A	B	C	D
7.	Manual or Inadvertant SCRAM	X	X	X	X
8	Loss of Plant Instrument/Service Air System			X	X
9.	Inadvertant Start-Up of HPCI Pumps	X	X	X	X
10.	Inadvertant Start-Up of Idle Recirculation Loop Pump	X	X	X	X
11.	Recirculation Loop Flow Control Failure-Increasing			X	X
12.	Recirculation Loop Flow Control Failure-Decreasing			X	X
13.	Recirculation Loop Pump Trips – One or Both			X	X
14.	Inadvertent MSIV Closure – One or Four Valves			X	X
15.	Inadvertent Operation of One Safety/Relief Valve			X	X
16.	Continuous Control Rod Withdrawal Error - During Start-Up - During Refueling	 X	 X		
17.	Continuous Control Rod Withdrawal Error - At Power			X	X
18.	RHRS – Shutdown Cooling Failure – Loss of Cooling	X	X	X	X
19.	RHRS – Shutdown Cooling Failure – Increased Cooling	X	X	X	X
20.	Loss of All Feedwater Flow			X	X
21.	Loss of One Feedwater Heater				X

HISTORICAL INFORMATION

TABLE 15A.6-2

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

	BWR Operating States			
Types of Operation and Events	A	B	C	D
22. Feedwater Controller Failure – Maximum Demand	X	X	X	X
23. Pressure Regulator Failure – Open			X	X
24. Pressure Regulator Failure – Closed			X	X
25. Main Turbine Trips – With Bypass				X
26. Loss of Main Condenser Vacuum			X	X
27. Main Generator Trip (Load Rejection) With Bypass				X
28. Loss of Plant Normal On-site AC Power – Auxiliary Transformer Loss	X	X	X	X
29. Loss of Plant Normal Off-site AC Power – Grid Connection Loss	X	X	X	X

TABLE 15A.2-3

ABNORMAL (UNEXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
30	Main Generator Trip (Load Rejection) with Bypass System Failure	15A.6-30	15.2.2
31	Main Turbine Trip With Bypass System Failure	15A.6-31	15.2.3
32	Inadvertent Loading and Operation of a Fuel Assembly In An Improper Position	15A.6-32	15.4.7
33	NOT USED	--	--
34	NOT USED	--	--
35	NOT USED	--	--
36	NOT USED	--	--
37	NOT USED	--	--
38	Recirculation Loop Pump Seizure	15A.6-38	15.3.3
39	Recirculation Loop Pump Shaft Break	15A.6-39	15.3.4

TABLE 15A.6-3

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

ABNORMAL (UNEXPECTED) OPERATIONAL TRANSIENTS

Types of Operation and Events		BWR Operating States			
		A	B	C	D
30.	Main Generator Trip (Load Rejection) – Without Bypass				X
31.	Main Turbine Trip - Without Bypass				X
32.	Inadvertant Loading and Operation of a Fuel Assembly in an Improper Position	X	X	X	X
33.	NOT USED				
34.	NOT USED				
35.	NOT USED				
36.	NOT USED				
37.	NOT USED				
38.	Recirculation Loop Pump Seizure			X	X
39.	Recirculation Loop Pump Shaft Break			X	X

TABLE 15A.2-4

DESIGN BASIS (POSTULATED) ACCIDENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
40	Control Rod Drop Accident	15A.6-40	15.4.9
41	Fuel Handling Accident	15A.6-41	15.7.4
42	Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within the RPCB Inside Containment	15A.6-42	15.6.5
43	Small, Large, Steam and Liquid Piping Breaks Outside Containment	15A.6-43	15.6.4
44	Instrument Line Break Outside Drywell	15A.6-44	15.6.2
45	Feedwater Line Break Outside Containment	15A.6-45	15.6.6
46	Gaseous Radwaste System Leak or Failure	15A.6-46	15.7.1
47	Ambient Charcoal Off-Gas Treatment System Failure	15A.6-47	15.7.1
48	Liquid Radwaste System Leak or Failure	15A.6-48	15.7.2
49	Liquid Radwaste System Storage Tank Failure	15A.6-49	15.7.3

TABLE 15A.6-4

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

DESIGN BASIS (POSTULATED) ACCIDENTS

	BWR Operating States			
Types of Operation and Events	A	B	C	D
40. Control Rod Drop Accident				X
41. Fuel Handling Accident	X	X	X	X
42. Loss of Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within RPCB Inside Containment			X	X
43. Steam System Piping Break Outside Secondary Containment			X	X
44. Instrument Line Break Inside Secondary Containment			X	X
45. Feedwater Line Break Outside Containment			X	X
46. Gaseous Radwaste System Leak or Failure			X	X
47. Ambient Charcoal Off-Gas Treatment System Failure			X	X
48. Liquid Radwaste System Leak or Failure	X	X	X	X
49. Liquid Radwaste System Storage Tank Failure	X	X	X	X

TABLE 15A.2-5

SPECIAL (PLANT CAPABILITY) EVENTS

Cross-Correlation References

NSQA Event No.	Event Description	NSQA Event Figure No.	Safety Analysis Section No.
50	Spent Fuel Cask Drop	15A.6-50	15.7.5
51	Reactor Shutdown From Anticipated Transient Without SCRAM (ATWS)	15A.6-51	15.8
52	Reactor Shutdown From Outside Main Control Room	15A6-52	7.5
53	Reactor Shutdown Without Control Rods	15A.6-53	9.3.5

TABLE 15A.6-5

PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE

SPECIAL (PLANT CAPABILITY) EVENTS

		BWR Operating States			
	Types of Operation and Events	A	B	C	D
50.	Spent Fuel Cask Drop	X	X	X	X
51.	Reactor Shutdown from Anticipated Transient - Without SCRAM (ATWS)	X	X	X	X
52.	Reactor Shutdown - From Outside Main Control Room	X	X	X	X
53.	Reactor Shutdown - Without Control Rods	X	X	X	X

TABLE 15A.2-6

PLANT EVENT CATEGORY: PLANNED (NORMAL) OPERATION

UNACCEPTABLE RESULTS CRITERIA

	UNACCEPTABLE RESULTS
1-1.	Release of radioactive material to the environs that exceeds the limits of either 10CFR20 or 10CFR50.
1-2.	Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded.
1-3.	Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.
1-4.	Existence of a plant condition not considered by plant safety analyses.

TABLE 15A.2-7

PLANT EVENT CATEGORY: ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

UNACCEPTABLE RESULTS CRITERIA

	UNACCEPTABLE RESULTS
2-1.	Release of radioactive material to the environs that exceeds the limits of 10CFR20.
2-2.	Any fuel failure calculated as a direct result of the transient analyses.
2-3.	Nuclear system stress exceeding that allowed for transients by applicable industry codes.
2-4.	Containment stresses exceeding that allowed for transients by applicable industry codes when containment is required.

TABLE 15A.2-8

PLANT EVENT CATEGORY: ABNORMAL (UNEXPECTED) OPERATIONAL TRANSIENTS

UNACCEPTABLE RESULTS CRITERIA

	UNACCEPTABLE RESULTS
3-1.	Radioactive material release exceeding a small fraction of 10CFR100.
*3-2.	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
3-3.	Nuclear system stresses exceeding that allowed for transients by applicable industry codes.
3-4.	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

*Failure of the fuel barrier means gross core-wide fuel cladding perforations.

TABLE 15A.2-9

PLANT EVENT CATEGORY: DESIGN BASIS (POSTULATED) ACCIDENTS

UNACCEPTABLE RESULTS CRITERIA

	UNACCEPTABLE RESULTS
4-1.	Radioactive material release exceeding the guideline values of 10CFR100.
**4-2.	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
4-3.	Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.
4-4.	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.
4-5.	Overexposure to radiation of plant main control room personnel.

**Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

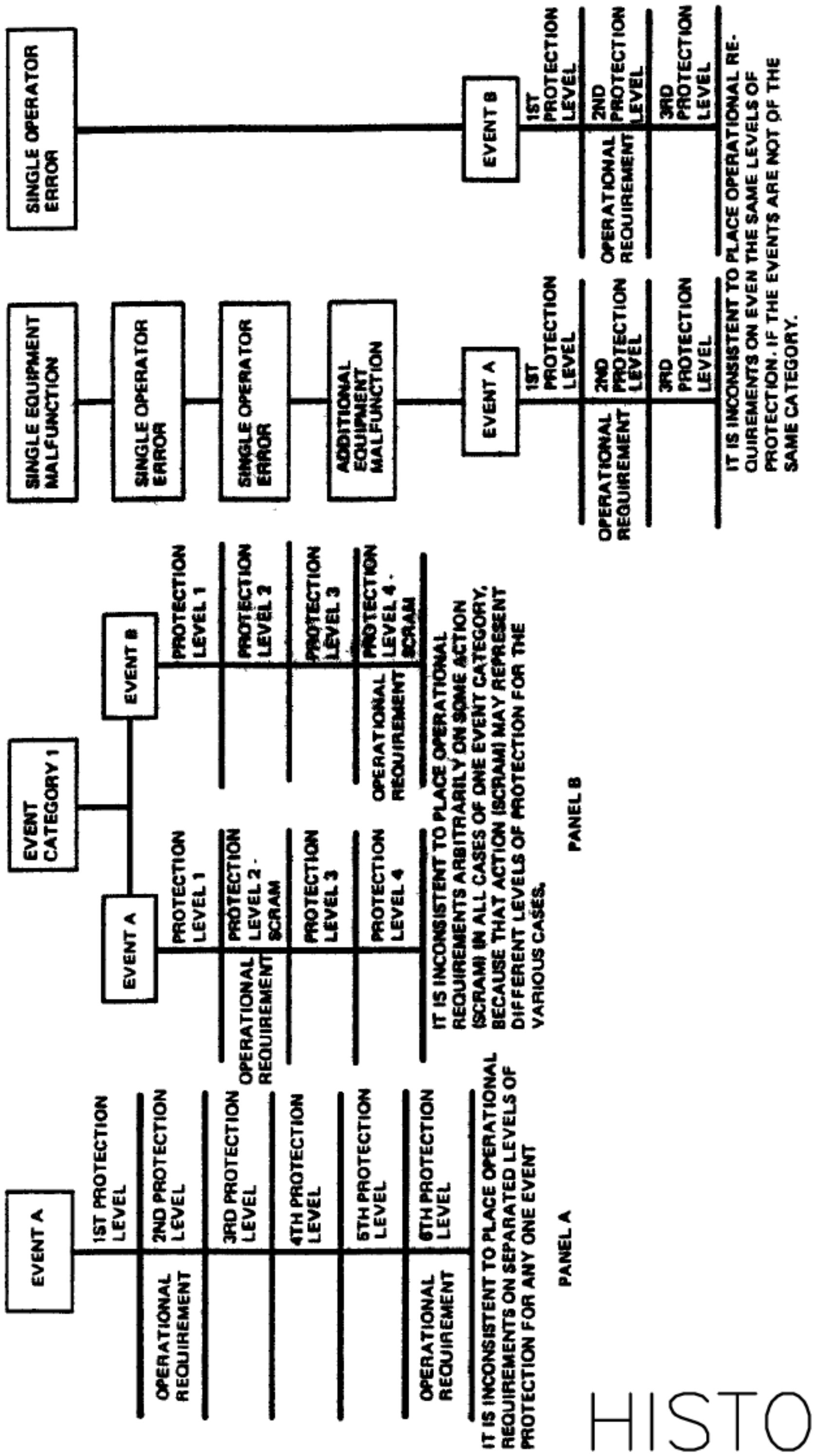
TABLE 15A.2-10

PLANT EVENT CATEGORY: SPECIAL (PLANT CAPABILITY) EVENTS

UNACCEPTABLE RESULTS CONSIDERATIONS

Special Events Considered	
A.	Reactor shutdown from outside control room
B.	Reactor shutdown without control rods
C.	Reactor shutdown with anticipated transient without scram (ATWS)

Capability Demonstration	
5-1.	Ability to shut down reactor by manipulating controls and equipment outside the main control room.
5-2.	Ability to bring the reactor to the cold shutdown condition from outside the main control room.
5-3.	Ability to shut down the reactor independent of control rods.



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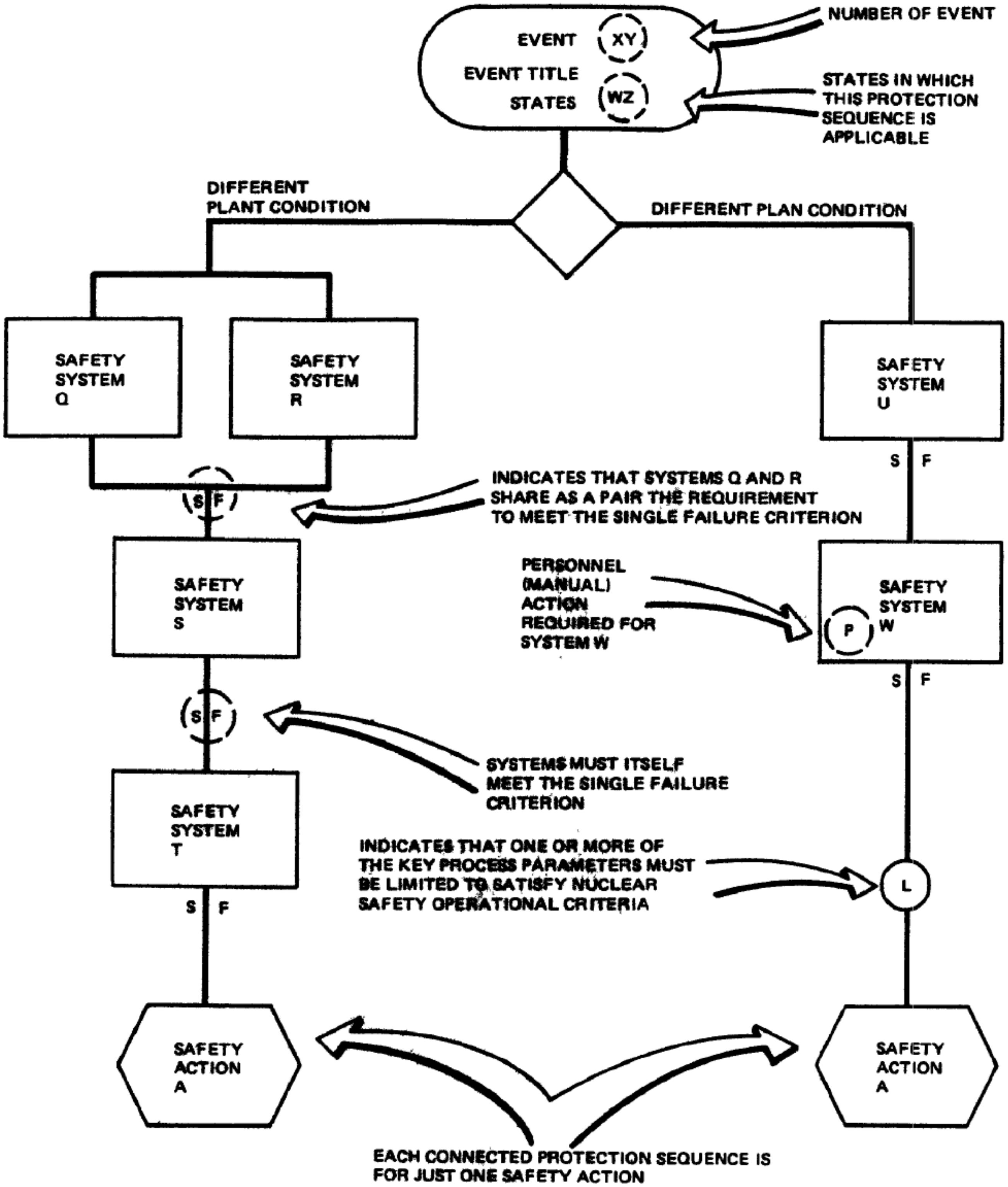
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

POSSIBLE INCONSISTENCIES IN THE SELECTION OF NUCLEAR SAFETY OPERATIONAL REQUIREMENTS

FIGURE 15A.2-1, Rev 55

AutoCAD: Figure Fsar 15A_2_1.dwg



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FINAL SAFETY ANALYSIS REPORT

FORMAT FOR PROTECTION SEQUENCE DIAGRAMS

FIGURE 15A.4-1, Rev 55

AutoCAD: Figure Fsar 15A_4_1.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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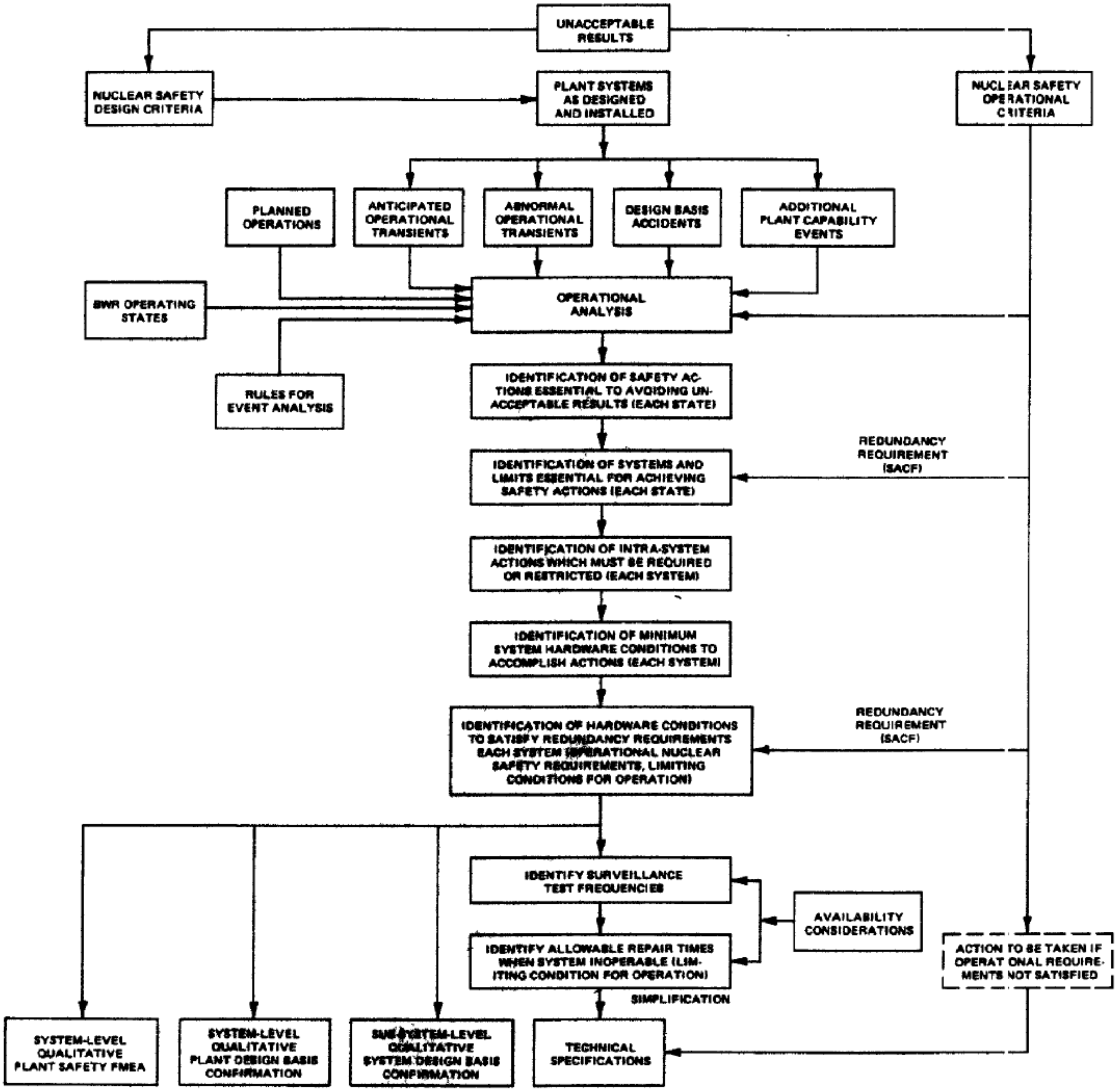
FINAL SAFETY ANALYSIS REPORT

SAFETY SYSTEM AUXILIARIES

FIGURE 15A.6-1, Rev 55

AutoCAD: Figure Fsar 15A_6_1.dwg



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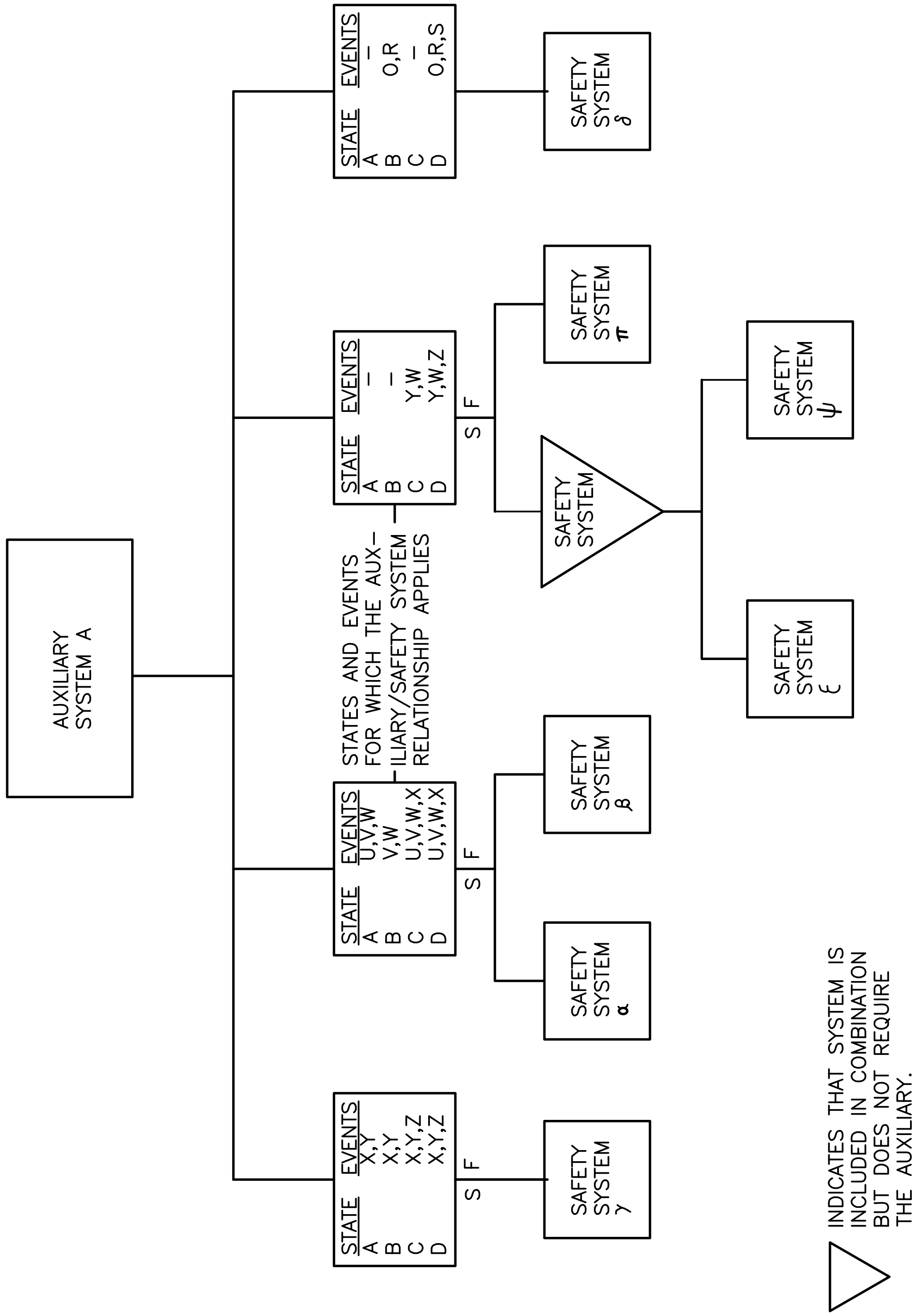
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

HISTORICAL

METHODS USED TO DERIVE NSO REQUIREMENTS SYSTEM & SUBSYSTEM LEVEL QUALITATIVE FMEA & DESIGN BASIS CONFIRMATION AUDITS & TECH SPECS

FIGURE 15A.2-2, Rev 55

AutoCAD: Figure Fsar 15A_2_2.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FORMAT FOR SAFETY SYSTEM
AUXILIARY DIAGRAMS

FIGURE 15A.4-2, Rev 55

AutoCAD: Figure Fsar 15A_4_2.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

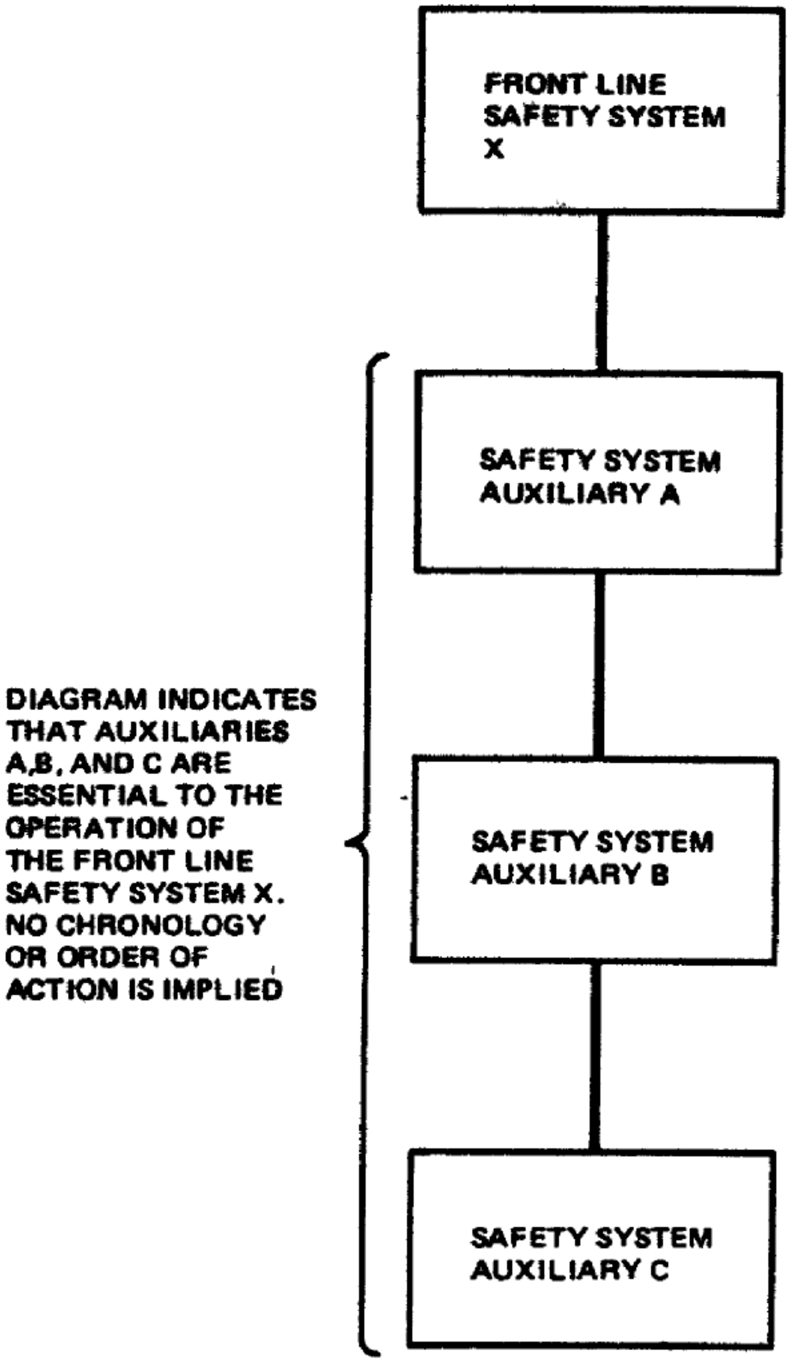
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAFETY SYSTEM AUXILIARIES

FIGURE 15A.6-2, Rev 55

AutoCAD: Figure Fsar 15A_6_2.dwg



HISTORICAL

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FORMAT FOR COMMONALITY
OF AUXILIARY DIAGRAMS

FIGURE 15A.4-3, Rev 55

AutoCAD: Figure Fsar 15A_4_3.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAFETY ACTION SEQUENCE
FOR PLANNED OPERATIONS
IN STATE A

FIGURE 15A.6-3, Rev 55

AutoCAD: Figure Fsar 15A_6_3.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAFETY ACTION SEQUENCES
FOR PLANNED OPERATIONS
IN STATE B

FIGURE 15A.6-4, Rev 55

AutoCAD: Figure Fsar 15A_6_4.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
SAFETY ACTION SEQUENCES FOR PLANNED OPERATIONS IN STATE C FIGURE 15A.6-5, Rev 55

AutoCAD: Figure Fsar 15A_6_5.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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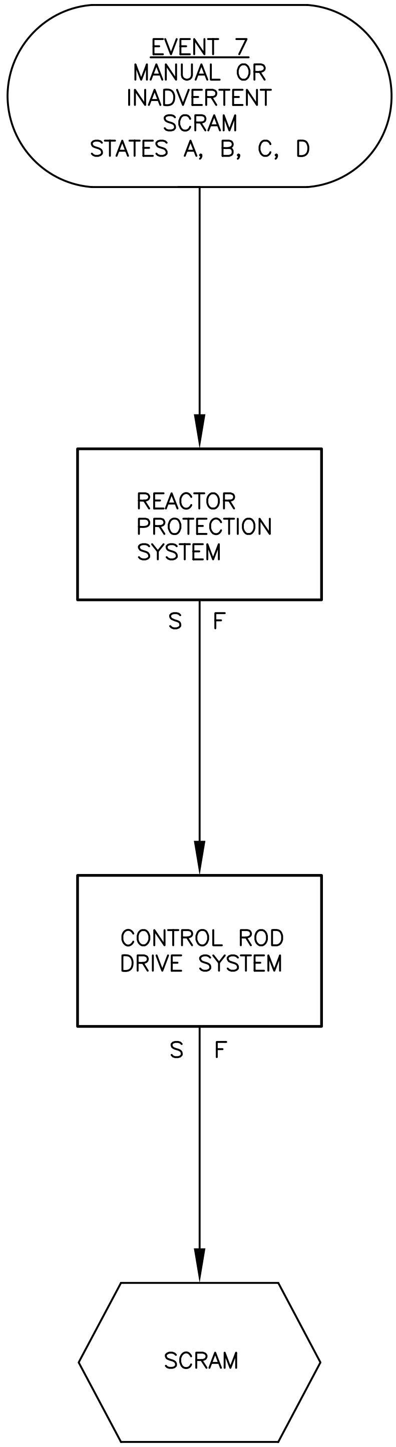
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAFETY ACTION SEQUENCES
FOR PLANNED OPERATIONS
IN STATE D

FIGURE 15A.6-6, Rev 55

AutoCAD: Figure Fsar 15A_6_6.dwg



HISTORICAL

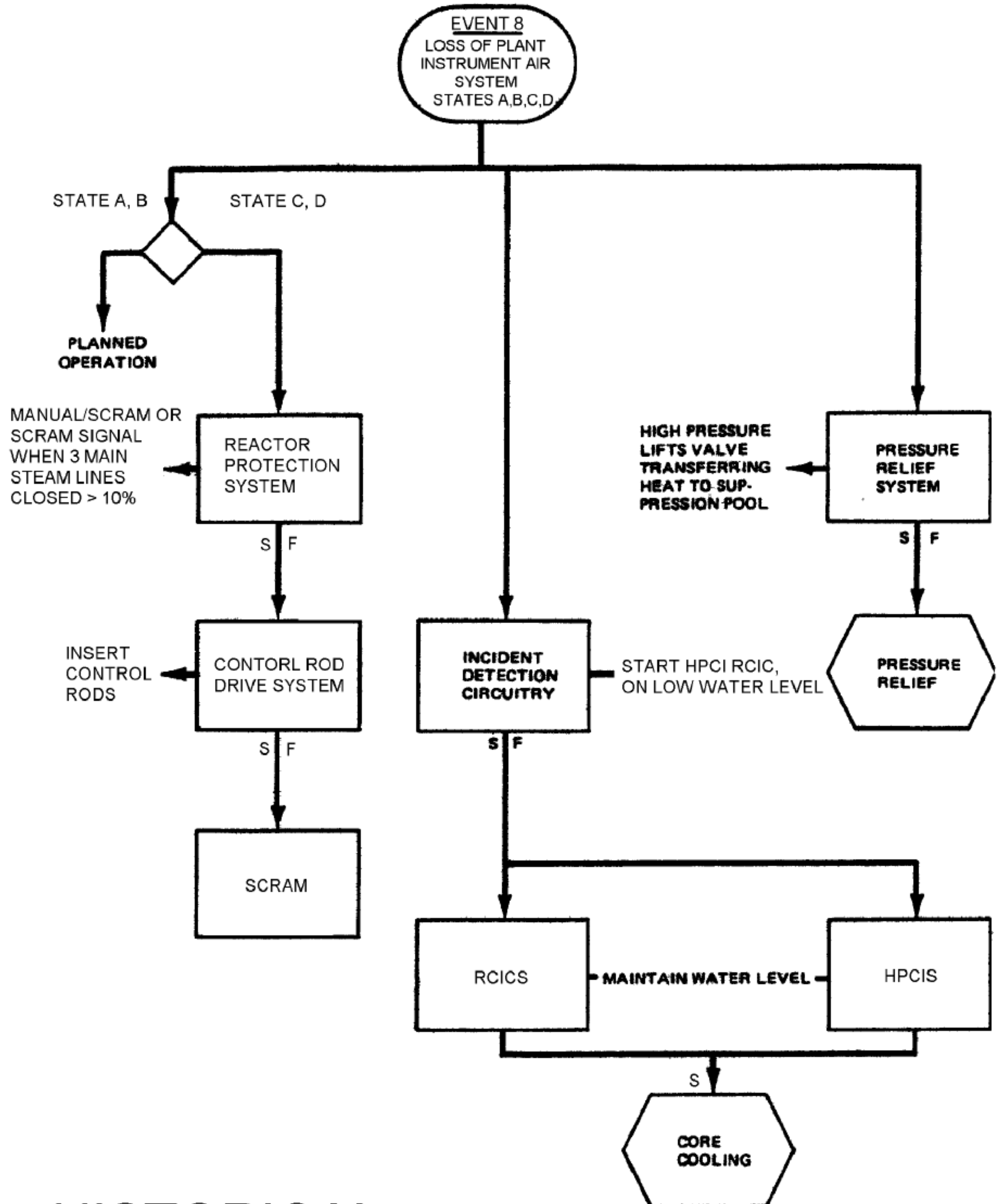
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
MANUAL OR INADVERTENT SCRAM

FIGURE 15A.6-7, Rev 55

AutoCAD: Figure Fsar 15A_6_7.dwg



HISTORICAL

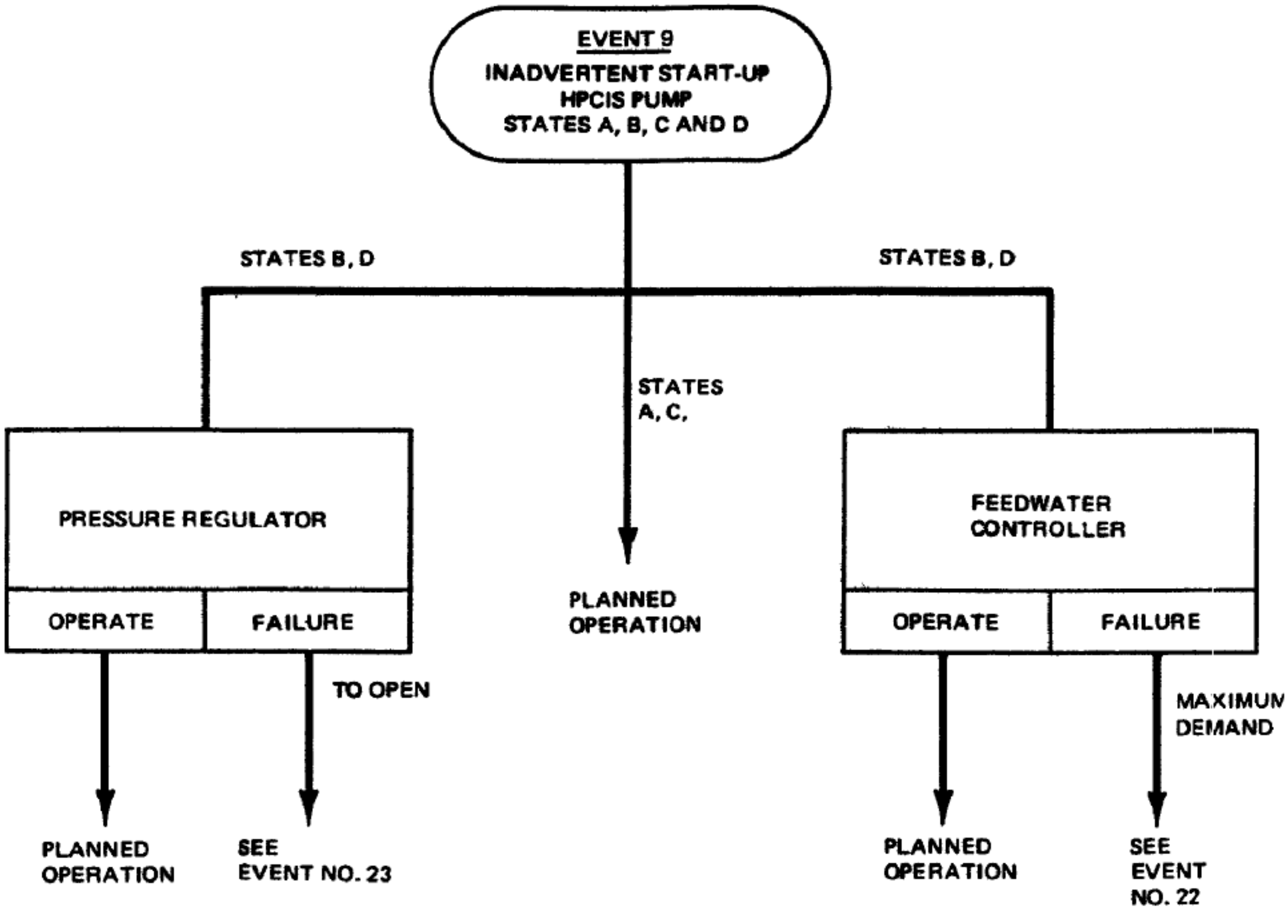
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR LOSS OF PLANT INSTRUMENT AIR SYSTEM

FIGURE 15A.6-8, Rev 55

AutoCAD: Figure Fsar 15A_6_8.dwg



HISTORICAL

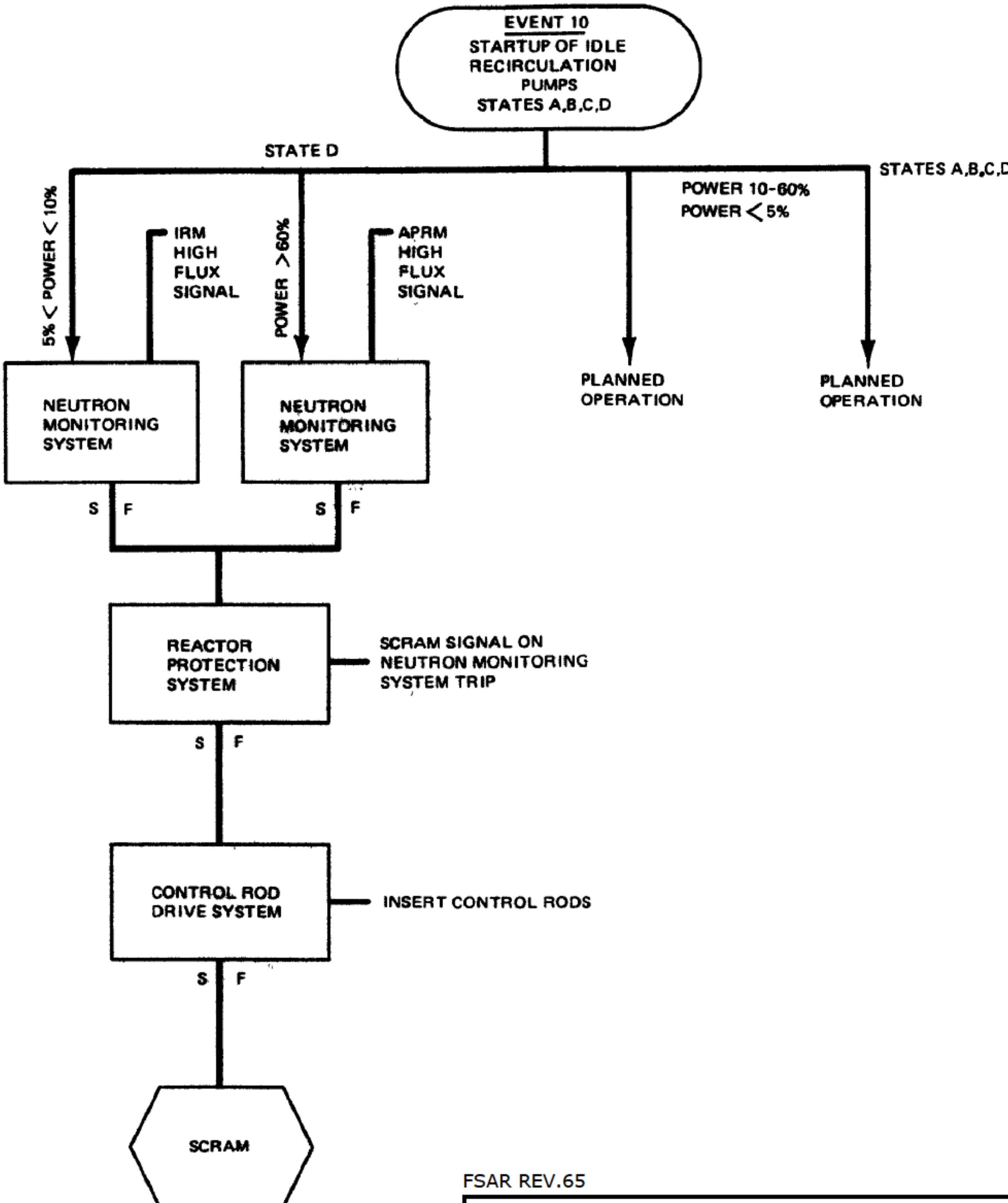
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
INADVERTENT START-UP
OF HPCI'S PUMP

FIGURE 15A.6-9, Rev 55

AutoCAD: Figure Fsar 15A_6_9.dwg



FSAR REV.65

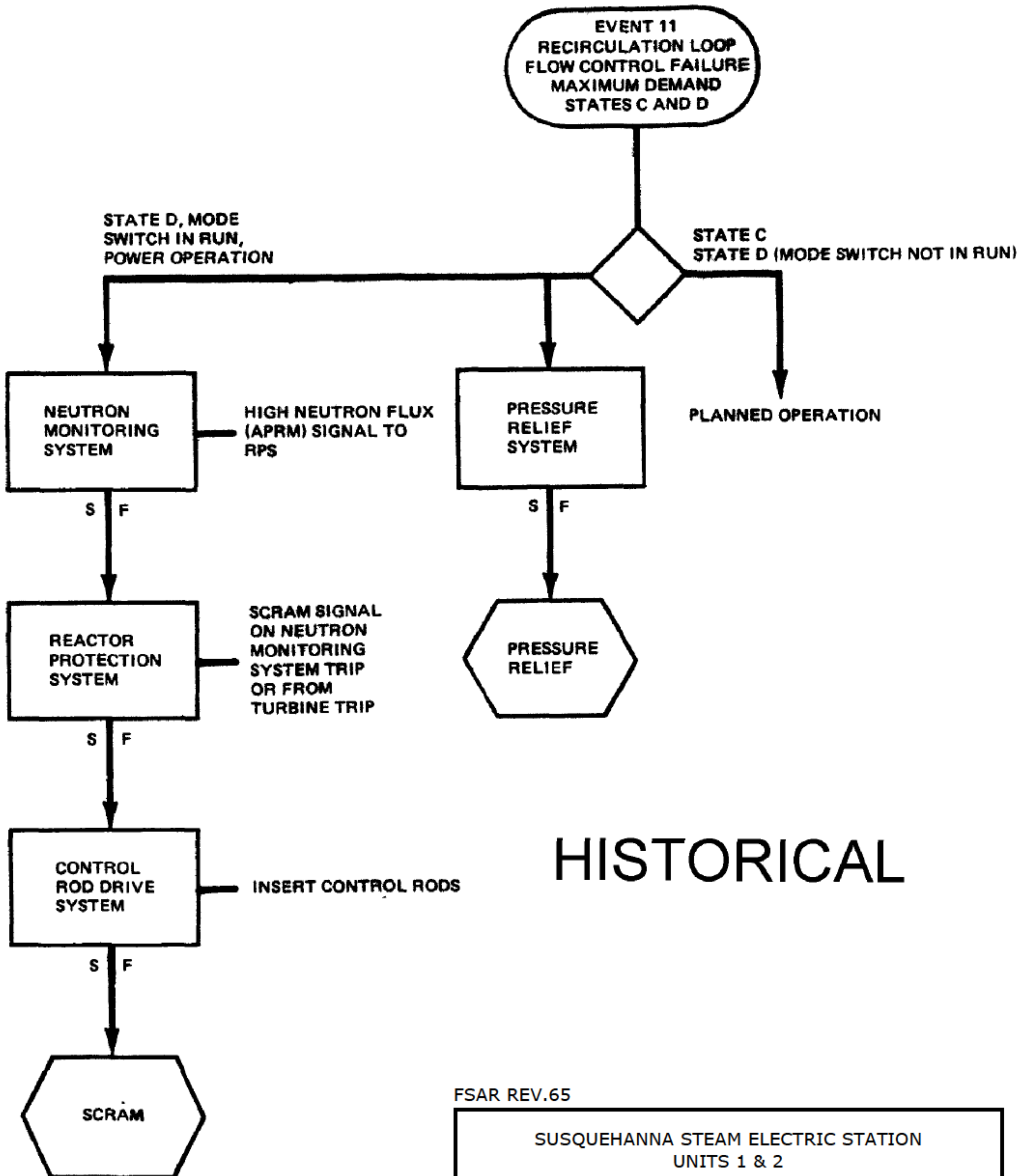
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

HISTORICAL

PROTECTION SEQUENCES FOR
INADVERTENT START-UP OF IDLE
RECIRCULATION LOOP PUMP

FIGURE 15A.6-10, Rev 55

AutoCAD: Figure Fsar 15A_6_10.dwg



HISTORICAL

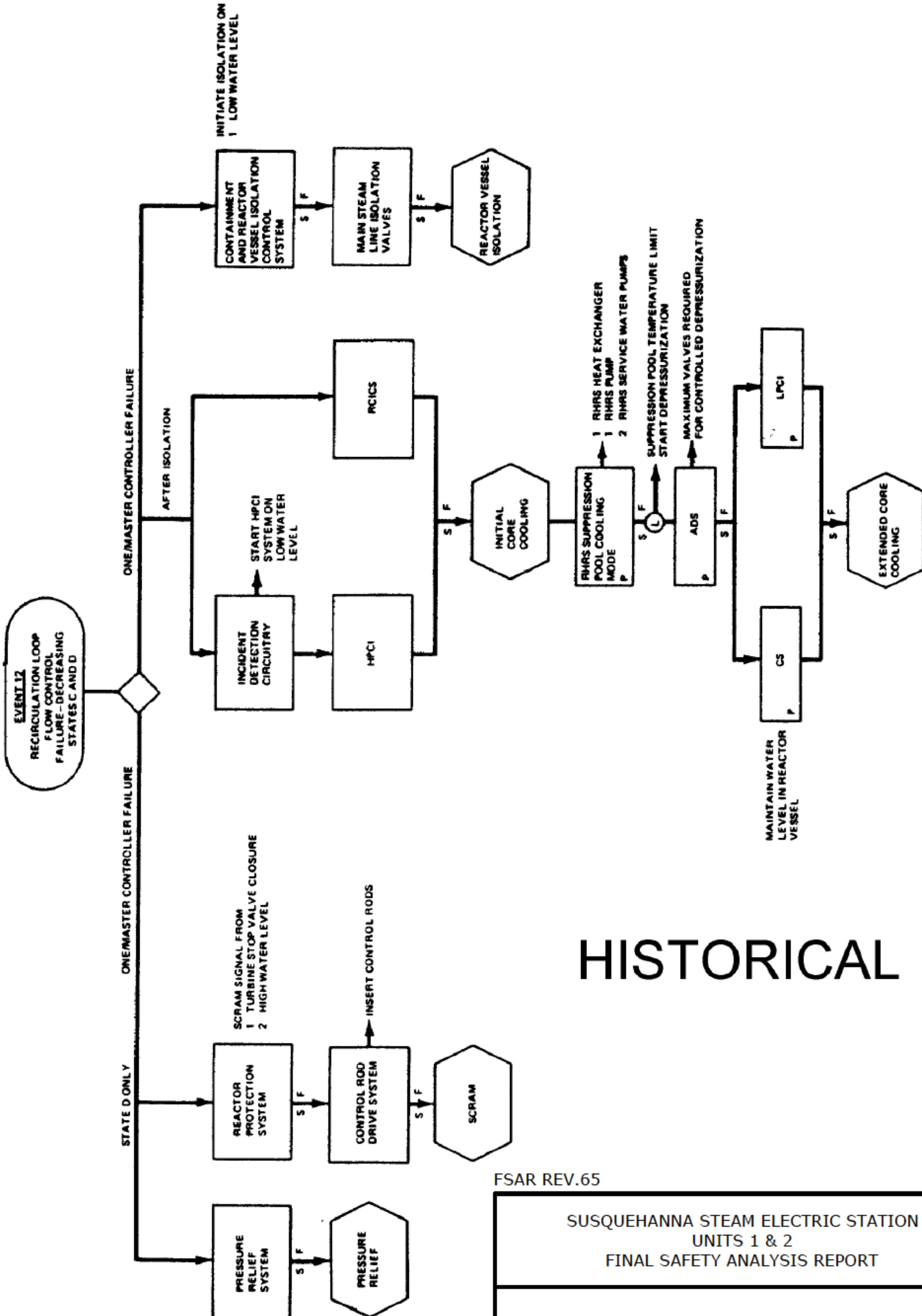
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
RECIRCULATION LOOP FLOW
CONTROL FAILURE-MAXIMUM
DEMAND

FIGURE 15A.6-11, Rev 55

AutoCAD: Figure Fsar 15A_6_11.dwg



HISTORICAL

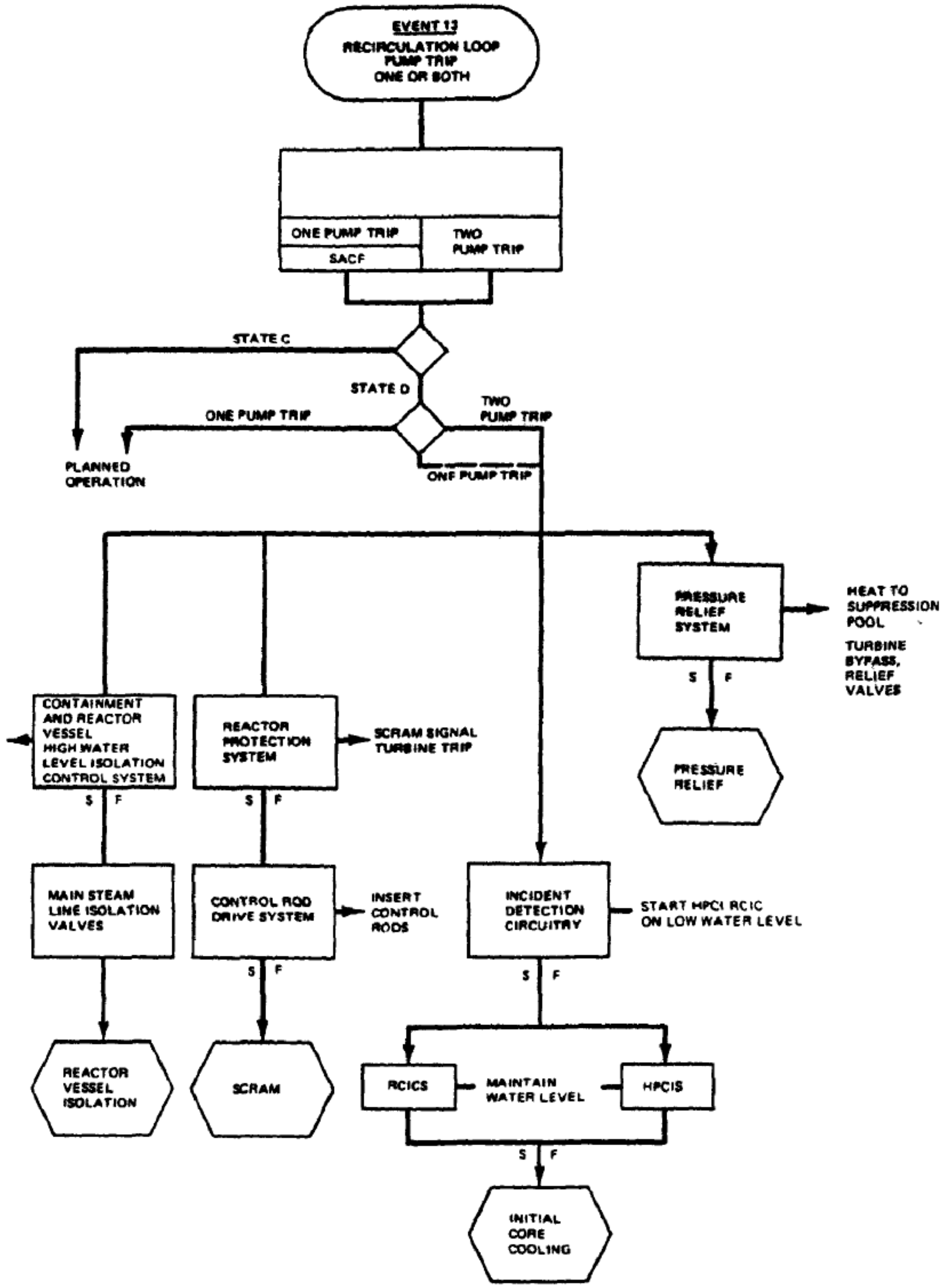
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
RECIRCULATION LOOP FLOW
CONTROL FAILURE-DECREASING

FIGURE 15A.6-12, Rev 55

AutoCAD: Figure Fsar 15A_6_12.dwg



FSAR REV.65

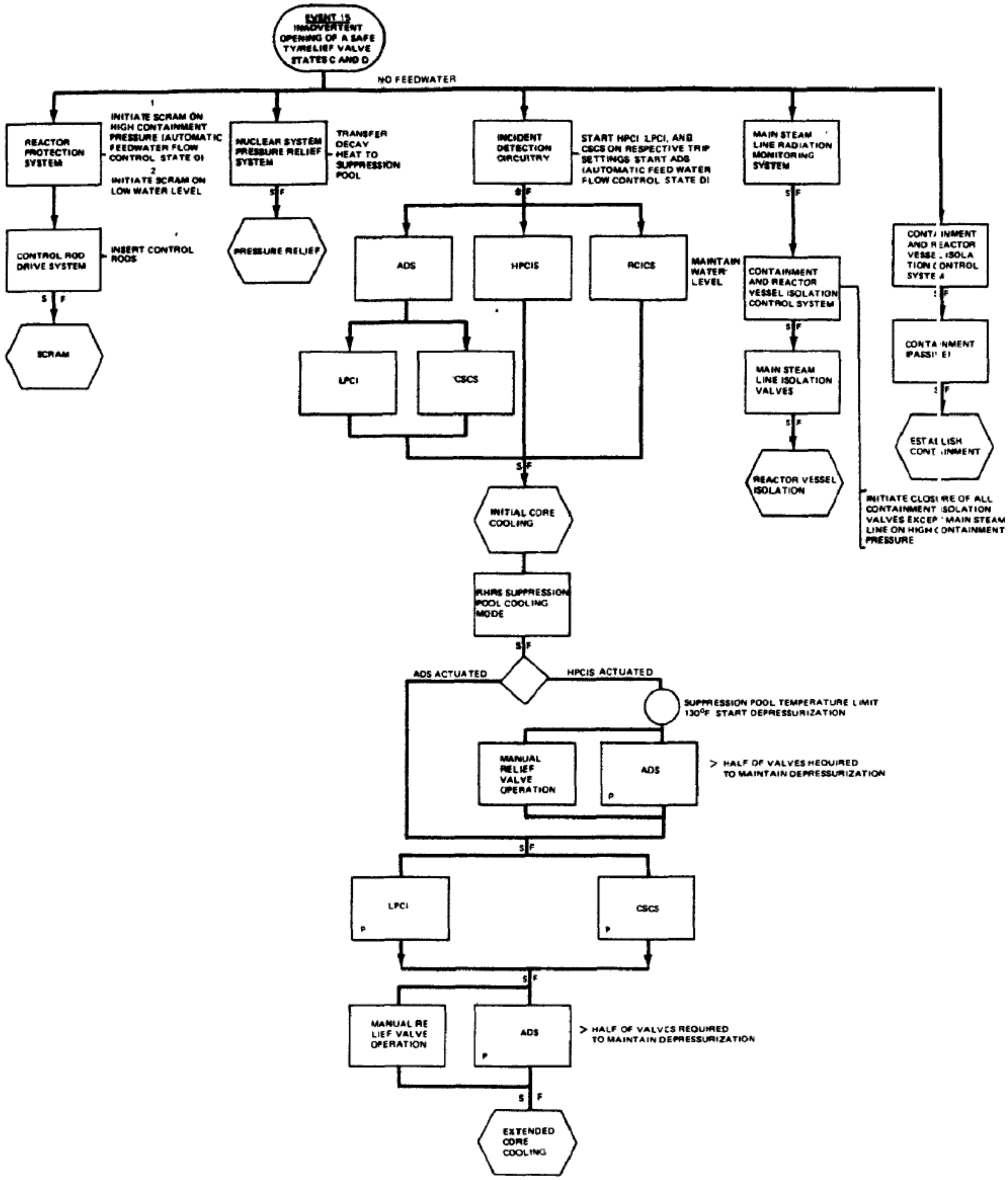
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION LOOP PUMP TRIP-ONE OR BOTH

FIGURE 15A.6-13, Rev 55

AutoCAD: Figure Fsar 15A_6_13.dwg



FSAR REV.65

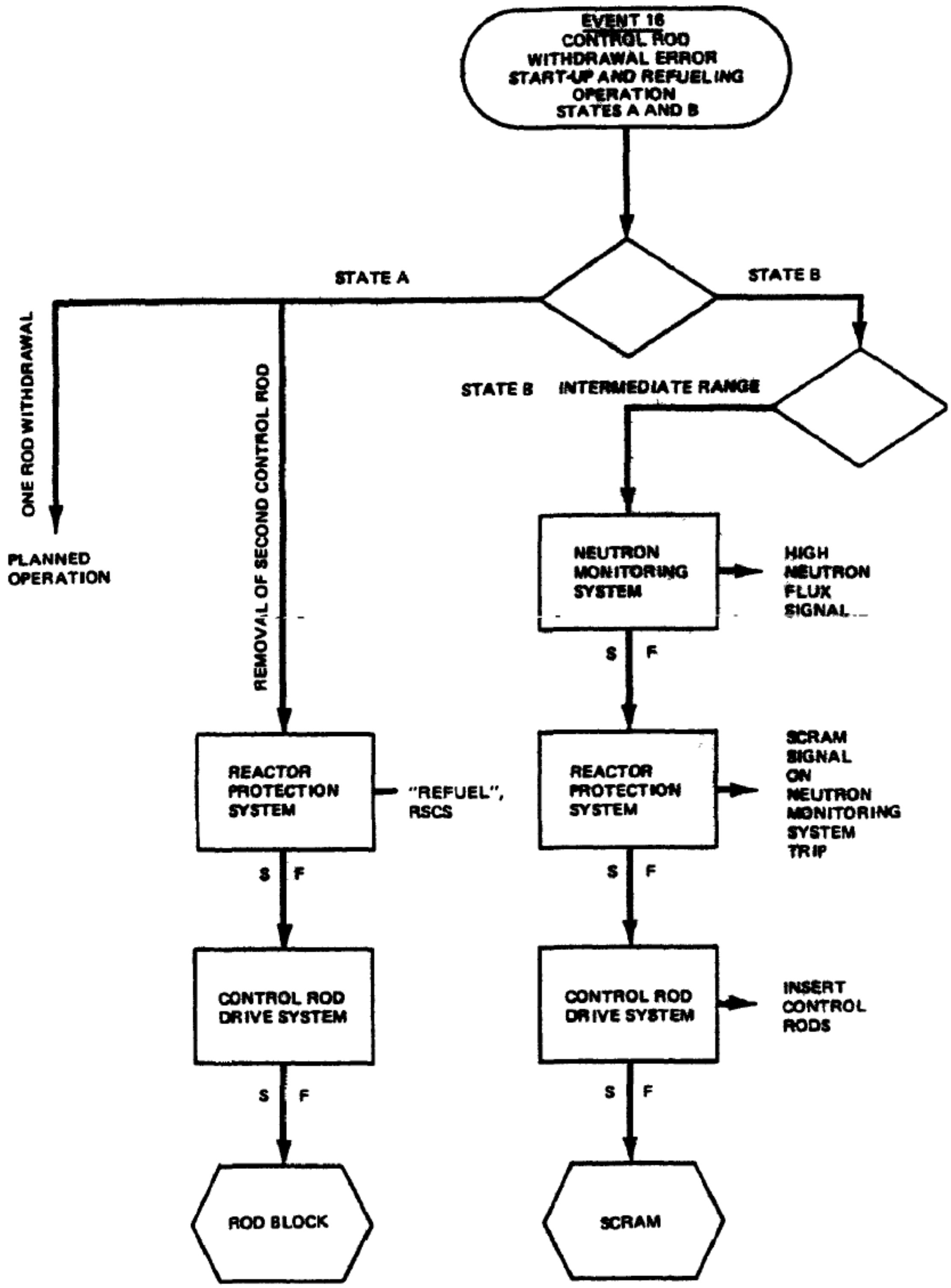
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR INADVERTENT OPENING OF A SAFETY/RELIEF VALVE

FIGURE 15A.6-15, Rev 55

AutoCAD: Figure Fsar 15A_6_15.dwg

HISTORICAL



HISTORICAL

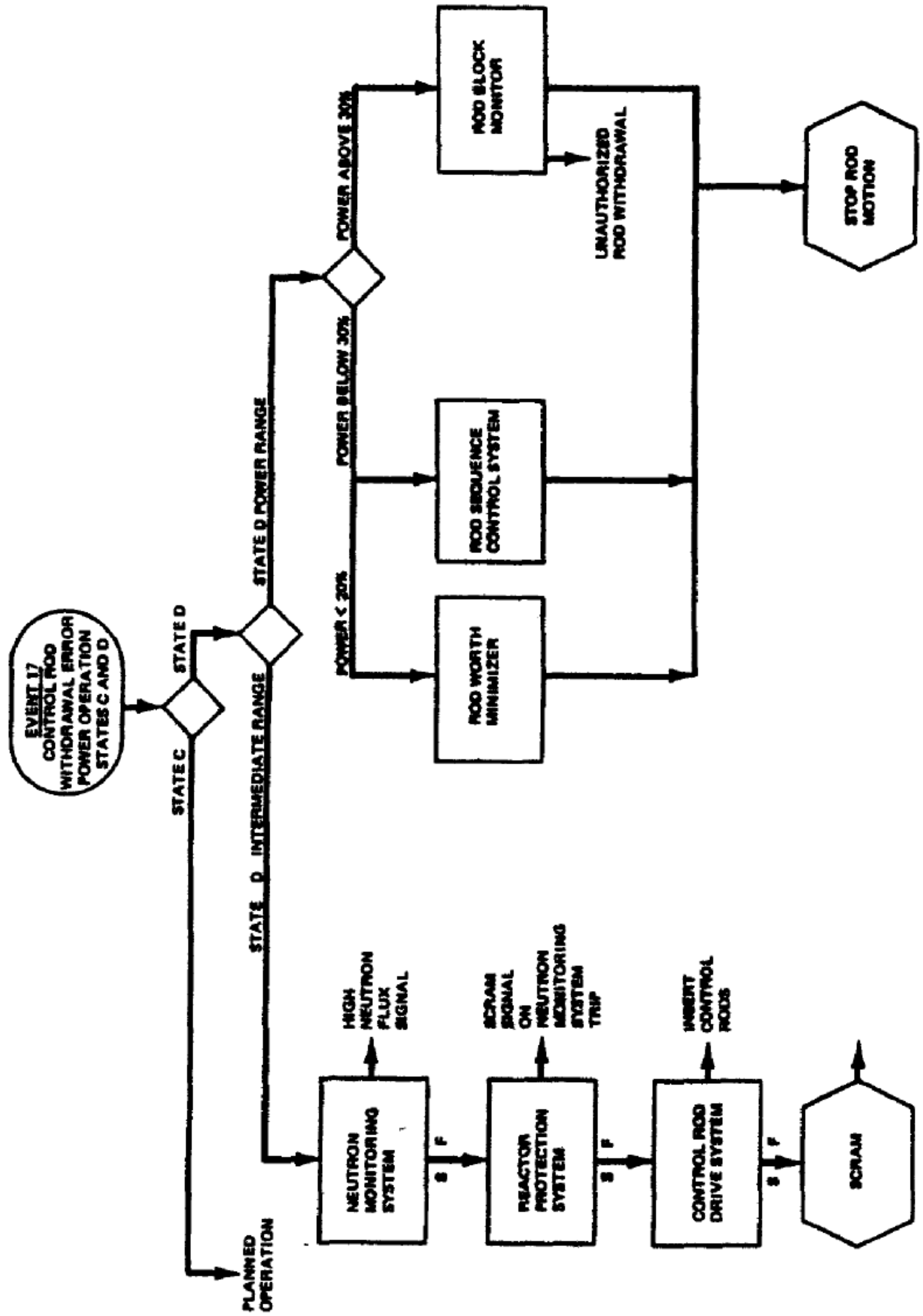
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
CONTROL ROD WITHDRAWAL
ERROR-STARTUP AND
REFUELING OPERAITON

FIGURE 15A.6-16, Rev 55

AutoCAD: Figure Fsar 15A_6_16.dwg



HISTORICAL

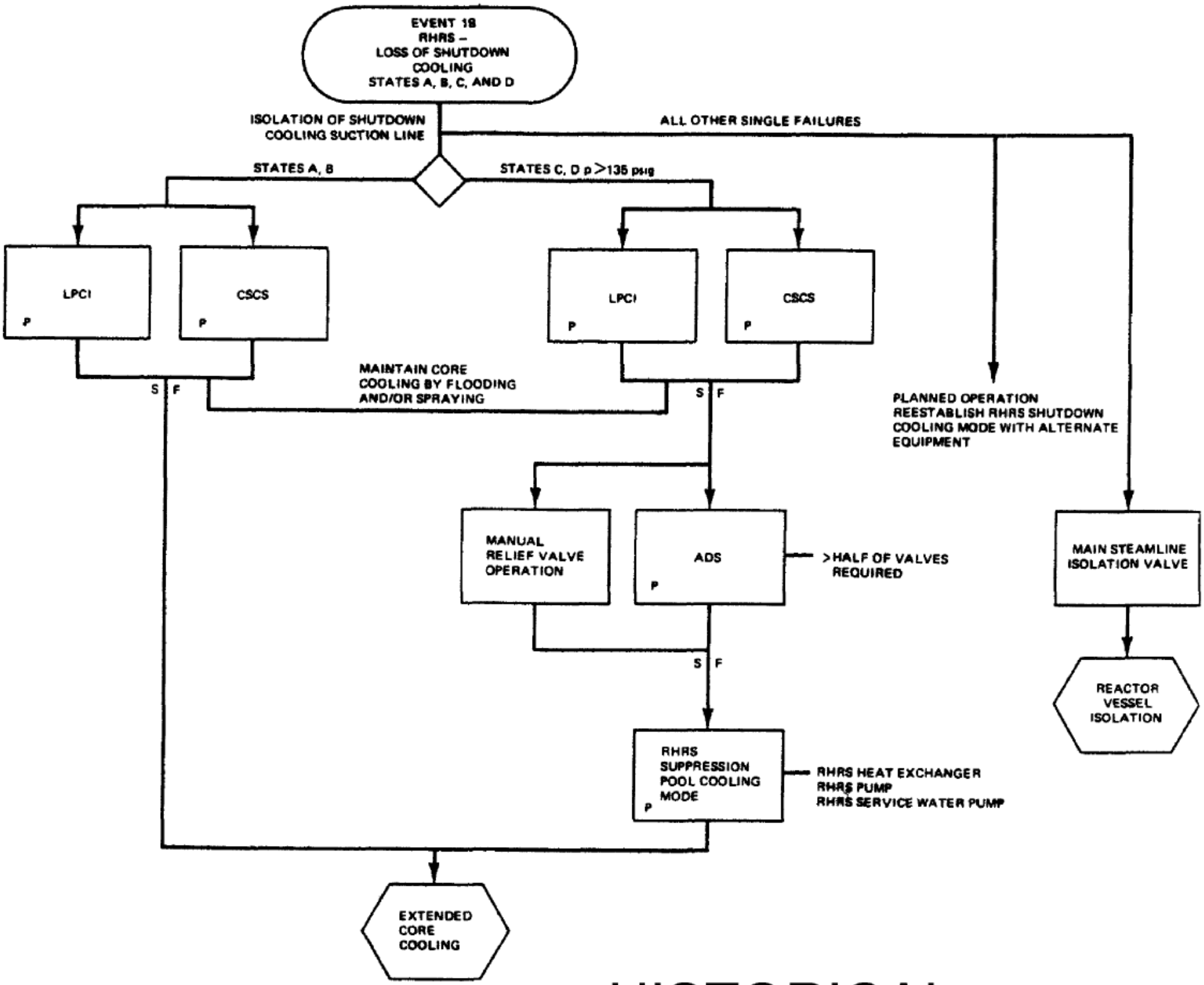
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
CONTROL ROD WITHDRAWAL
ERROR-POWER OPERATION

FIGURE 15A.6-17, Rev 55

AutoCAD: Figure Fsar 15A_6_17.dwg



HISTORICAL

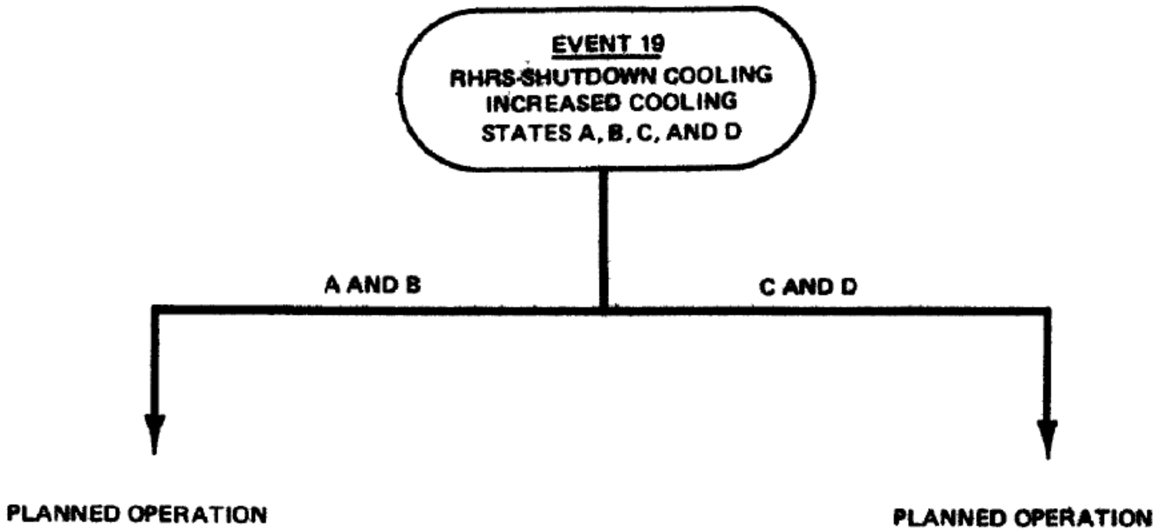
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
RHR'S - LOSS OF
SHUTDOWN COOLING FAILURE

FIGURE 15A.6-18, Rev 55

AutoCAD: Figure Fsar 15A_6_18.dwg



HISTORICAL

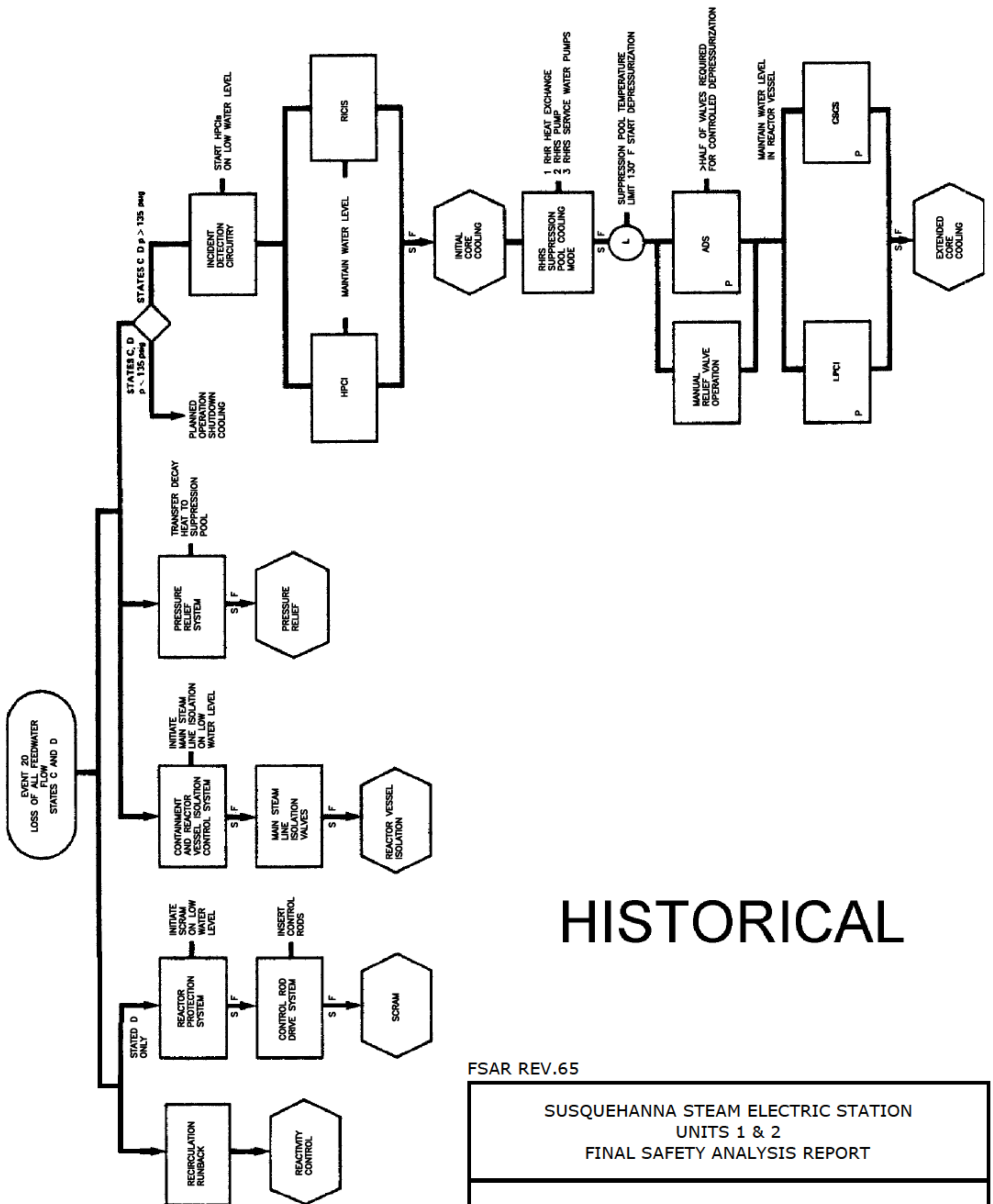
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RHR'S - SHUTDOWN COOLING
FAILURE-
INCREASED COOLING

FIGURE 15A.6-19, Rev 55

AutoCAD: Figure Fsar 15A_6_19.dwg



HISTORICAL

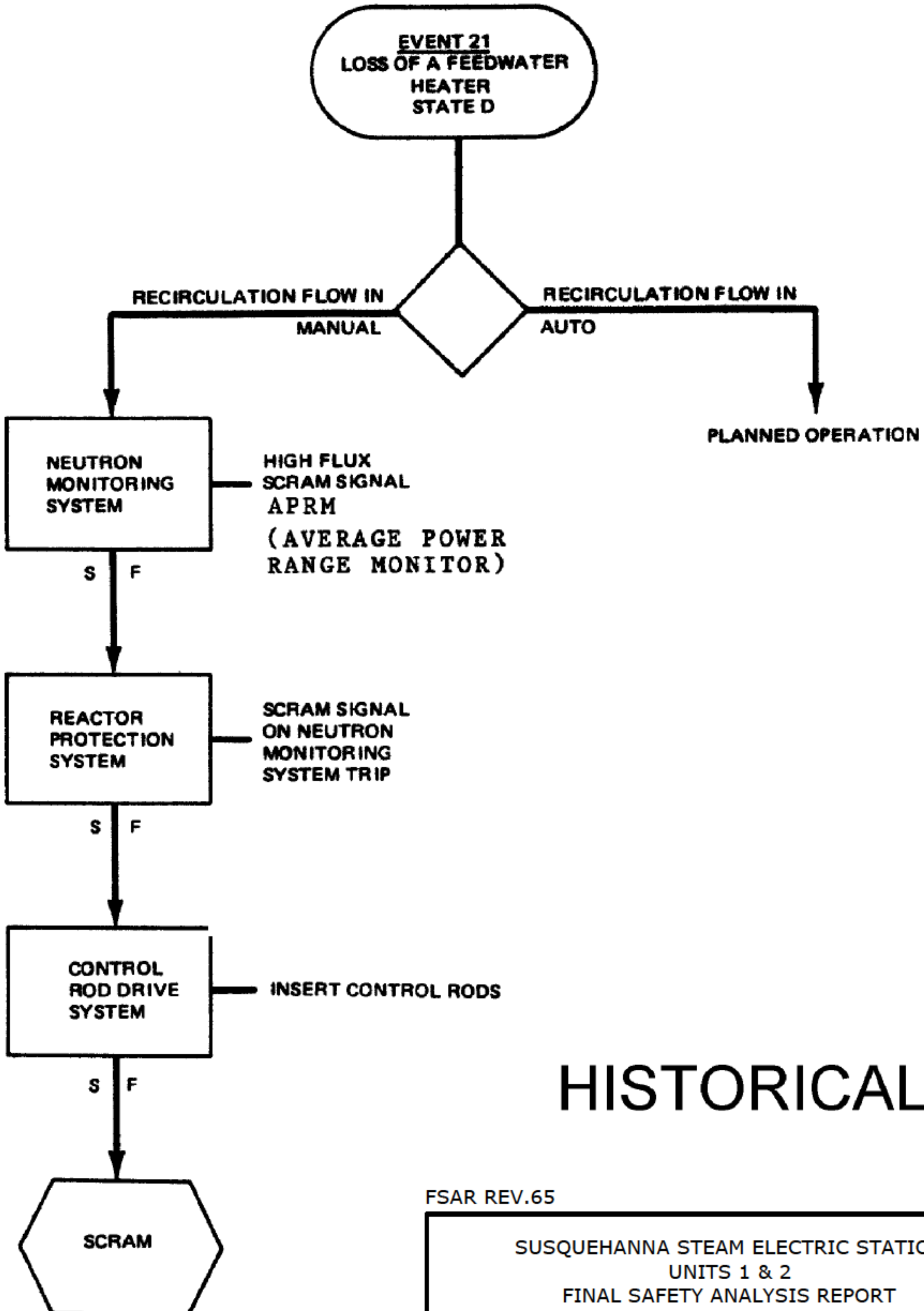
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
LOSS OF FEEDWATER FLOW

FIGURE 15A.6-20, Rev 55

AutoCAD: Figure Fsar 15A_6_20.dwg



HISTORICAL

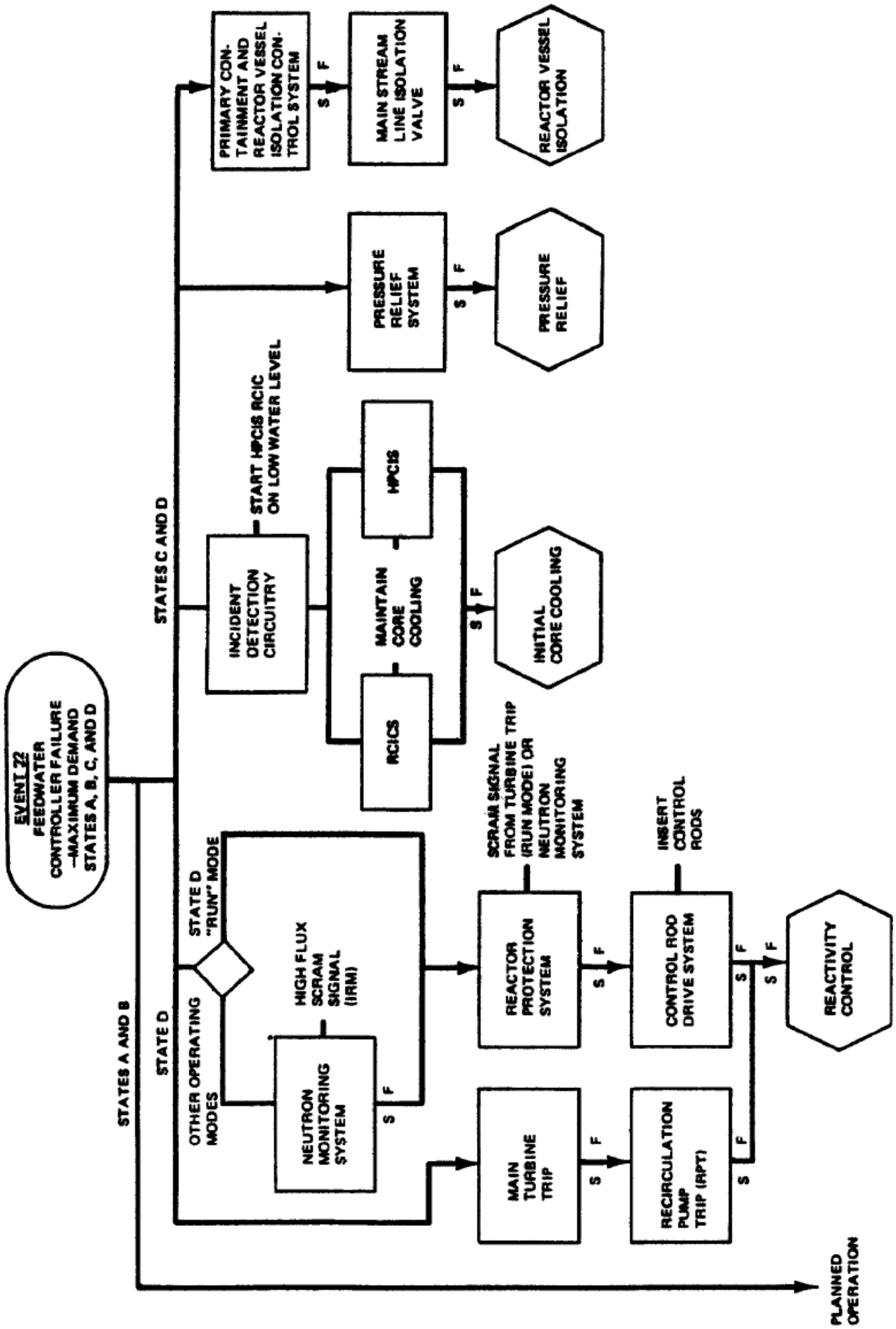
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
LOSS OF A FEEDWATER HEATER

FIGURE 15A.6-21, Rev 55

AutoCAD: Figure Fsar 15A_6_21.dwg



FSAR REV.65

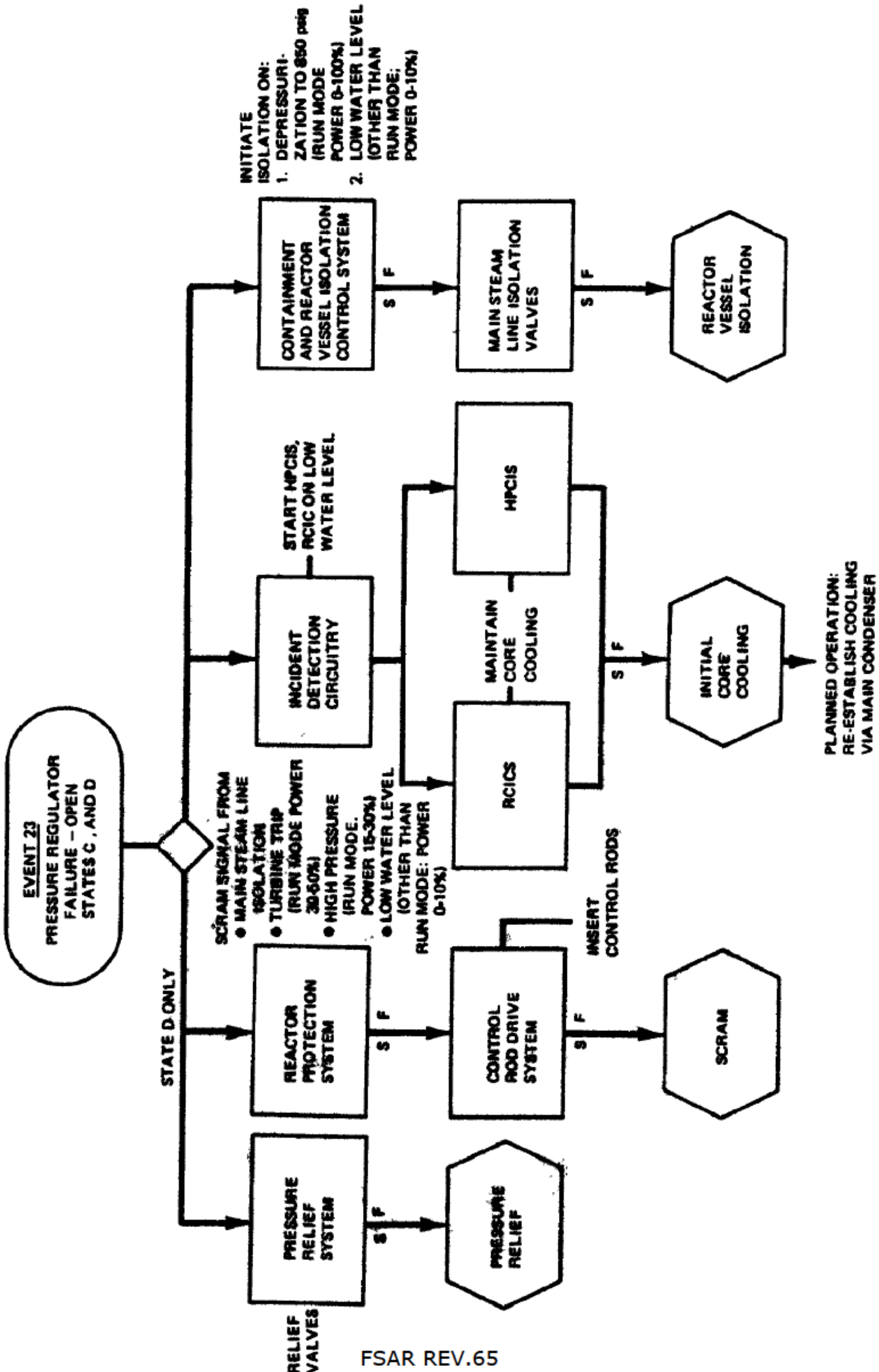
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
FEEDWATER CONTROLLER FAILURE
MAXIMUM DEMAND

FIGURE 15A.6-22, Rev 55

AutoCAD: Figure Fsar 15A_6_22.dwg

HISTORICAL



FSAR REV.65

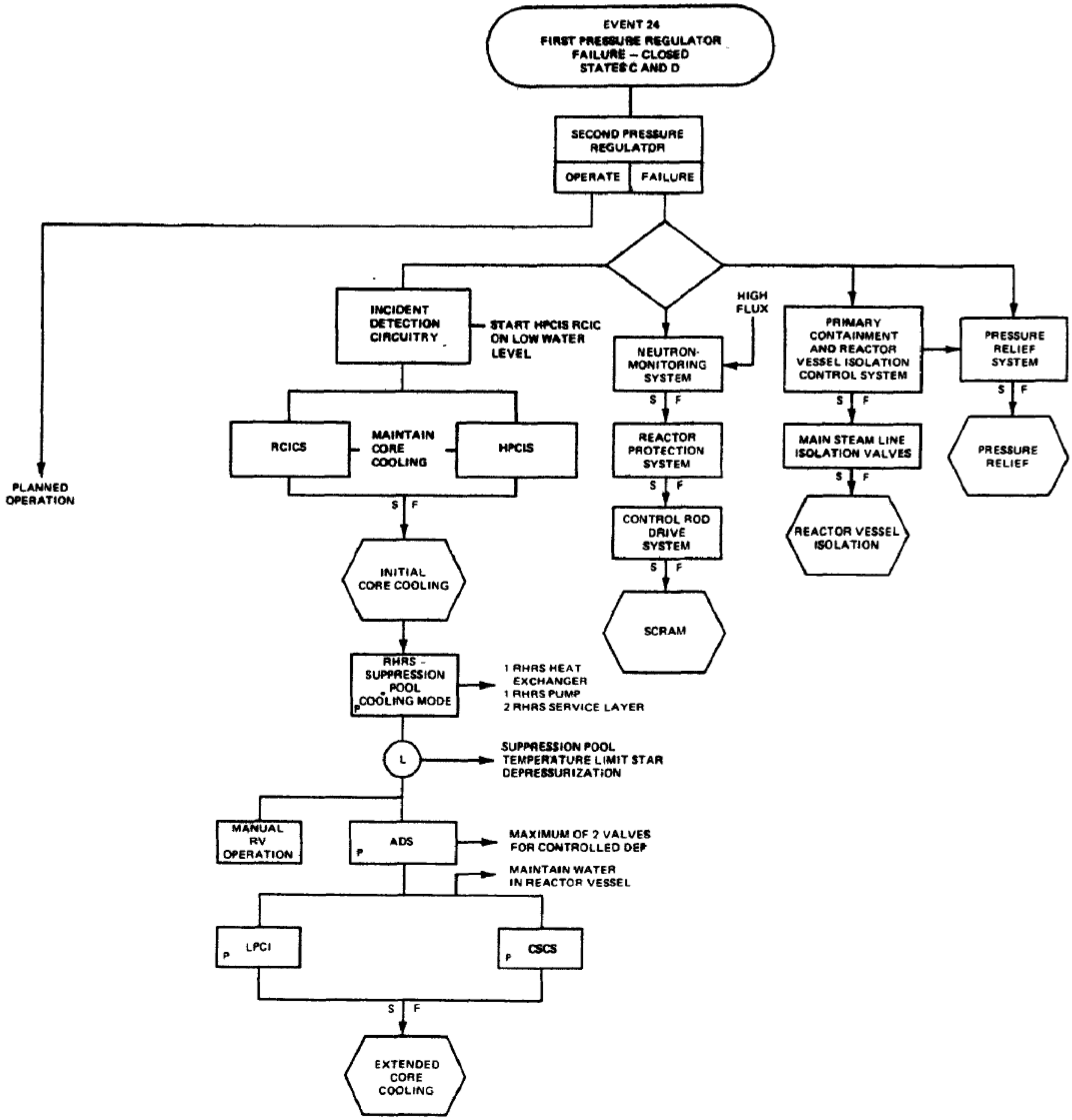
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
PRESSURE REGULATOR
FAILURE - OPEN

FIGURE 15A.6-23, Rev 55

AutoCAD: Figure Fsar 15A_6_23.dwg

HISTORICAL



FSAR REV.65

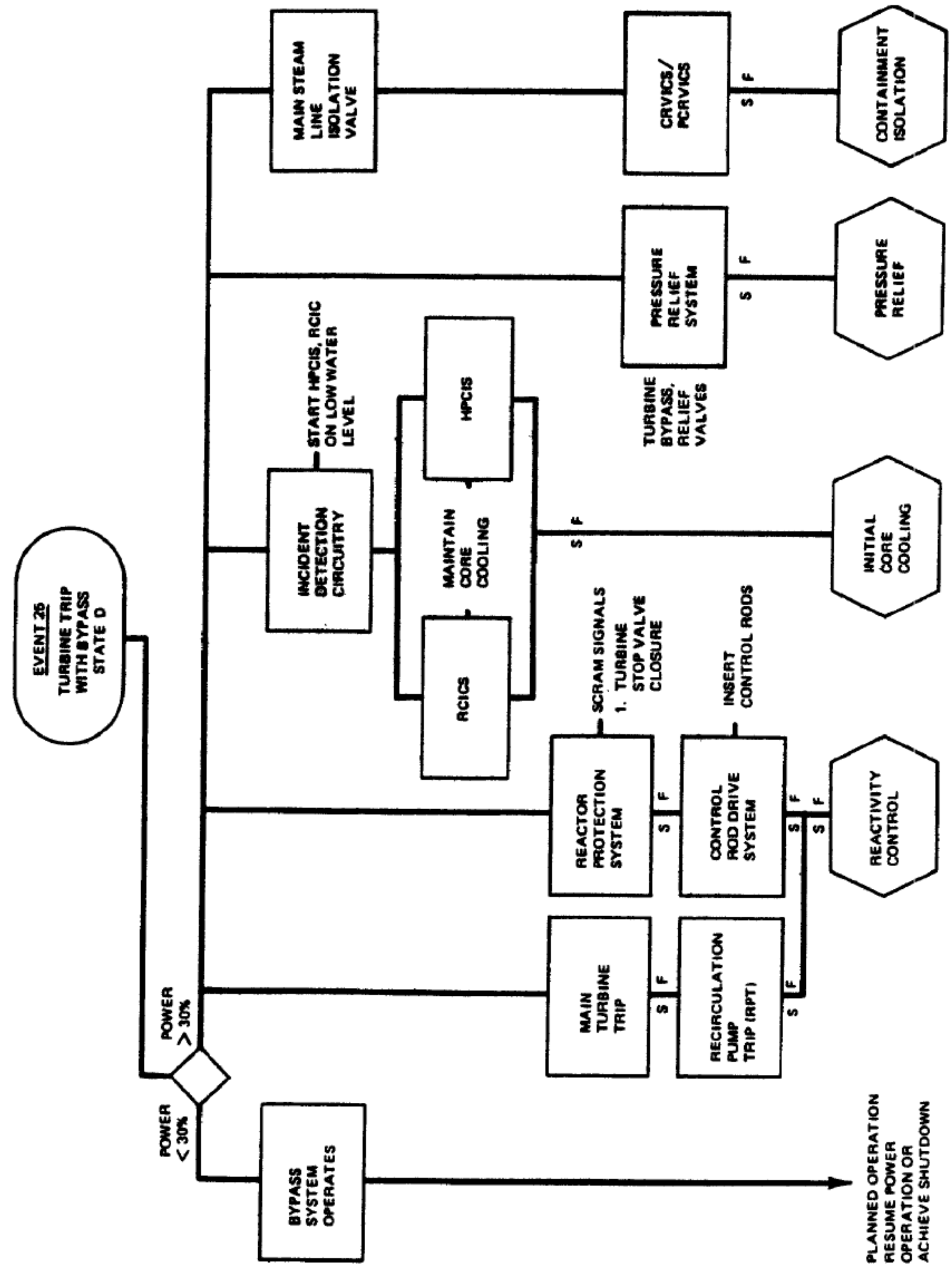
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR PRESSURE REGULATOR FAILURE - CLOSED

FIGURE 15A.6-24, Rev 55

AutoCAD: Figure Fsar 15A_6_24.dwg



HISTORICAL

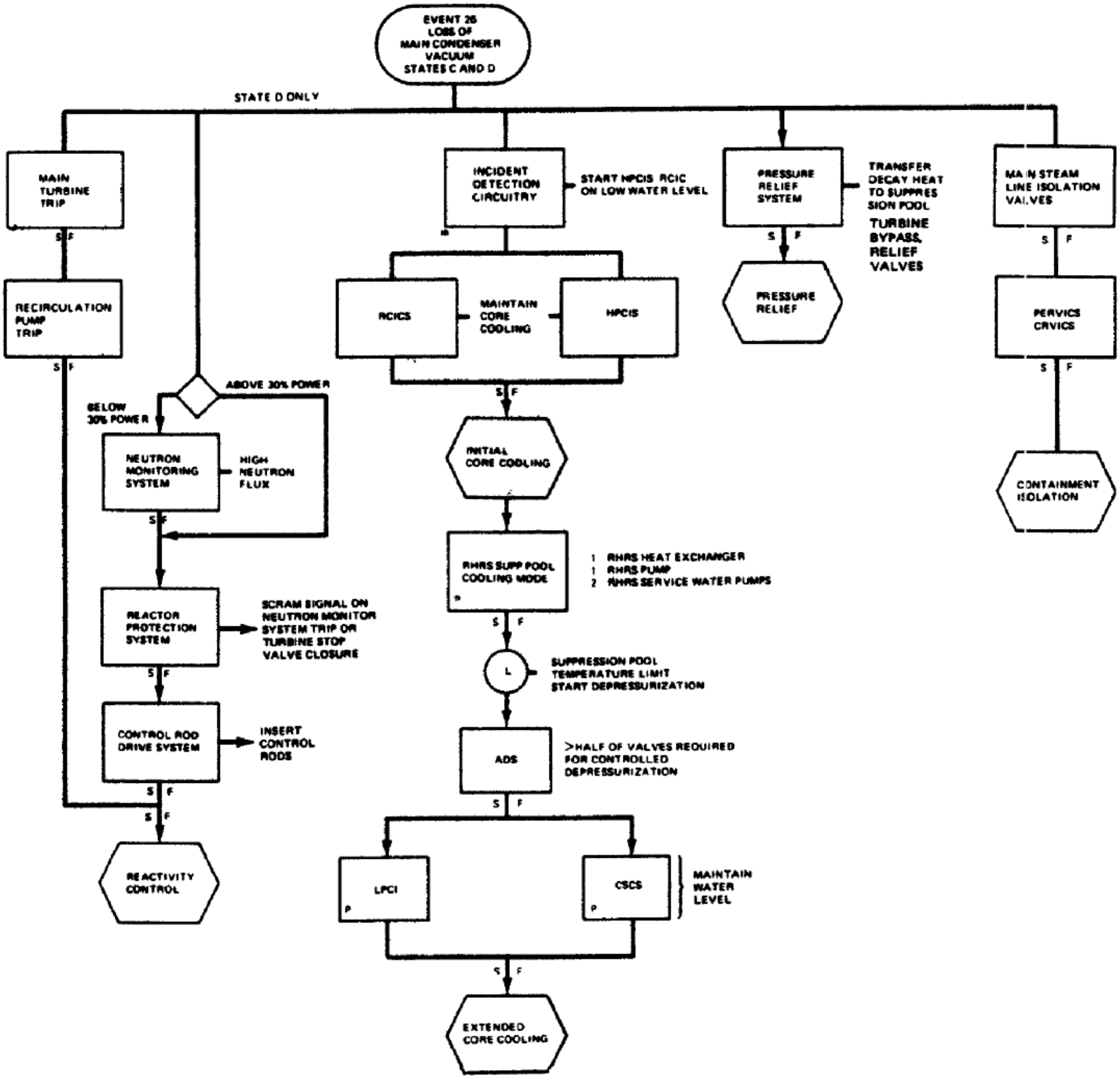
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
MAIN TURBINE TRIP
WITH BYPASS

FIGURE 15A.6-25, Rev 55

AutoCAD: Figure Fsar 15A_6_25.dwg



HISTORICAL

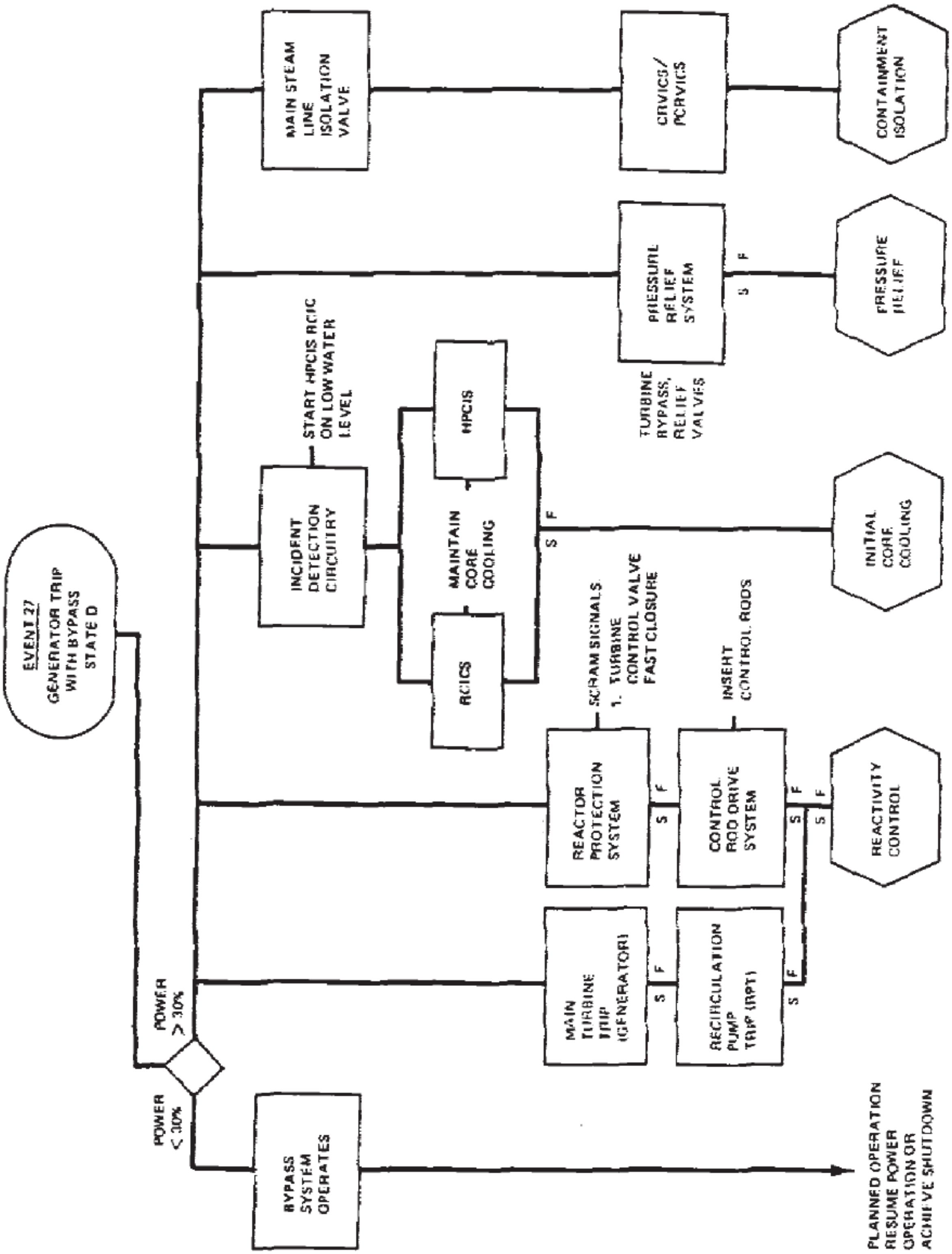
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCEs FOR
LOSS OF MAIN
CONDENSER VACUUM

FIGURE 15A.6-26, Rev 55

AutoCAD: Figure Fsar 15A_6_26.dwg



FSAR REV.65

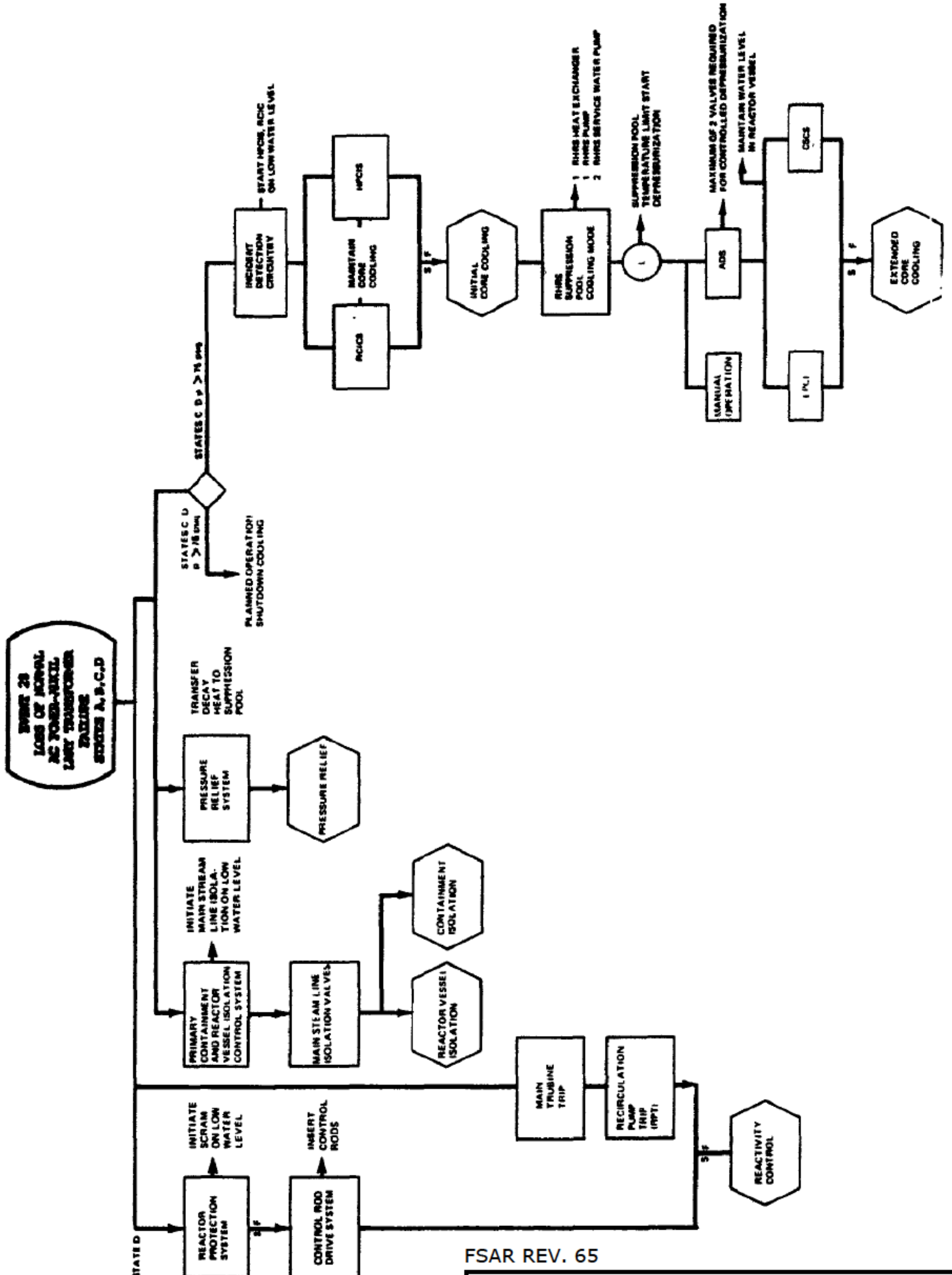
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
MAIN GENERATOR TRIP
WITH BYPASS

FIGURE 15A.6-27, Rev 55

AutoCAD: Figure Fsar 15A_6_27.dwg



HISTORICAL

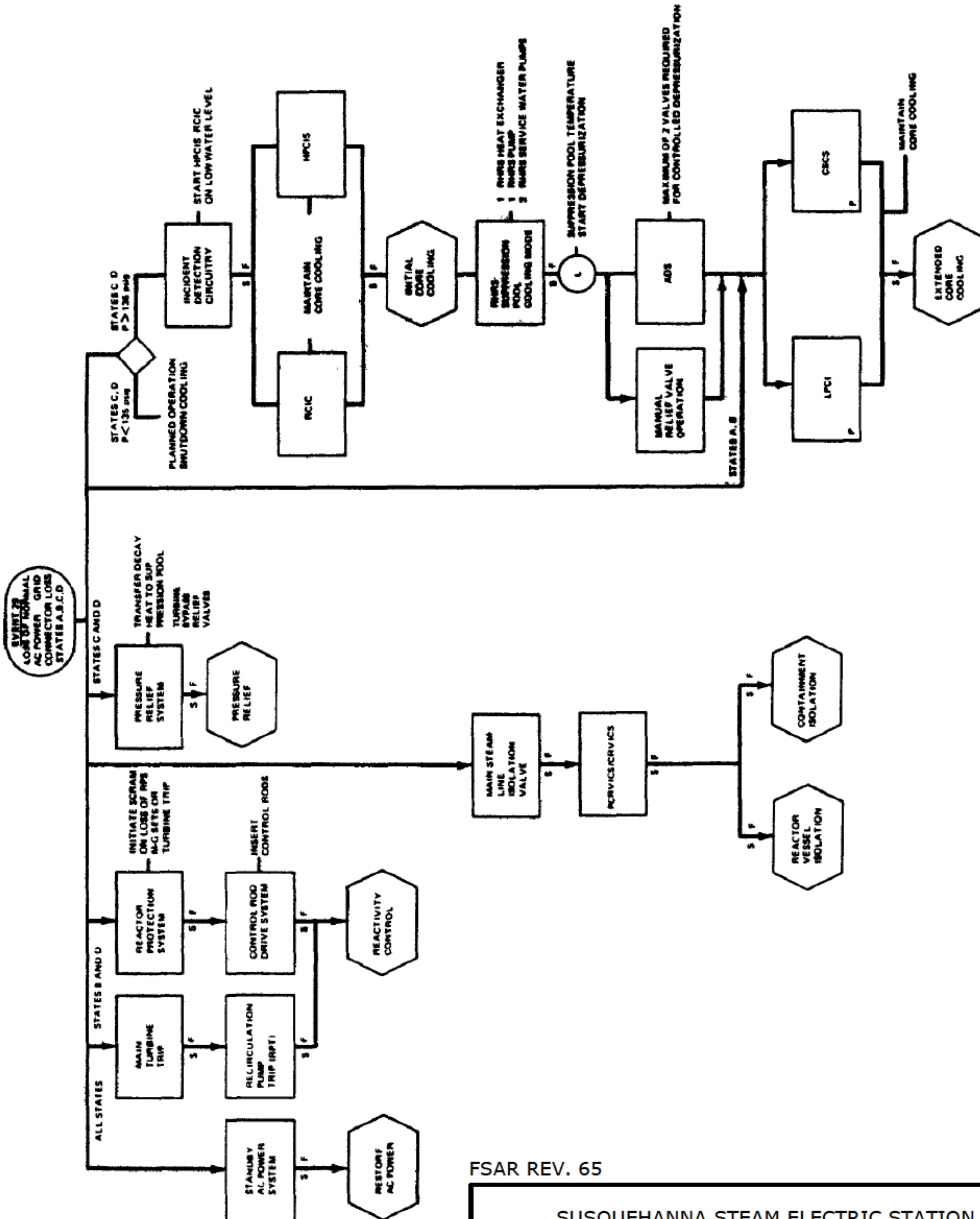
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR LOSS
OF NORMAL AC POWER
AUX TRANSFORMER FAILURE

FIGURE 15A.6-28, Rev 55

AutoCAD: Figure Fsar 15A_6_28.dwg



FSAR REV. 65

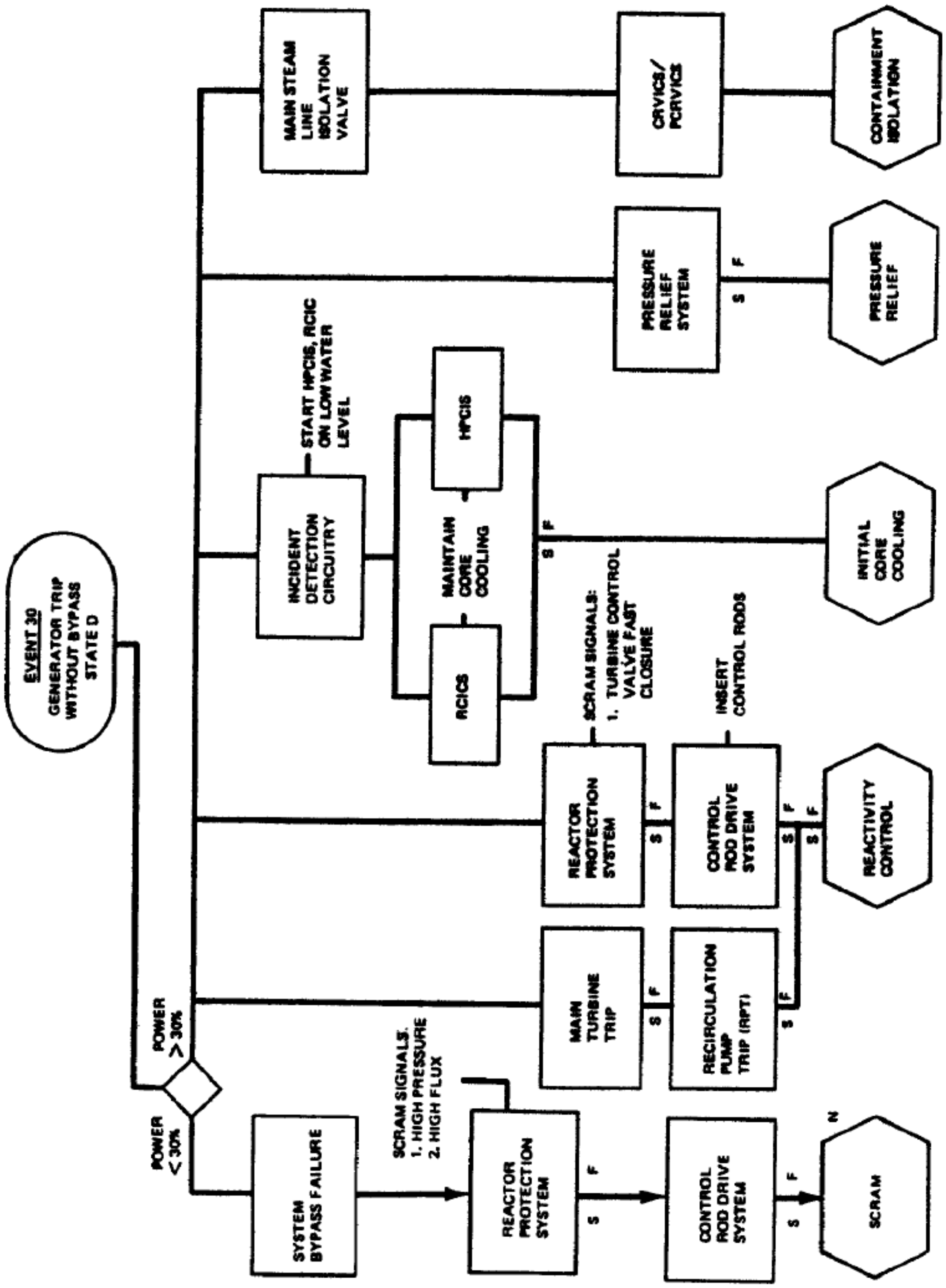
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
LOSS OF NORMAL AC POWER-
GRID CONNECTION LOSS

FIGURE 15A.6-29, Rev 55

AutoCAD: Figure Fsar 15A_6_29.dwg

HISTORICAL



HISTORICAL

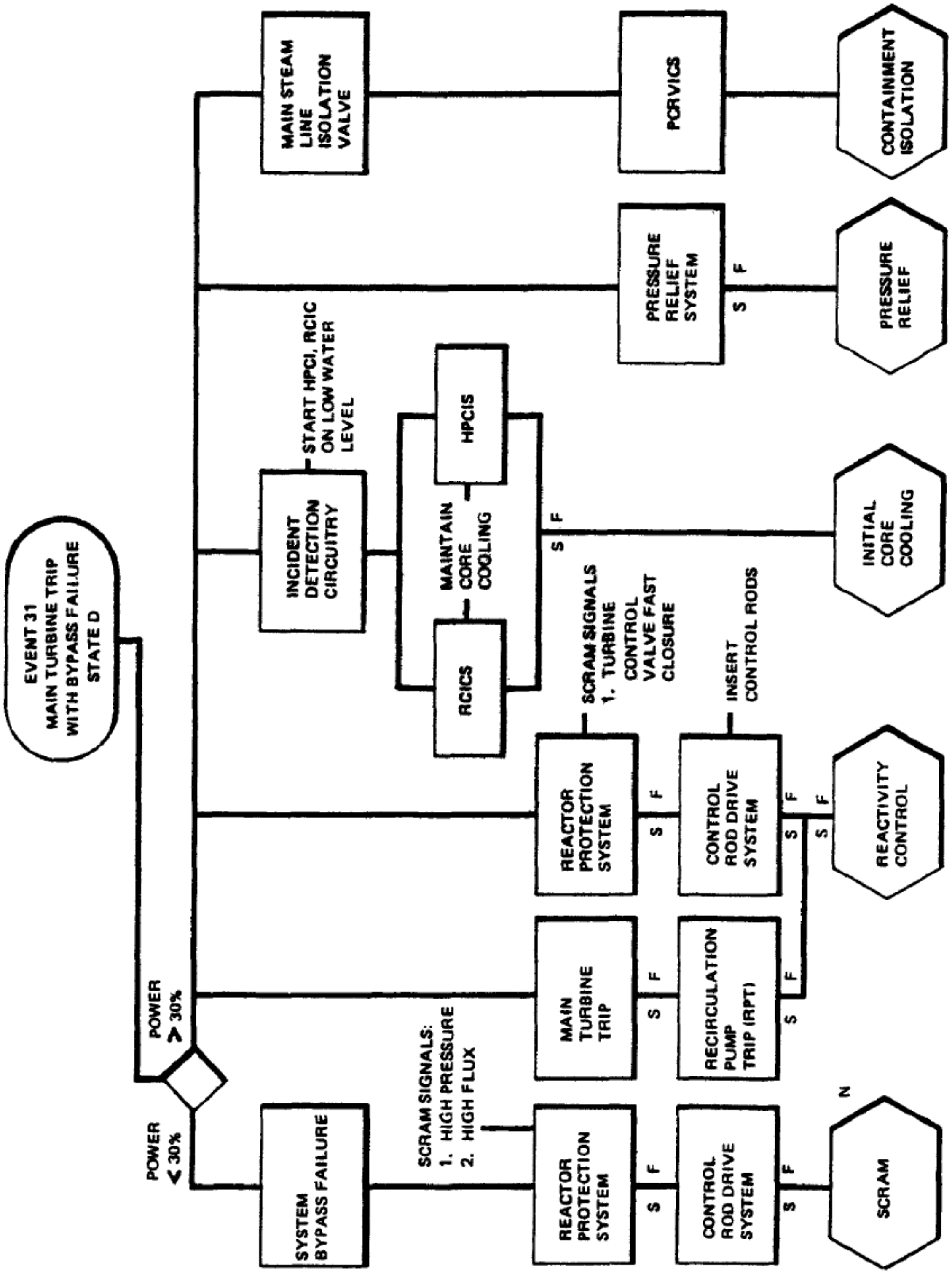
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES
MAIN GENERATOR TRIP -
WITHOUT BYPASS

FIGURE 15A.6-30, Rev 55

AutoCAD: Figure Fsar 15A_6_30.dwg

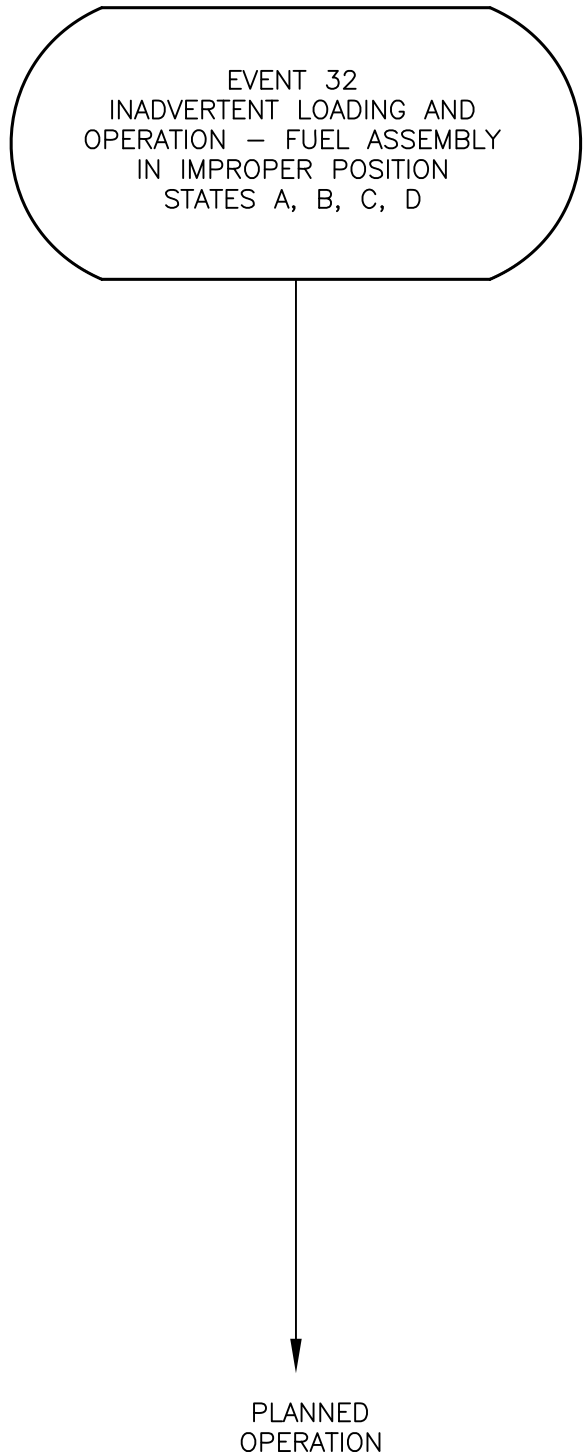


FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
PROTECTION SEQUENCES MAIN TURBINE TRIP - WITH BYPASS FAILURE FIGURE 15A.6-31, Rev 55

AutoCAD: Figure Fsar 15A_6_31.dwg



FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
INADVERTENT LOADING AND
OPERATIONS OF FUEL ASSEMBLY
IN IMPROPER POSITION

FIGURE 15A.6-32, Rev 55

AutoCAD: Figure Fsar 15A_6_32.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure intentionally left blank FIGURE 15A.6-33, Rev. 55

AutoCAD Figure 15A_6_33.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15A.6-34, Rev. 55

AutoCAD Figure 15A_6_34.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15A.6-35, Rev. 55

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15A.6-36, Rev. 55

AutoCAD Figure 15A_6_36.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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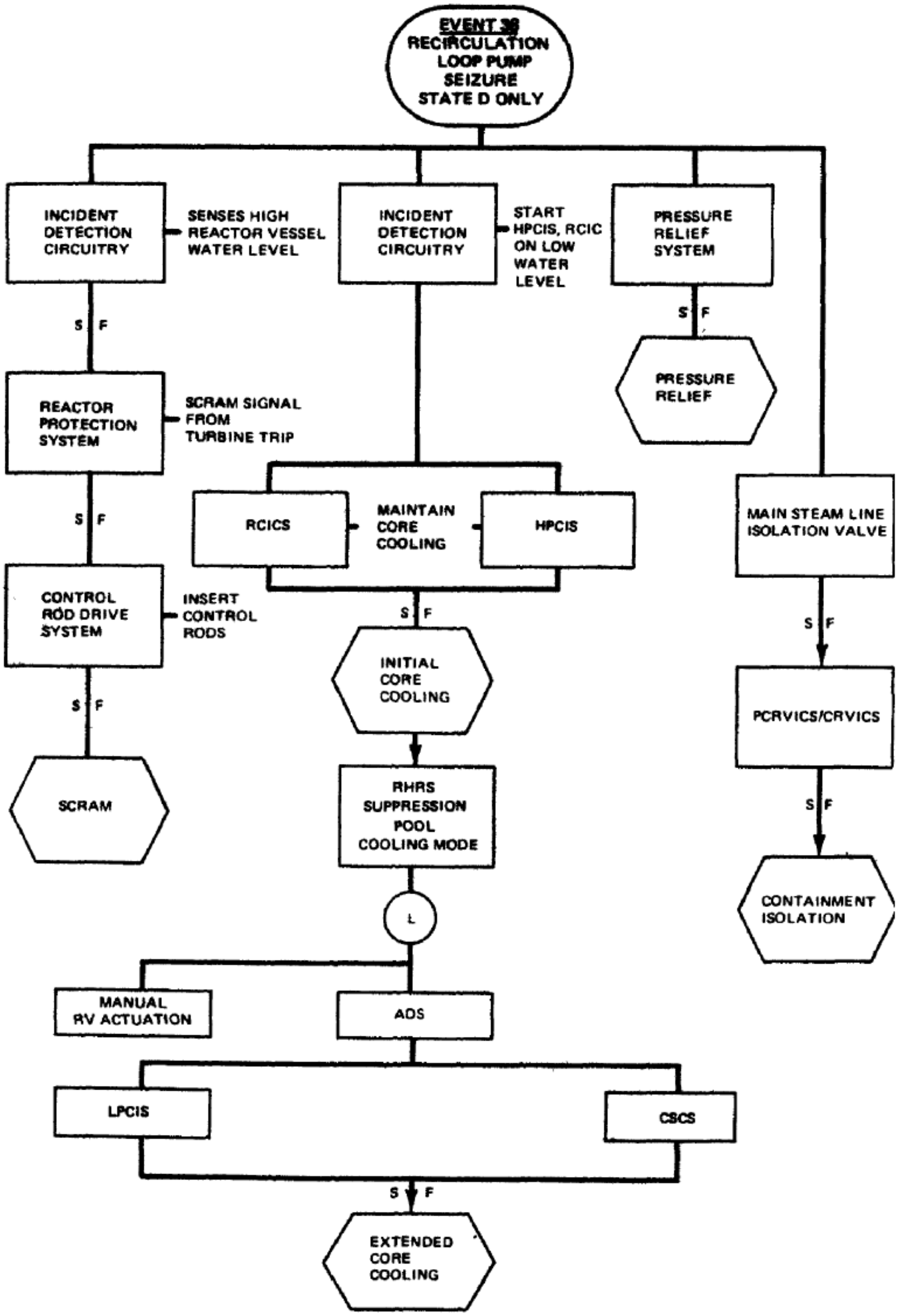
FIGURE 15A.6-37, Rev. 55



HISTORICAL

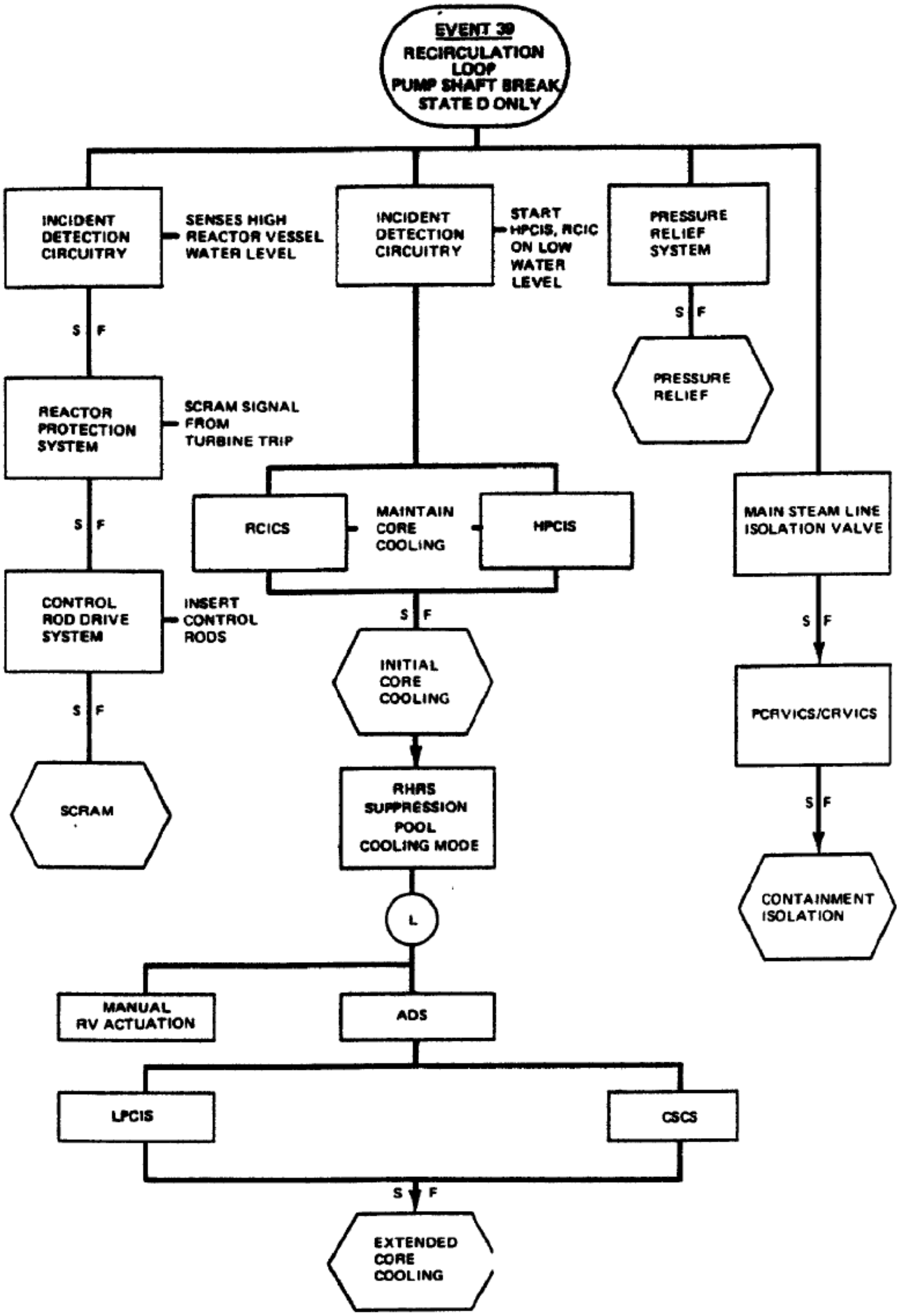
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
RECIRCULATION LOOP PUMP
SEIZURE

FIGURE 15A.6-38, Rev 55

AutoCAD: Figure Fsar 15A_6_38.dwg



FSAR REV. 65

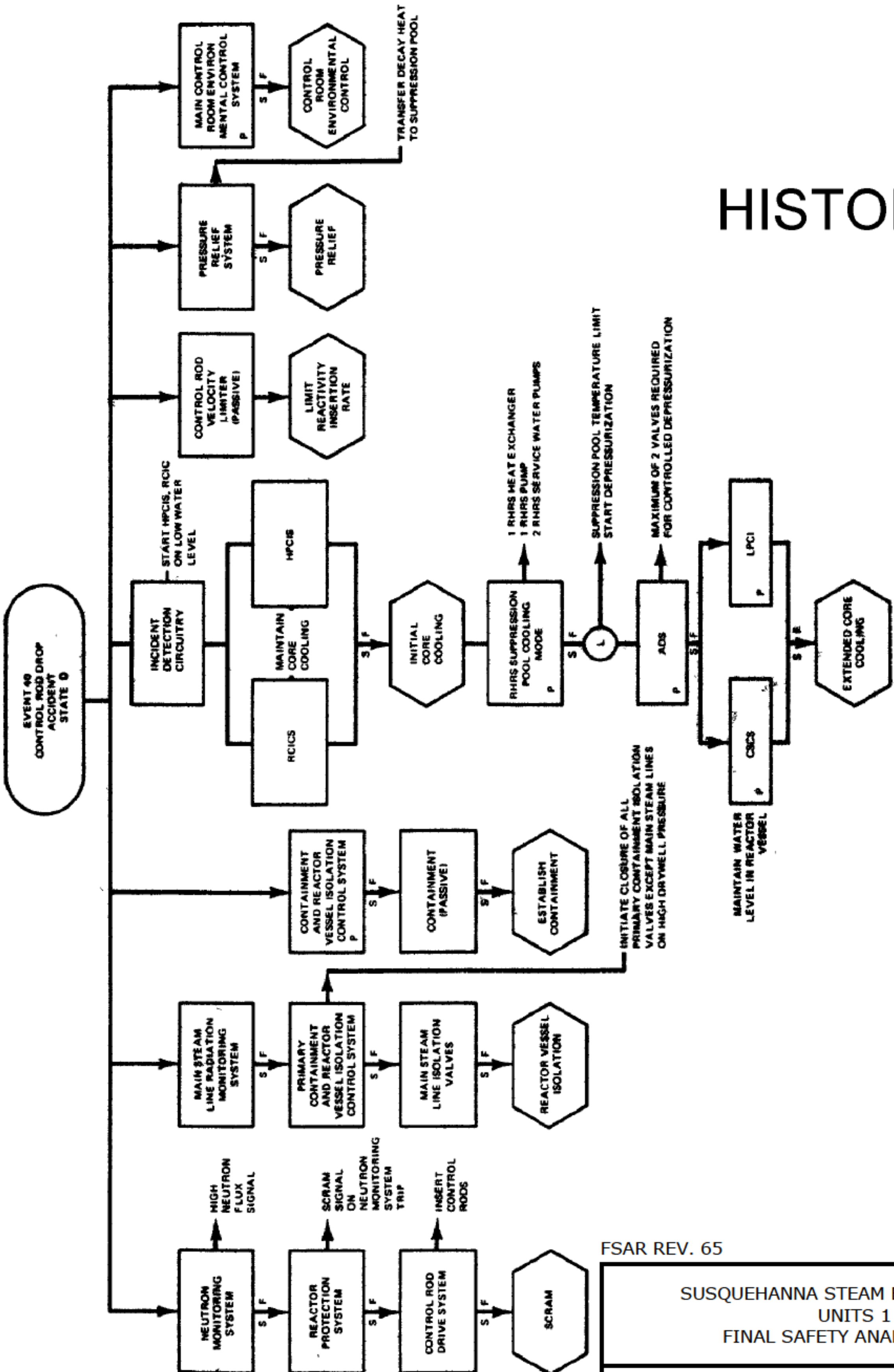
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
RECIRCULATION LOOP
PUMP SHAFT BREAK

FIGURE 15A.6-39, Rev 55

AutoCAD: Figure Fsar 15A_6_39.dwg

HISTORICAL



HISTORICAL

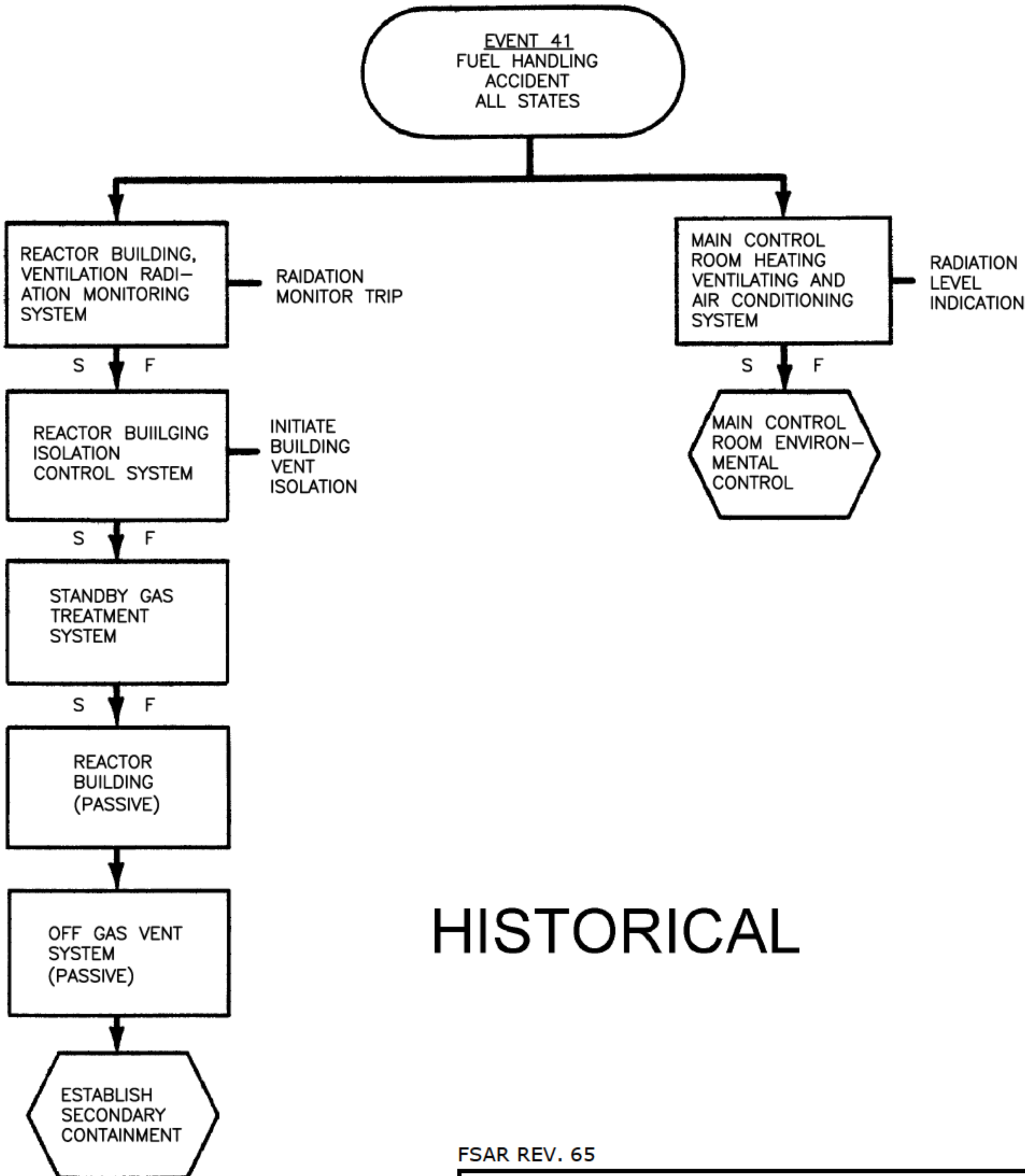
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
CONTROL ROD
DROP ACCIDENT

FIGURE 15A.6-40, Rev 55

AutoCAD: Figure Fsar 15A_6_40.dwg



HISTORICAL

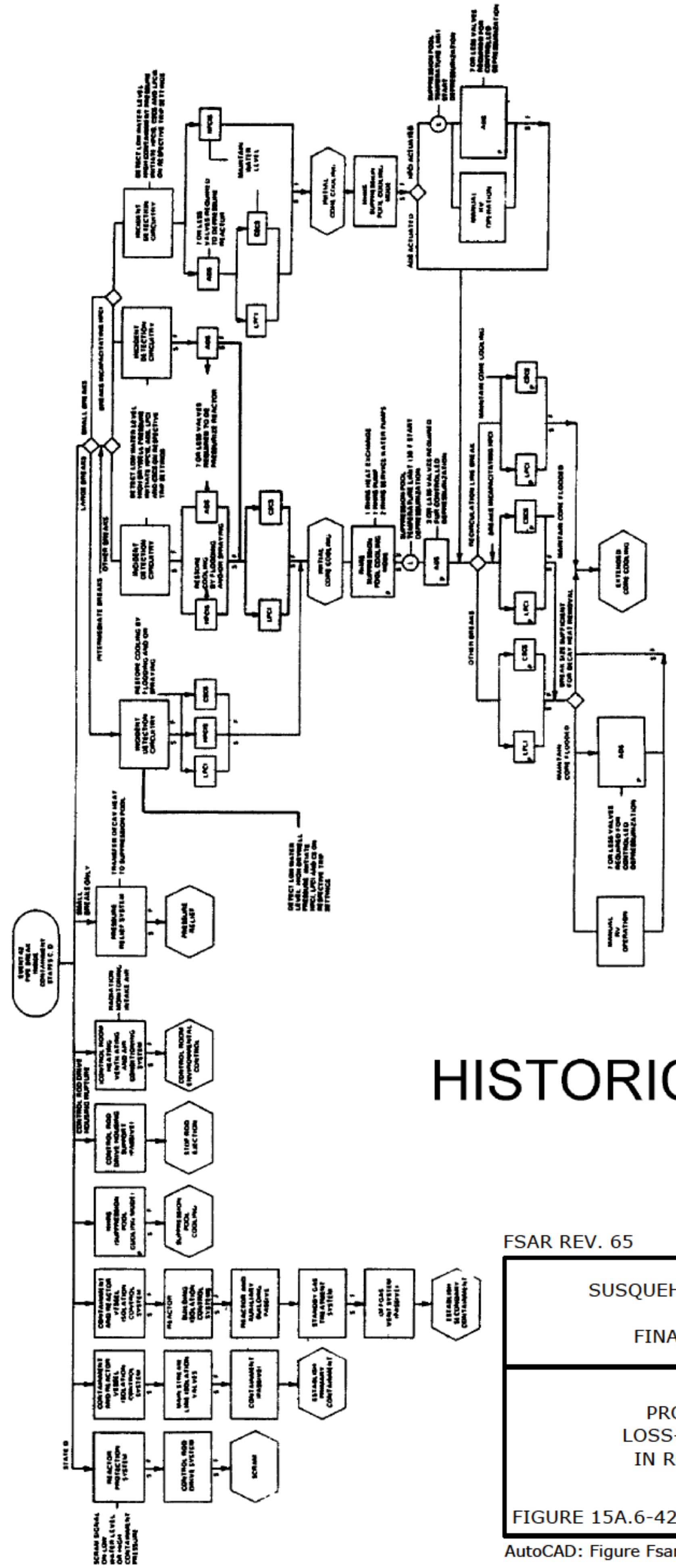
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES
FOR
FUEL HANDLING ACCIDENT

FIGURE 15A.6-41, Rev 55

AutoCAD: Figure Fsar 15A_6_41.dwg



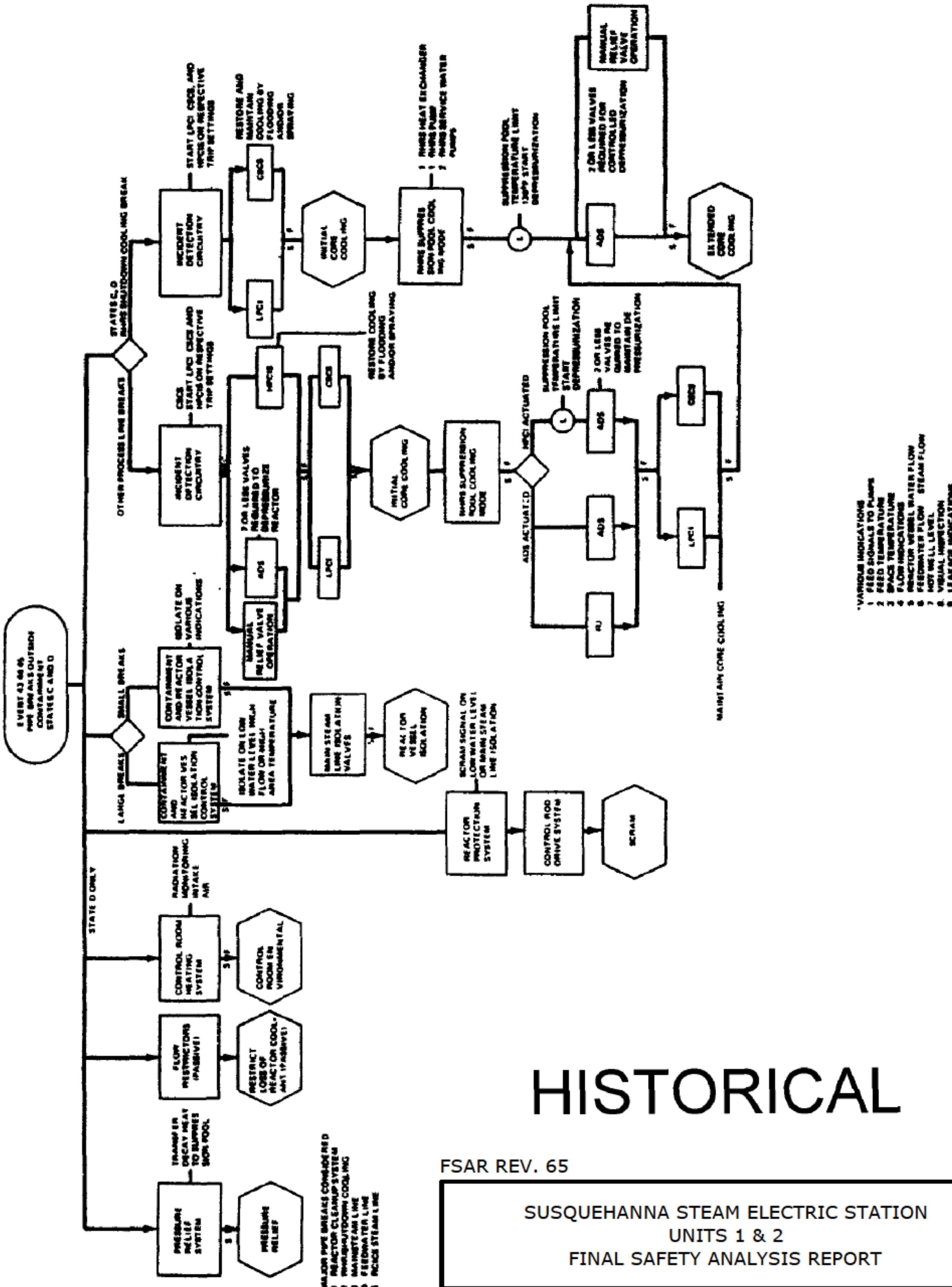
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
LOSS-OF-COOLANT PIPING BREAKS
IN RCPB - INSIDE CONTAINMENT

FIGURE 15A.6-42, Rev 55

AutoCAD: Figure Fsar 15A_6_42.dwg



HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SYSTEM FOR LIQUID,
STEAM, LARGE, SMALL PIPING
BREAKS OUTSIDE CONTAINMENT

FIGURE 15A.6-43, Rev 55

AutoCAD: Figure Fsar 15A_6_43.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15A.6-44, Rev. 55

AutoCAD Figure 15A_6_44.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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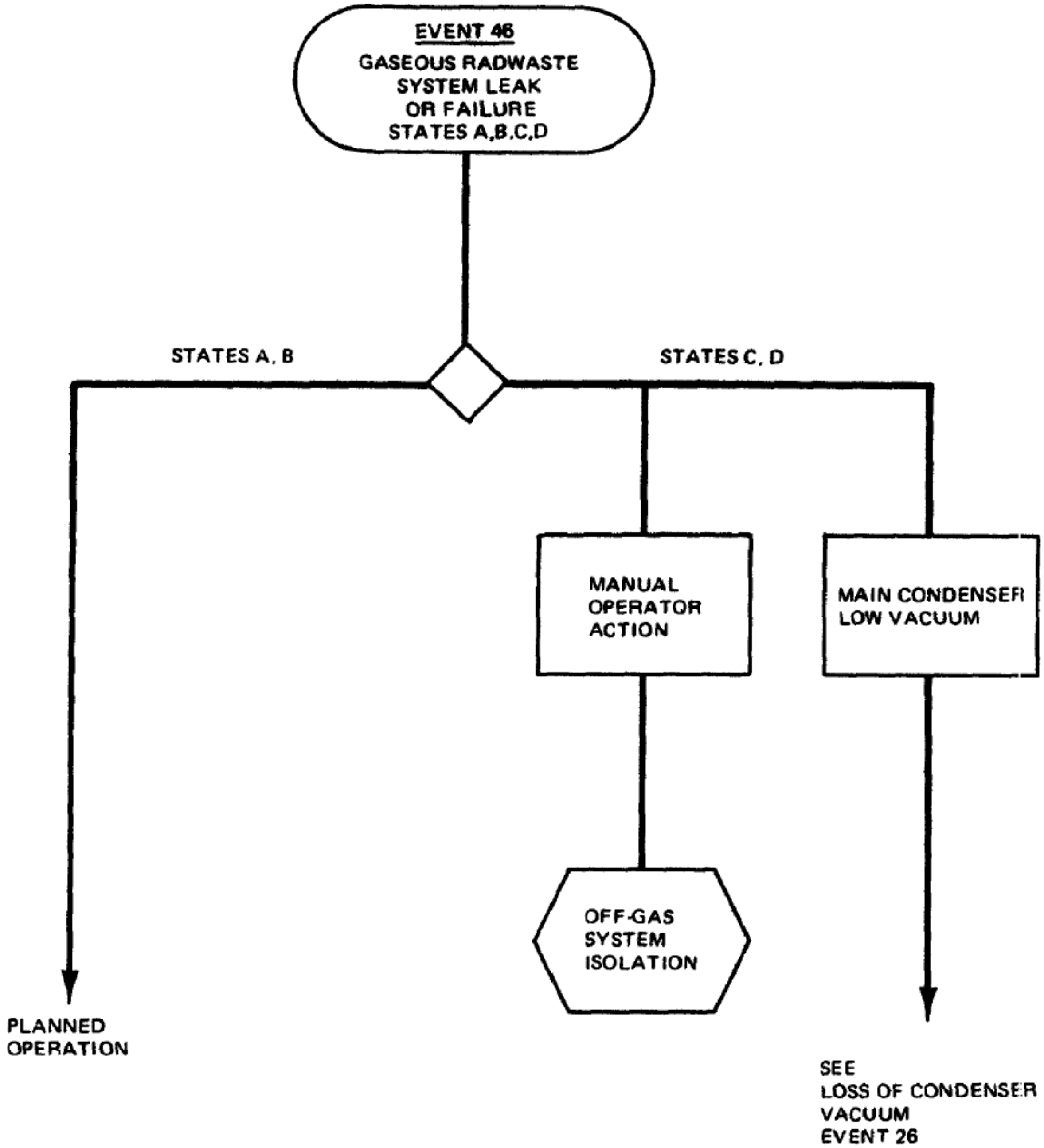
FIGURE 15A.6-45, Rev. 55

AutoCAD Figure 15A_6_45.doc



HISTORICAL

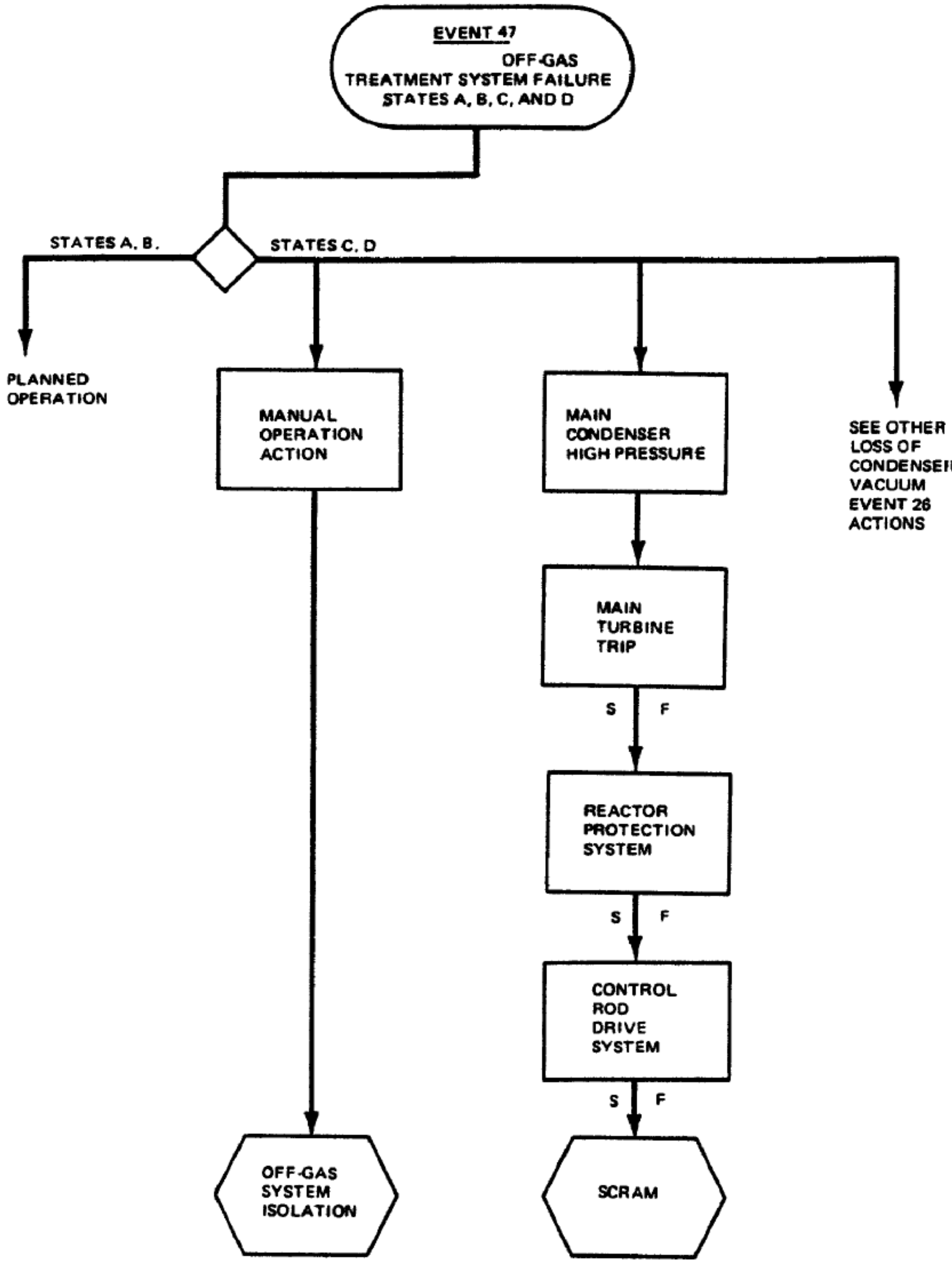
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
GASEOUS RADWASTE SYSTEM
LEAK OR FAILURE

FIGURE 15A.6-46, Rev 55

AutoCAD: Figure Fsar 15A_6_46.dwg



FSAR REV. 65

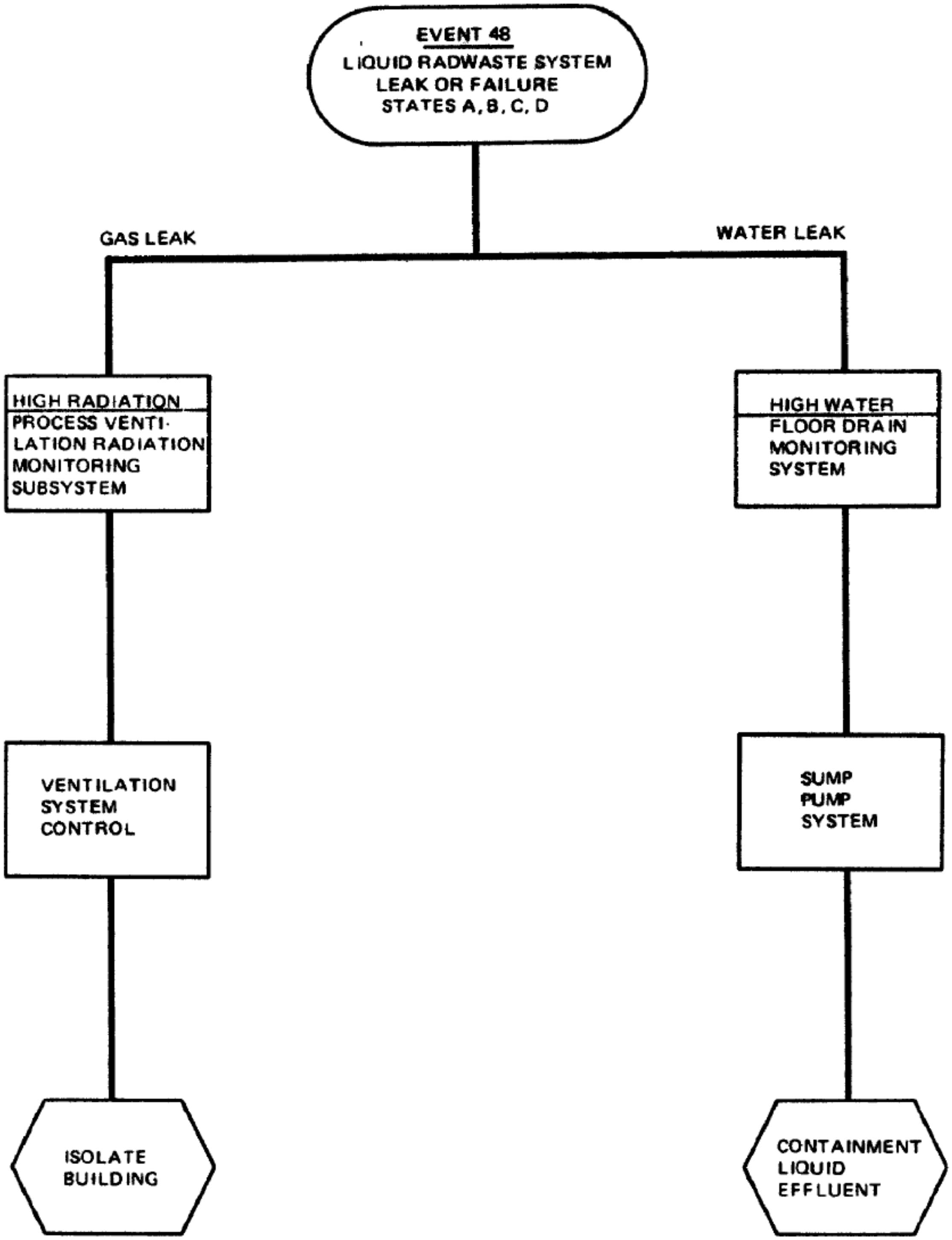
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
OFF-GAS TREATMENT
SYSTEM FAILURE

FIGURE 15A.6-47, Rev 55

AutoCAD: Figure Fsar 15A_6_47.dwg



HISTORICAL

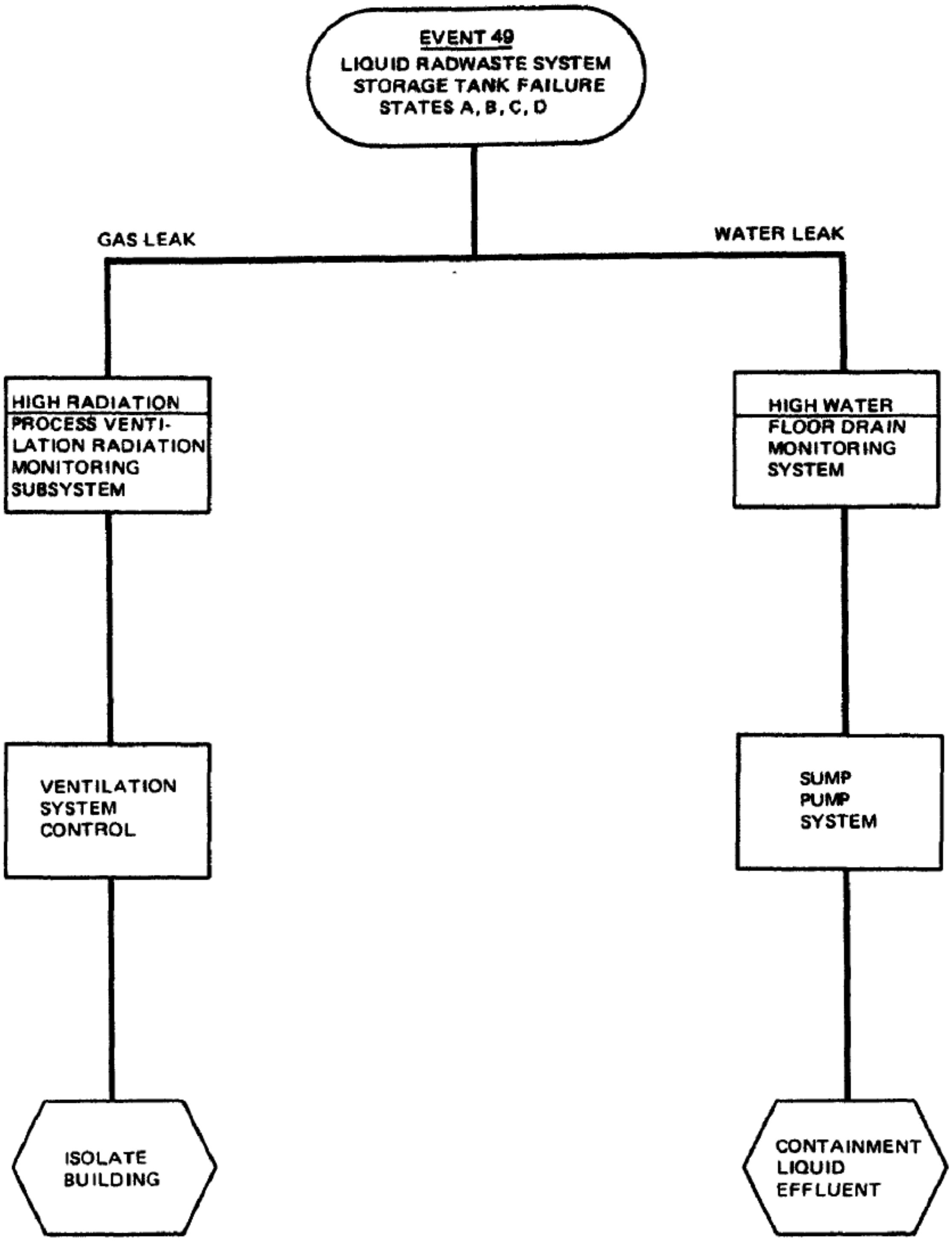
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
LIQUID RADWASTE SYSTEM
LEAK OR FAILURE

FIGURE 15A.6-48, Rev 55

AutoCAD: Figure Fsar 15A_6_48.dwg



HISTORICAL

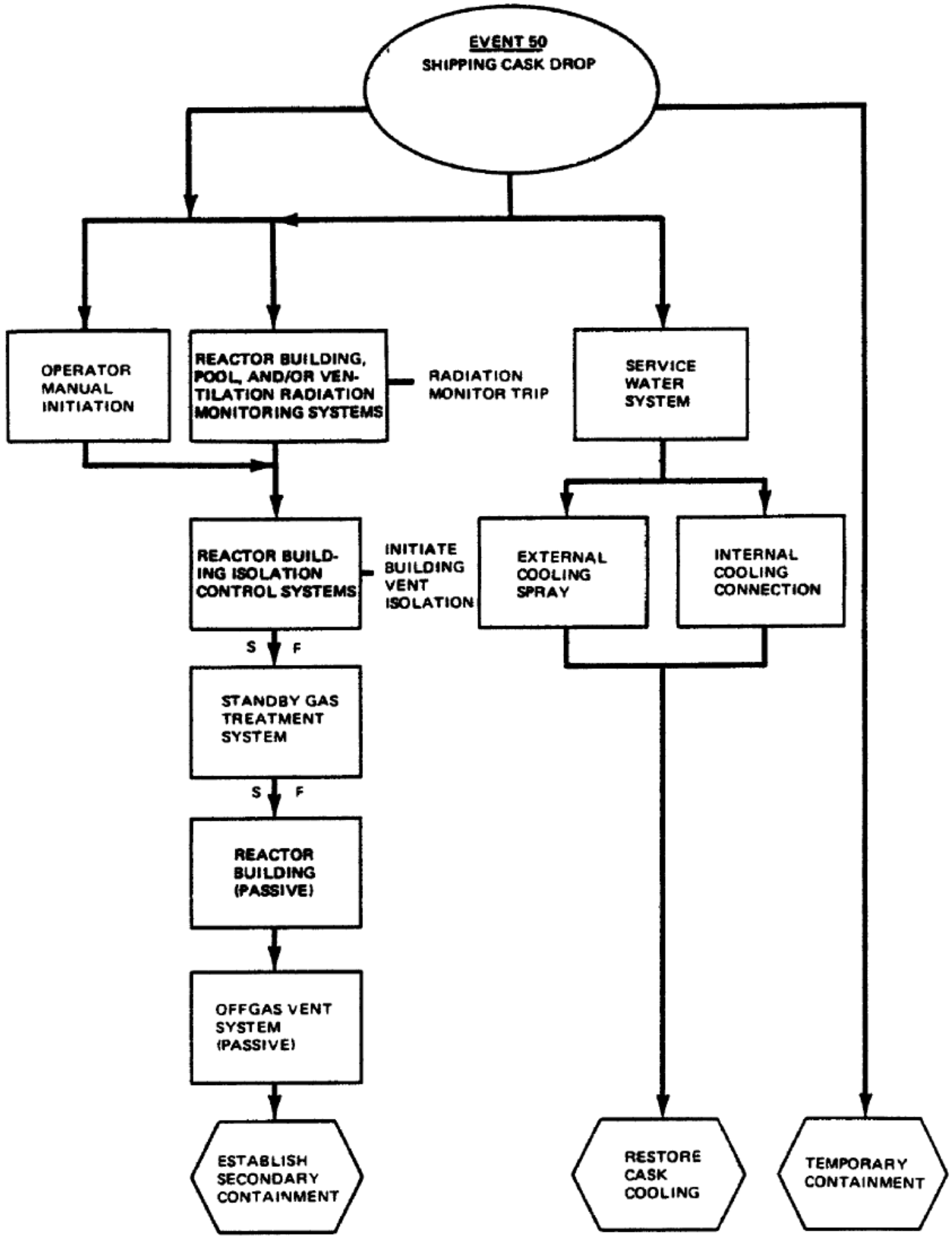
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
LIQUID RADWASTE SYSTEM
STORAGE TANK FAILURE

FIGURE 15A.6-49, Rev 55

AutoCAD: Figure Fsar 15A_6_49.dwg



FSAR REV. 65

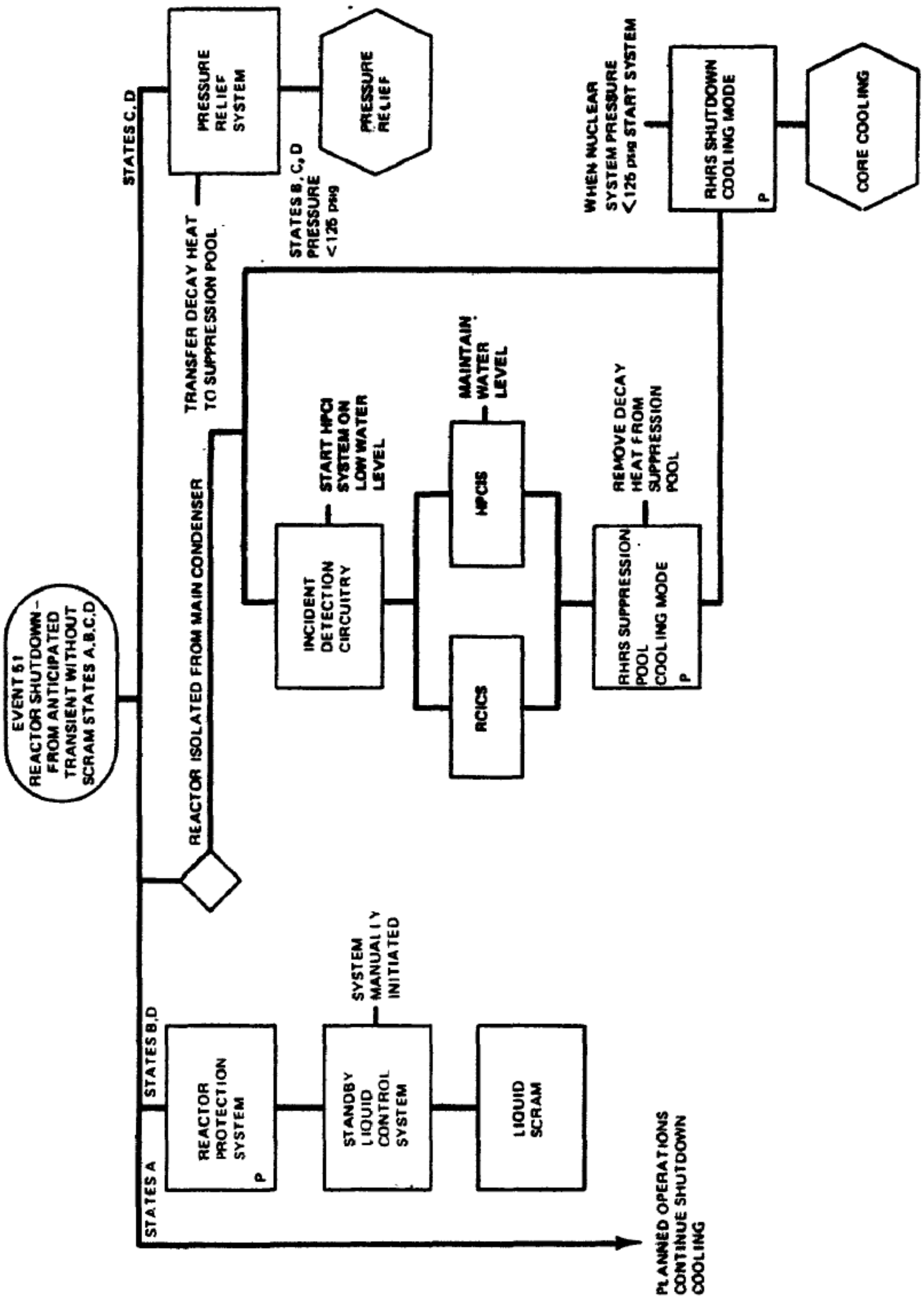
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE
FOR
SPENT FUEL CASK DROP

FIGURE 15A.6-50, Rev 55

AutoCAD: Figure Fsar 15A_6_50.dwg



FSAR REV. 65

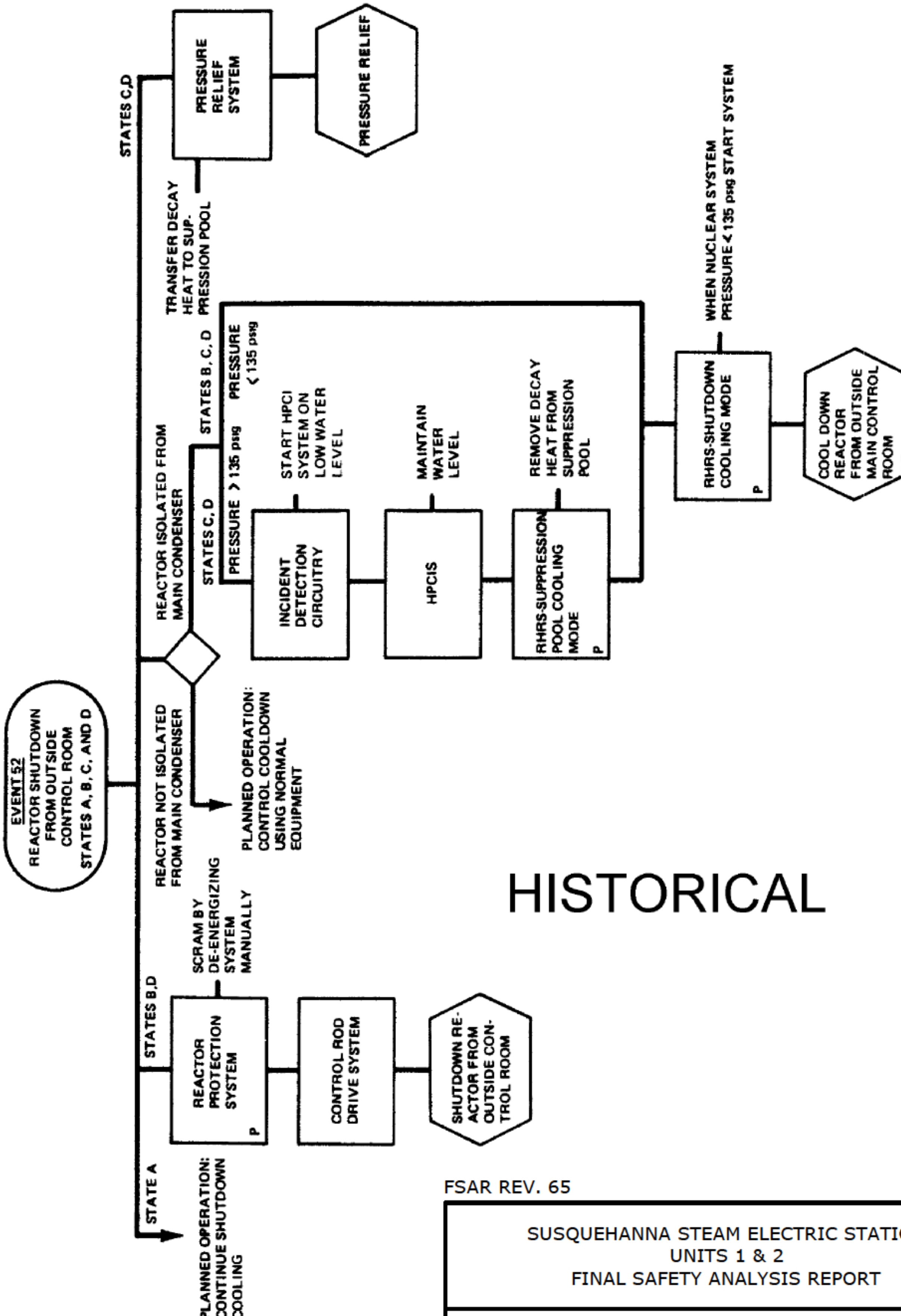
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
REACTOR SHUTDOWN - FROM
ANTICIPATED TRANSIENT
WITHOUT SCRAM

FIGURE 15A.6-51, Rev 55

AutoCAD: Figure Fsar 15A_6_51.dwg



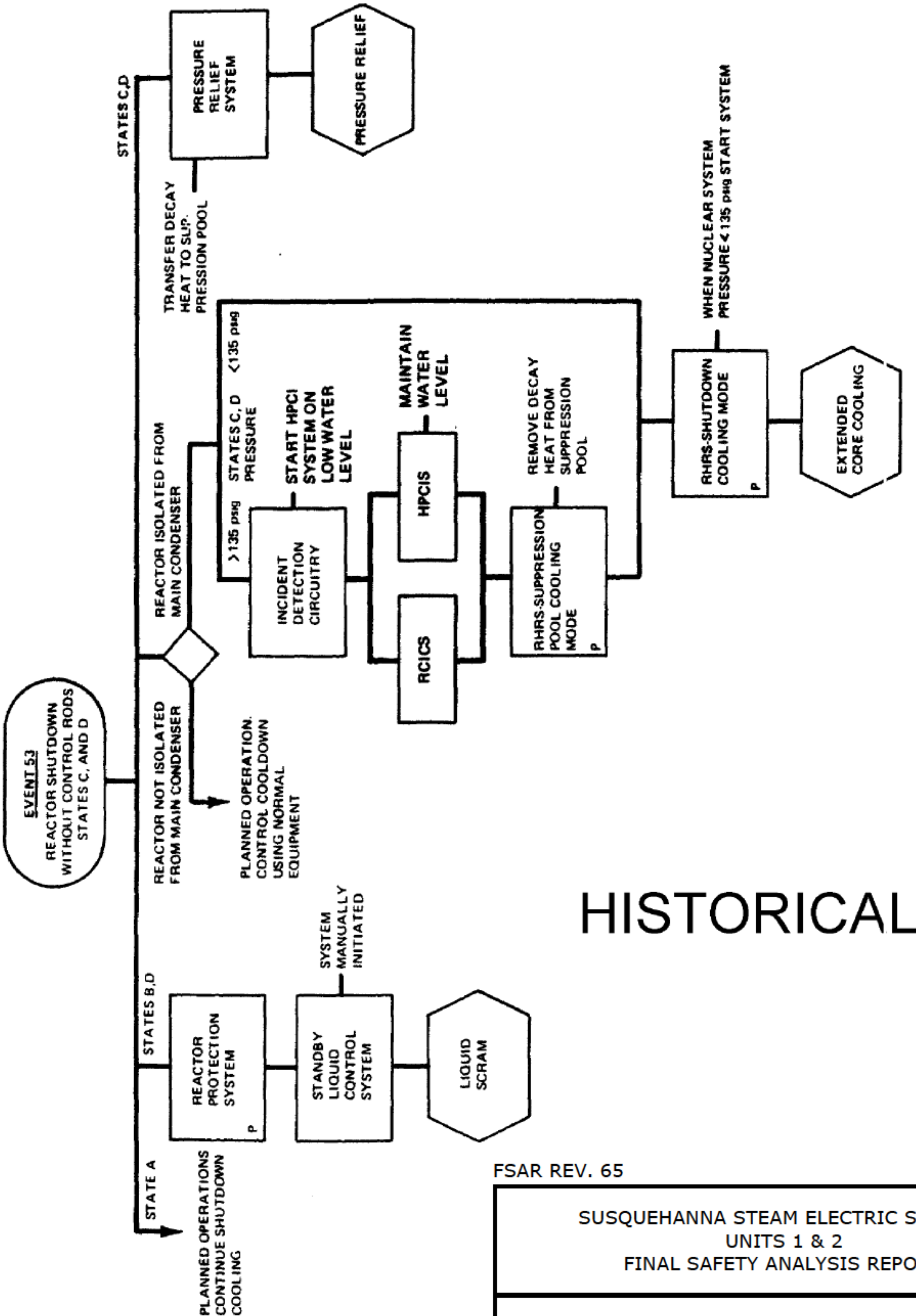
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
REACTOR SHUTDOWN - FROM
OUTSIDE MAIN CONTROL ROOM

FIGURE 15A.6-52, Rev 55

AutoCAD: Figure Fsar 15A_6_52.dwg



HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCE FOR
REACTOR SHUTDOWN -
WITHOUT CONTROL RODS

FIGURE 15A.6-53, Rev 55

AutoCAD: Figure Fsar 15A_6_53.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-54, Rev. 49

AutoCAD Figure 15A_6_54.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-55, Rev. 49

AutoCAD Figure 15A_6_55.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-56, Rev. 49

AutoCAD Figure 15A_6_56.doc

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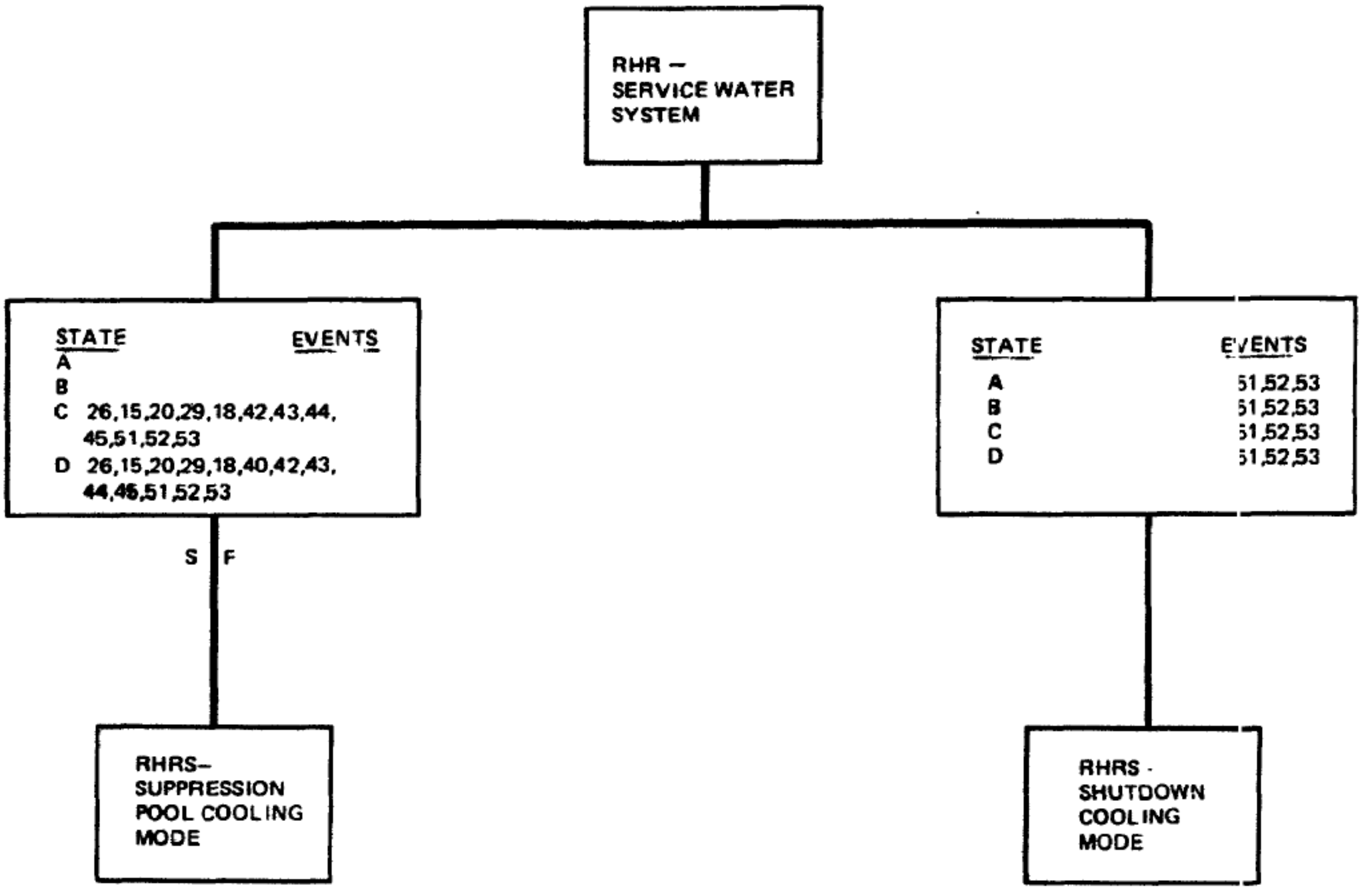
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-57, Rev. 49

AutoCAD Figure 15A_6_57.doc



NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 51, 52, 53

FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - RHR SERVICE
WATER SYSTEM

FIGURE 15A.6-58, Rev 55

AutoCAD: Figure Fsar 15A_6_58.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Deleted FIGURE 15A.6-59, Rev. 49

AutoCAD Figure 15A_6_59.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-60, Rev. 49

AutoCAD Figure 15A_6_60.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-61, Rev. 49

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-62, Rev. 49

AutoCAD Figure 15A_6_62.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-63, Rev. 49

AutoCAD Figure 15A_6_63.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-64, Rev. 49

AutoCAD Figure 15A_6_64.doc

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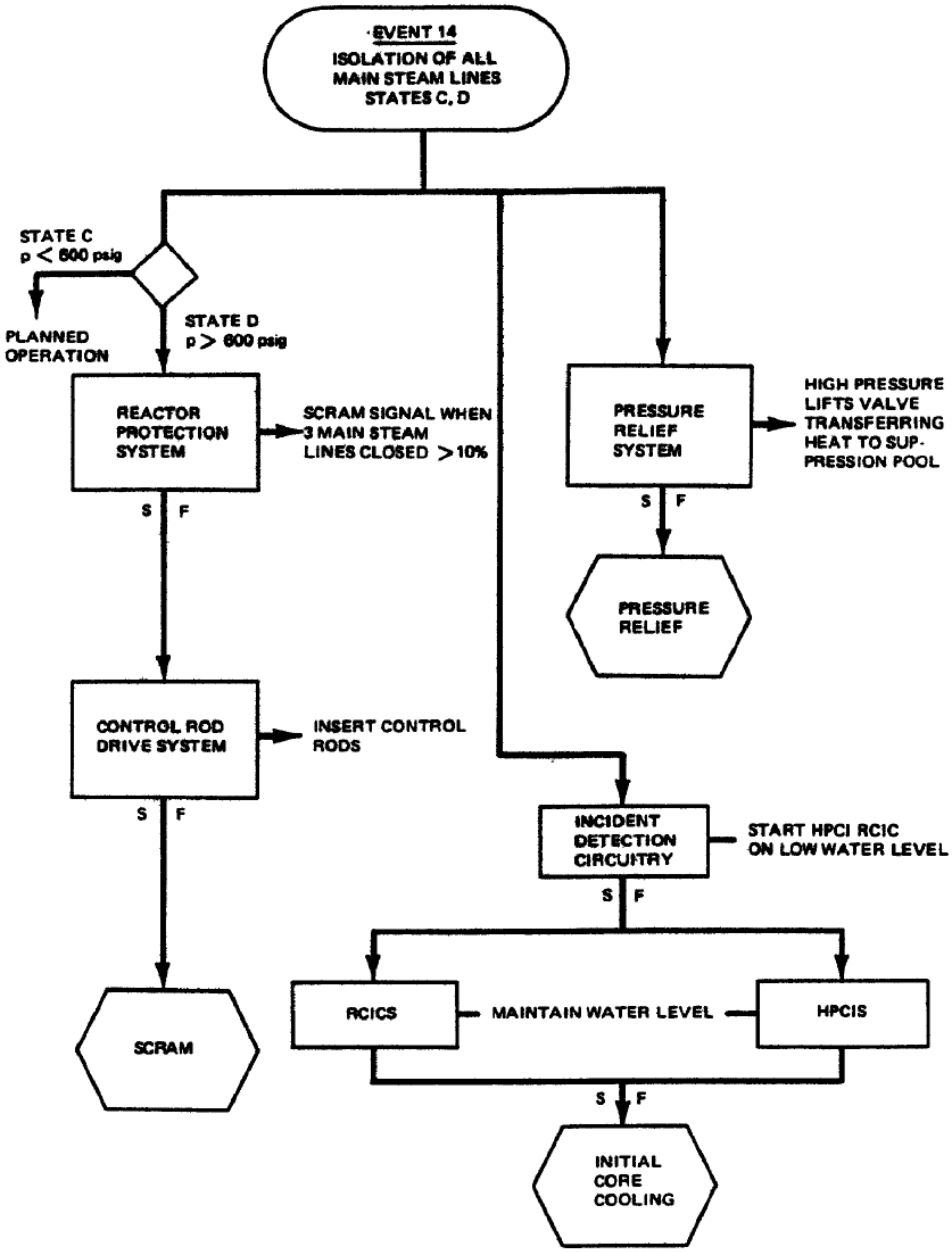
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15A.6-65, Rev. 49

AutoCAD Figure 15A_6_65.doc



FSAR REV.65

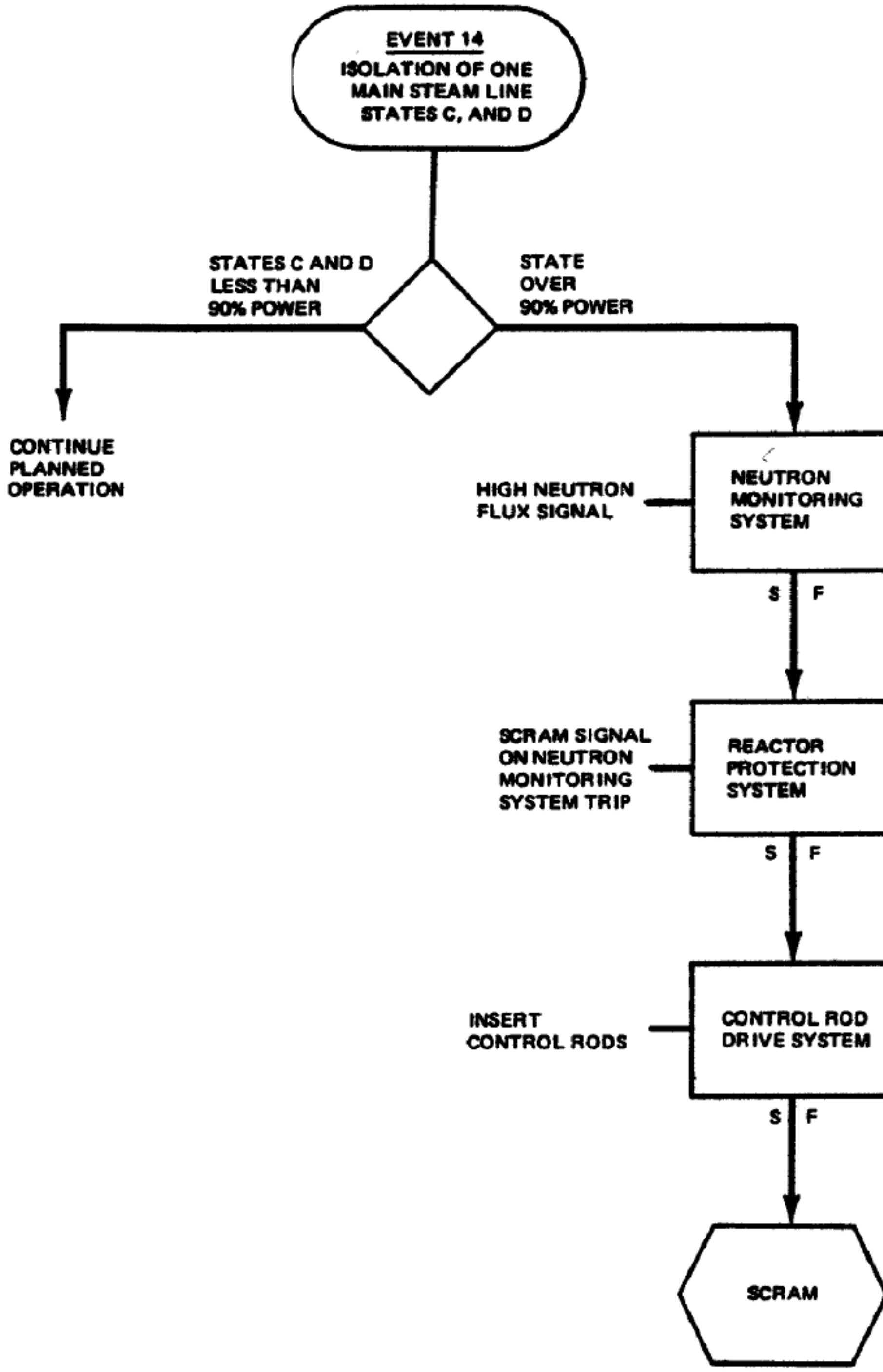
HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
ISOLATION OF ALL
MAIN STEAMLINES

FIGURE 15A.6-14A, Rev 55

AutoCAD: Figure Fsar 15A_6_14A.dwg



HISTORICAL

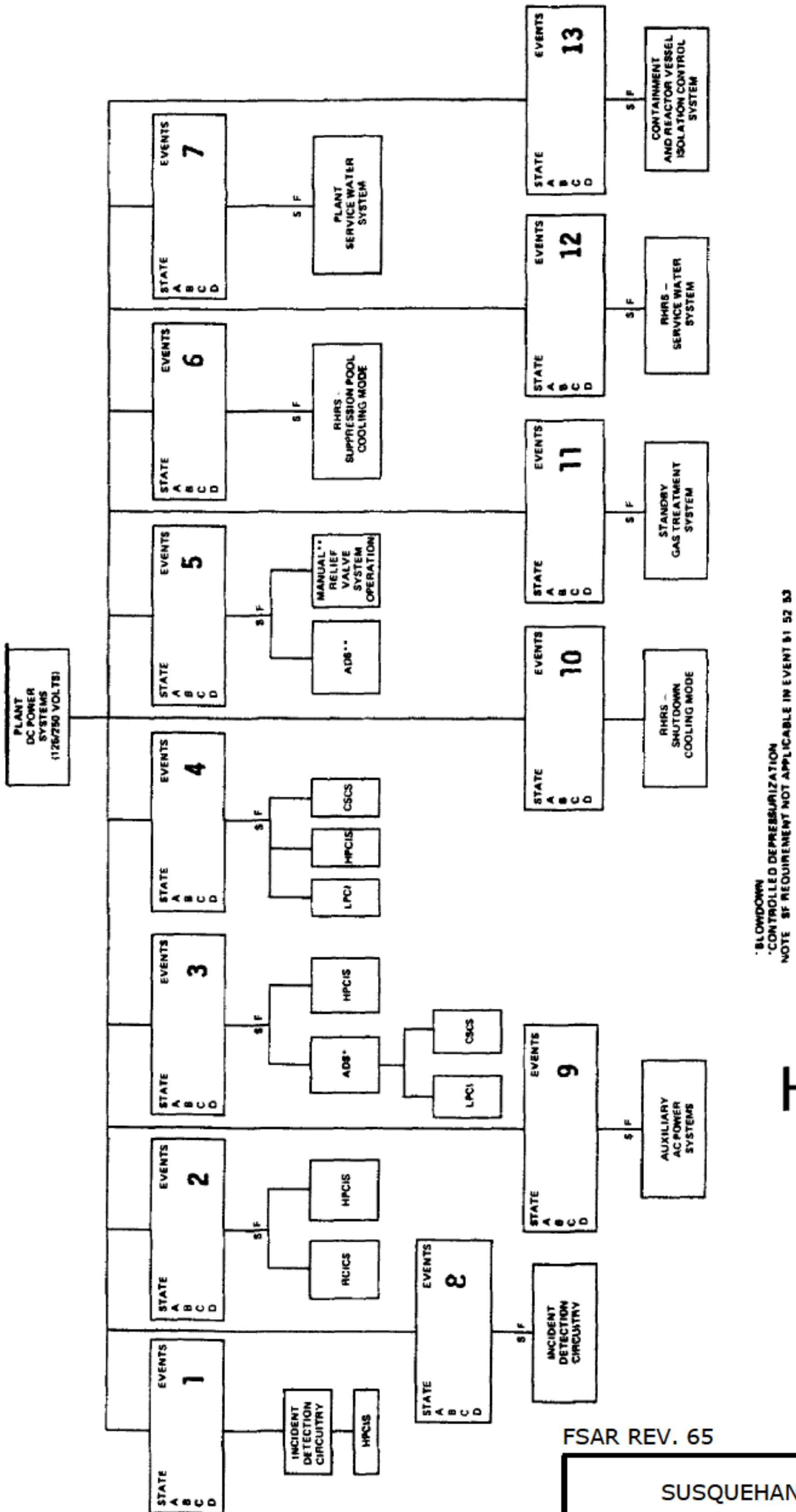
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROTECTION SEQUENCES FOR
ISOLATION OF THE ONE
MAIN STEAMLINE

FIGURE 15A.6-14B, Rev 55

AutoCAD: Figure Fsar 15A_6_14B.dwg



HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - DC POWER SYSTEMS
(125/250 VOLTS)

FIGURE 15A.6-54-1, Rev 55

AutoCAD: Figure Fsar 15A_6_54_1.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-54

ITEM	STATE	EVENTS
1	A	
	B	
	C	51, 52, 53
	D	51, 52, 53
2	A	
	B	
	C	14, 26, 23, 20, 29
	D	14, 26, 23, 20, 29, 40
3	A	
	B	
	C	15, 42, 43, 44, 45
	D	15, 42, 43, 44, 45
4	A	29, 18
	B	29, 18
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 40, 18, 42, 43, 44, 45
5	A	
	B	
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 40, 18, 42, 43, 44, 45
6	A	
	B	
	C	26, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
7	A	29, 18, 41
	B	29, 18, 41
	C	26, 23, 15, 20, 29, 18, 41, 42, 43, 44, 45, 51, 52, 53
	D	26, 23, 15, 20, 29, 18, 40, 41, 42, 43, 44, 45, 51, 52, 53
8	A	
	B	
	C	14, 15, 20, 23, 26, 29, 42, 43, 44, 45, 51, 52, 53
	D	14, 15, 20, 23, 26, 29, 40, 42, 43, 44, 45, 51, 52, 53

FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-54
PAGE 1 OF 2

FIGURE 15A.6-54-2, Rev 55

AutoCAD: Figure Fsar 15A_6_54_2.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-54

9	A	29, 18, 41
	B	29, 18, 41, 51, 52, 53
	C	14, 23, 26, 15, 20, 29, 18, 42, 41, 43, 44, 45, 51, 52, 53
	D	14, 23, 26, 15, 20, 19, 18, 40, 42, 41, 43, 44, 45, 51, 52, 53
10	A	52, 53
	B	52, 53
	C	51, 52, 53
	D	51, 52, 53
11	A	41
	B	41
	C	41, 42
	D	41, 42
12	A	
	B	52, 53
	C	14, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53
	D	14, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
13	A	
	B	
	C	23, 15, 20, 42, 43, 44, 45
	D	23, 15, 20, 40, 42, 43, 44, 45

HISTORICAL

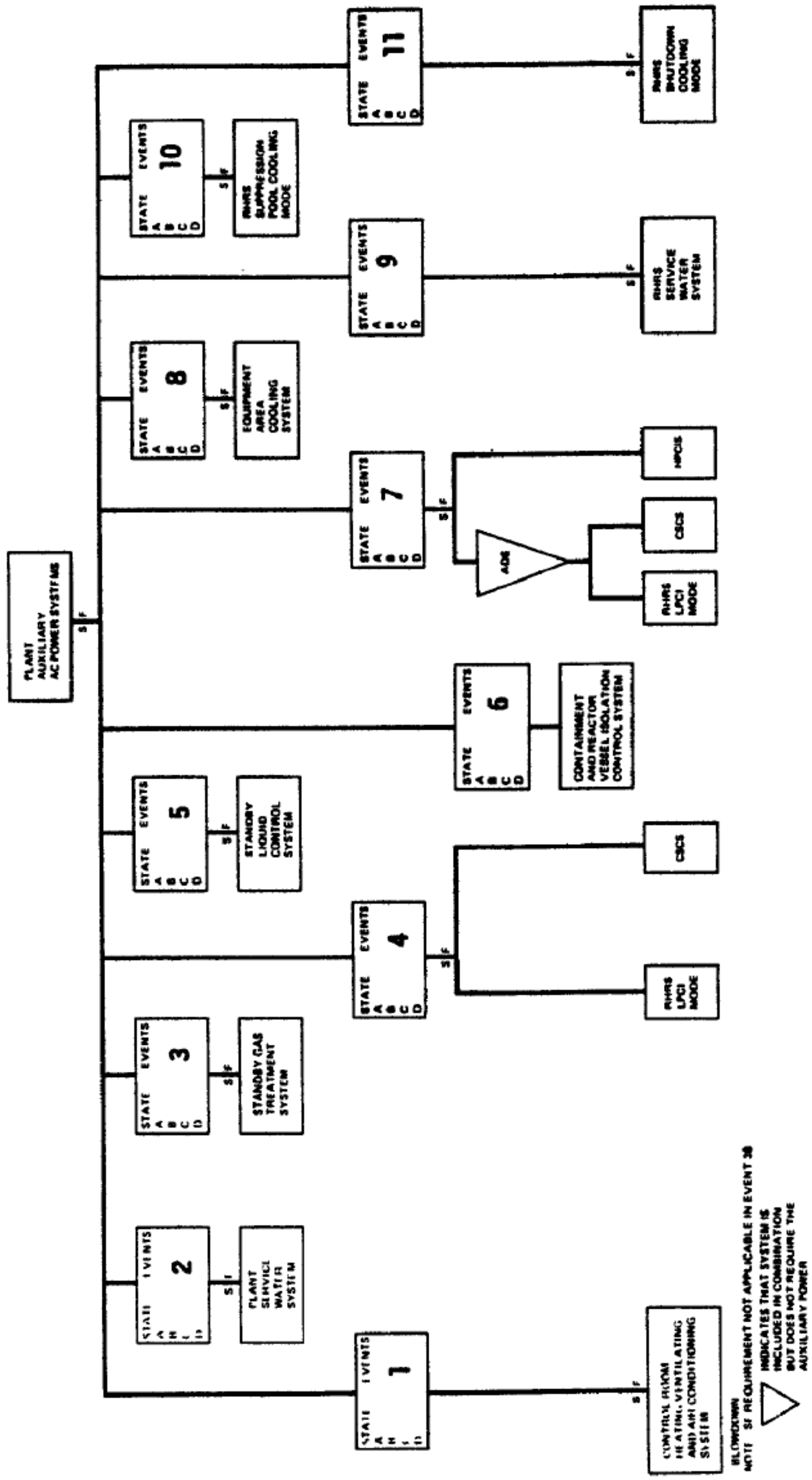
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-54
PAGE 2 OF 2

FIGURE 15A.6-54-3, Rev 55

AutoCAD: Figure Fsar 15A_6_54_3.dwg



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - AC POWER SYSTEMS
(120/480/4160 VOLTS)

FIGURE 15A.6-55-1, Rev 55

AutoCAD: Figure Fsar 15A_6_55_1.dwg

HISTORICAL

EVENT IDENTIFICATION FOR FIGURE 15A.6-55

ITEM	STATE	EVENTS
1	A B C D	41 41 41, 42, 43, 44, 45 41, 42, 43, 44, 45
2	A B C D	18, 41, 51, 52, 53 18, 41, 51, 52, 53 14, 26, 23, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53 14, 26, 23, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
3	A B C D	41 41 41, 42 41, 42
4	A B C D	 26, 15, 20, 29, 18, 42, 43, 44, 45 26, 15, 20, 29, 18, 40, 42, 43, 44, 45
5	A B C D	 51, 52, 53 51, 52, 53
6	A B C D	 23, 15, 20, 42, 43, 44, 45 23, 15, 20, 40, 42, 43, 44, 45
7	A B C D	 15, 42, 43, 44, 45 15, 42, 43, 44, 45
8	A B C D	29, 18 29, 18 14, 26, 23, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53 14, 26, 23, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53

FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-55
PAGE 1 OF 2

FIGURE 15A.6-55-2, Rev 55

AutoCAD: Figure Fsar 15A_6_55_2.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-55

9	A B C D	 52, 53 26, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53 26, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
10	A B C D	 26, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53 26, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
11	A B C D	52, 53 52, 53 51, 52, 53 51, 52, 53

HISTORICAL

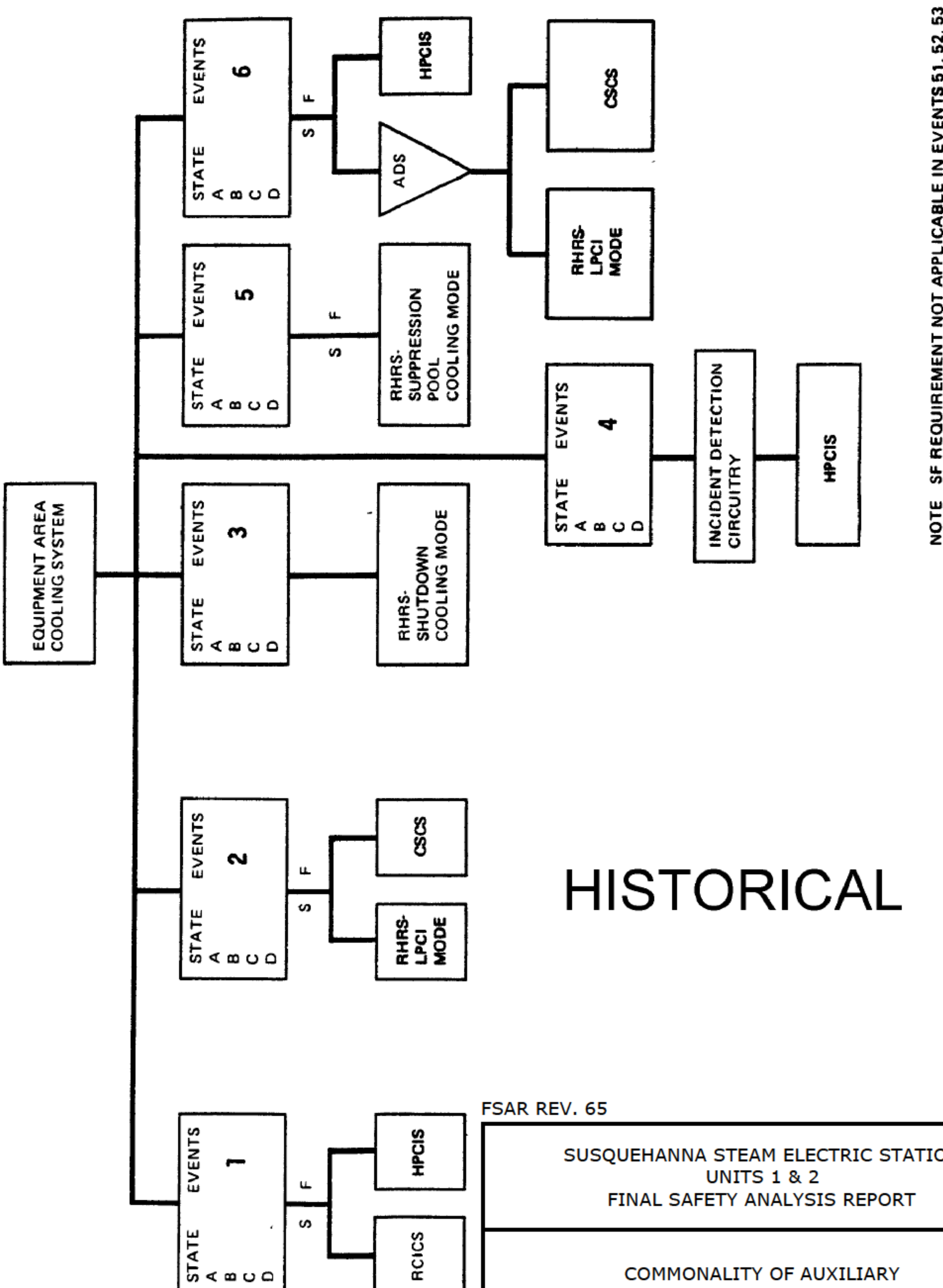
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-55
PAGE 2 OF 2

FIGURE 15A.6-55-3, Rev 55

AutoCAD: Figure Fsar 15A_6_55_3.dwg



HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - EQUIPMENT AREA
CODING SYSTEM

FIGURE 15A.6-56-1, Rev 55

AutoCAD: Figure Fsar 15A_6_56_1.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-56

ITEM	STATE	EVENTS
1	A	
	B	
	C	14, 26, 23, 20, 29, 42
	D	14, 26, 23, 20, 29, 40, 42
2	A	29, 18
	B	29, 18
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45
3	A	
	B	51, 52, 53
	C	51, 52, 53
	D	51, 52, 53
4	A	
	B	
	C	51, 52, 53
	D	51, 52, 53
5	A	
	B	
	C	26, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
6	A	
	B	
	C	15, 42, 43, 44, 45
	D	15, 42, 43, 44, 45

HISTORICAL

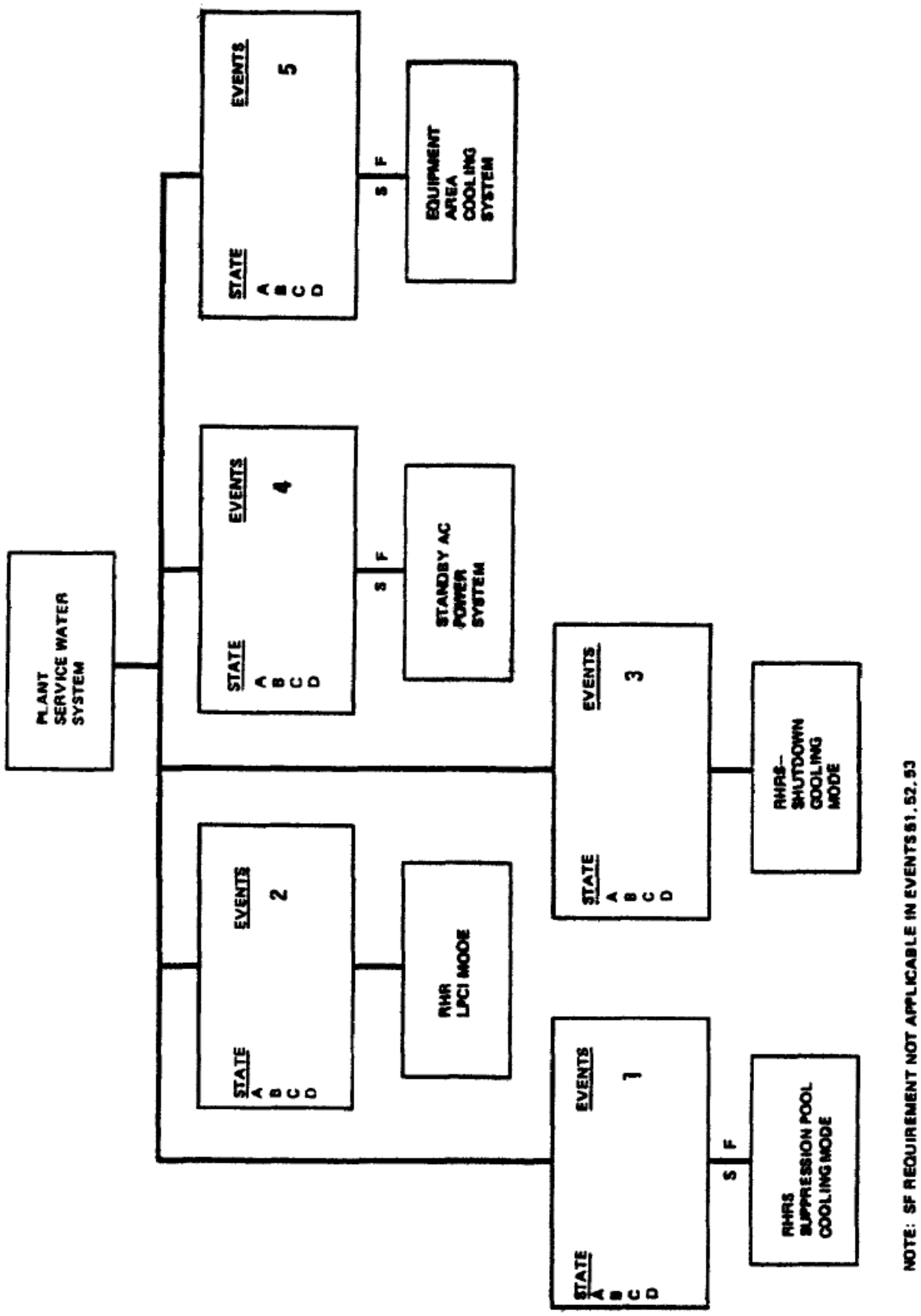
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-56

FIGURE 15A.6-56-2, Rev 55

AutoCAD: Figure Fsar 15A_6_56_2.dwg



FSAR REV. 65

HISTORICAL

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - PLANT SERVICE
WATER SYSTEM

FIGURE 15A.6-57-1, Rev 55

AutoCAD: Figure Fsar 15A_6_57_1.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-57

ITEM	STATE	EVENTS
1	A	
	B	
	C	26, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53
2	A	29, 18
	B	29, 18
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45
3	A	51, 52, 53
	B	51, 52, 53
	C	51, 52, 53
	D	51, 52, 53
4	A	29
	B	29
	C	29, 42, 43, 44, 45
	D	29, 42, 43, 44, 45
5	A	29, 18
	B	29, 18
	C	14, 26, 23, 15, 20, 29, 18, 42, 43, 44, 45, 51, 52, 53
	D	14, 26, 23, 15, 20, 29, 18, 40, 42, 43, 44, 45, 51, 52, 53

HISTORICAL

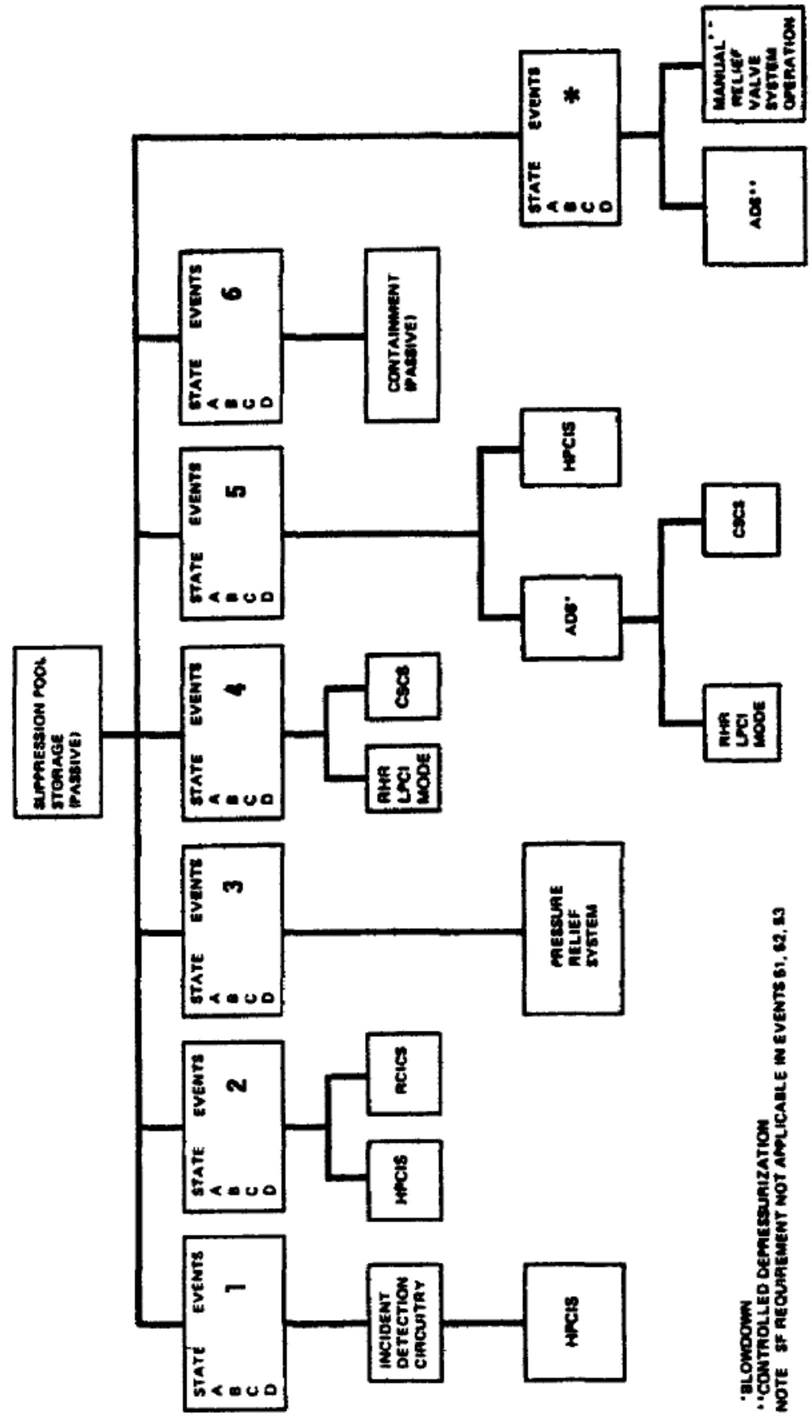
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-57

FIGURE 15A.6-57-2, Rev 55

AutoCAD: Figure Fsar 15A_6_57_2.dwg



HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMMONALITY OF AUXILIARY
SYSTEMS - SUPPRESSION
POOL STORAGE

FIGURE 15A.6-59-1, Rev 55

AutoCAD: Figure Fsar 15A_6_59_1.dwg

EVENT IDENTIFICATION FOR FIGURE 15A.6-59

ITEM	STATE	EVENTS
1	A	
	B	
	C	51, 52, 53
	D	51, 52, 53
2	A	
	B	
	C	14, 26, 23, 20, 29
	D	14, 26, 23, 20, 29, 40
3	A	
	B	
	C	14, 26, 23, 20, 29, 22, 42, 43, 44, 45, 51, 52, 53
	D	30, 25, 14, 26, 23, 20, 29, 22, 40, 42, 43, 44, 45, 51, 52, 53, 31, 27
4	A	29, 18
	B	29, 18
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45
5	A	
	B	
	C	15, 42, 43, 44, 45
	D	15, 42, 43, 44, 45
6	A	
	B	23, 18,
	C	26, 23, 15, 20, 18, 42, 43, 44, 45, 51, 52, 53
	D	26, 23, 15, 20, 18, 40, 42, 43, 44, 45, 51, 52, 53
7	A	
	B	
	C	26, 15, 20, 29, 18, 42, 43, 44, 45
	D	26, 15, 20, 29, 18, 40, 42, 43, 44, 45

HISTORICAL

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EVENT IDENTIFICATION
FOR FIGURE 15A.6-59

FIGURE 15A.6-59-2, Rev 55

AutoCAD: Figure Fsar 15A_6_59_2.dwg

APPENDIX 15B

ACCIDENT DOSE MODEL DESCRIPTIONS

15B.1 OFFSITE DOSE MODEL

This discussion describes the models used to calculate offsite radiological doses that would result from releases of radioactivity due to various postulated accidents.

The following assumptions are used for offsite dose evaluations:

a) The direct dose contribution offsite from any post-accident onsite source point is negligible compared with the direct dose due to immersion in the post-accident effluent cloud.

b) All radioactivity releases are treated as ground level releases regardless of the point of discharge.

c) Isotopic data including decay constants and dose conversion factors are listed in Table 15B-2. The isotopic data listed in Table 15B-2 is obtained from the RADTRAD (Reference 15B-4) computer code which is used to evaluate the radiological consequences of accidents. These dose conversion factors are used to calculate immersion and inhalation doses and are derived from Federal Guidance Report Nos. 11 and 12 (References 15B-6 and 15B-7).

The acceptance criteria for the offsite doses is in terms of Rem TEDE. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity. The definition of these doses is given in 10CFR20.1003.

The models used to evaluate offsite doses for accidents are as follows:

Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi / Q)_j \qquad \text{(EQ. 15B-1)}$$

where:

D_{im} = Immersion (EDE) dose (rem)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-sec)

R_{ij} = Amount of isotope i released during time period j (Ci)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (sec/m^3)

Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi/Q)_j \qquad \text{(EQ. 15B-2)}$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

R_{ij} = Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/sec)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (sec/m^3)

Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15B.2 CONTROL ROOM HABITABILITY ENVELOPE DOSE MODEL

This discussion describes the models used to calculate control room habitability envelope (CRHE) radiological doses that would result from releases of radioactivity due to various postulated accidents.

The acceptance criteria for CRHE doses is in terms of Rem TEDE. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity. The definition of these doses is given in 10CFR20.1003. The total CRHE TEDE dose is the sum of the EDE and the CEDE doses for all CRHE post-accident radiation sources.

The design basis for the CRHE is to provide adequate radiation protection to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. This basis is consistent with 10CFR50.67. Radiation protection for the CRHE is provided by radiation shielding and by an emergency ventilation system.

The CRHE radiation shielding is designed to reduce gamma radiation shine from both normal and post-accident radiation sources to levels consistent with the requirements of 10CFR20 or 10CFR50.67.

The post-accident emergency ventilation system is designed to preclude entrance of unfiltered air to the control room and to maintain outleakage of air from this zone with respect to other plant ventilation zones and the air outside the plant.

Details of control room emergency ventilation system design and instrumentation are discussed in Subsection 9.4.1 and Section 6.4.

During emergency operation, 5810 +/- 10% cfm filtered outside air is supplied to the control structure. In addition to the intake of air through the filter system, some air will enter the control building due to ingress/egress of personnel and via infiltration from other identified leakage paths. An infiltration rate of 10 scfm has been assumed for ingress/egress and 500 cfm for the other unidentified leakage. Credit for operation of the CRHE emergency ventilation system is only taken for the DBA-LOCA and fuel handling/equipment handling accidents.

Under accident conditions, radiation doses to control room personnel may result from several sources. While in the control room, personnel are exposed to beta and gamma radiation from gaseous fission products that enter after an accident via the ventilation system or from unfiltered air entering the control room. In addition, personnel may be subject to gamma shine dose from fission products in the containment and reactor building, from contained system sources and from fission products in the atmosphere outside the control room.

To evaluate the capability of the control room ventilation system and radiation shielding to keep doses within the specified criteria, control room doses are evaluated for each of these dose contributors. This analysis includes control room doses from the following radiation sources:

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Radiation shine from the external radioactive plume released from the facility,
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from radioactive material in buildings adjacent to the control structure; includes containment, reactor building and turbine building,
- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., piping, components and radioactive material buildup in HVAC filters.

The short term accident χ/Q's for the SSES Control Room Habitability Envelope (CRHE) were calculated using the methodology provided in NUREG/CR-6331 - ARCON96 (Reference 15B-1) and Regulatory Guide 1.194 (Reference 15B-3). The ARCON96 code uses hourly meteorological data and recently developed methods for estimating χ/Q’s in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than five percent of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration.

The specific locations requiring ARCON96 χ/Qs for use in the applicable radiological evaluations were:

Turbine Building Unit 1 exhaust vent.
Turbine Building Unit 2 exhaust vent.
Standby Gas Treatment System exhaust vent.
Reactor Building Unit 2 Exhaust Vent

The atmospheric dispersion factors (χ/Q) used in the accident control room dose evaluation are listed in Table 15B-1.

The models used to evaluate CRHE doses for accidents are as follows:

15B.2.1 CRHE IMMERSION DOSES - IN-LEAKAGE OF RADIOACTIVITY

The dose to an individual in the Control Room Habitability Envelope (CRHE) from the in-leakage of radioactivity is calculated based on the time integrated concentration in the control room compartment. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

CRHE immersion doses are calculated using the RADTRAD computer code (Reference 15B-4) and the control room atmospheric dispersion factors given in Table 15B-1. The dose models and methodology are as follows:

Immersion Dose (Effective Dose Equivalent)

Due to the finite volume of air contained in the CRHE, the immersion dose for an operator occupying the main control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 15B-6).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_{i} DCF_i \sum_{j} (IAR)_{ij} O_j \qquad \text{(EQ. 15B-3)}$$

where:

D_{im} = Immersion (EDE) dose (rem)

GF = Geometry factor = $1173/V^{0.338}$

V = Volume of the CRHE (ft^3)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-sec)

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-sec/m^3)

O_j = Fraction of time period j that the operator is assumed to be present Table 15B-1

Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j O_j \qquad \text{(EQ. 15B-4)}$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-sec/m^3)

$(BR)_j$ = Breathing rate during time period j (m^3/sec)

O_j = Fraction of time period j that the operator is assumed to be present

Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses. THE CRHE dose acceptance criteria is given as 5 Rem TEDE. The TEDE (total effective dose equivalent) is defined as the sum of the external dose equivalent (EDE) from external contamination plus the committed effective dose equivalent (CEDE) from internal contamination in NRC Regulatory Issue Summary 2003-04, Use of the Effective Dose Equivalent in Place of the Deep Dose Equivalent in Dose Assessments (Reference 15B-5).

In order to take credit for the radiation shielding effects of the control structure floors, the EDE portion of the TEDE is adjusted by the ratio of the geometry factor GF for 518,000 ft^3 (volume of CRHE) to the GF for 110,000 ft^3 (volume of control room) or

GF = 1173. / (518,000)$^{0.338}$ = 13.74

GF = 1173. / (110,000)$^{0.338}$ = 23.19

and the resulting ratio = 0.59.

15B.2.2 CRHE DIRECT SHINE DOSES

Unprotected doses outside the control room for a DBA-LOCA are calculated using the RADTRAD computer code (Reference 15B-4) and the control room atmospheric dispersion factors given in Table 15B-1. These results serve as input to evaluate the direct shine dose from the post-LOCA effluent cloud.

The direct shine to the control structure from the post-LOCA effluent cloud is evaluated by applying dose reduction factors for control structure radiation shielding to the unprotected whole body gamma dose outside the control room. The cloud shine dose is calculated as follows:

Cloud Shine Dose = D_γ^I (unprotected) X RF where:

Cloud Shine Dose	=	Direct dose inside the control structure from the post-LOCA effluent cloud (rem)
D_{γ}^{I} (unprotected)	=	unprotected whole body gamma immersion dose outside the control room (rem)
RF	=	Direct dose reduction factor for control structure radiation shielding.
	=	$B\ e^{-\mu x}$

where:

B	=	buildup factor for shielding configuration
μ	=	total linear attenuation factor (cm^{-1})
x	=	thickness of concrete shielding provided by control structure (cm)

Conservatively assuming an average gamma energy of 1.0 Mev, for ordinary concrete:

μ(1.0 Mev) = $0.149\ cm^{-1}$

The control structure provides a minimum of 2.5 ft of concrete as radiation shielding for the control room from the post-LOCA effluent cloud. For 2.5' of concrete, $Be^{-\mu x} = 3.34 \times 10^{-4}$.

The direct shine CRHE doses from post-accident contained radiation sources are evaluated using source specific shielding design calculations. This includes radiation shine from radioactive material in buildings adjacent to the control structure (containment, reactor building and turbine building) and radiation shine from radioactive material in systems and components inside or external to the CRHE (e.g., piping, components and radioactive material buildup in HVAC filters). Dose rates are evaluated as a function of time post-accident using source terms based on activity transport and leakage assumptions and then are integrated to obtain an effective dose equivalent (rem EDE) for the duration of the accident. Direct shine doses results are combined with immersion and inhalation doses to obtain a total post-accident rem TEDE in the CRHE.

15B.3 REFERENCES

15B-1 NUREG/CR-6331, “Atmospheric Relative Concentrations In Building Wakes”, Revision 1, May 1997 (ARCON96 Computer Code).

15B-2 "Alternative Radiological Source terms For Evaluating Design Basis Accidents At Nuclear Power Reactors”, USNRC Regulatory Guide 1.183, Rev. 0 July 2000.

15B-3 “Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants”, USNRC Regulatory Guide 1.194, June 2003.

15B-4 NUREG/CR-6604, “RADTRAD: A Simplified Model For RADionuclide Transport And Removal And Dose Assessment”, and Supplement 1, 6/8/99.

15B-5 NRC Regulatory Issue Summary 2003-04, “Use Of Effective Dose Equivalent In Place Of Deep Dose Equivalent In Dose Assessments”, 2/13/2003.

15B-6 K.G. Murphy and K.M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference.

15B-7 Federal Guidance Report No. 11, “Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhalation, Submersion, And Ingestion”, 1988 (US Environmental Protection Agency).

15B-8 Federal Guidance Report No. 12, “External Exposure To Radionuclides In Air, Water, And Soil”, 1993 (US Environmental Protection Agency).

TABLE 15.B-1
CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS FOR DESIGN BASIS ACCIDENTS
χ/Q (sec/m^3)

	CRHE χ/Q's (sec/m^3) without Occupancy Correction Factors (1)				
Time Period	0 to 2 hours	2 to 8 hours	8 to 24 hours	1 to 4 days	4 to 30 days
Release Point	RB Unit 2 CRHE Outside Air Intake Location (2)				
TB Unit 1 Exhaust Vent	1.09E-03	8.01E-04	2.89E-04	1.72E-04	1.50E-04
TB Unit 2 Exhaust Vent	1.21E-03	8.76E-04	3.16E-04	1.92E-04	1.61E-04
SGTS Exhaust Vent	1.16E-03	8.64E-04	3.09E-04	1.87E-04	1.60E-04
RB Unit 2 Exhaust Vent	2.29E-03	2.05E-03	8.56E-04	6.13E-04	4.77E-04
Release Point	Outside Control Building Location (3)				
TB Unit 1 Exhaust Vent	4.03E-03	3.61E-03	1.56E-03	1.12E-03	8.71E-04
TB Unit 2 Exhaust Vent	4.72E-03	4.25E-03	1.84E-03	1.32E-03	1.03E-03
SGTS Exhaust Vent	4.15E-03	3.61E-03	1.57E-03	1.12E-03	8.86E-04
RB Unit 2 Exhaust Vent	NA				

NOTES:

(1) Occupancy Correction Factors (Reference 15B-2
1.0 0-24 hrs
0.6 1-4 days
0.4 4-30 days

(2) Values to be used for dose internal to CRHE.

(3) Values to be used for dose external to CRHE. RB Unit Exhaust Vent not used for external cloud dose.

TABLE 15B-2
PHYSICAL DATA FOR ISOTOPES (1)

DOSE CONVERSION FACTORS

Isotope	Half Life (sec)	Whole Body DCF ($Sv\text{-}m^3/Bq\text{-}sec$)	Inhaled Thyroid DCF (Sv/Bq)	Inhaled Effective DCF (Sv/Bq)
Co-58	6.12E+06	4.76E-14	8.72E-10	2.94E-09
Co-60	1.66E+08	1.26E-13	1.62E-08	5.91E-08
Kr-85	3.38E+08	1.19E-16	0.00E+00	0.00E+00
Kr-85m	1.61E+04	7.48E-15	0.00E+00	0.00E+00
Kr-87	4.58E+03	4.12E-14	0.00E+00	0.00E+00
Kr-88	1.02E+04	1.02E-13	0.00E+00	0.00E+00
Rb-86	1.61E+06	4.81E-15	1.33E-09	1.79E-09
Sr-89	4.36E+06	7.73E-17	7.96E-12	1.12E-08
Sr-90	9.19E+08	7.53E-18	2.69E-10	3.51E-07
Sr-91	3.42E+04	4.92E-14	9.93E-12	4.55E-10
Sr-92	9.76E+03	6.79E-14	3.92E-12	2.18E-10
Y-90	2.30E+05	1.90E-16	5.17E-13	2.28E-09
Y-91	5.06E+06	2.60E-16	8.50E-12	1.32E-08
Y-92	1.27E+04	1.30E-14	1.05E-12	2.11E-10
Y-93	3.64E+04	4.80E-15	9.26E-13	5.82E-10
Zr-95	5.53E+06	3.60E-14	1.44E-09	6.39E-09
Zr-97	6.08E+04	4.43E-14	2.32E-11	1.17E-09
Nb-95	3.04E+06	3.74E-14	3.58E-10	1.57E-09
Mo-99	2.38E+05	7.28E-15	1.52E-11	1.07E-09
Tc-99m	2.17E+04	5.89E-15	5.01E-11	8.80E-12
Ru-103	3.39E+06	2.25E-14	2.57E-10	2.42E-09
Ru-105	1.60E+04	3.81E-14	4.15E-12	1.23E-10
Ru-106	3.18E+07	1.04E-14	1.72E-09	1.29E-07
Rh-105	1.27E+05	3.72E-15	2.88E-12	2.58E-10
Sb-127	3.33E+05	3.33E-14	6.15E-11	1.63E-09
Sb-129	1.56E+04	7.14E-14	9.72E-12	1.74E-10
Te-127	3.37E+04	2.42E-16	1.84E-12	8.60E-11
Te-127m	9.42E+06	1.47E-16	9.66E-11	5.81E-09
Te-129	4.18E+03	2.75E-15	5.09E-13	2.09E-11
Te-129m	2.90E+06	3.34E-15	1.56E-10	6.48E-09
Te-131m	1.08E+05	7.46E-14	3.67E-08	1.76E-09
Te-132	2.82E+05	1.03E-14	6.28E-08	2.55E-09

TABLE 15B-2
PHYSICAL DATA FOR ISOTOPES (1)

DOSE CONVERSION FACTORS

Isotope	Half Life (sec)	Whole Body DCF (Sv-m^3/Bq-sec)	Inhaled Thyroid DCF (Sv/Bq)	Inhaled Effective DCF (Sv/Bq)
I-131	6.95E+05	1.82E-14	2.92E-07	8.89E-09
I-132	8.28E+03	1.12E-13	1.74E-09	1.03E-10
I-133	7.49E+04	2.94E-14	4.86E-08	1.58E-09
I-134	3.16E+03	1.30E-13	2.88E-10	3.55E-11
I-135	2.38E+04	8.29E-14	8.46E-09	3.32E-10
Xe-133	4.53E+05	1.56E-15	0.00E+00	0.00E+00
Xe-135	3.27E+04	1.19E-14	0.00E+00	0.00E+00
Cs-134	6.51E+07	7.57E-14	1.11E-08	1.25E-08
Cs-136	1.13E+06	1.06E-13	1.73E-09	1.98E-09
Cs-137	9.47E+08	2.73E-14	7.93E-09	8.63E-09
Ba-139	4.96E+03	2.17E-15	2.40E-12	4.64E-11
Ba-140	1.10E+06	8.58E-15	2.56E-10	1.01E-09
La-140	1.45E+05	1.17E-13	6.87E-11	1.31E-09
La-141	1.42E+04	2.39E-15	9.40E-12	1.57E-10
La-142	5.55E+03	1.44E-13	8.74E-12	6.84E-11
Ce-141	2.81E+06	3.43E-15	2.55E-11	2.42E-09
Ce-143	1.19E+05	1.29E-14	6.23E-12	9.16E-10
Ce-144	2.46E+07	2.77E-15	2.92E-10	1.01E-07
Pr-143	1.17E+06	2.10E-17	1.68E-18	2.19E-09
Nd-147	9.49E+05	6.19E-15	1.82E-11	1.85E-09
Np-239	2.04E+05	7.69E-15	7.62E-12	6.78E-10
Pu-238	2.77E+09	4.88E-18	3.86E-10	7.79E-05
Pu-239	7.59E+11	4.24E-18	3.75E-10	8.33E-05
Pu-240	2.06E+11	4.75E-18	3.76E-10	8.33E-05
Pu-241	4.54E+08	7.25E-20	9.15E-12	1.34E-06
Am-241	1.36E+10	8.18E-16	1.60E-09	1.20E-04
Cm-242	1.41E+07	5.69E-18	9.41E-10	4.67E-06
Cm-244	5.72E+08	4.91E-18	1.01E-09	6.70E-05

Notes:

(1) All isotopic data contained in this Table is obtained from Reference 15B-4.

TABLE 15B-3

BREATHING RATES

Time Period	Breathing Rate (m^3sec)
0 - 8	3.47-4[1]
8 - 24	1.75-4
24 - 720	2.32-4

[1] 3.47-4 = 3.47×10^{-4}

TABLE 15B-4

BETA SKIN DOSE CONVERSION FACTORS

THIS TABLE HAS BEEN DELETED

APPENDIX 15C

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT

CYCLE SPECIFIC DATA

TABLE 15C.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 1 CYCLE 20

Section	Figure	Description[1]	Maximum Neutron Flux % of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux,%	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1		Loss of Feedwater Heater	NOTE 5	NOTE 5	NOTE 5	NOTE 5	NOTE 5	0.12	Moderate	0	0 sec
15.1.2	15C.1.2-1	Feedwater Controller Failure (100% Power, 108 Mlb_m/hr, Max Allowable Scram Time) EOC RPT Operable	222	1247	1268	1257	118	0.27	Moderate	14	4 sec estimate
15.1.3	15C.1.3-1	Pressure Regulator Failure – Open	102	1106	1129	1106	103	0.01	Moderate	2	See Text
15.1.4		Inadvertent Opening of Safety or Relief Valves	See Text						Moderate		
15.1.6		RHR Shutdown Cooling Malfunction	See Text						Moderate		
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Failure – Closed	See Text						Moderate		
15.2.2		Generator Load Reject – Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.2	15C.2.2-1	Generator Load Reject- Without Bypass (100% Power, 108 Mlb_m/hr, max allowable Scram Time) EOC RPT Operable	258	1263	1287	1306	117	0.27	Moderate	14	10 sec estimate
15.2.3		Turbine Trip – Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.3	15C.2.2-1	Turbine Trip – Without Bypass (100% Power, 108 Mlb_m/hr, Max Allowable Scram Time) EOC RPT Operable	258	1263	1287	1306	117	0.27	Moderate	14	10 sec estimate
15.2.4		Inadvertent MSIV Closure	See Text and Appendix 15E						Moderate		

TABLE 15C.0-1 (Cont'd)
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 1 CYCLE 20

Section	Figure	Description[1]	Maximum Neutron Flux % of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux,%	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.2.5		Loss of Condenser Vacuum	See Text and Appendix 15E						Moderate		
15.2.6		Loss of Auxiliary Power Transformer	See Text and Appendix 15E						Moderate		
15.2.6		Loss of All Grid Connections	See Text and Appendix 15E						Moderate		
15.2.7		Loss of All Feedwater Flow	See Text and Appendix 15E						Moderate		
15.2.8		Feedwater Piping Break	See Section 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		<u>DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE</u>									
15.3.1		Trip of One Recirculation Pump Motor	See Text and Appendix 15E						Moderate		
15.3.2		Trip of Both Recirculation Pump Motors	See Text and Appendix 15E						Moderate		
15.3.3	15C.3.3-1 & 15C.3.3-3	Seizure of One Recirculation Pump (Single Loop Operation)	67	1035	1070	1035	67	0.33	Limiting Fault		
15.3.4		Recirculation Pump Shaft Break	See Text						Limiting Fault		

TABLE 15C.0-1 (Cont'd)
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 1 CYCLE 20

Section	Figure	Description[1]	Maximum Neutron Flux % of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux,%	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.4		<u>REACTIVITY AND POWER ANOMALIES</u>									
15.4.1.1		RWE – Refueling	See Text						Infrequent		
15.4.1.2		RWE – Startup	See Text						Infrequent		
15.4.2		RWE - At Power, 108 Mlbs/hr, Bypass Operable	See Text	Note 5	Note 5	Note 5	Note 5	0.22	Moderate		
15.4.3		Control Rod Maloperation	See Subsections 15.4.1 and 15.4.2								
15.4.4		Startup of Idle Recirculation Loop	See Text and Appendix 15E						Moderate		
15.4.5		Recirculation Flow Controller Failure[3]	See Text	NOTE 5	NOTE 5	NOTE 5	NOTE 5	0.34	Moderate		
15.4.7		Misplaced Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		
15.4.7		Rotated Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		

TABLE 15C.0-1 (Cont'd)
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 1 CYCLE 20

Section	Figure	Description[1]	Maximum Neutron Flux % of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux,%	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.5		INCREASE IN REACTOR INVENTORY									
15.5.1		Inadvertent HPCI Pump Start (at 60% power)	See Text and Appendix 15E					0.39	Moderate		
15.5.3		BWR Transients That Increase Reactor Coolant Inventory	See Sections 15.1 and 15.2								

Notes

1. Unless otherwise stated, the plant initial condition listed in this table for transients is: 100% Power, 108 M bs/hr Flow, EOC-Reactor Pump Trip Operable, Bypass Operable, Realistic Scram Time.

2. Minimum MCPR operating limit for Single Loop Operation, see Text.

3. Recirculation Flow Controller Failure transients are initiated from low power/low flow conditions. This one started at 62 Mlbs/hr flow with main steam bypass operable.

4. Steam line pressure is at the turbine stop valve for events in which the turbine trips. For other transients the steam line pressure is assumed to be no higher than the reactor vessel dome pressure.

5. These Anticipated Operational Occurrences are analyzed as steady-state events.

TABLE 15C.1.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATING
UNIT 1

TIME, SECONDS	EVENT
0	Initiate a 100°F temperature reduction into the feedwater system.
2	Initial effect of unheated feedwater starts to raise core power level and steam flow, (Transport delay in feedwater piping is neglected).
≈40 (estimate)	APRM high neutron flux alarm sounds.
≈60 (estimate)	Reactor variables settle into new steady state, (below Scram trip point).
600 (estimate)	Operator begins to reduce core flow.
	The above times are estimates. This event is a relatively slow transient and the analysis was performed as a series of steady-state calculations.

TABLE 15C.1.2-1

SEQUENCE OF EVENTS FOR FEEDWATER
CONTROLLER FAILURE, MAXIMUM DEMAND
UNIT 1 CYCLE 20

TIME, SECONDS	EVENT
0	Initiate simulated failure of 127% upper limit on feedwater flow.
22.660	L8 vessel level setpoint trips main turbine and feedwater pumps.
22.730	Reactor scram trip actuated from main turbine stop valve position switch.
22.760	Bypass Valves actuated
22.835	Recirculation pump trip (RPT) actuated by stop valve position switch.
25.269	Second group of safety/relief valves activate due to high pressure. (First group out of service)
25.709	Third group of safety valves activate.

Initial Conditions:

Power	= 100%
Flow	= 108 Mlbs/hr
Bypass	= Operable
RPT	= Operable
Scram Time	= Maximum Allowable
Exposure	= EOC

TABLE 15C.1.3-1

SEQUENCE OF EVENTS FOR PRESSURE REGULATOR FAILURE - OPEN UNIT 1

TIME, SECONDS	EVENT
0	Initial conditions, maximum limit on steam flow to turbine.
0.2	Main turbine bypass valves full open
10.82	Main steamline isolation trip occurs.
11.63	Initiation of scram trip signal, 0.06 seconds after the Main steam isolation valves reach 85% open position.
15.50	Pressure in reactor vessel reaches a minimum and starts to increase.
15.82	MSIV's are fully closed.
48 (est)	Relief valves at lowest setting start to cycle to remove decay heat.

TABLE 15C.2.2-1

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS UNIT 1 CYCLE 20

TIME, SECONDS	EVENTS
~0	Turbine-generator detection of loss electrical load.
0	Generator lockout relays act to initiate turbine control valve fast closure.
0.000	Turbine control valves closure on GLR (Generator Load Reject)
0.070	Scram initiated; TCV (Turbine Control Valve) fast closure (Trip oil pressure low)
0.175	A&B RPT: Turbine Control Valve fast closure
	Group 1 safety valves out of service.
1.96	Group 2 safety valves activate due to high pressure.
2.20	Group 3 safety valves activate due to high pressure.

Initial Conditions

Power: 100%	Flow: 108 Mlbs/hr
Bypass: Inoperable	Scram: Maximum Allowable
RPT: Operable	

TABLE 15C.3.3-1

PUMP SEIZURE ACCIDENT FROM TWO LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 1

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
0.8	Jet Pump Diffuser Flow Reverses in Seized Loop
1.31	Minimum CPR

Note: Figures include a 0.5 second null transient.

TABLE 15C.4.2-1

SEQUENCE OF EVENTS - RWE IN POWER RANGE
UNIT 1

ELAPSED TIME	EVENT
0	Core is assumed to be at rated conditions.
0	Operator selects and withdraws the maximum worth control rod.
~1 sec	The total core power and the local power in the vicinity of the control rod increase.
~5 sec	The operator ignores warning and continues withdrawal.
~15 sec	The RBM system indicates excessive localized peaking.
~15 sec	The operator ignores warning and continues withdrawal.
~20 sec	The RBM system initiates a rod block inhibiting signal, credit is taken for this signal. Further control rod withdrawal is blocked.
~40 sec	Reactor core stabilizes at higher core power level.
~60 sec	Operator attempts to re-insert control rod to reduce core power level.
~80 sec	Core stabilizes at rated conditions.

TABLE 15C.4.5-1

SEQUENCE OF EVENTS FOR RECIRCULATION FLOW CONTROLLER FAILURE UNIT 1

TIME, SECONDS	EVENT
0	Master Flow Controller fails initiating a slow run-up of both reactor recirculation pumps
~220	Reactor high flux scram (analytical setpoint, 122%).
~220	Two relief valves open at 1120.7 psia.
~230	Two relief valves reseat at 1045.7 psia.

This sequence of events is for the event initiated from:

Power	=	69%
Flow	=	60 Mlbs/hr
Bypass	=	Inoperable
Exposure	=	EOC

TABLE 15C.4.7-1

UNIT 1

SEQUENCE OF EVENTS FOR MISLOADED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in the wrong position.
2. Subsequently, the bundle intended for this position is placed in the position of the previous bundle.
3. During core verification procedure, error is not observed.
4. Plant is brought to full power operation without detecting misplaced bundle.
5. Plant continues to operate.

SEQUENCE OF EVENTS FOR ROTATED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in its proper location but rotated either 90° or 180° from its proper orientation.
2. During core verification procedure this error is not observed.
3. Plant is brought to full power operation without detecting rotated bundle.
4. Plant continues to operate.

TABLE 15C.4.9-1

SEQUENCE OF EVENTS FOR CONTROL ROD DROP ACCIDENT UNIT 1

APPROXIMATE ELAPSED TIME	EVENT
	Reactor is operating at rod density pattern of up to 50%.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the Banked Position Withdrawal Sequence (BPWS).
	Decoupled control rod sticks in the fully inserted or in an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
$<$1 second	Reactor goes on a positive period and initial power increase is terminated by the Doppler effect.
$<$1 second	APRM 120% power signal scrams the reactor.
$<$5 seconds	Scram terminates the accident.

TABLE 15C.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS UNIT 1 CYCLE 20

1.	Thermal Power Level, MWT Rated Value Analysis Value	3952 (100%) 4031(102%)
2.	Steam Flow, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.624
3.	Maximum Core Flow, Mlbs/hr	108.0(3)
4.	Feedwater Flow Rate, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.592
5.	Feedwater Temperature, °F (At 100% Power and 100 Mlbs/hr)	403.3
6.	Vessel Dome Pressure,psig (At 100% Power and 100 Mlbs/hr)	1035.7
7.	Vessel Core Pressure,psig at Channel exit (At 100% Power and 100 Mlbs/hr)	1047.4
8.	Turbine bypass Capacity, % Rated	21.5%
9.	Core Coolant Inlet Enthalpy, BTU/lb (At 100% Power and 100 Mlbs/hr)	523.6(2)
10.	Turbine Inlet Pressure, psia	976.3
11.	Fuel Types	ATRIUM-10
12.	Core Average Gap Conductance, BTU/hr-ft^2-°F	500 to 1600(1)
13.	Core Leakage Flow,%	~10%(2)
14.	Required MCPR Operating Limit	See Unit 1 COLR (FSAR section 16.3 – TRMs)
15.	MCPR Safety Limit	See Table 15C.0-3
16.	Doppler Coefficient	See Note 4

TABLE 15C.0-2 (Cont'd)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 1 CYCLE 20

17.	Void Coefficient		See Note 4
18.	Core Average Rated Void Fraction		See Note 4
19.	Scram Reactivity Analysis Data		See Note 4
20.	Control Rod Scram Times		Table 15C.0-5
21.	Jet Pump Ratio		2.1
22.	Safety Relief Valve Capacity (16 Valves) Percent of Rated Steam Flow		87%
23.	Relief Function Delay, sec		0.1
24.	Relief Function Response, sec		≤0.15
25a.	Relief Mode Set Points for Safety/Relief Valves, psig		2 @ 1106 4 @ 1116 4 @ 1126 3 @ 1136 3 @ 1146
26b.	Safety mode Set Points for Safety/Relief valves, psig		2 @ 1175 6 @ 1195 8 @ 1205
26.	Number of Valve Groups Simulated		3
27.	High Flux Trip, % Rated Analysis set point		122
28.	High Pressure Trip, Analysis Set Point, psig		1105
29.	Vessel Level Trips, Nominal Setpoints Inches Above(+), Below (-) Dryer Skirt Bottom, (See Note 5)	High Level Low Level Low Low Level Low Low Low Level	(L8)≤ 54 (L4)≤ 30 (L3)≥ 13 (L2)≥ -38 (L1)≥-129
30.	APRM Thermal Trip, Analytical Set Point,% Rated		118

TABLE 15C.0-2 (Cont'd)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 1 CYCLE 20

31.	Recirculation Pump Trip Delay, sec	0.175
32.	Recirculation Pump Trip Inertia for Analysis, lb_m-ft^2	16,800

NOTES

1. Gap conductance for reactor system behavior is determined for the fuel types within the core as a function of power and exposure. The hot bundle gap conductance is based on the fuel type that is expected to be limiting. It is also determined based on the initial hot bundle power and exposure.

2. Inlet enthalpy and leakage flow are determined for each initial condition analyzed.

3. Core flow shown is the maximum. It is varied depending on the initial conditions being analyzed.

4. The physics characteristics are based on initial conditions determined from a 3-D simulation of the core over a range of power, flow, and pressure conditions. For certain transient analyses this data is transferred and collapsed for use in a 1-D reactor core/system transient simulation model of SSES unit 1.

5. Analytical limits for level setpoints include drift and uncertainty allowances.

TABLE 15C.3.3-2

PUMP SEIZURE ACCIDENT FROM SINGLE LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 1

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
N/A	Jet Pump Diffuser Flow Reverses in Seized Loop
1.84	Minimum CPR

Table 15C.4.7-2

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TABLE 15C.4.9-2

CONTROL ROD DROP ACCIDENT
UNIT 1 CYCLE 20

Cycle Exposure, GWD/MTU	BOC
Control Rod Sequence	B
Rod Group	1
Dropped Rod Location	14-47
Dropped Rod Worth	10.24 mk (from 00 to 48)
Number of Fuel Rods with Fuel Enthalpy above 170 cal/gm	$\leq$2000
Peak Deposited Enthalpy, cal/gm	191.3

TABLE 15C.0-3

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
UNIT 1 CYCLE 20

MCPRSL for Two Loop Operation	MCPRSL for Single Loop Operation
1.09	1.12

Table Rev. 58

Table 15C.3.3-3

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Table 15C.4.9-3

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TABLE 15C.0-4

UNIT 1

MINIMUM MCPR REQUIREMENT
FOR
SINGLE LOOP OPERATION

MCPR Safety Limit	1.08	1.09	1.10	1.11	1.12	1.13
Minimum MCPR Requirement	1.41	1.42	1.43	1.44	1.45	1.46

(Based on Analysis of Pump Seizure Accident in Single Loop Operation)

MINIMUM MCPR REQUIREMENT
FOR
TWO LOOP OPERATION

MCPR Safety Limit	1.07	1.08	1.09	1.10	1.11
Minimum MCPR Requirement	1.29	1.30	1.31	1.32	1.33

(Based on Analysis of Pump Seizure Accident in Two Loop Operation)

TABLE 15C.3.3-4

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Table 15C.4.9-4

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TABLE 15C.0-5

AVERAGE SCRAM INSERTION TIMES
UNIT 1 CYCLE 20

Control Rod Position	Average Scram Time to Position (Seconds)	
	Realistic	Maximum Allowable
45	0.470	0.520
39	0.630	0.860
25	1.500	1.910
5	2.700	3.440
Scram Time Fraction	0.0	1.0

The times listed above do not include rod tip delay

Table 15C.3.3-5

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Table 15C.4.9-5

This Table Has Been Deleted

TABLE 15C.3.3-6

THIS TABLE HAS BEEN DELETED

Table 15C.4.9-6

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Table 15C.3.3-7

This Table Has Been Deleted

Table 15C.4.9-7

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Table 15C.3.3-8

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Table 15C.4.9-8

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Table 15C.0-1A

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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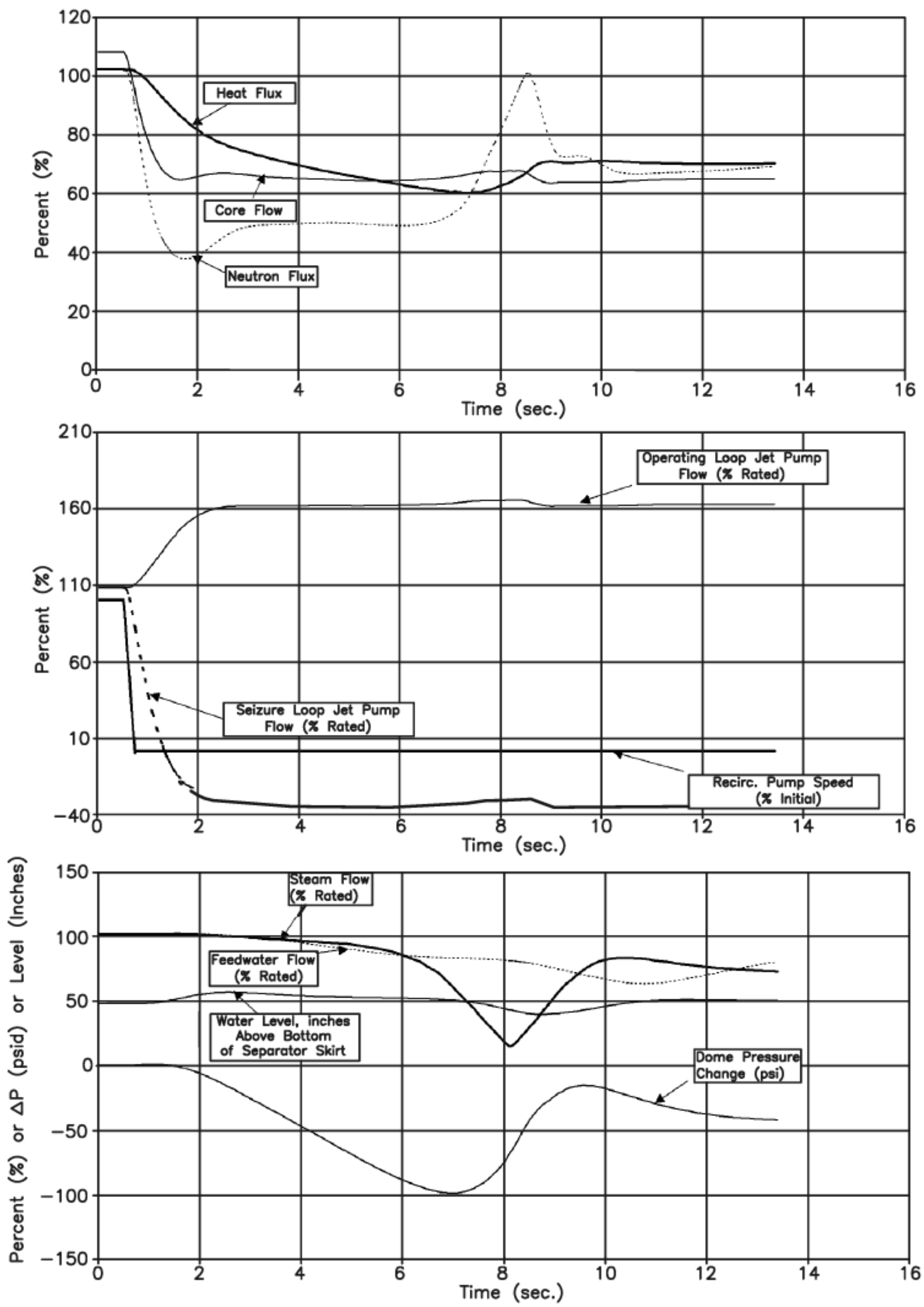
FIGURE 15C.0-1, Rev. 57



FSAR REV.65

*100% Rated Core Power is Assumed to be 3952 MWth

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
TWO LOOP OPERATION
4031 MWth*/108Mlbm/HR
CORE RESPONSE

FIGURE 15C.3.3-1, Rev 59

AutoCAD: Figure Fsar 15C_3_3_1.dwg

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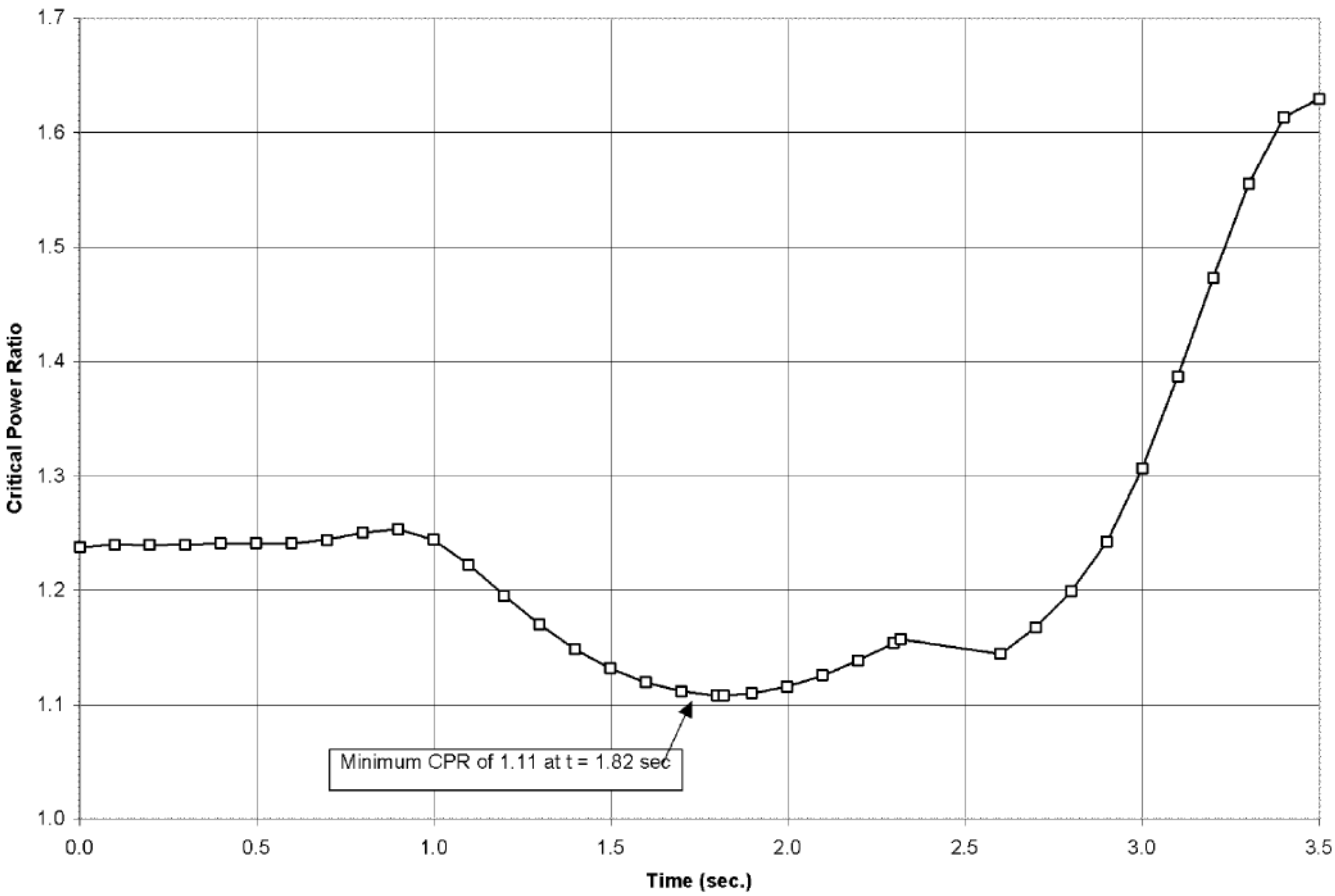
RENUMBERED TO 15.4-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Renumbered to 15.4-1
FIGURE 15C.4.9-1, Rev. 55

AutoCAD Figure 15C_4_9_1.doc



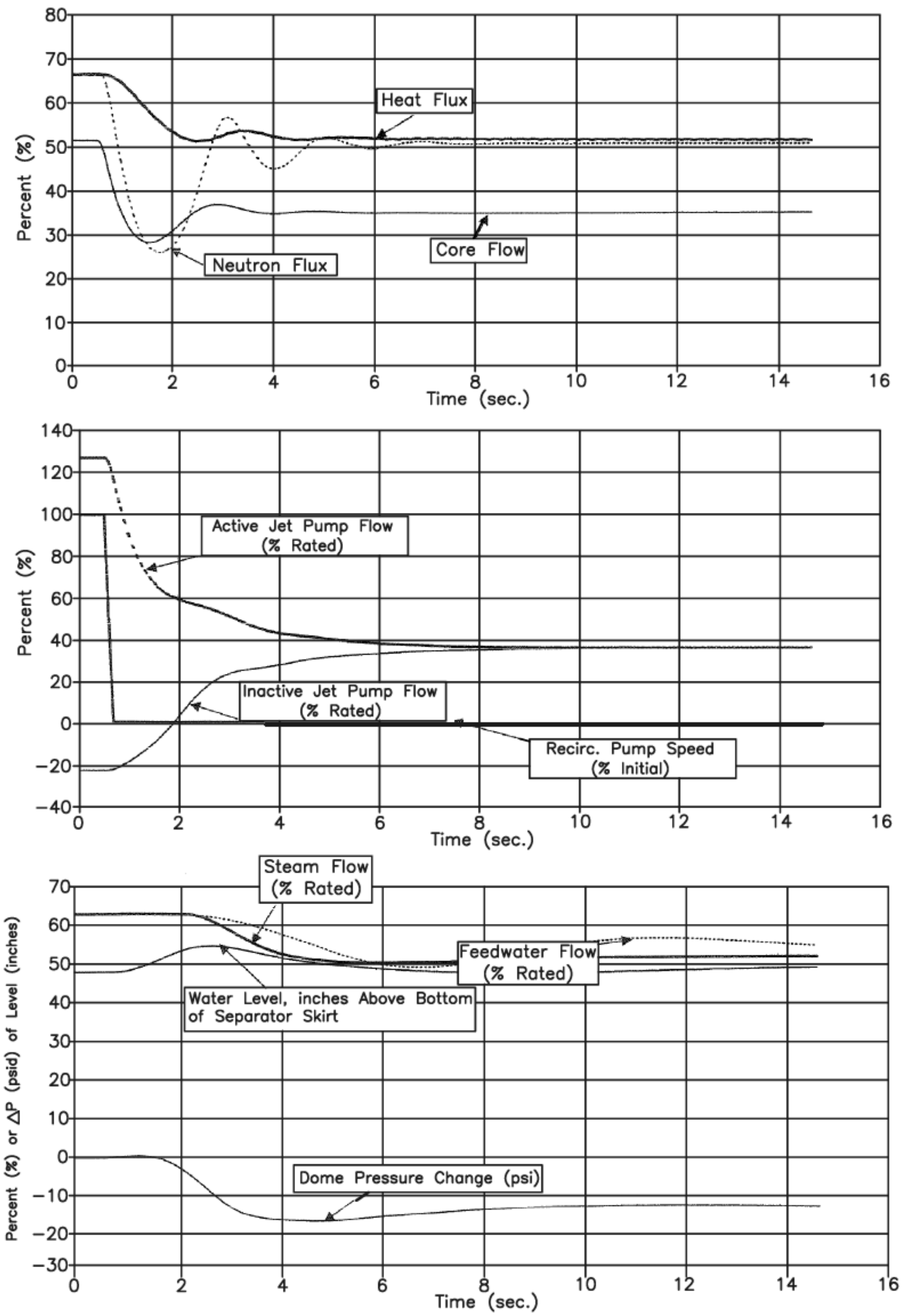
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
TWO LOOP OPERATION
4031 MWth/108Mlbm/HR
TYPICAL CPR RESPONSE

FIGURE 15C.3.3-2, Rev 59

AutoCAD: Figure Fsar 15C_3_3_2.dwg



FSAR REV.65

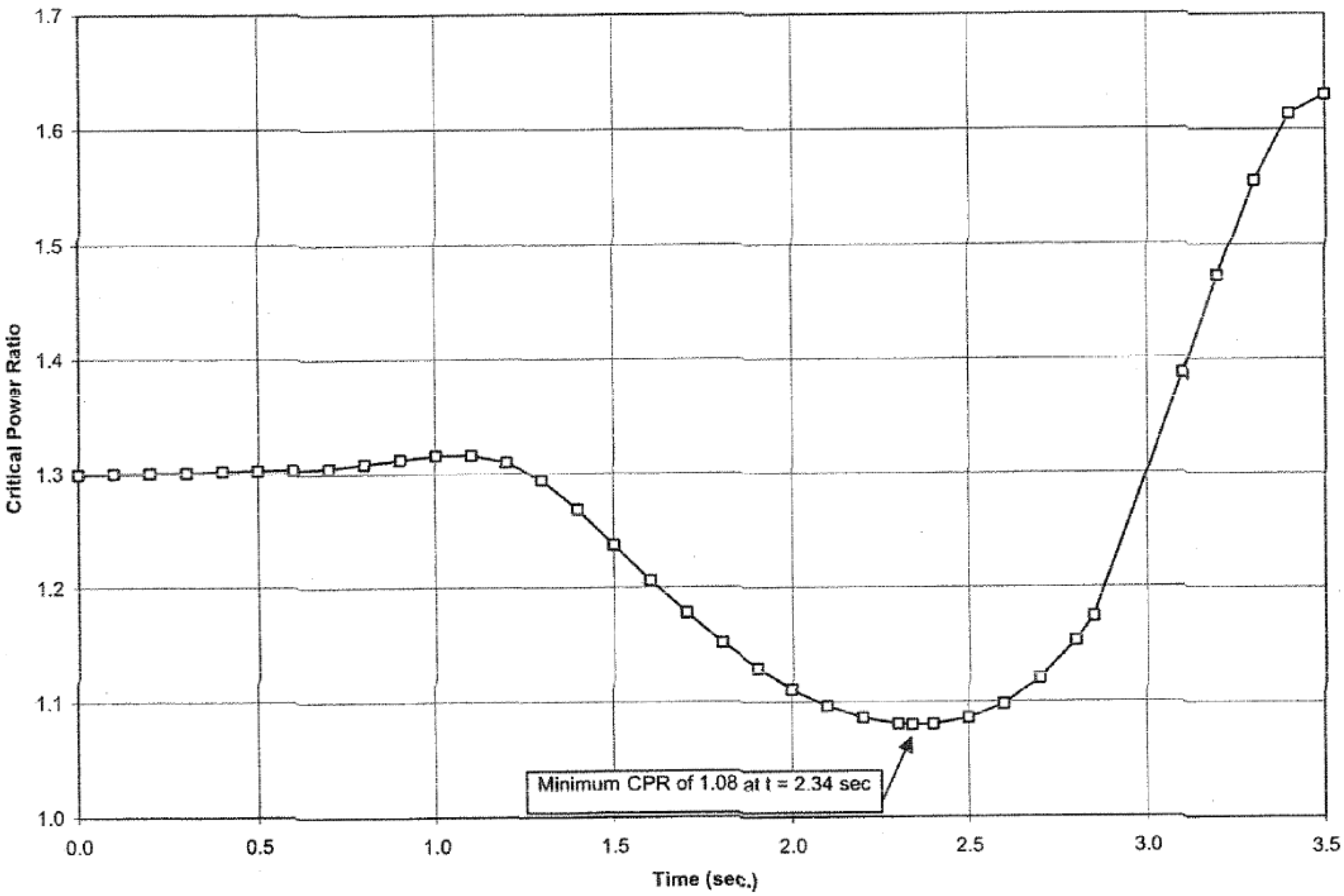
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1
FINAL SAFETY ANALYSIS REPORT

*100% Rated Core Power is Assumed to be 3952 MWth

PUMP SEIZURE ACCIDENT
SINGLE LOOP OPERATION
2652 MWth*/52Mlbm/HR
CORE RESPONSE

FIGURE 15C.3.3-3, Rev 1

AutoCAD: Figure Fsar 15C_3_3_3.dwg



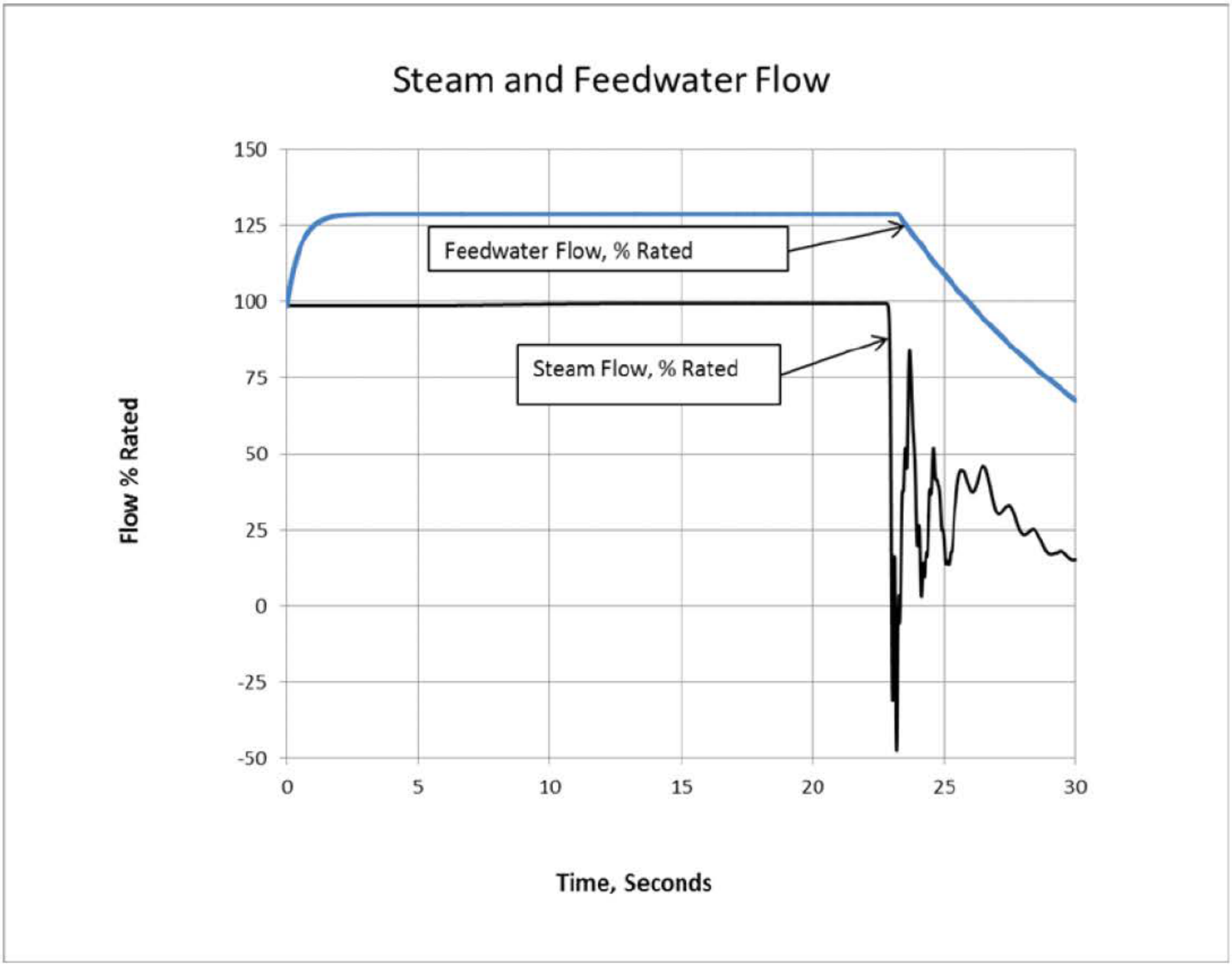
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
SINGLE LOOP OPERATION
2652 MWth/52 Mlbm/HR
TYPICAL CPR RESPONSE

FIGURE 15C.3.3-4, Rev 1

AutoCAD: Figure Fsar 15C_3_3_4.dwg



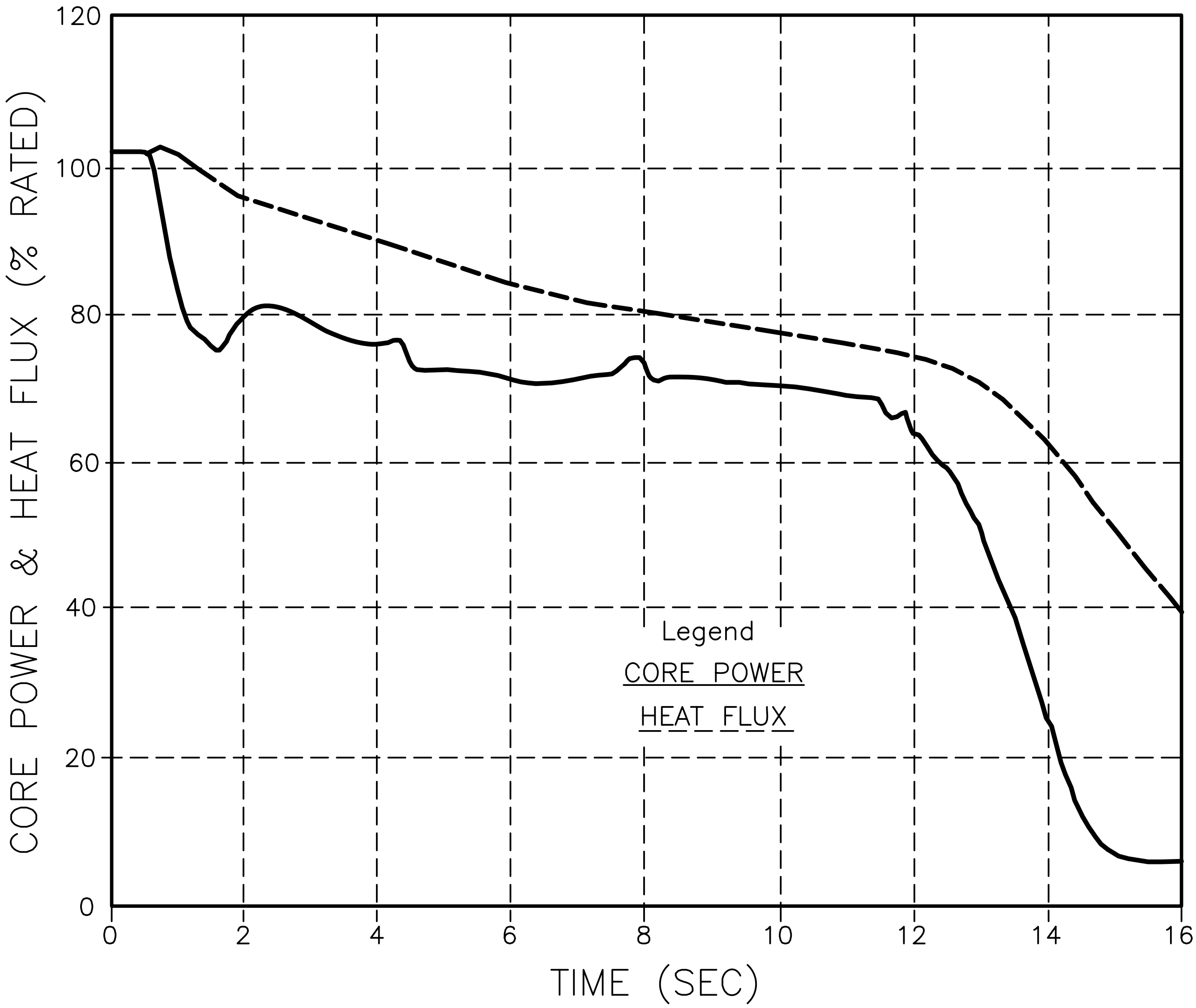
FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15C.1.2-1-1, Rev 64

AutoCAD: Figure Fsar 15C_1_2_1_1.dwg



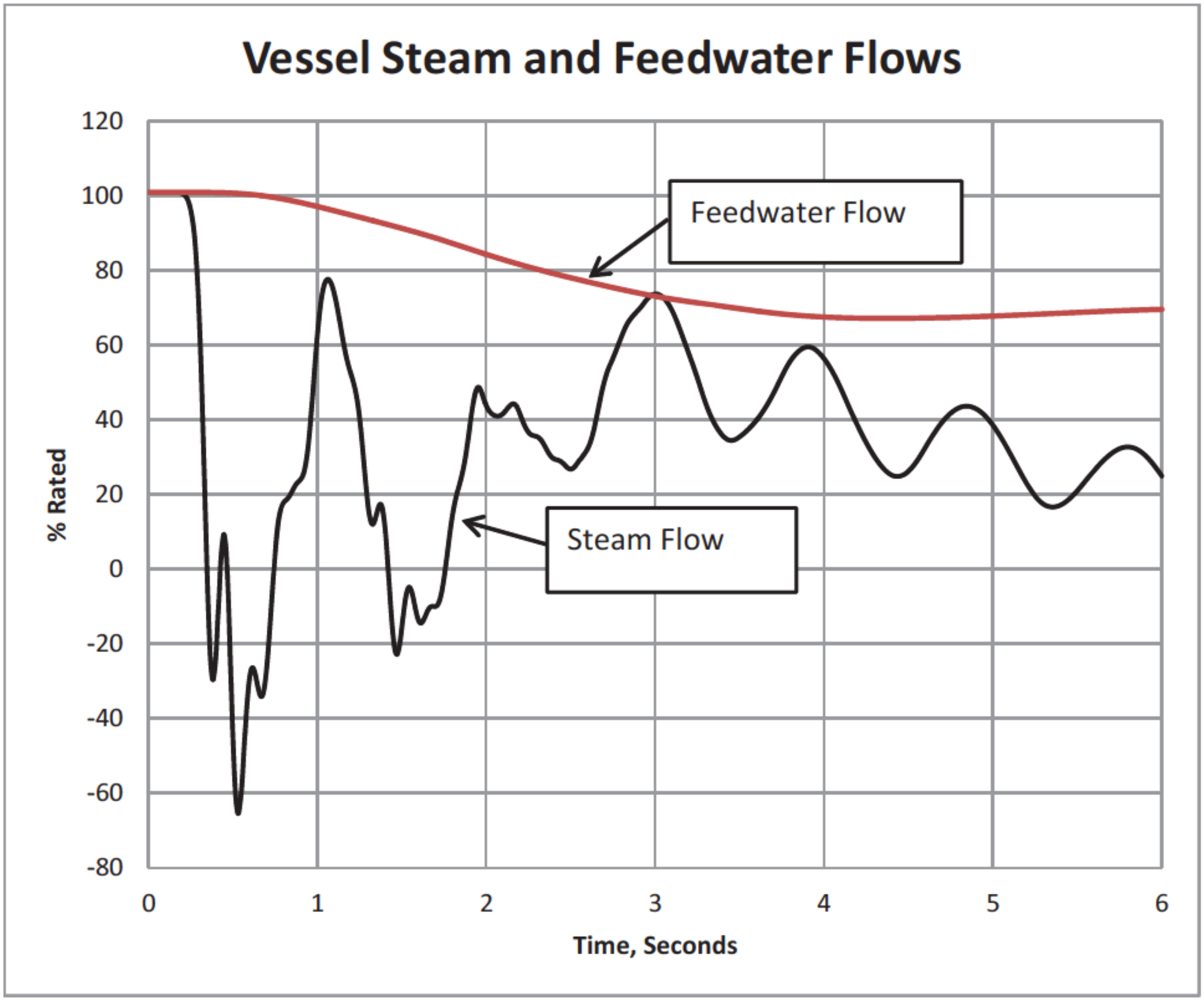
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 MWt

FIGURE 15C.1.3-1-1, Rev 55

AutoCAD: Figure Fsar 15C_1_3_1_1.dwg



FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15C.2.2-1-1, Rev 63

AutoCAD: Figure Fsar 15C_2_2_1_1.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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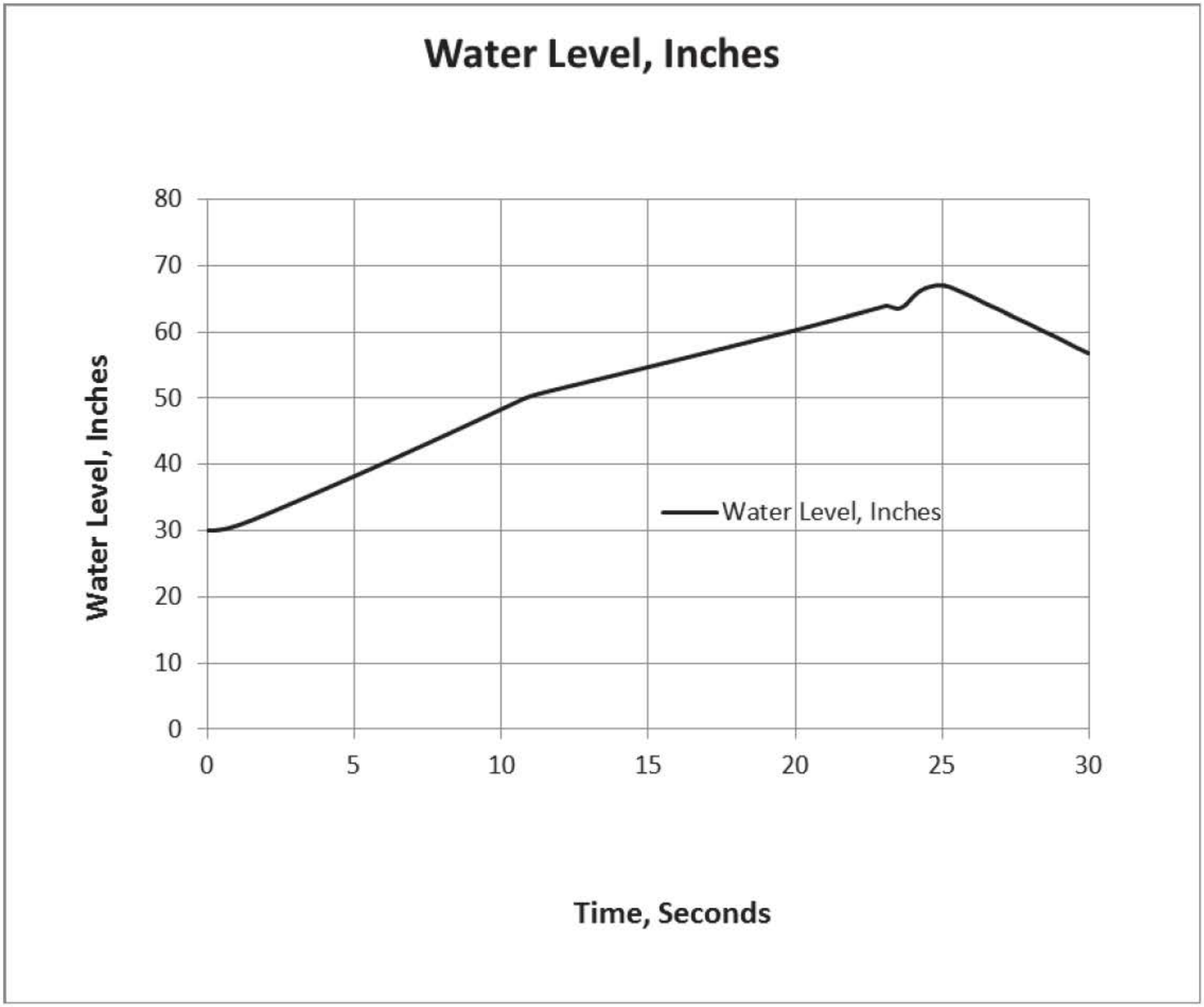
FIGURE 15C.4.5-1-1, Rev. 56

AutoCAD Figure 15C_4_5_1_1.doc



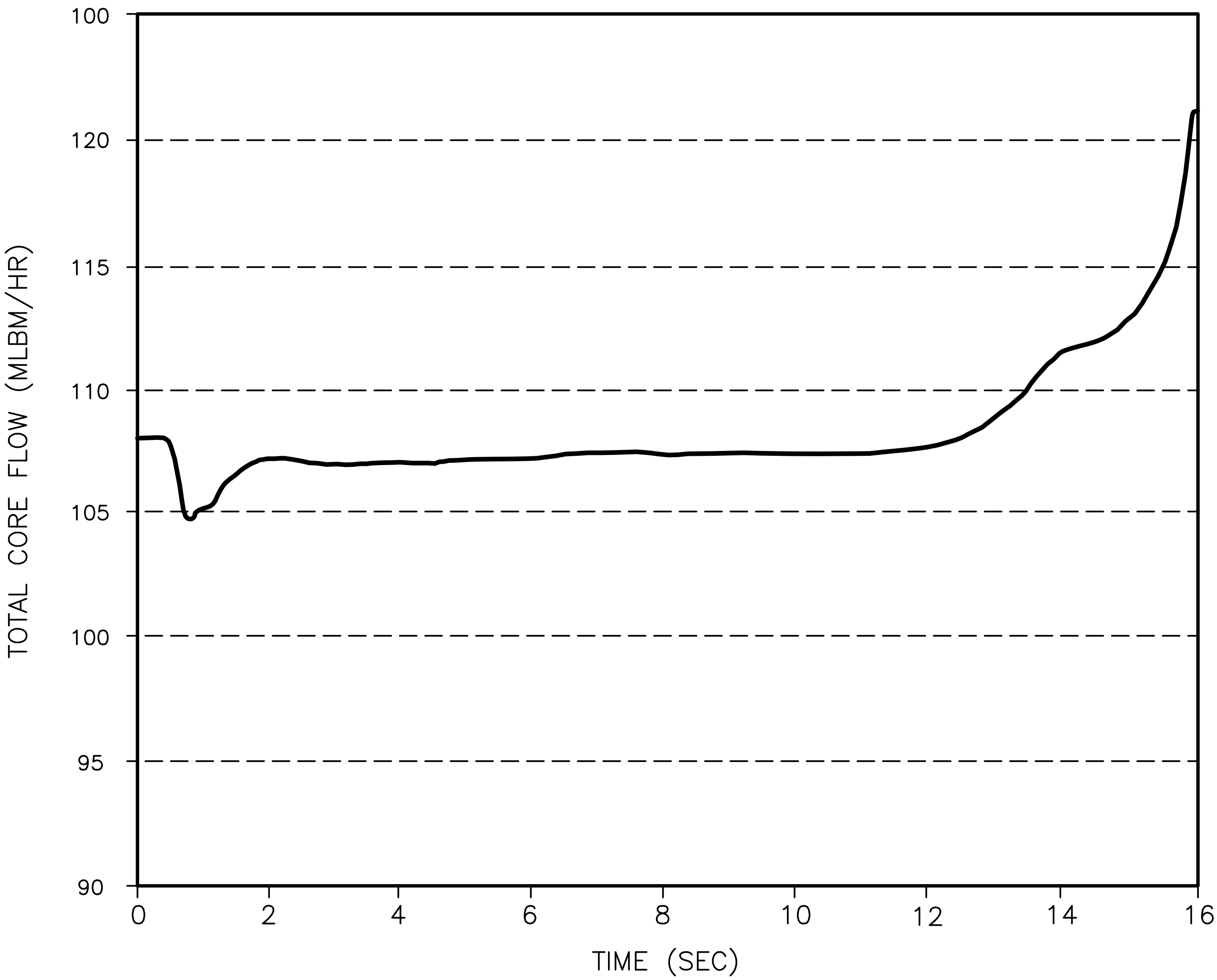
FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15C.1.2-1-2, Rev 64

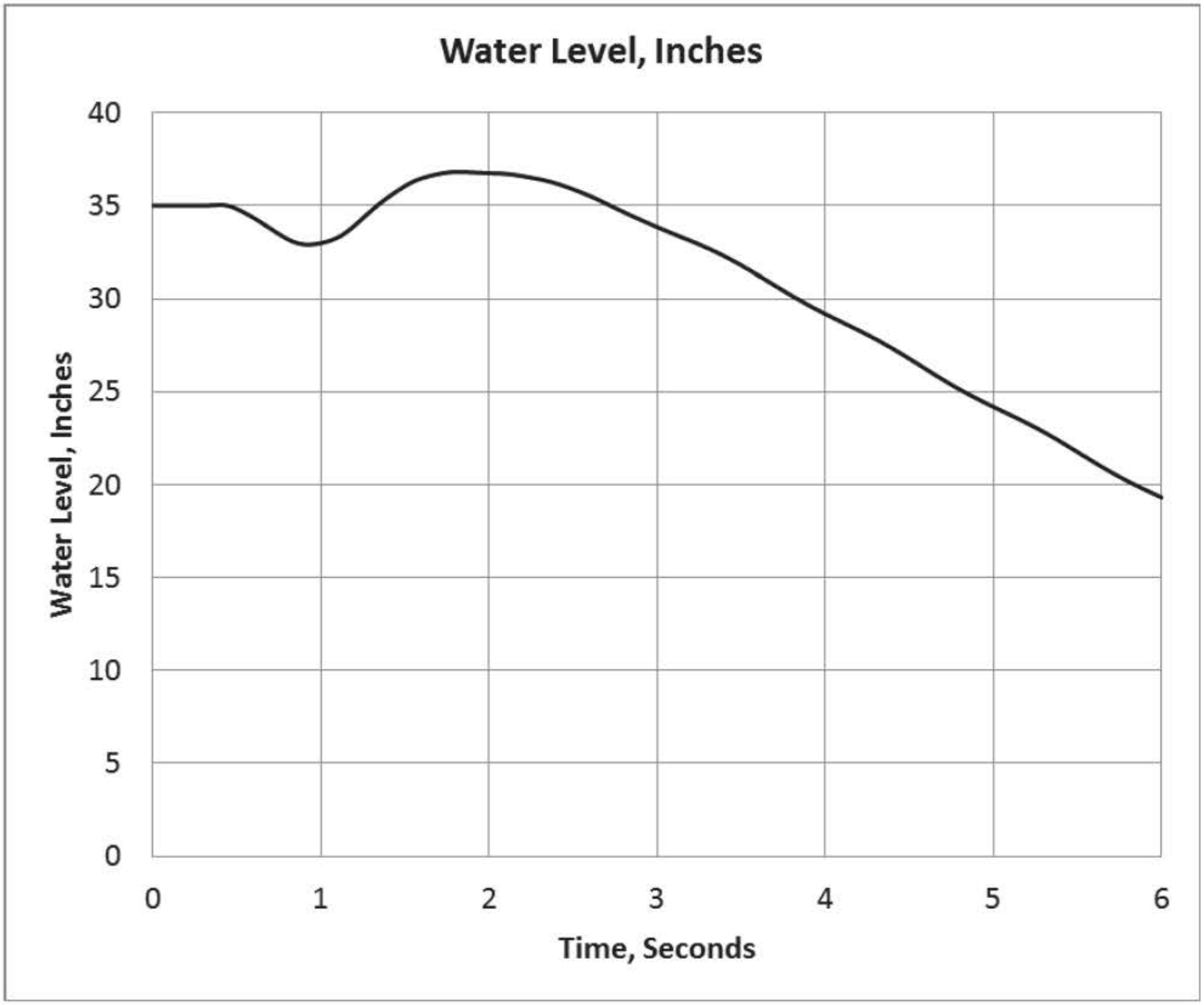
AutoCAD: Figure Fsar 15C_1_2_1_2.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
PRESSURE REGULATOR FAILURE STEAM FLOW AT 130% OF STEAM FLOW AT 3441 MWt FIGURE 15C.1.3-1-2, Rev 55

AutoCAD: Figure Fsar 15C_1_3_1_2.dwg



FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15C.2.2-1-2, Rev 63

AutoCAD: Figure Fsar 15C_2_2_1_2.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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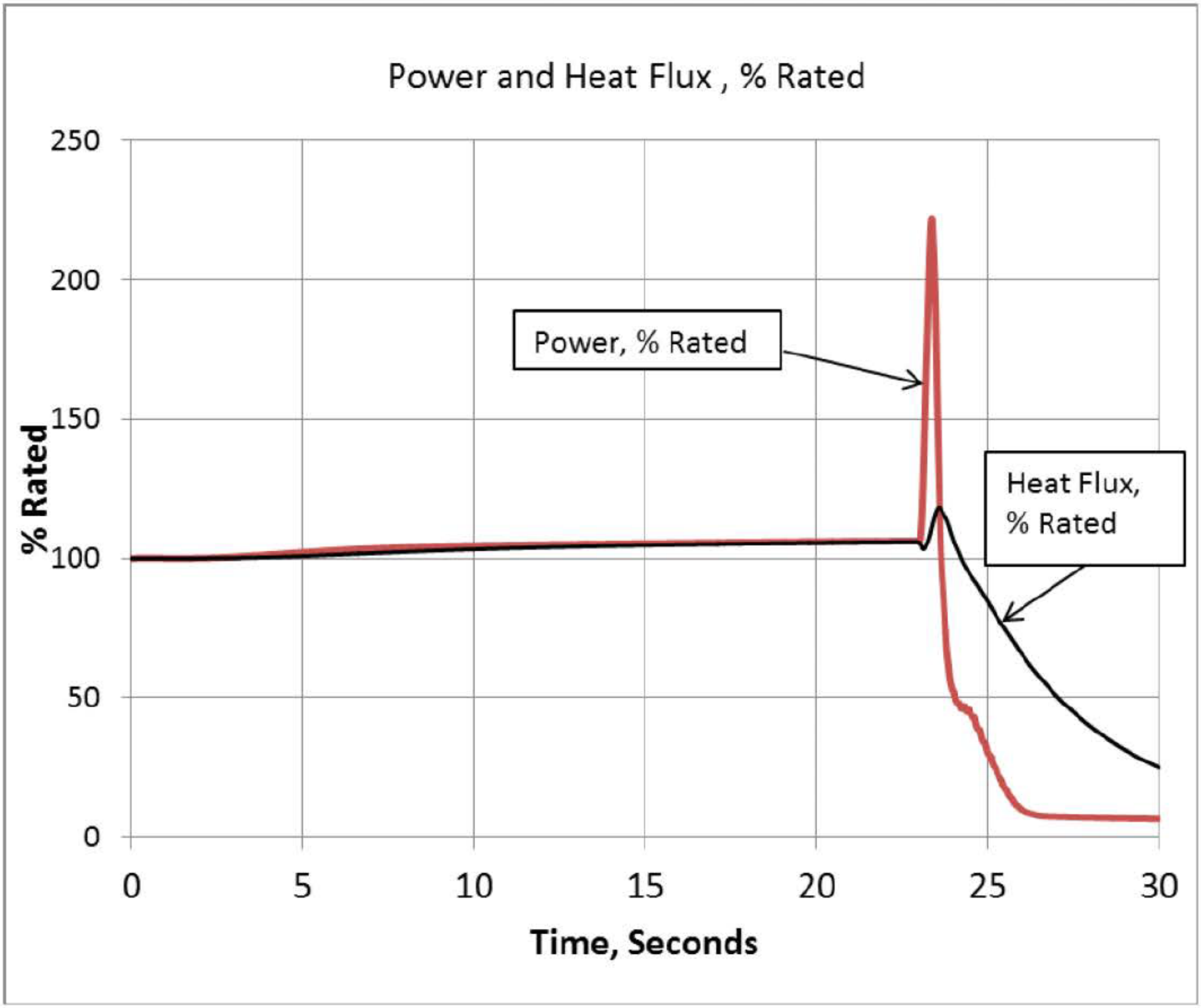
FIGURE 15C.4.5-1-2, Rev. 56

AutoCAD Figure 15C_4_5_1_2.doc



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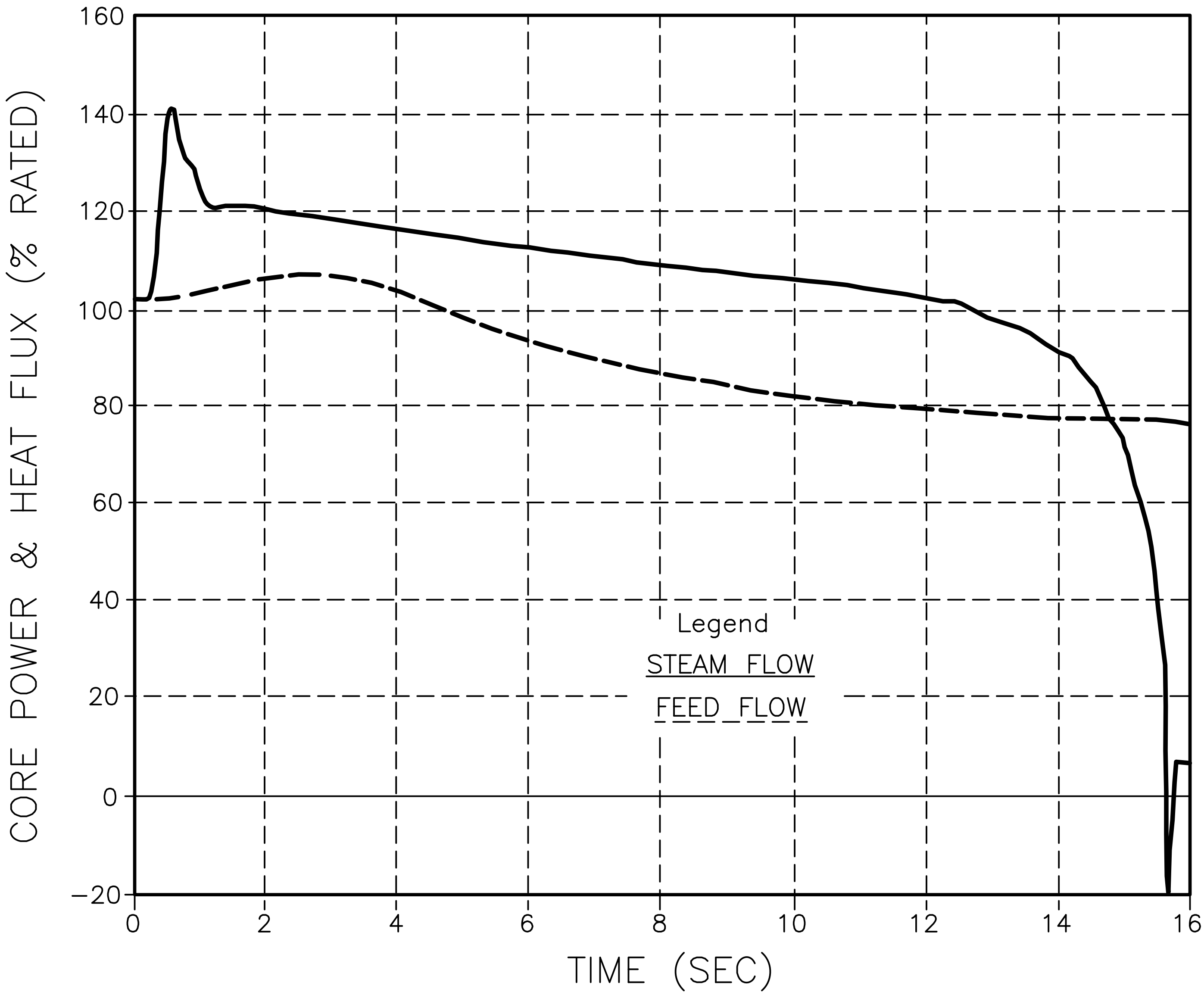
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15C.1.2-1-3, Rev 64

AutoCAD: Figure Fsar 15C_1_2_1_3.dwg



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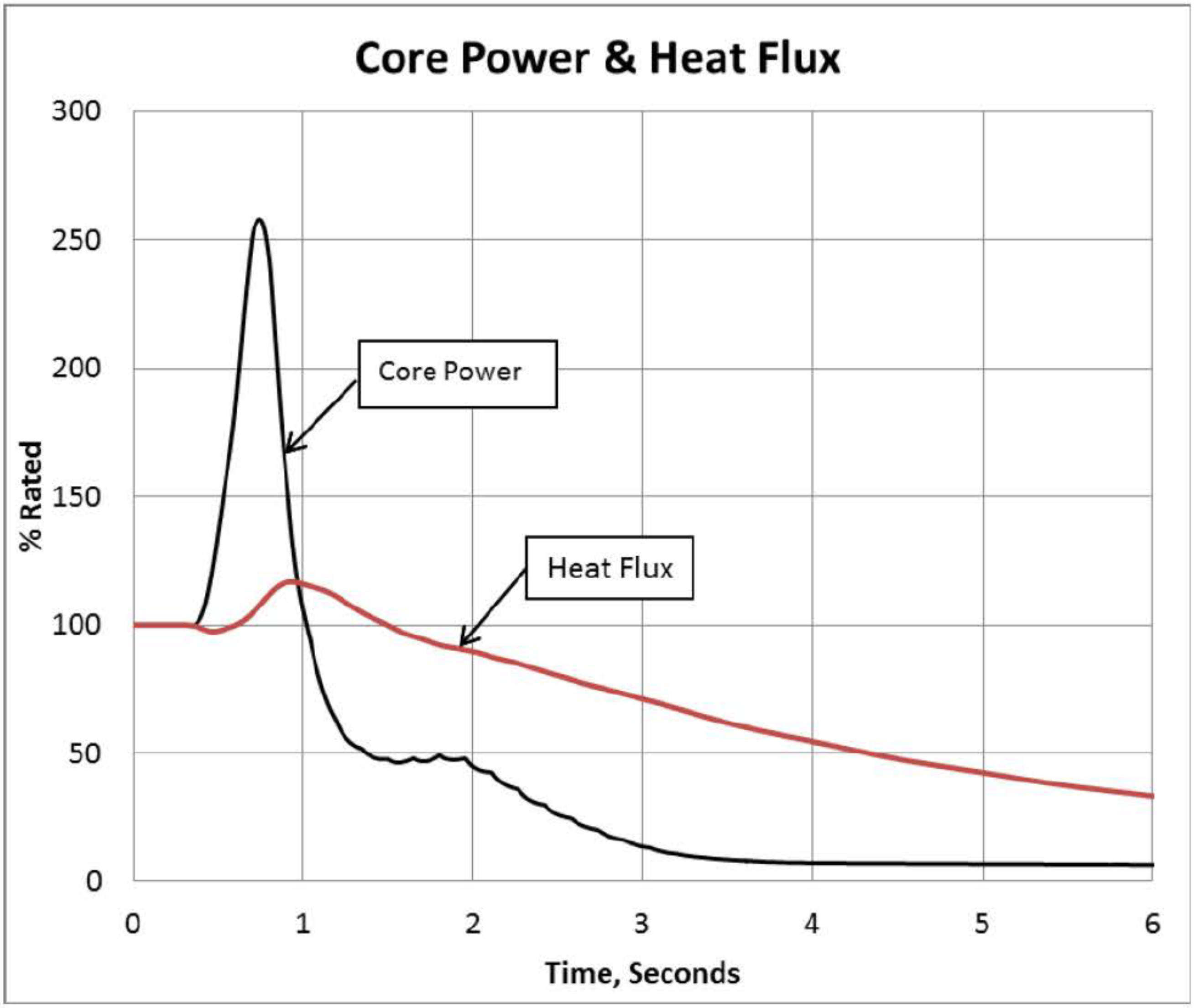
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 MWt

FIGURE 15C.1.3-1-3, Rev 55

AutoCAD: Figure Fsar 15C_1_3_1_3.dwg



FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15C.2.2-1-3, Rev 63

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Deleted FIGURE 15C.4.5-1-3, Rev. 56

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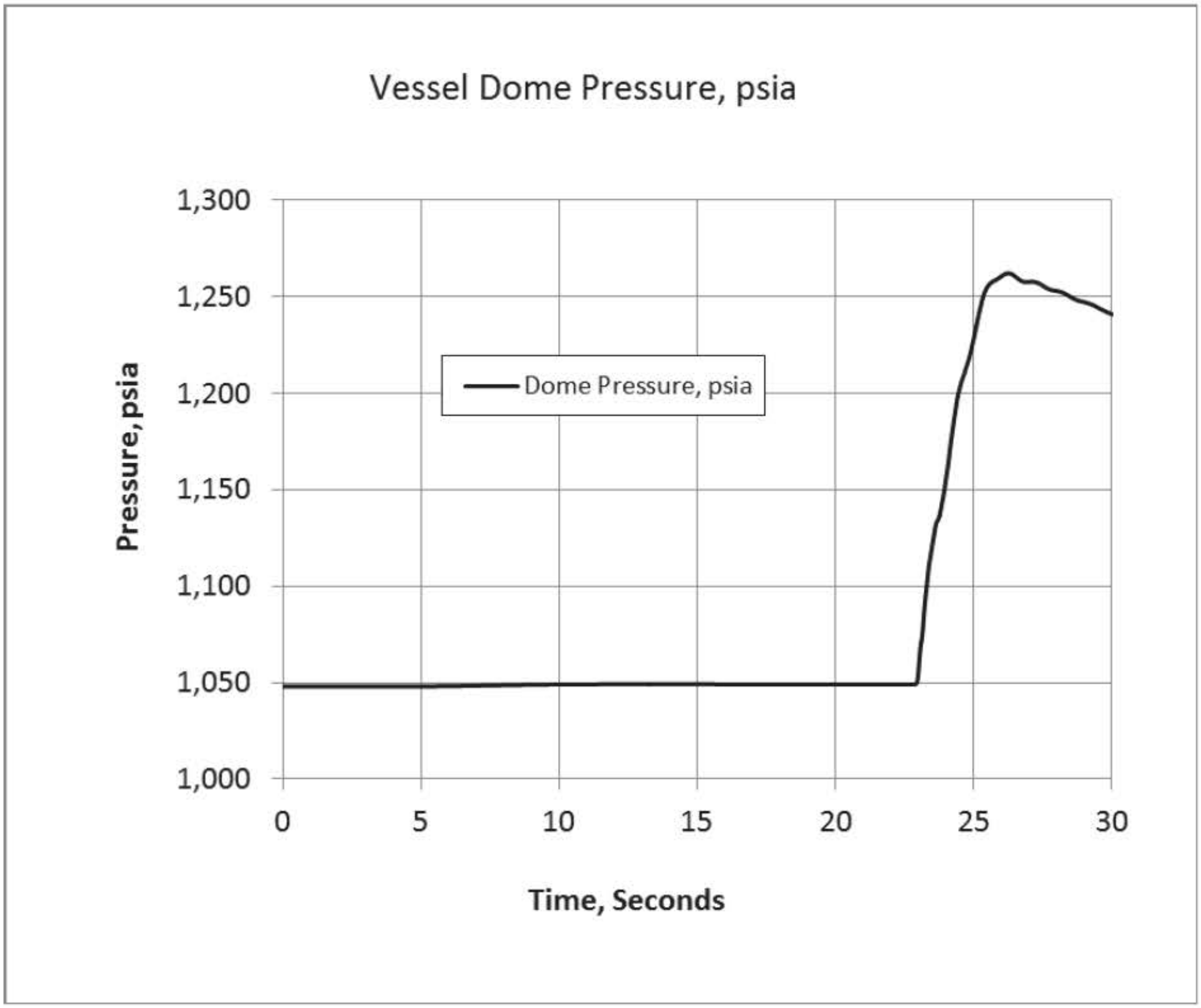

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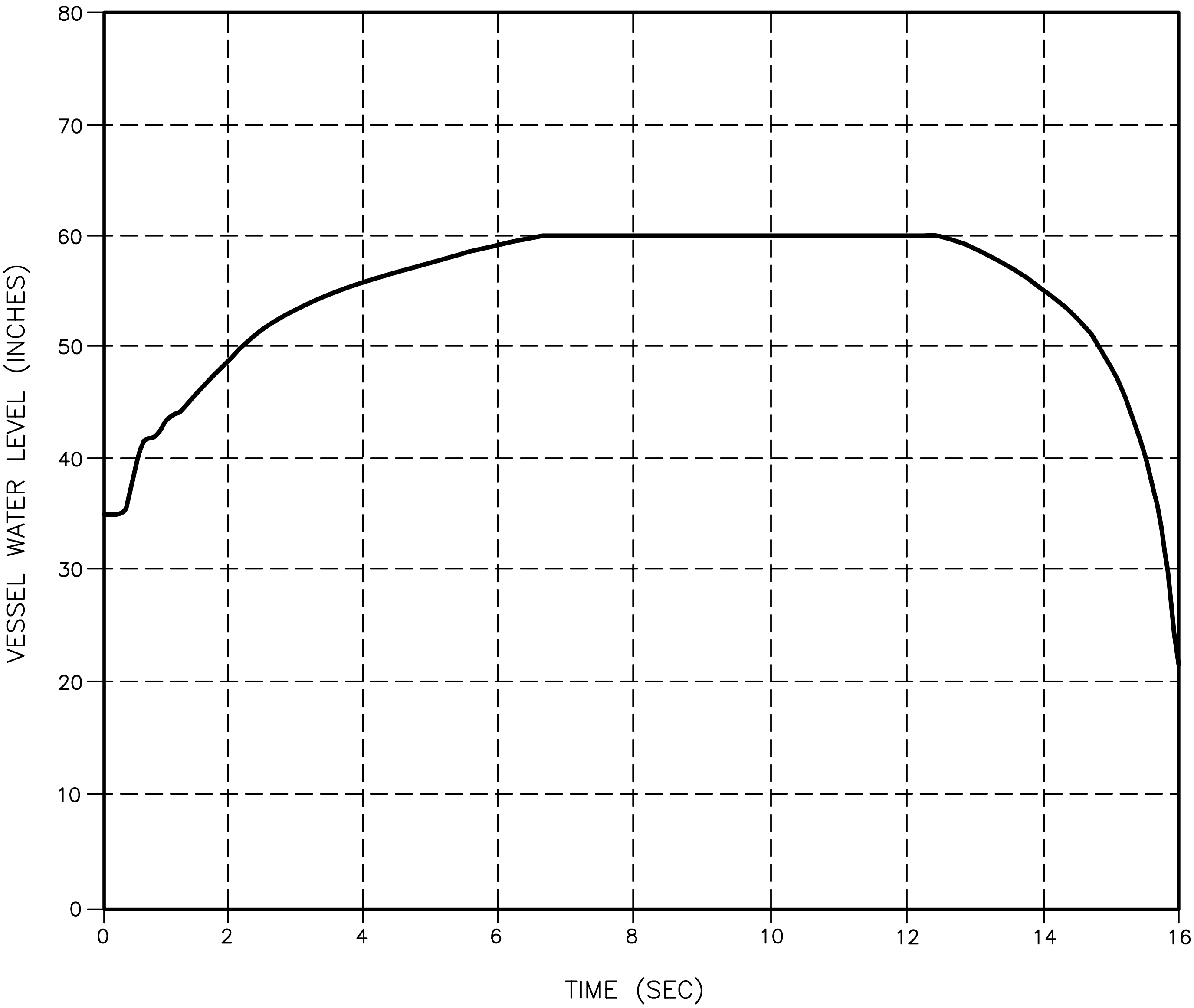
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15C.1.2-1-4, Rev 64

AutoCAD: Figure Fsar 15C_1_2_1_4.dwg



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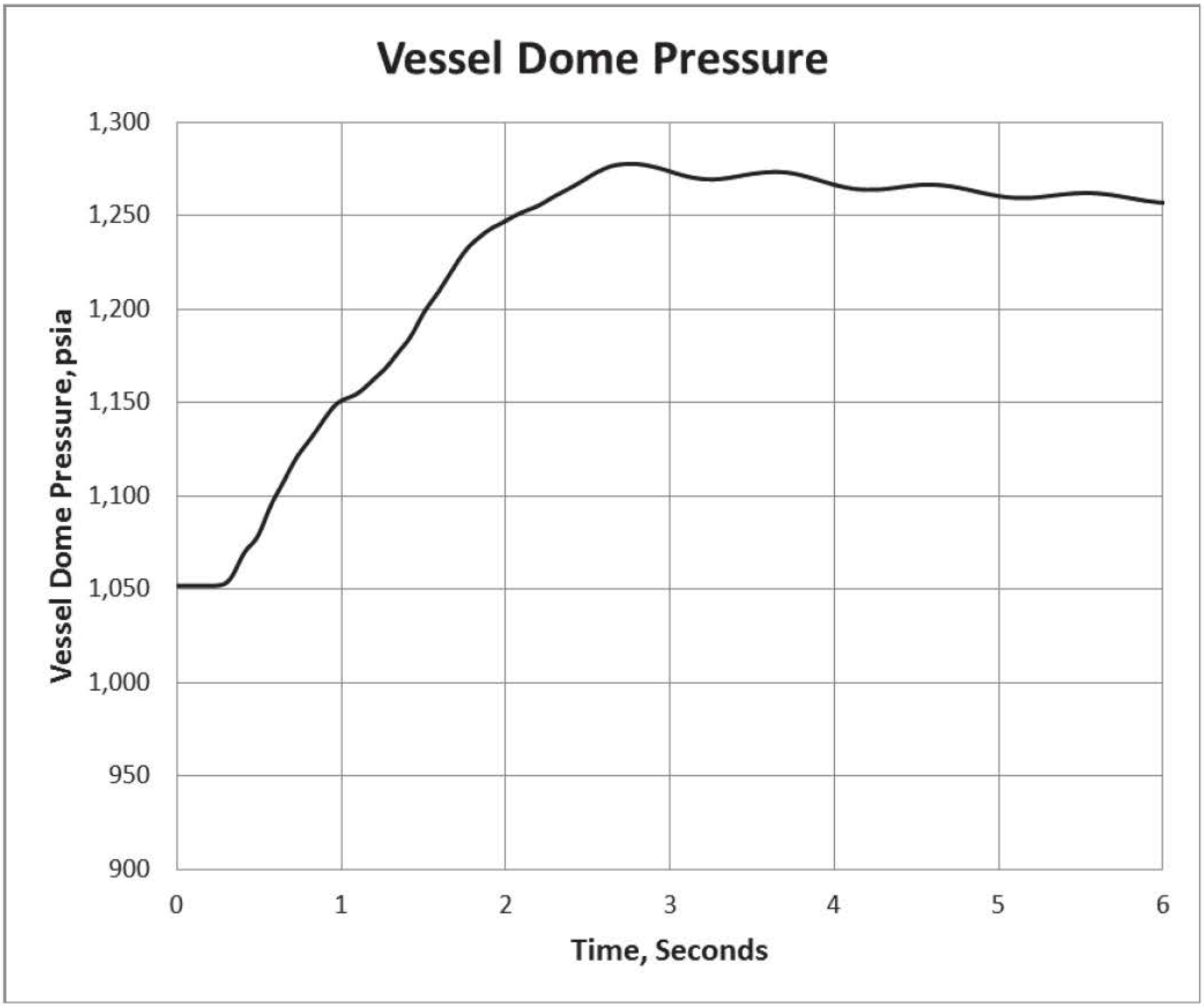
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 MWt

FIGURE 15C.1.3-1-4, Rev 55

AutoCAD: Figure Fsar 15C_1_3_1_4.dwg



FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15C.2.2-1-4, Rev 63

AutoCAD: Figure Fsar 15C_2_2_1_4.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15C.4.5-1-4, Rev. 56

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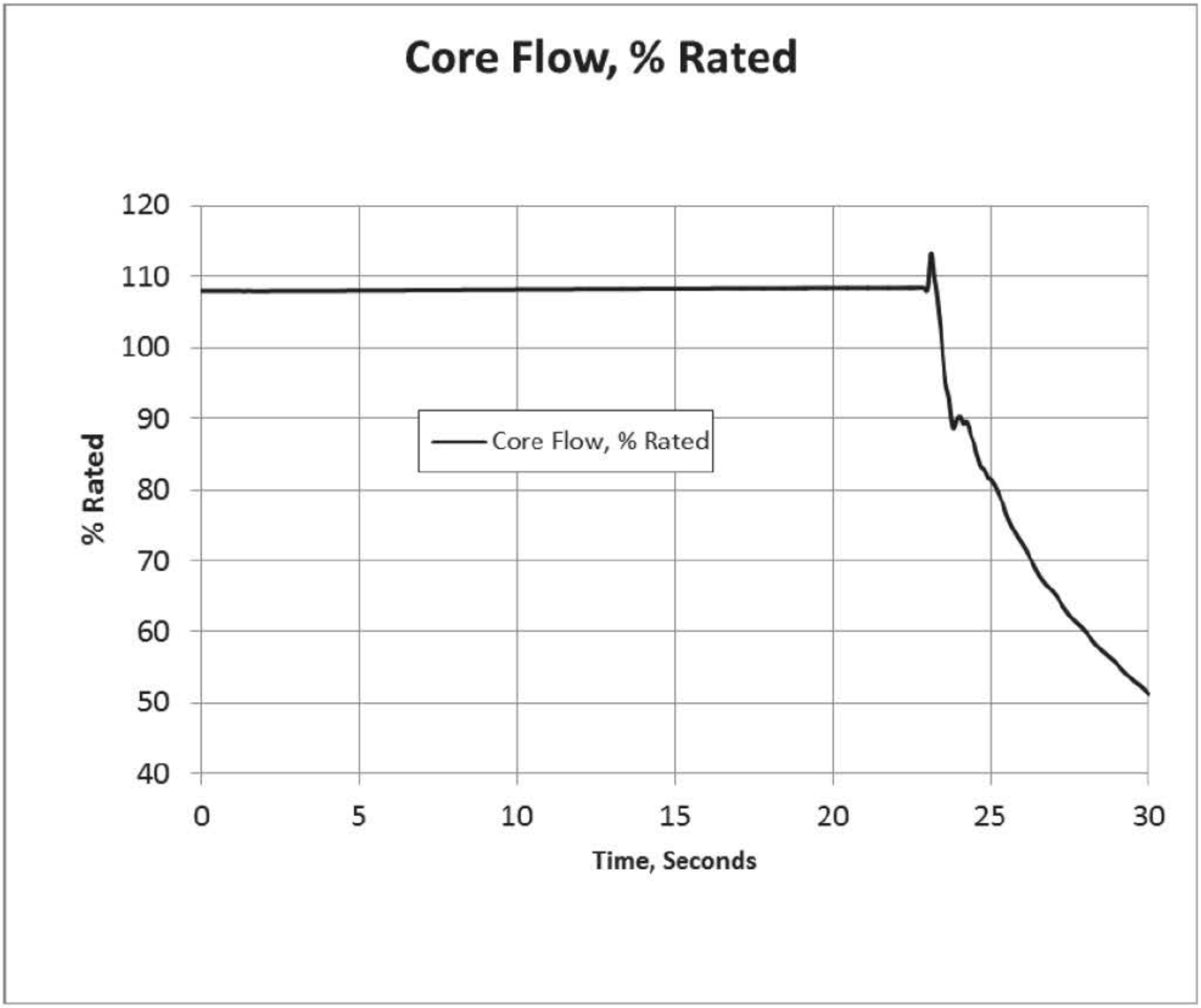


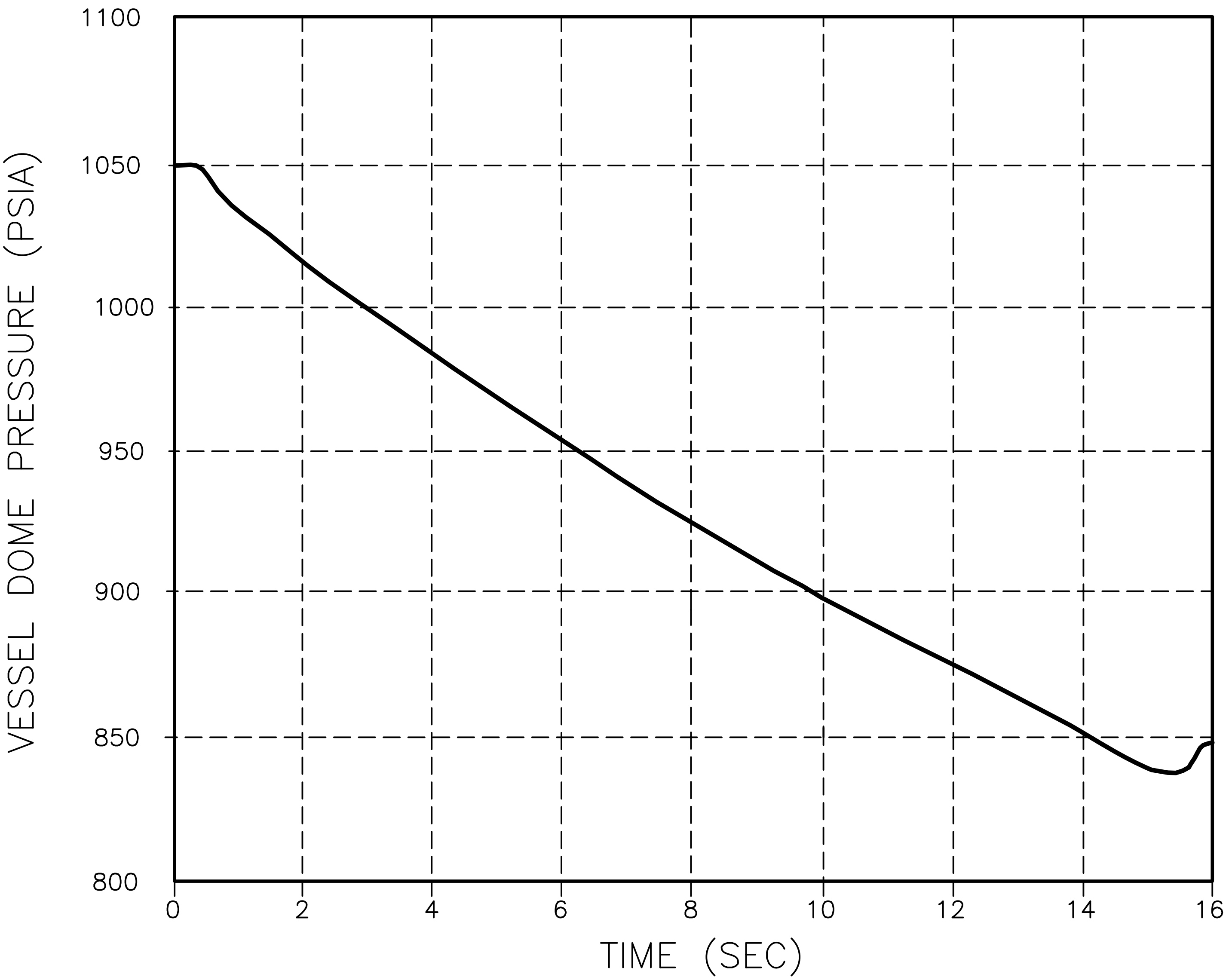
FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15C.1.2-1-5, Rev 64

AutoCAD: Figure Fsar 15C_1_2_1_5.dwg



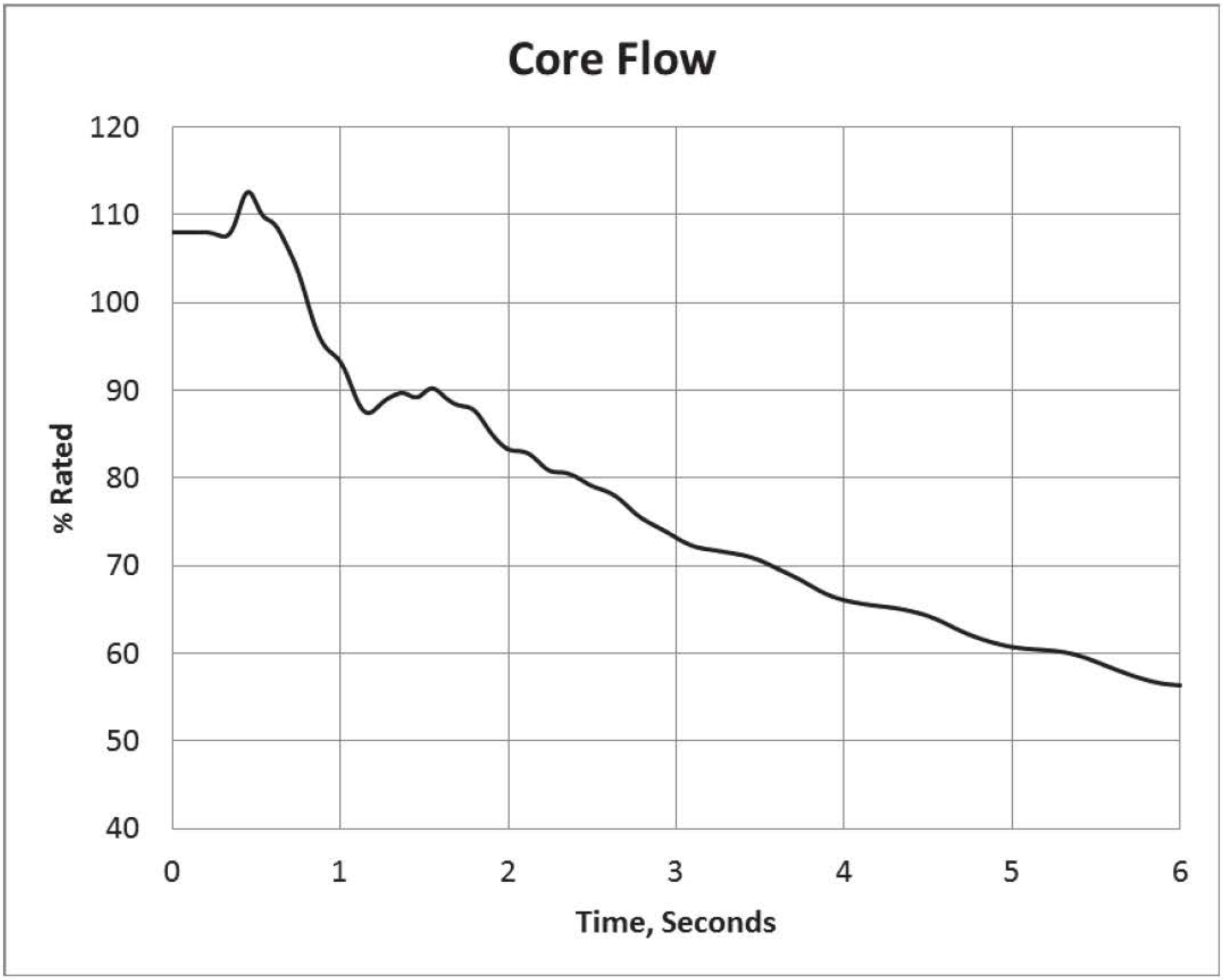
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 MWt

FIGURE 15C.1.3-1-5, Rev 55

AutoCAD: Figure Fsar 15C_1_3_1_5.dwg



FSAR REV.68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 CYCLE 20
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15C.2.2-1-5, Rev 63

AutoCAD: Figure Fsar 15C_2_2_1_5.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15C.4.5-1-5, Rev. 56

AutoCAD Figure 15C_4_5_1_5.doc

APPENDIX 15D

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2

FINAL SAFETY ANALYSIS REPORT

CYCLE SPECIFIC DATA

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE 19

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1		Loss of Feedwater Heater	NOTE 5	NOTE 5	NOTE 5	NOTE 5	NOTE 5	0.12	Moderate	0	0 sec
15.1.2	15D.1.2-1	Feedwater Controller Failure (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	225	1247	1268	1257	118	0.27	Moderate	14	4 sec estimate
15.1.3	15D.1.3-1	Pressure Regulator Failure - Open	102	1106	1129	1106	103	0.01	Moderate	2	See Text
15.1.4		Inadvertent Opening of Safety or Relief Valves	See Text						Moderate		
15.1.6		RHR Shutdown Cooling Malfunction	See Text						Moderate		
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Failure – Closed	See Text						Moderate		
15.2.2		Generator Load Reject – Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.2	15D.2.2-1	Generator Load Reject- Without Bypass (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	265	1263	1287	1307	117	0.27	Moderate	14	10 sec estimate
15.2.3		Turbine Trip - Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.3	15D.2.2-1	Turbine Trip – Without Bypass (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	265	1263	1287	1307	117	0.27	Moderate	14	10 sec estimate
15.2.4		Inadvertent MSIV Closure	See Text and Appendix 15E						Moderate		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE 19

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.2.5		Loss of Condenser Vacuum	See Text and Appendix 15E						Moderate		
15.2.6		Loss of Auxiliary Power Transformer	See Text and Appendix 15E						Moderate		
15.2.6		Loss of All Grid Connections	See Text and Appendix 15E						Moderate		
15.2.7		Loss of All Feedwater Flow	See Text and Appendix 15E						Moderate		
15.2.8		Feedwater Piping Break	See Section 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		<u>DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE</u>									
15.3.1		Trip of One Recirculation Pump Motor	See Text and Appendix 15E						Moderate		
15.3.2		Trip of Both Recirculation Pump Motors	See Text and Appendix 15E						Moderate		
15.3.3	15D.3.3-1 through 15D.3.3-4	Seizure of One Recirculation Pump (Single Loop Operation)	67	1035	1070	1035	67	0.33	Limiting Fault		
15.3.4		Recirculation Pump Shaft Break	See Text						Limiting Fault		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE 19

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.4		REACTIVITY AND POWER ANOMALIES									
15.4.1.1		RWE – Refueling	See Text						Infrequent		
15.4.1.2		RWE – Startup	See Text						Infrequent		
15.4.2		RWE - At Power, 108 Mlbs/hr, Bypass Operable	See Text	Note 5	Note 5	Note 5	Note 5	0.23	Moderate		
15.4.3		Control Rod Maloperation	See Subsections 15.4.1 and 15.4.2								
15.4.4		Startup of Idle Recirculation Loop	See Text and Appendix 15E						Moderate		
15.4.5		Recirculation Flow Controller Failure[(3)]	See Text	Note 5	Note 5	Note 5	Note 5	0.33	Moderate		
15.4.7		Misplaced Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		
15.4.7		Rotated Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE 19

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.5		INCREASE IN REACTOR INVENTORY									
15.5.1		Inadvertent HPCI Pump Start @ 60% Power	See Text and Appendix 15E					0.31	Moderate		
15.5.3		BWR Transients That Increase Reactor Coolant Inventory	See Sections 15.1 and 15.2								

Notes

1. Unless otherwise stated, the plant initial condition listed in this table for transients is: 102% Power, 108 M bs/hr Flow, EOC-Reactor Pump Trip Operable, Bypass Operable, Realistic Scram Time.
2. Minimum MCPR operating limit for Single Loop Operation, see Text.
3. Recirculation Flow Controller Failure analyses are initiated from low power/low flow conditions. This one started at 60 Mlbs/hr flow with main steam bypass operable.
4. Steam line pressure is at the turbine stop valve for events in which the turbine trips. For other transients the steam line pressure is assumed to be no higher than the reactor vessel dome pressure.
5. These Anticipated Operational Occurrences are analyzed as steady-state events.

TABLE 15D.1.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATING
UNIT 2

TIME, SECONDS	EVENT
0	Initiate a 100°F temperature reduction into the feedwater system.
2	Initial effect of unheated feedwater starts to raise core power level and steam flow, (Transport delay in feedwater piping is neglected).
≈40 (estimate)	APRM high neutron flux alarm sounds.
≈60 (estimate)	Reactor variable settle into new steady state, (below Scram trip point).
600 (estimate)	Operator begins to reduce power (reduce core flow and/or insert normal sequence control rods) to restore plant operation within normal power-flow conditions.

The above times are estimates. This event is a relatively slow transient and the analysis was performed as a series of steady-state calculations

TABLE 15D.1.2-1

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND UNIT 2 CYCLE 19

TIME, SECONDS	EVENT
0	Initiate simulated failure of 130% upper limit on feedwater flow.
22.66	L8 vessel level setpoint trips main turbine and feedwater pumps.
22.73	Reactor scram trip actuated from main turbine stop valve position switch.
22.76	Bypass Valves actuated
22.84	Recirculation pump trip (RPT) actuated by stop valve position switch.
25.27	Second group of safety/relief valves activate. (First group out of service)
25.66	Third group of safety/relief valves activate

Initial Conditions:

Power	= 100%
Flow	= 108 Mlbs/hr
Bypass	= Operable
RPT	= Operable
Scram Time	= Maximum Allowable
Exposure	= EOC

TABLE 15D.1.3-1

SEQUENCE OF EVENTS FOR PRESSURE REGULATOR FAILURE - OPEN
UNIT 2

TIME, SECONDS	EVENT
0	Initial conditions, maximum limit on steam flow to turbine.
0.2	Main turbine bypass valves full open
10.82	Main steamline isolation trip occurs
11.63	Initiation of scram trip signal, 0.06 seconds after the Main Steam Isolation Valves reach 85% open position.
15.50	Pressure in the reactor reaches a minimum and starts to increase.
15.82	MSIV's are fully closed
48 (est)	Relief valves at lowest setting start to cycle to remove decay heat.

Table Rev. 64

TABLE 15D.2.2-1

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS
UNIT 2 CYCLE 19

TIME, SECONDS	EVENTS
~0	Turbine-generator detection of loss electrical load.
0	Generator lockout relays act to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0.001	Turbine control valves (TCV) close on GLR, (Generator Load Reject)
0.070	Initiate scram on TCV fast closure (Trip oil pressure-low).
0.106	Turbine control valves closed.
0.175	EOC-Reactor Pump Trip initiated.
	Group 1 safety valves out of service.
1.96	Group 2 safety valves actuated.
2.20	Group 3 safety valves actuated.

Initial Conditions

Power: 100%
Flow: 108 Mlbs/hr
Bypass: Inoperable
Scram: Maximum Allowable Time
RPT: Operable

TABLE 15D.3.3-1

PUMP SEIZURE ACCIDENT FROM TWO LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 2

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
0.8	Jet Pump Diffuser Flow Reverses in Seized Loop
1.30	Minimum CPR

Note: Figures include a 0.5 second null transient

TABLE 15D.4.2-1

SEQUENCE OF EVENTS - RWE IN POWER RANGE
UNIT 2

ELAPSED TIME	EVENT
0	Core is assumed to be at rated conditions.
0	Operator selects and withdraws the maximum worth control rod.
~1 sec	The total core power and the local power in the vicinity of the control rod increase.
~5 sec	The operator ignores warning and continues withdrawal.
~15 sec	The RBM system indicates excessive localized peaking.
~15 sec	The operator ignores warning and continues withdrawal.
~20 sec	The RBM system initiates a rod block inhibiting signal, credit is taken for this signal. Further control rod withdrawal is blocked.
~40 sec	Reactor core stabilizes at higher core power level.
~60 sec	Operator attempts to re-insert control rod to reduce core power level.
~80 sec	Core stabilizes at rated conditions.

TABLE 15D.4.5-1

SEQUENCE OF EVENTS FOR RECIRCULATION FLOW CONTROLLER FAILURE
UNIT 2

TIME, SECONDS	EVENT
0	Master Flow Controller fails initiating a slow run-up of both reactor recirculation pumps
~220	Two relief valves open at 1120.7 psia.
~220	Reactor high flux scram (analytical setpoint, 122%).
~230	Two relief valves reseat at 1045.7 psia.

This sequence of events is for the event initiated from:

INITIAL CONDITIONS

Power	=	69%
Flow	=	60M lbs/hr
Bypass	=	Inoperable
Exposure	=	EOC

TABLE 15D.4.7-1

UNIT 2

SEQUENCE OF EVENTS FOR MISLOADED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in the wrong position.
2. Subsequently, the bundle intended for this position is placed in the position of the previous bundle.
3. During core verification procedure, error is not observed.
4. Plant is brought to full power operation without detecting misplaced bundle.
5. Plant continues to operate.

SEQUENCE OF EVENTS FOR ROTATED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in its proper location but rotated either 90° or 180° from its proper orientation.
2. During core verification procedure this error is not observed.
3. Plant is brought to full power operation without detecting rotated bundle.
4. Plant continues to operate.

TABLE 15D.4.9-1

SEQUENCE OF EVENTS FOR CONTROL ROD DROP ACCIDENT
UNIT 2

APPROXIMATE ELAPSED TIME	EVENT
	Reactor is operating at a rod density pattern of up to 50%.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the Bank Position Withdrawal Sequence (BPWS).
	Decoupled control rod sticks in the fully inserted or in an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1 second	Reactor goes on a positive period and initial power increase is terminated by the Doppler effect.
<1 second	APRM 120% power signal scrams the reactor
<5 seconds	Scram terminates the accident.

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2 CYCLE 19

1.	Thermal Power Level, MWT Rated Value Analysis Value	3952 (100%) 4031 (102%)
2.	Steam Flow, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.624
3.	Maximum Core Flow, Mlbs/hr	108.0(3)
4.	Feedwater Flow Rate, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.592
5.	Feedwater Temperature,°F (At 100% Power and 100 Mlbs/hr)	403.3
6.	Vessel Dome Pressure, psig (At 100% Power and 100 Mlbs/hr)	1035.7
7.	Vessel Core Pressure, psig at Channel Exit (At 100% Power and 100 Mlbs/hr)	1047.4
8.	Turbine Bypass Capacity, % Rated	21.5%
9.	Core Coolant Inlet Enthalpy, BTU/lb (At 100% Power and 100 Mlbs/hr)	523.6
10.	Turbine Inlet Pressure, psia	976.3
11.	Fuel Types	ATRIUM™-10
12.	Core Average Gap Conductance, BTU/hr-ft^2-°F	500 to 1700(1)
13.	Core Leakage Flow,%	~10%(2)
14.	Required MCPR Operating Limit	See Unit 2 COLR (FSAR section 16.3 – TRMs)
15.	MCPR Safety Limit	See Table 15D.0-3
16.	Doppler Coefficient	See Note 4

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2 CYCLE 19

17.	Void Coefficient		See Note 4
18.	Core Average Rated Void Fraction		See Note 4
19.	Scram Reactivity Analysis Data		See Note 4
20.	Control Rod Scram Times		Table 15D.0-5
21.	Jet Pump Ratio		2.1
22.	Safety Relief Valve Capacity (16 Valves) Percent of Rated Steam Flow		87%
23.	Relief Function Delay, sec		0.1
24.	Relief Function Response, sec		≤0.15
25-a.	Relief Mode Set Points for Safety/Relief Valves, psig		2 @ 1106 3 @ 1136 4 @ 1116 3 @ 1146 4 @ 1126
25-b.	Safety Mode Set Points for Safety/Relief Valves, psig		2 @ 1175 6 @ 1195 8 @ 1205
26.	Number of Valve Groups Simulated		3
27.	High Flux Trip, % Rated Analysis set point		122
28.	High Pressure Trip, Analysis Set Point, psig		1105
29.	Vessel Level Trips, Nominal Setpoints Inches Above(+), Below (-) Dryer Skirt Bottom, (See Note 5)	High Level Low Level Low Low Level Low Low Low Level	(L8)≤ 54 (L4)≤ 30 (L3)≥ 13 (L2)≥ -38 (L1)≥-129
30.	APRM Thermal Trip, Analytical Set Point,% Rated		118

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2 CYCLE 19

31.	Recirculation Pump Trip Delay, sec	0.175
32.	Recirculation Pump Trip Inertia for Analysis,lb_m-ft^2	16,800

NOTES

1. Gap conductance for reactor system behavior is determined for the fuel types within the core as a function of power and exposure. The hot bundle gap conductance is based on the fuel type that is expected to be limiting. It is also determined based on the initial hot bundle power and exposure.

2. Inlet enthalpy and leakage flow are determined for each initial condition analyzed.

3. Core flow shown is the maximum. It is varied depending on the initial conditions being analyzed.

4. The physics characteristics are based on initial conditions determined from a 3-D simulation of the core over a range of power, flow, and pressure conditions. For certain transient analyses this data is transferred and collapsed for use in a 1-D reactor core/system transient simulation model of SSES unit 2.

5. Analytical limits for level setpoints include drift and uncertainty allowances.

TABLE 15D.3.3-2

PUMP SEIZURE ACCIDENT FROM SINGLE LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 2

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
N/A	Jet Pump Diffuser Flow Reverses in Seized Loop
1.84	Minimum CPR

Table Rev. 1

Table 15D.4.7-2

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TABLE 15D.4.9-2 CONTROL ROD DROP ACCIDENT UNIT 2 CYCLE 19	
Cycle Exposure, MWD/MTU	18,896
Control Rod Sequence	B
Rod Group	1
Dropped Rod Location	14-47
Dropped Rod Worth	11.35 mk
Number of Fuel Rods with Fuel Enthalpy above 170 cal/gm	<2000
Peak deposited Enthalpy, cal/gm	<269.4

TABLE 15D.0-3

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
UNIT 2 CYCLE 19

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
FOR
SINGLE LOOP OPERATION

1.11

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
FOR
TWO LOOP OPERATION

1.08

Table 15D.3.3-3

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Table 15D.4.9-3

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TABLE 15D.0-4

UNIT 2

MINIMUM MCPR REQUIREMENT
FOR
SINGLE LOOP OPERATION

MCPR Safety Limit	1.08	1.09	1.10	1.11	1.12	1.13	1.14
Minimum MCPR Requirement	1.41	1.42	1.43	1.44	1.45	1.46	1.47

(Based on Analysis of Pump Seizure Accident in Single Loop Operation)

MINIMUM MCPR REQUIREMENT
FOR
TWO LOOP OPERATION

MCPR Safety Limit	1.07	1.08	1.09	1.10	1.11
Minimum MCPR Requirement	1.29	1.30	1.31	1.32	1.33

(Based on Analysis of Pump Seizure Accident in Two Loop Operation)

Table 15D.3.3-4

This Table Has Been Deleted

Table 15D.4.9-4

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Table Rev. 64

TABLE 15D.0-5

AVERAGE SCRAM INSERTION TIMES
UNIT 2 CYCLE 19

Control Rod Position	Average Scram Time to Position (seconds)	
	Realistic	Maximum Allowable
45	0.470	0.520
39	0.630	0.860
25	1.500	1.910
5	2.700	3.440
Scram Time Fraction	0.0	1.0

The times listed above do not include Rod Tip Delay

Table 15D.3.3-5

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Table 15D.4.9-5

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Table 15D.3.3-6

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Table 15D.4.9-6

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TABLE 15D.3.3-7

DOSES FOR THE SINGLE LOOP AND TWO LOOP PUMP SEIZURE EVENT
UNIT 2

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Table 15D.4.9-7

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Table 15D.3.3-8

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Table 15D.4.9-8

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Table 15D.0-1A

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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FIGURE 15D.0-1, Rev. 58

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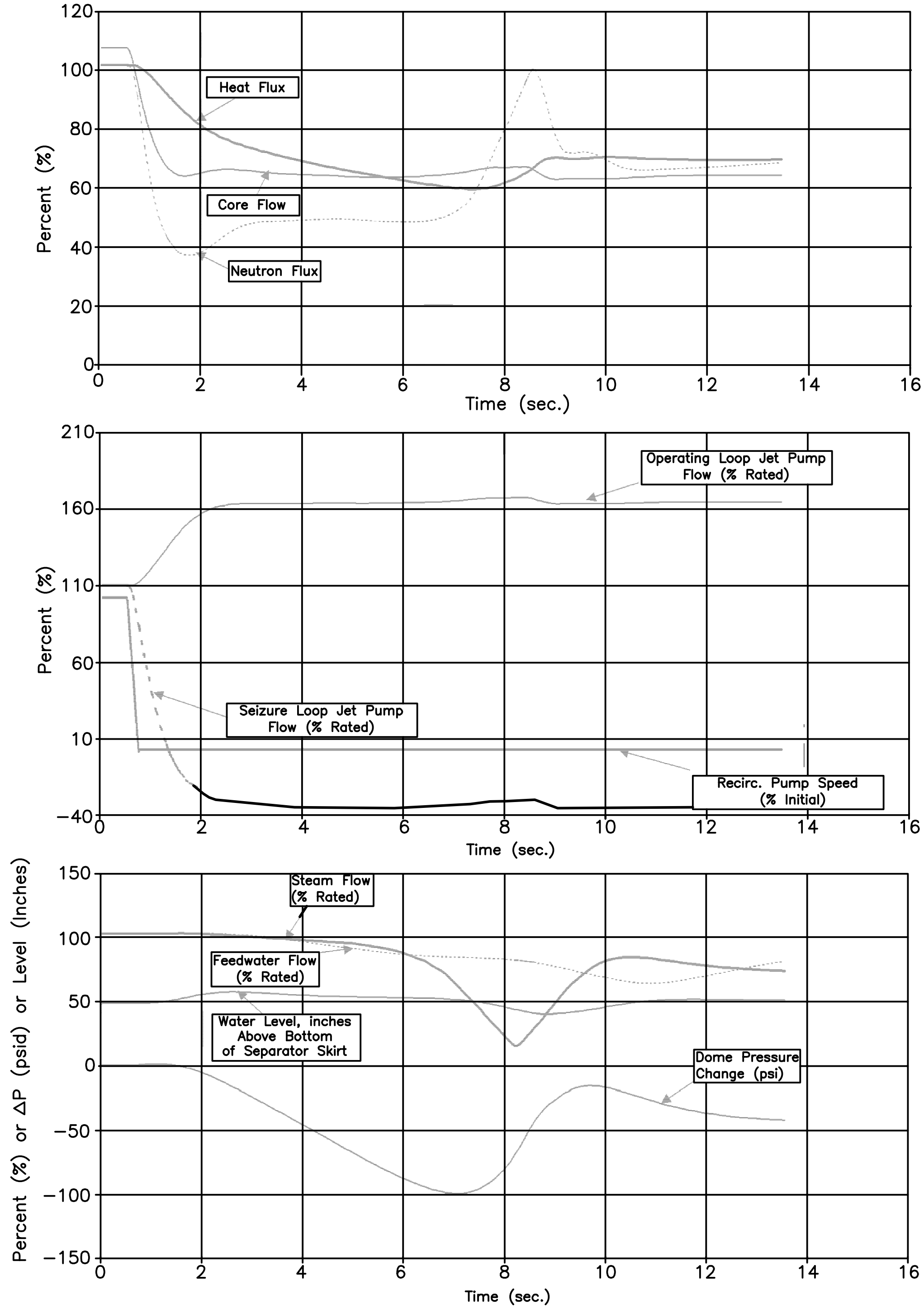

FSAR REV.65

*100% Rated Core Power is Assumed to be 3952 MWth

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
TWO LOOP OPERATION
4031MWth*/108Mlbm/HR
CORE RESPONSE

FIGURE 15D.3.3-1, Rev 61

AutoCAD: Figure Fsar 15D_3_3_1.dwg

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RENUMBERED TO 15.4-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Renumbered to 15.4-1

FIGURE 15D.4.9-1, Rev. 55

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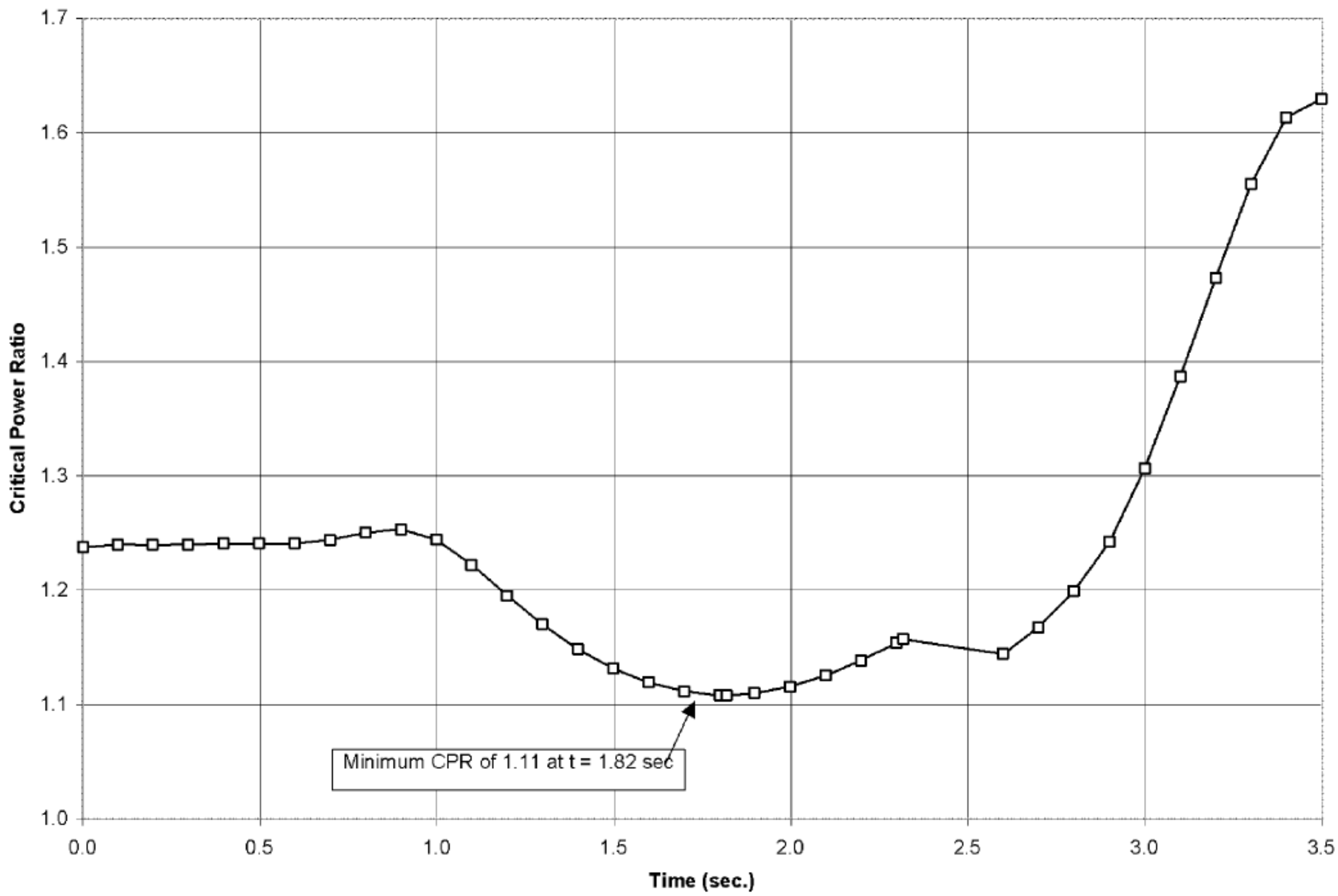

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
TWO LOOP OPERATION
4031 MWth/108 Mlbm/hr
TYPICAL CPR RESPONSE

FIGURE 15D.3.3-2, Rev 63

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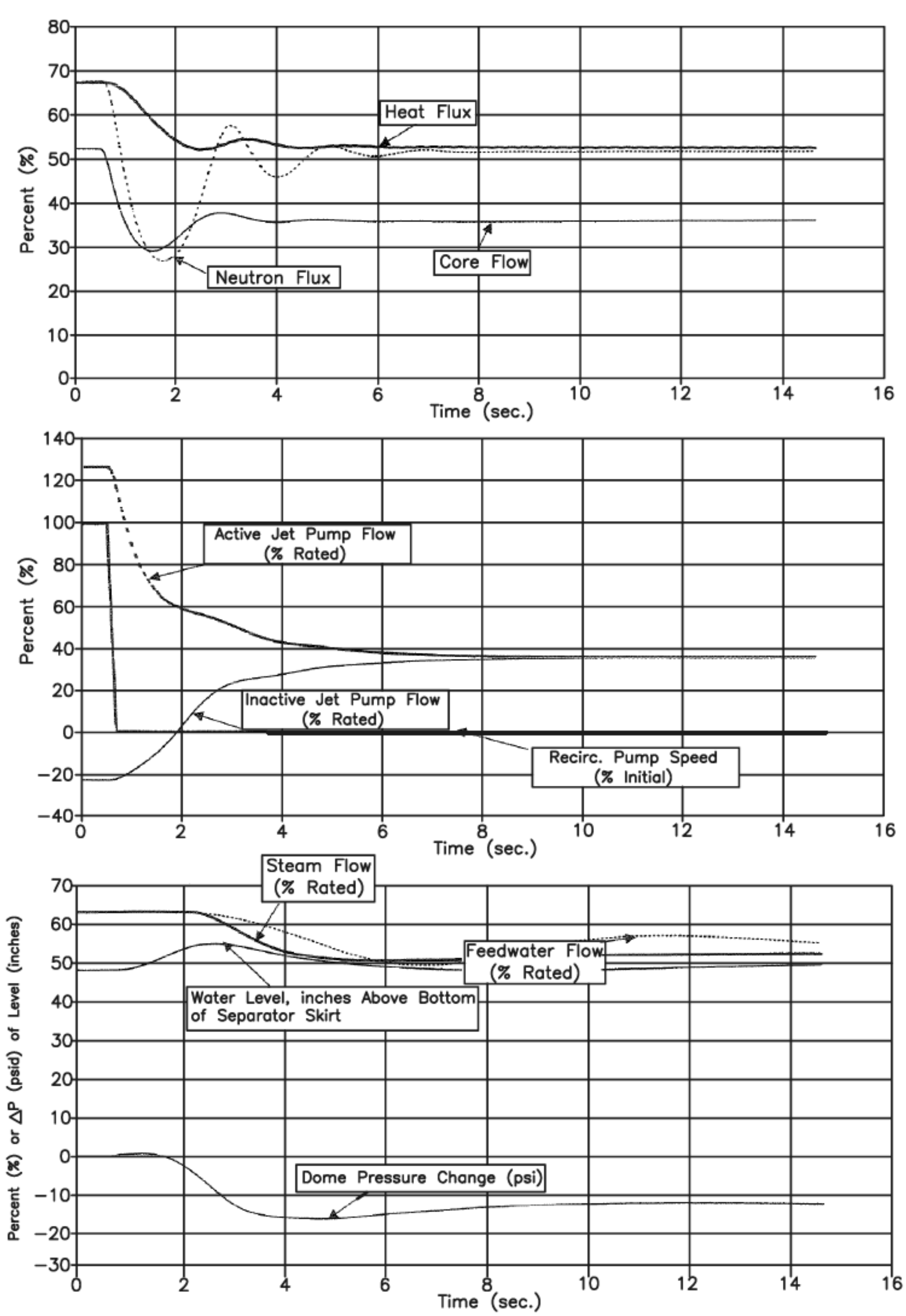

FSAR REV.65

*100% Rated Core Power is Assumed to be 3952 MWth

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
SINGLE LOOP OPERATION
2652 MWth*/52 Mlbm/hr
CORE RESPONSE

FIGURE 15D.3.3-3, Rev 1

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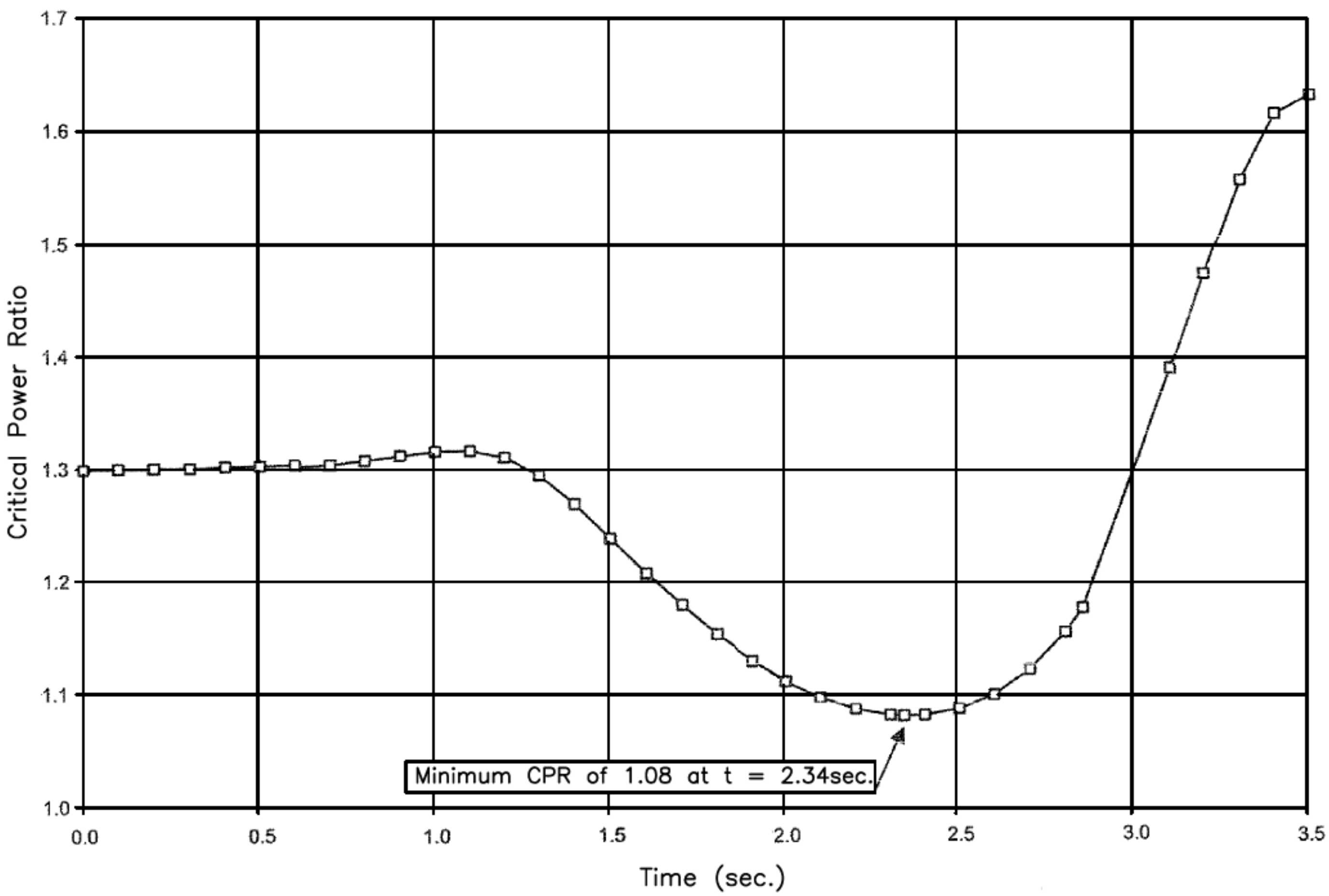

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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
SINGLE LOOP OPERATION
2652 MWt/52 Mlbm/hr
TYPICAL CPR RESPONSE

FIGURE 15D.3.3-4, Rev 3

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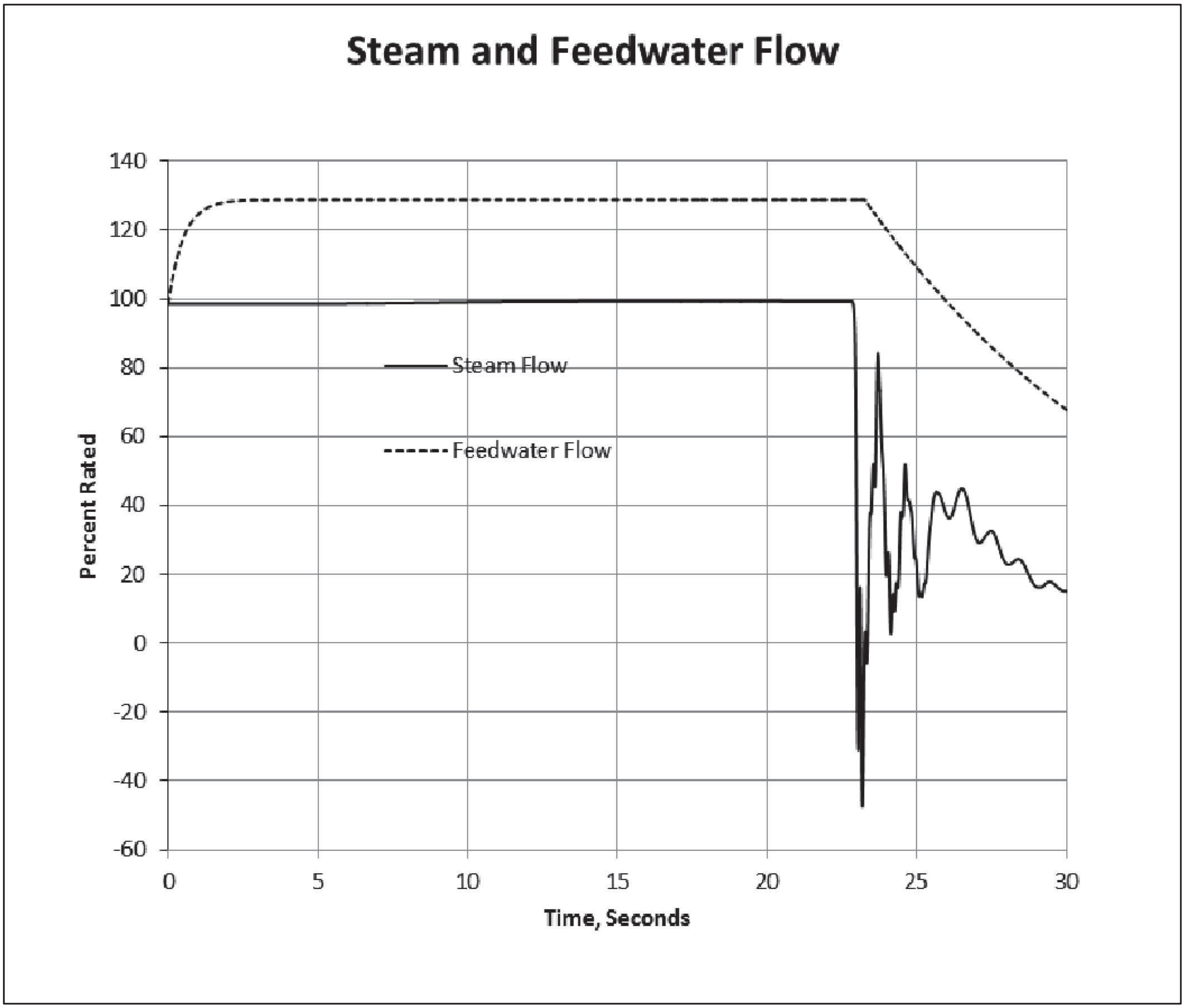

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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 18
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15D.1.2-1-1, Rev 64

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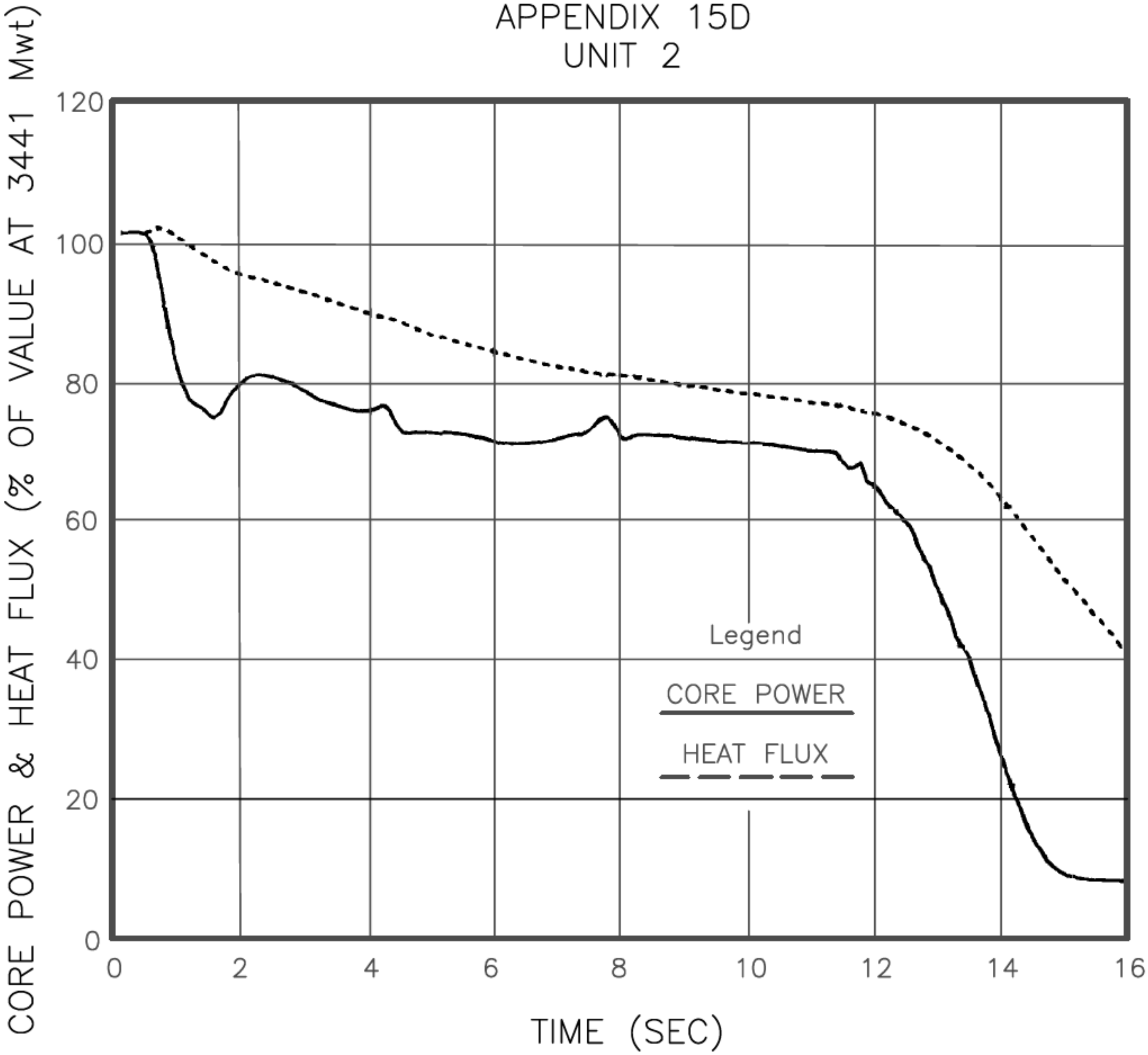

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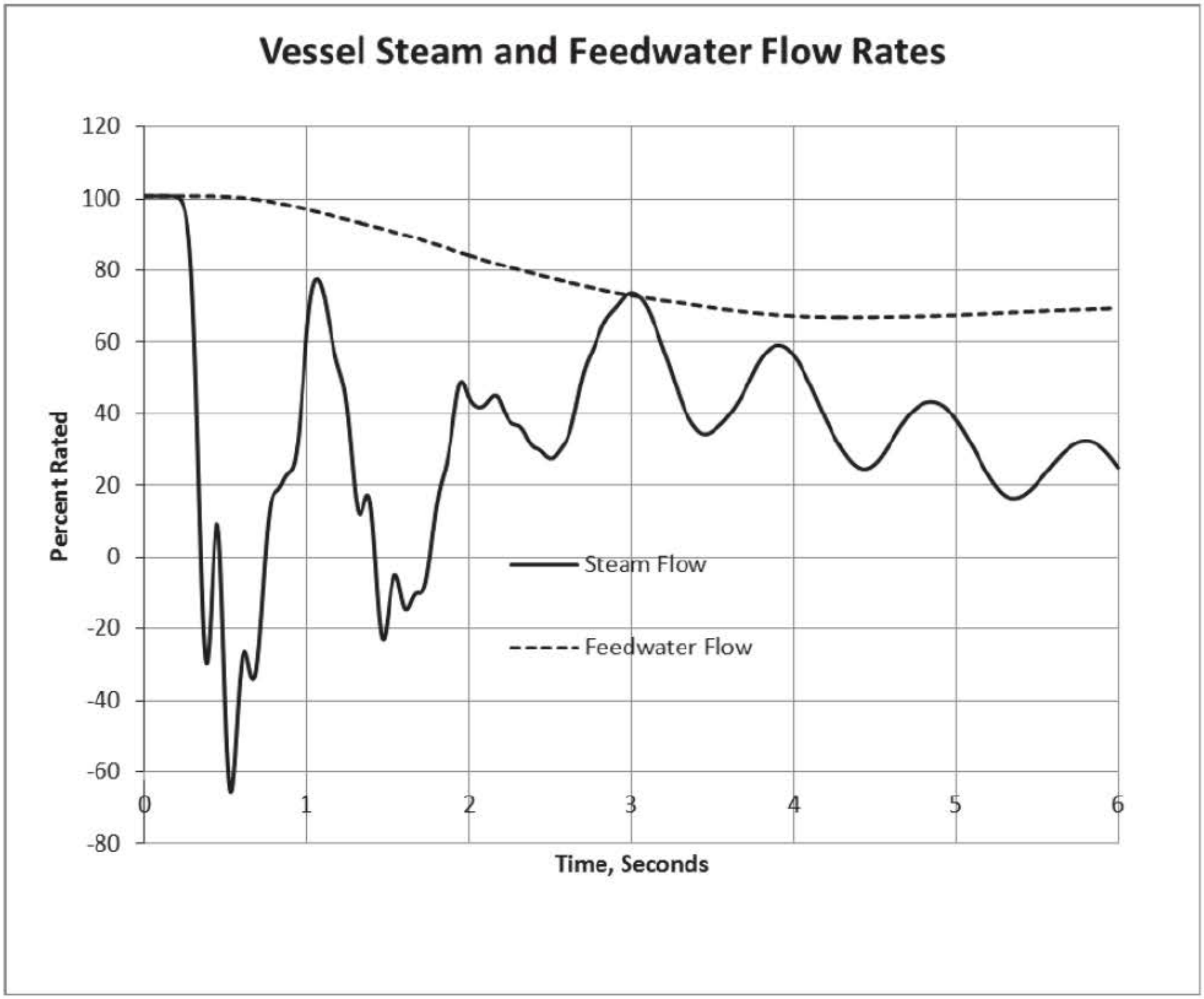
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE-
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 Mwt

FIGURE 15D.1.3-1-1, Rev 56

AutoCAD: Figure Fsar 15D_1_3_1_1.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 19
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15D.2.2-1-1, Rev 65

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15D.4.5-1-1, Rev. 58

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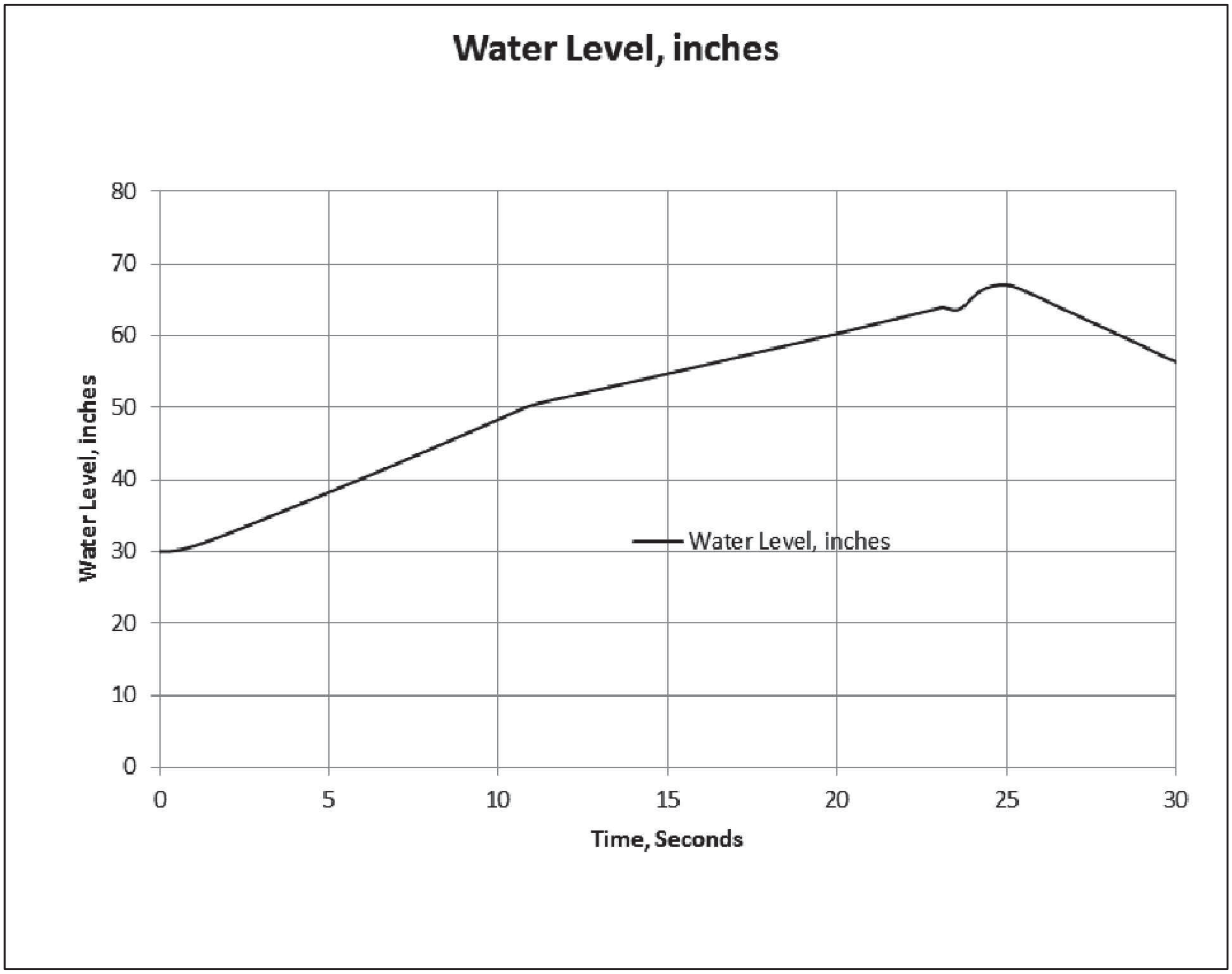

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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 18
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15D.1.2-1-2, Rev 63

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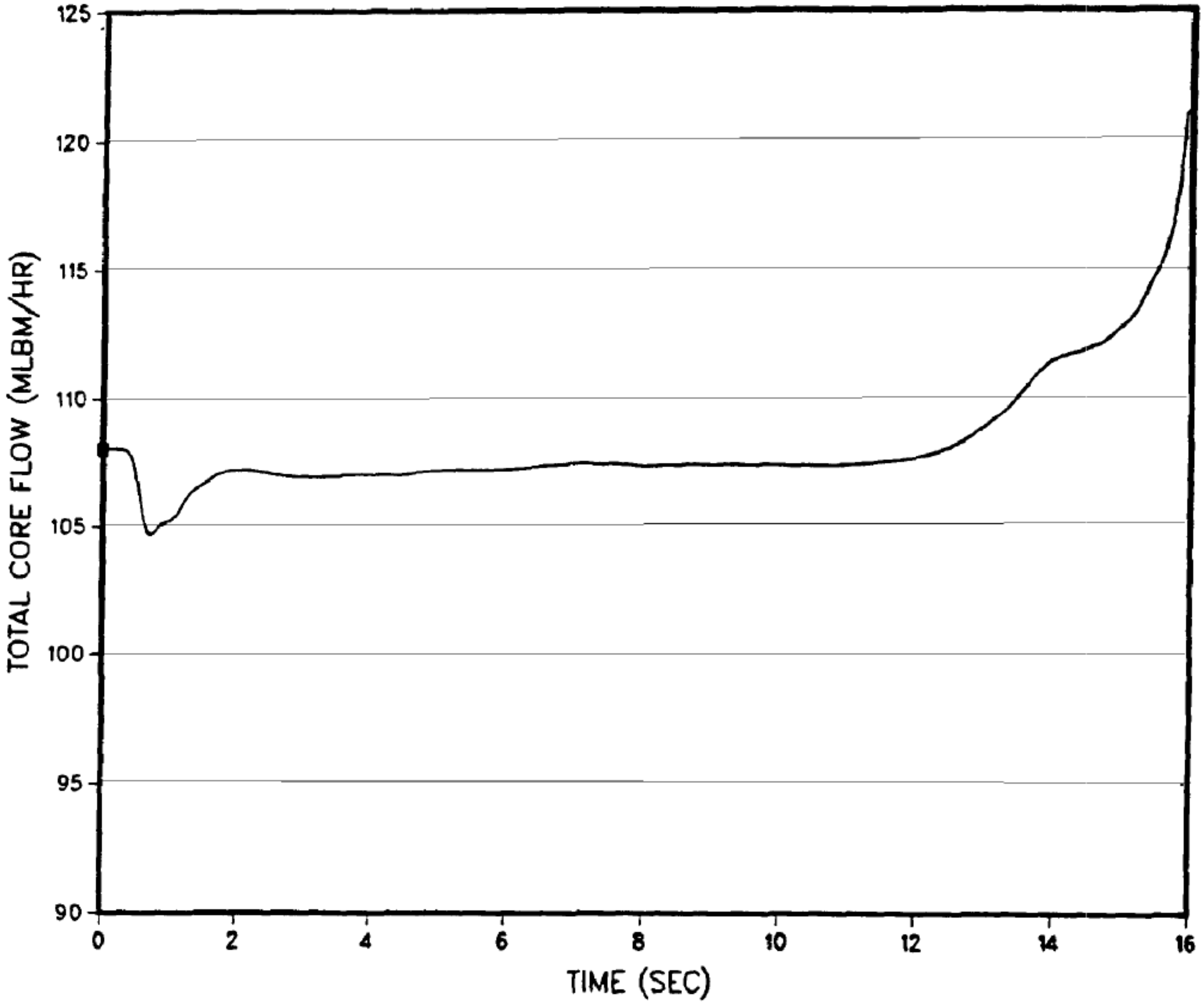

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE-
STEAM FLOW AT 130% OF
STEAM FLOW AT 3441 Mwt

FIGURE 15D.1.3-1-2, Rev 55

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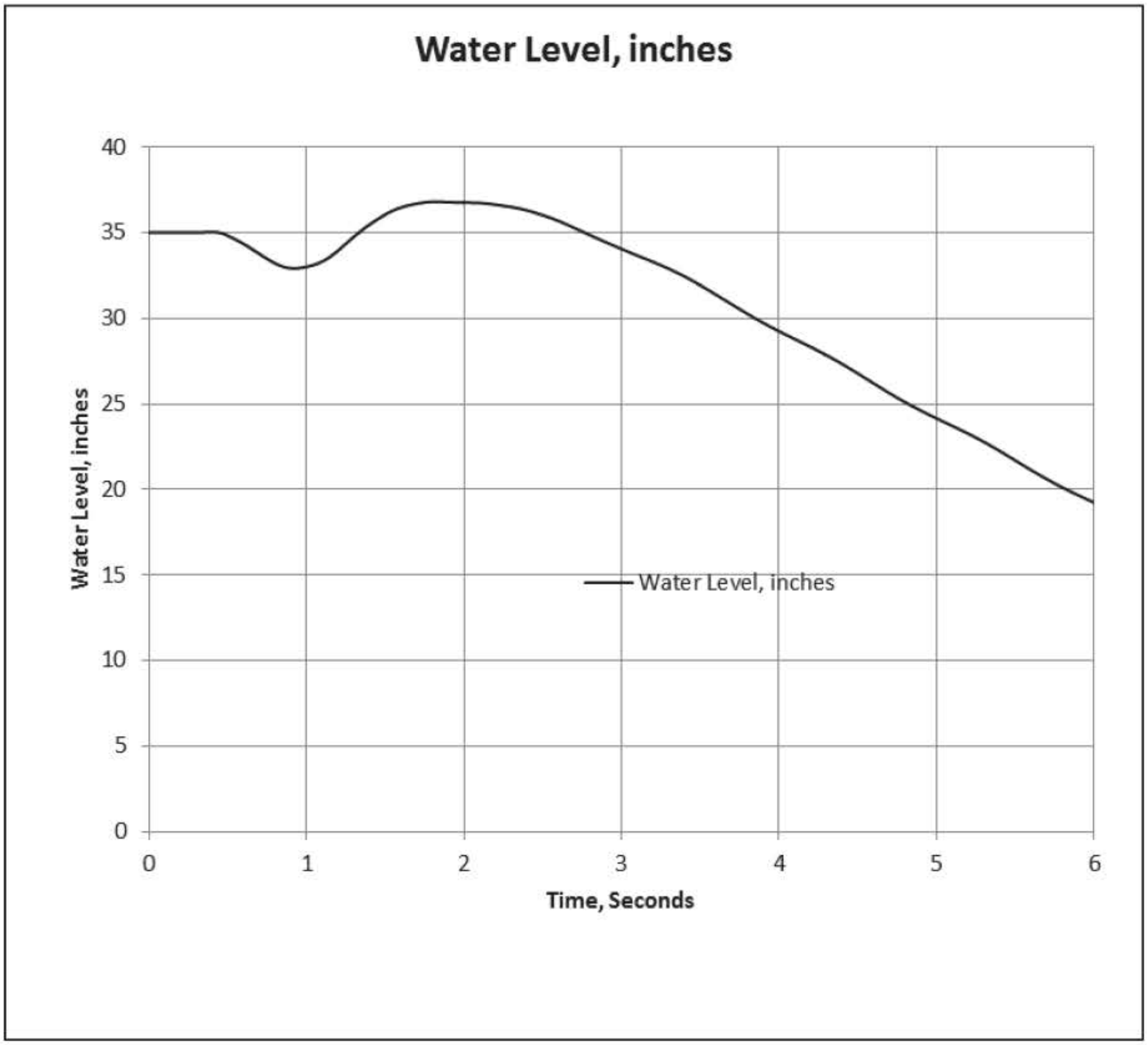

FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 19
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15D.2.2-1-2, Rev 64

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15D.4.5-1-2, Rev. 58

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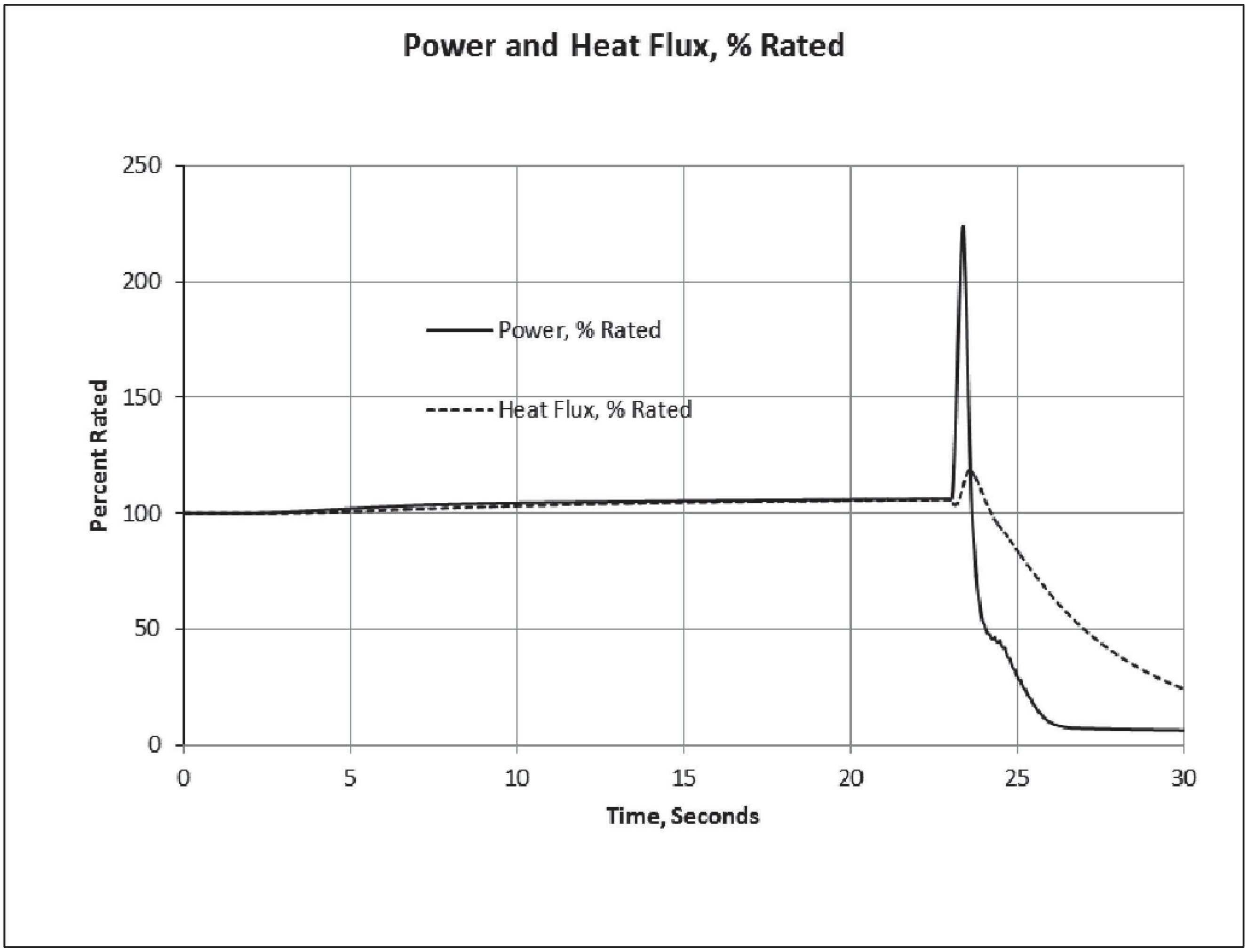


FSAR REV.67

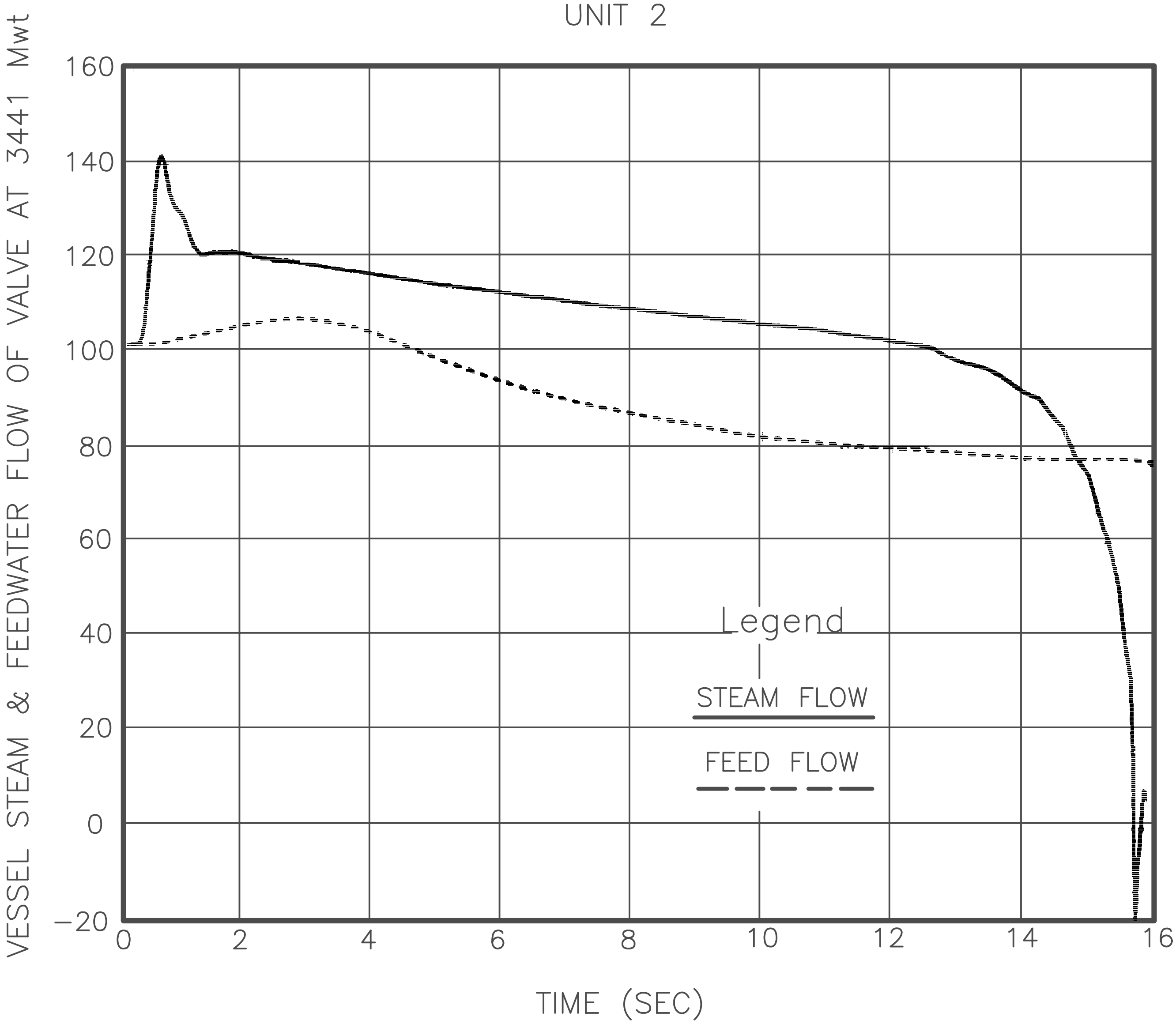
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 18
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15D.1.2-1-3, Rev 64

AutoCAD: Figure Fsar 15D_1_2_1_3.dwg

SSES–FSAR
APPENDIX 15D
UNIT 2



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
PRESSURE REGULATOR FAILURE- STEAM FLOW AT 130% OF STEAM FLOW AT 3441 Mwt FIGURE 15D.1.3-1-3, Rev 56

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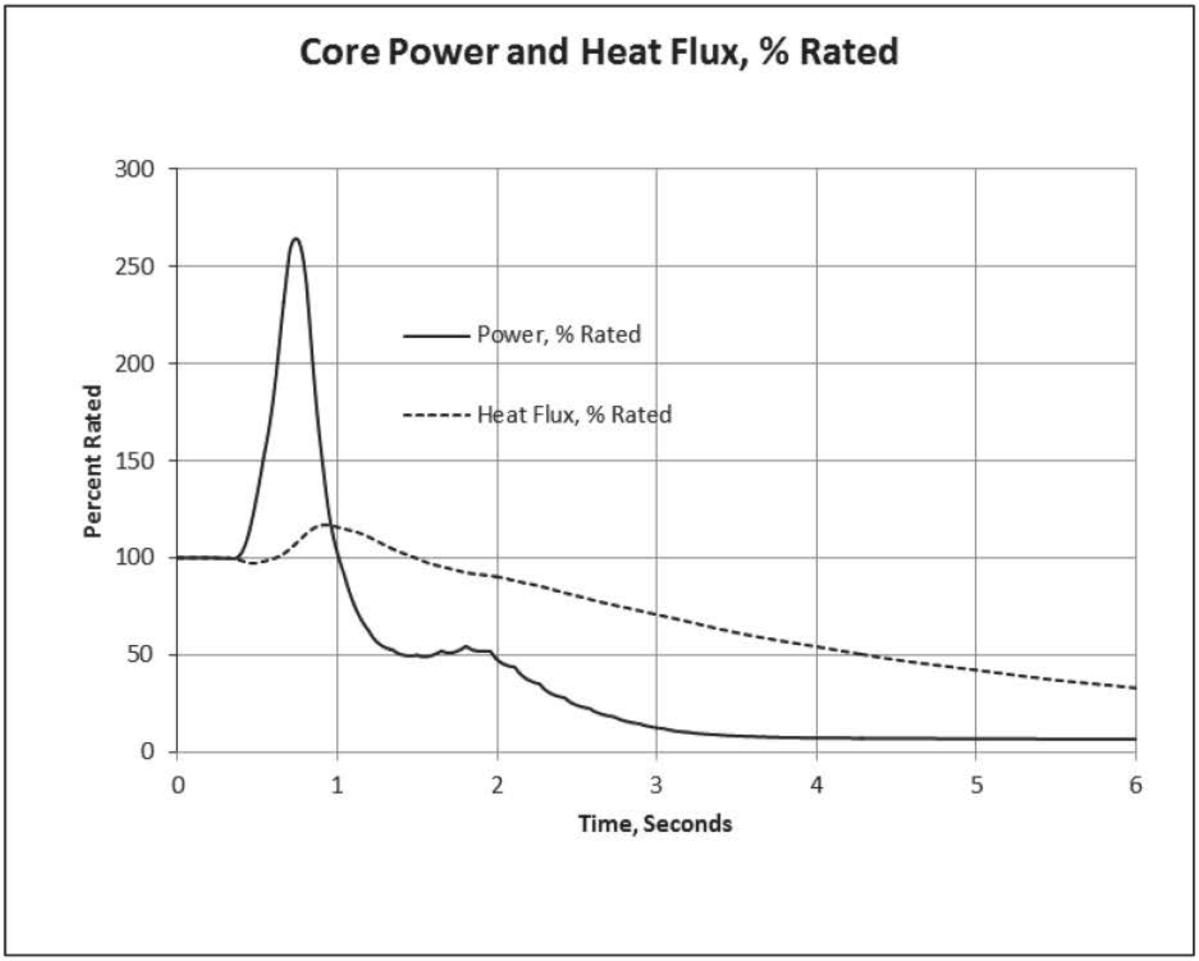

FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 19
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15D.2.2-1-3, Rev 65

AutoCAD: Figure Fsar 15D_2_2_1_3.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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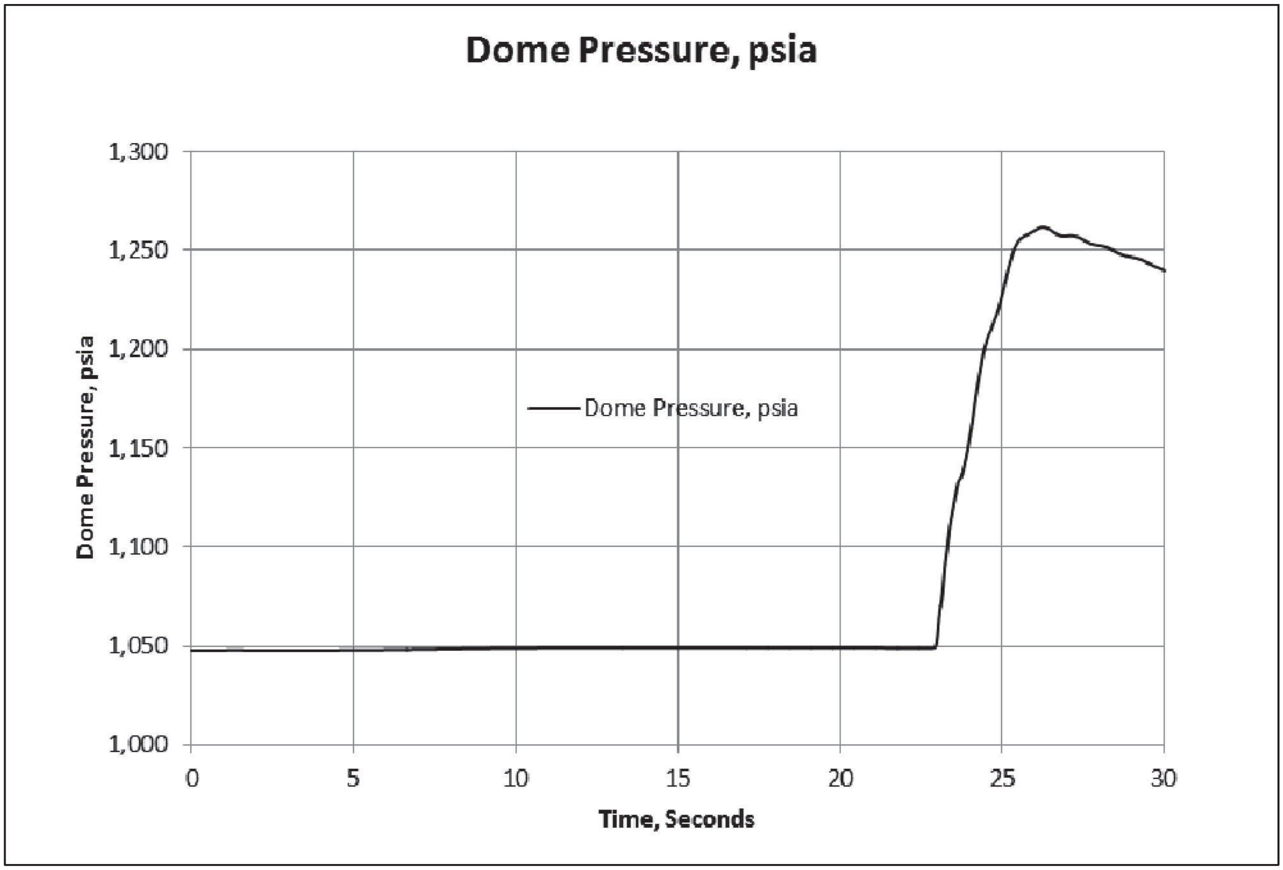
FIGURE 15D.4.5-1-3, Rev. 58

AutoCAD Figure 15D_4_5_1_3.doc



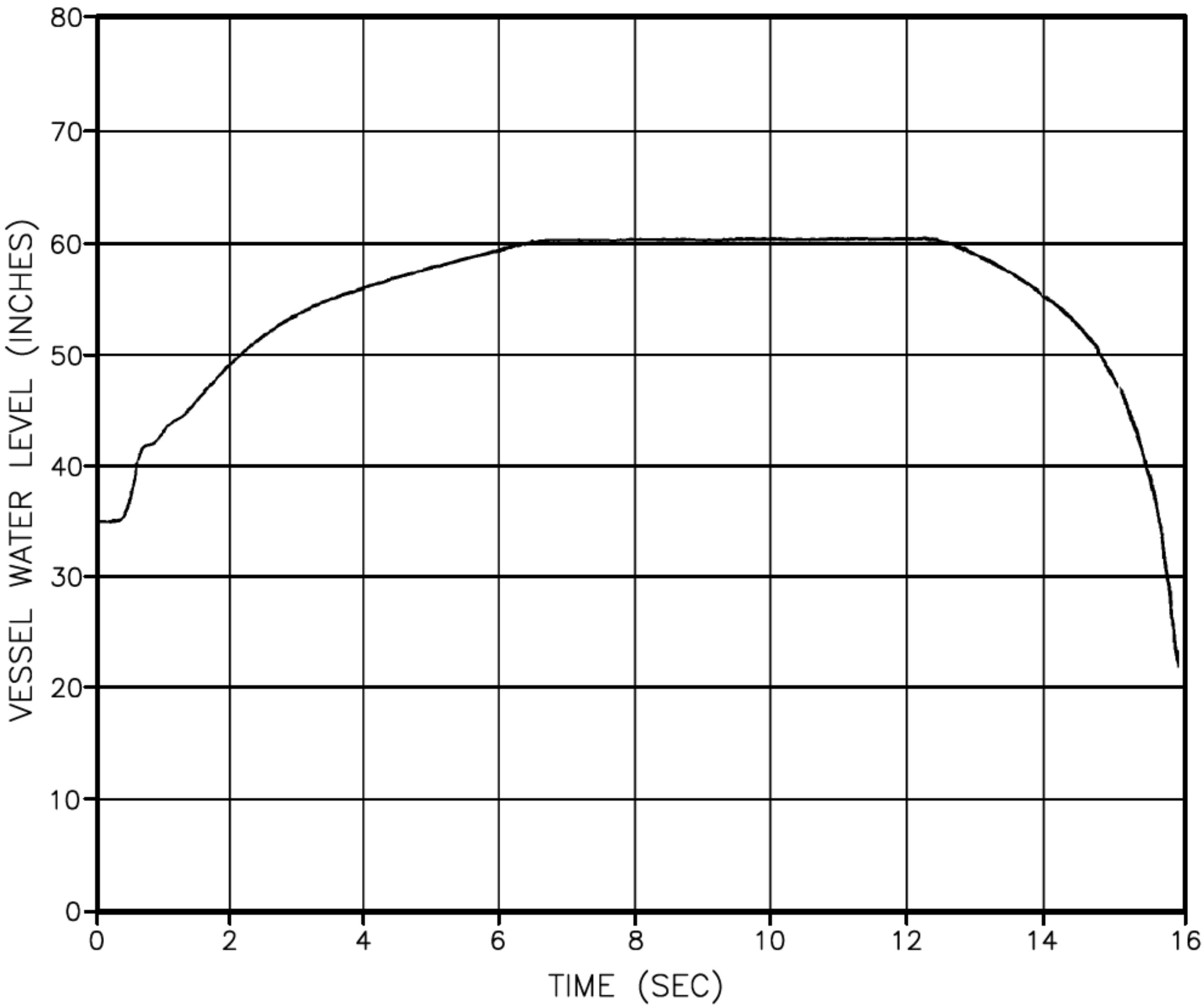
FSAR REV.67

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 18
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FIGURE 15D.1.2-1-4, Rev 63

AutoCAD: Figure Fsar 15D_1_2_1_4.dwg



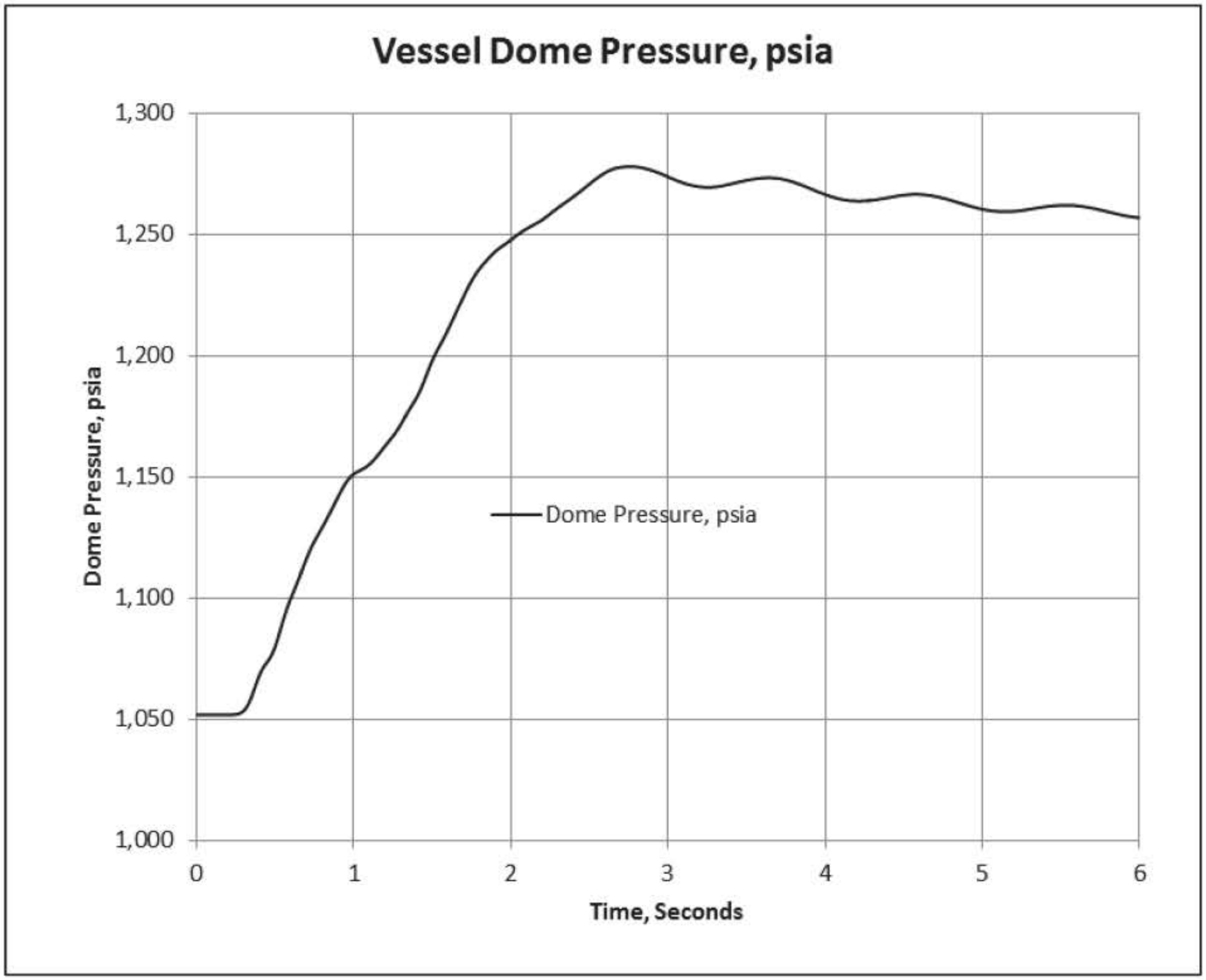
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE-
STEAM FLOW AT 130% OF STEAM
FLOW AT 3441 Mwt

FIGURE 15D.1.3-1-4, Rev 55

AutoCAD: Figure Fsar 15D_1_3_1_4.dwg



FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 19
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15D.2.2-1-4, Rev 64

AutoCAD: Figure Fsar 15D_2_2_1_4.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 15D.4.5-1-4, Rev. 58

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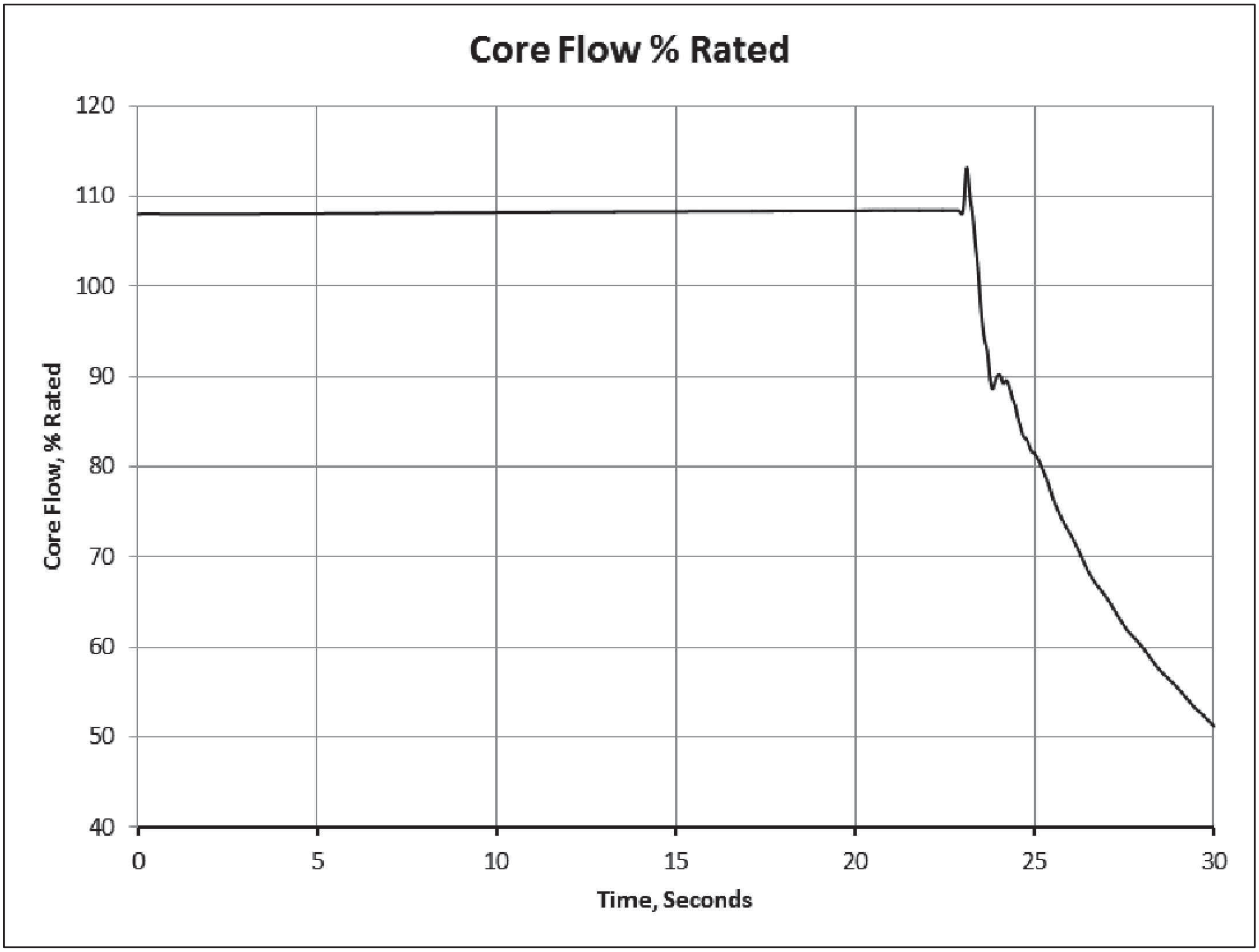
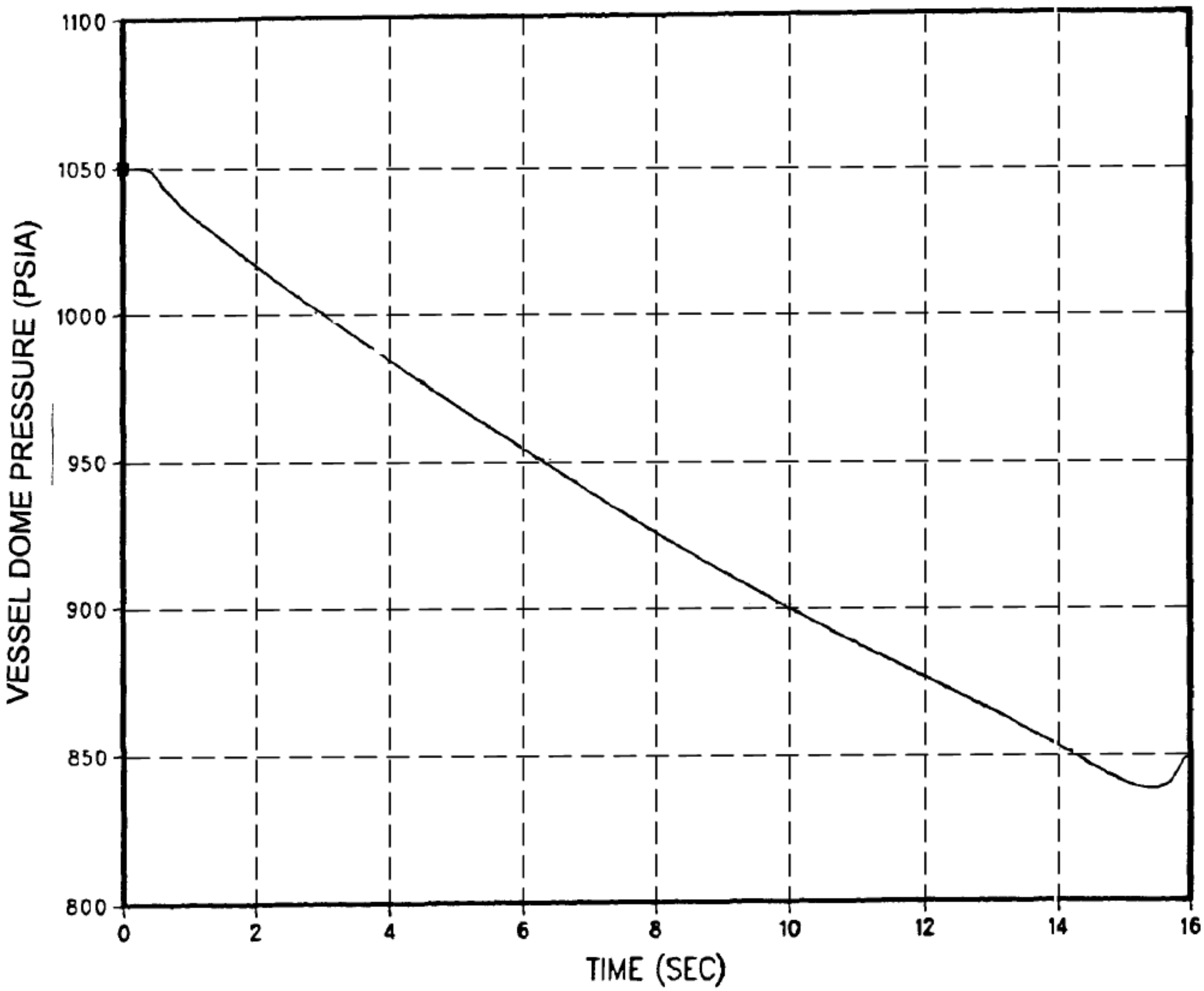


FSAR REV.67

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 18 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND, WITH HIGH WATER LEVEL TRIP FIGURE 15D.1.2-1-5, Rev 63

AutoCAD: Figure Fsar 15D_1_2_1_5.dwg



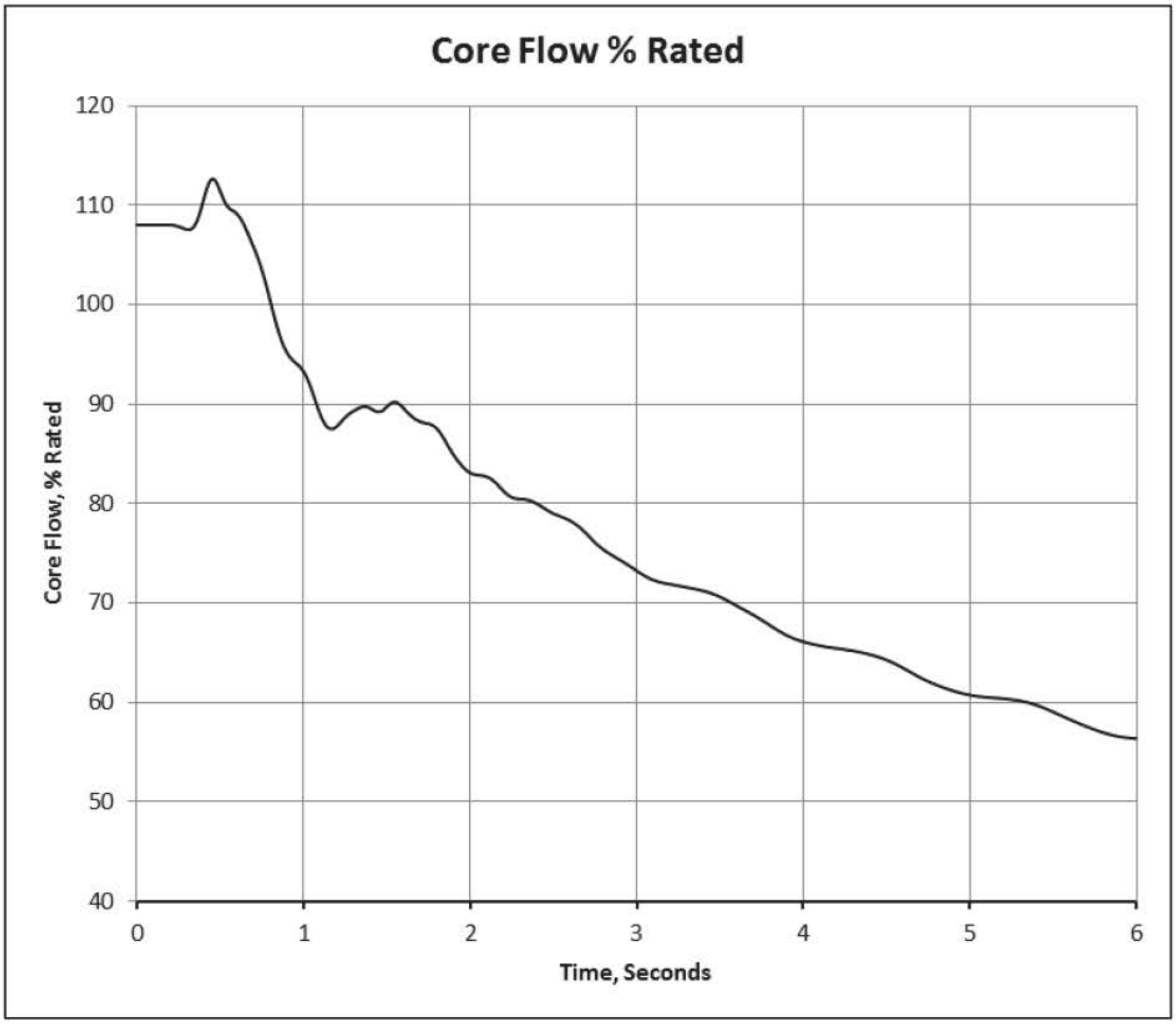
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE-
STEAM FLOW AT 130% OF RATED

FIGURE 15D.1.3-1-5, Rev 54

AutoCAD: Figure Fsar 15D_1_3_1_5.dwg



FSAR REV. 68

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 19
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FIGURE 15D.2.2-1-5, Rev 64

AutoCAD: Figure Fsar 15D_2_2_1_5.dwg

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 15D.4.5-1-5, Rev. 58

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APPENDIX 15E

INITIAL CORE FOR SSES UNITS 1 AND 2

NON-LIMITING EVENTS

TABLE 15E.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNITS 1 AND 2 NON-LIMITING EVENTS (VALUES ARE FOR THE INITIAL CORES ONLY)

Section	Figure	Description	Maximum Neutron Flux	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	ΔCPR[(1)]	Frequency Category*	Number of Valves - 1st Blowdown	Duration of Blowdown
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1		Loss of Feedwater Heater	See Text and Appendices 15C and 15D for current cycle limits								
15.1.2		Feedwater Controller Failure	See Text and Appendices 15C and 15D for current cycle limits								
15.1.3		Pressure Regulator Failure - Open	See Text and Appendices 15C and 15D for current cycle limits								
15.1.4		Inadvertent Opening of Safety or Relief Valves	See Text								
15.1.6		RHR Shutdown Cooling Malfunction	See Text								
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Failure - Closed	See Text								
15.2.2	15E.2.2-1	Generator Load Reject - Bypass On	281.7	1154	1179	1153	109.7	0.11	a	16	9 sec
15.2.2		Generator Load Reject- Bypass Off	See Text and Appendices 15C and 15D for current cycle limits								
15.2.3	15E.2.3-1	Turbine Trip - Bypass On	167.2	1143	1167	1132	101.4	0.09	a	16	8 sec

TABLE 15E.0-1 (Cont'd)
RESULTS SUMMARY OF TRANSIENT EVENTS
UNITS 1 AND 2 NON-LIMITING EVENTS (VALUES ARE FOR THE INITIAL CORES ONLY EXCEPT AS NOTED)

Section	Figure	Description	Maximum Neutron Flux	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	ΔCPR[(1)]	Frequency Category*	Number of Valves - 1st Blowdown	Duration of Blowdown
15.2.3		Turbine Trip - Bypass Off	See Text and Appendices 15C and 15D for current cycle limits								
15.2.4	15E.2.4-1	Inadvertent MSIV Closure (3)	144.3	1250	1281	12501	100.8	0.11	a	12	16 sec estimated
15.2.5	15E.2.5-1	Loss of Condenser Vacuum	167.5	1140	1165	1131	101.3	<0.09	a	13	20 sec
15.2.6	15E.2.6-1	Loss of Auxiliary Power Transformer	104.5	1145	1160	1140	100.1	~0.0	a	16	16 sec
15.2.6	15E.2.6-2	Loss of All Grid Connections	107.2	1140	1161	1130	100.1	~0.0	a	13	17 sec
15.2.7	15E.2.7-1	Loss of All Feedwater Flow (3)	102.0	1040	1081	1029	102.0	~0.0	a	0	0 sec
15.2.8		Feedwater Piping Break	See Section 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		<u>DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE</u>									
15.3.1	15E.3.1-1	Trip of One Recirculation Pump Motor	103.6	1015	1053	998	100.0	~0.0	a	0	0 sec
15.3.2	15E.3.2-1	Trip of Both Recirculation Pump Motors	103.5	1113	1127	1109	100.1	~0.0	a	10	28 sec
15.3.3		Seizure of One Recirculation Pump (Single Loop Operation)	See Text and Appendices 15C and 15D for current cycle limits						c		
15.3.4		Recirculation Pump Shaft Break	See Text						c		
15.4		<u>REACTIVITY AND POWER ANOMALIES</u>									
15.4.1.1		RWE – Refueling	See Text						b		

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TABLE 15E.0-1 (Cont'd)
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 1 AND 2 NON-LIMITING EVENTS (VALUES ARE FOR THE INITIAL CORES ONLY)

Section	Figure	Description	Maximum Neutron Flux	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	ΔCPR[1]	Frequency Category*	Number of Valves - 1st Blowdown	Duration of Blowdown
15.4.1.2		RWE - Startup	See Text						b		
15.4.2		RWE - At Power	See Text						a		
15.4.3		Control Rod Maloperation	See Subsections 15.4.1 and 15.4.2								
15.4.4	15E.4.4-1	Startup of Idle Recirculation Loop	323.4	973	988	967	134.9	(2)	a	0	0
15.4.5		Recirculation Flow Controller Failure	See Text and Appendices 15C and 15D for current cycle limits								
15.4.7		Misplaced Bundle Accident	See Text								
15.5		INCREASE IN REACTOR INVENTORY									
15.5.1	15E.5.1-1	Inadvertent HPCI Pump Start[3]	112	1039	1079	1028	112	0.18	a	0	0
15.5.3		BWR Transients That Increase Reactor Coolant Inventory	See Sections 15.1 and 15.2								

*Frequency
- a = Moderate
- b = Infrequent
- c = Limiting Faults

Notes

1. ΔCPRs values in this Table are for the initial cores for Units 1 and 2 and are non-limiting. SEE TEXT.
2. Event initiated from low power levels - Initial core MCPR safety limit 1.06 is not violated
3. The event was re-analyzed for 3952 MWt rated power. The results show that the vent is non-limiting and is therefore reported in this Appendix.

TABLE 15E.1.1-1

OPERATOR ACTIONS WHEN REACTOR SCRAM IS INCURRED

- Place Mode switch to shutdown
- Confirm all rods have inserted and power is decreasing
- Insert neutron monitoring detectors
- Confirm main turbines trip and main generator lockout.
- Monitor and control reactor pressure
- Monitor and control reactor water level
- Cool down the reactor per standard procedure (if required)
- Monitor entry condition parameters for other Emergency Operating Procedures

TABLE 15E.1.4-1

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY RELIEF VALVE OPENING

TIME (SEC.)	EVENT
0	Opening of 1 safety relief valve, reaches full flow and remains open throughout the event.
1,200	Reactor scrammed on high suppression pool temperature. Technical Specification limit of 110°F. Closure of all MSIVs. Two loops of RHR suppression pool cooling placed into service.
5,200	Reactor depressurization initiated on high suppression pool temperature Technical Specification limit of 120°F.
59,200	Reactor depressurized to 14.7 psia, terminating blowdown through safety relief valve.

TABLE 15E.1.6-1

SEQUENCE OF EVENTS FOR INADVERTENT RHR SHUTDOWN COOLING OPERATION

APPROXIMATE ELAPSED TIME	EVENT
0	Reactor at states B or D (of Appendix 15A) when RHR shutdown cooling inadvertently activated.
0-10 min.	Slow rise in reactor power.
±10 min.	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

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TABLE 15E.2.2-1

SEQUENCE OF EVENTS FOR FIGURE 15E2.2-1 GENERATOR LOAD REJECTION, BYPASS ON

TIME, SEC	EVENT
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Generator lockout relays act to initiate turbine control fast valve closure.
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0.016	Fast control valve closure (FCV) initiates scram trip
0.016	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
0.11	Turbine bypass valves start to open.
1.145	Group 1 relief valves actuated.
1.160	Group 2 relief valves actuated.
1.356	Group 3 relief valves actuated.
1.520	Group 4 relief valves actuated.
1.789	Group 5 relief valves actuated.

TABLE 15E.2.3-1

SEQUENCE OF EVENTS FOR TURBINE TRIP
WITH BYPASS OPERABLE FIGURE 15E.2.3-1

TIME, SEC	EVENT
0	Turbine trip initiates closure of main stop valves.
0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 80% open position and initiate a recirculation pump trip (RPT)
0.1	Turbine stop valves closed.
0.1	Turbine bypass valves start to open to regulate pressure.
1.7	Group 1 relief valves actuated.
1.9	Group 2 relief valves actuated.
2.2	Group 3 relief valves actuated.
2.4	Group 4 relief valves actuated.
2.6	Group 5 relief valves actuated.
4.3	L8 vessel level set point trips feedwater pumps.
10.1	Group 1 relief valves close.
44.8	Vessel water level decreases to L2 vessel level set point.
75 (est.)	HPCI/RCIC flow enters vessel not simulated).

TABLE 15E.2.4-1

SEQUENCE OF EVENTS FOR MSIV CLOSURE, FIGURE 15E.2.4-1

TIME, SEC	EVENT
0	Initiate closure of all main steam line isolation valves (MSIV).
0.30	MSIVs reach 90%* open.
0.36	MSIV position trip scram initiated.
-	Group 1 safety valves assumed to be out of service
3.7	Group 2 safety valves open.
3.8	Group 3 safety valves open.
~20	All safety valves reclose (estimate).
23.0	Group 1 pressure relief valves reopen.
29.0	Group 1 pressure relief valves reclose.
36.0	Group 1 pressure relief valves reopen.
40.0	Group 1 pressure relief valves reclose.

Valves opening based on safety settings is conservative
*Changed to 85% with no significant impact on transient results.

TABLE 15E.2.5-1

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

CAUSE	ESTIMATED VACUUM DECAY RATE
1. Failure or Isolation of Steam Jet Air Ejectors	<1 inch Hg/minute
2. Loss of Sealing Steam to Shaft Gland Seals	~1 to 2 inches Hg/minute
3. Opening of Vacuum Breaker Valves	~2 to 12 inches Hg/minute
4. Loss of One or More Circulating Water Pumps	~4 to 24 inches Hg/minute

TABLE 15E.2.6-1

LOSS OF AUXILIARY POWER
SEQUENCE OF EVENTS FOR FIGURE 15E.2.6-1

TIME, SEC	EVENT
0	Loss of auxiliary power transformer occurs.
0	Recirculation system pump motors are tripped.
0	Condensate booster pumps are tripped.
0	Condenser circulating water pumps tripped.
2.0	Closure of main steamline isolation valves initiated.
2.0	Reactor scram initiated.
4.0	Feedwater turbines tripped off.
4.4	Group 1 Safety/Relief valves actuated.
21	Group 1 Safety/Relief valves closed.
32	Initiate HPCI and RCIC operation (not simulated).

TABLE 15E.2.7-1

LOSS OF FEEDWATER FLOW
SEQUENCE OF EVENTS FOR FIGURE 15E.2.7-1

TIME, SEC	EVENT
0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays essentially to zero.
5.1	Recirculation pumps runback, low feed water flow
6.8	Vessel water level (L3) trip initiates scram trip.
52.8	Vessel water level (L2) trip initiates recirculation pump system trip.
52.8	Vessel water level (L2) trip initiates containment isolation.
82.8	Vessel water level (L2) trip initiates RCIC operation – (30 sec. delay) (HPCI not simulated)
	The MSIVs will not close until water level reaches L1. Water level is not expected to reach L1 during this event since RCIC initiates at L2.
	SRVs Do Not Open

TABLE 15E.2.9-1

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

TIME	EVENT
0	Reactor is operating at 102% of rated thermal power when LOP transient occurs initiating plant shutdown
0	Concurrently loss of one division of power occurs
10 min.	Controlled depressurization initiated (100°F/hr) and continues until vessel pressure reaches approximately 115 psia
15 min.	Operators initiate suppression pool cooling
140 min.	When vessel pressure reaches 115 psia, a failure in a shutdown cooling suction valve prevents operation of normal shutdown cooling
170 min.	Operator initiates core spray for use with alternate shutdown cooling. Operator opens ADS valves to achieve continuous core flow
6.5 hrs	Peak suppression pool temperature is attained

TABLE 15E.3.1-1

SEQUENCE OF EVENTS FOR TRIP OF ONE RECIRCULATION PUMP

TIME, SEC	EVENT
0	Trip of one recirculation pump initiated.
5.7	Diffuser flow decreases significantly in the tripped loop.
30.0	Core flow stabilizes at new equilibrium conditions.
42.0	Power level stabilizes at new equilibrium conditions

TABLE 15E.4.4-1

SEQUENCE OF EVENTS ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP FOR FIGURE 15E.4.4-1

TIME, SECOND	EVENT
0	Start pump motor.
9.0	Startup loop flow starts to increase significantly.
10.0	Reactor high flux scram initiated.
11.0	Vessel level reaches (L4) Low Level Alarm.
23.5	Vessel level reaches (L3) Low Level Scram Trip.
35.0	Diffuser flows and pressures begin to stabilize.
50.0	Vessel level begins to stabilize.

TABLE 15E.5.1-1

SEQUENCE OF EVENTS FOR INADVERTENT STARTUP OF HPCI

Time, sec.	Event
0.0	Simulate HPCI cold water injection
1.0	Full flow established for HPCI
	No reactor scram
	No recirculation pump trip
	No SRV flow
$\approx$50	Reactor variables settle into new steady state

TABLE 15E.0-2
INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNITS 1 AND 2 INITIAL CYCLES

1.	Thermal Power Level, MWT Rated Value (NBR) Analysis Value (104.4% NBR)	 3293 3439
2.	Steam Flow, Mlbs/hr Analysis Value (105% NBR)	 14.153
3.	Core Flow, Mlbs/hr	100.0
4.	Feedwater Flow Rate, lbs/sec Analysis Value	3921
5.	Feedwater Temperature,°F	386.9
6.	Vessel Dome Pressure, psig	1020.0
7.	Vessel Core Pressure, psig	1030.0
8.	Turbine bypass Capacity, %NBR	25%
9.	Core Coolant Inlet Enthalpy, BTU/lb	521.1
10.	Turbine Inlet Pressure, psig	960.0
11.	Fuel Lattice	8x8
12.	Core Average Gap Conductance, BTU/sec-ft^2-°F	0.1744
13.	Core Leakage Flow,%	9.85
14.	Required MCPR Operating Limit	Not Applicable to current cycles ΔCPRs are non-limiting
15.	MCPR Safety Limit	1.06
16.	Doppler Coefficient** Nominal EOC-1 (-)cents/°F Analysis Data	 0.2255 0.2142

TABLE 15E.0-2 (Cont'd)
INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNITS 1 AND 2 - INITIAL CYCLES

17.	Void Coefficient** Nominal EOC-1 (-)cents/°F Analysis Data for Power Increase Events Analysis Data for Power Decrease Events		 7.48 12.0 6.61
18.	Core Average Rated Void Fraction **		40.74
19.	Scram Reactivity Analysis Data		See Figure 15E.0-2
20.	Control Rod Drive Speed, Position Versus Time		See Figure 15E.0-2
21.	Jet Pump Ratio, M		1.84
22.	Safety Relief Valve Capacity, % NBR @ 1091 psig Manufacturer Number installed		 99.0 Crosby 16
23.	Relief Function Delay, sec		0.4
24.	Relief Function Response, sec		<0.15
25.	Set Points for Safety/Relief Valves, psig		2@1110 3@1140 4@1120 3@1150 4@1130
26.	Number of Valve Groups Simulated		5
27.	High Flux Trip, %NBR Analysis set point		125.3
28.	High Pressure Scram Set Point, psig		1071
29.	Vessel Level Trips, Inches Above(+), Below (-) Dryer Skirt Bottom, Tech Spec settings	High Level Low Level Low Low Level	(L8)≤ 58.7 (L4)≤ 30 (L3)≥ 12.5 (L2)≥ -38
30.	APRM Thermal Trip Set Point,%NBR		125.0

TABLE 15E.0-2 (Cont'd)
INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNITS 1 AND 2 - INITIAL CYCLES

31.	Recirculation Pump Trip Delay, sec	0.175
32.	Recirculation Pump Trip Inertia time constant for Analysis*, sec	4.5
	*The inertia time constant is defined by the expression: $t= (2\pi J_0 n)/gT_0$ Where: t = Inertia time constant, sec J_0 = Pump Motor Inertia, lb-ft^2 n = Rated pump speed, rps g = gravitational constant ft/sec^2 T_0 = Pump shaft torque, lb-ft	
**	Parameters used in REDY only. ODYN values are calculated within the code for equilibrium cycle conditions.	

TABLE 15E.1.4-2

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY ABOVE SUPPRESSION POOL (curies)

Isotope	Realistic	Design Basis
I-131	1.35E-03[1]	5.40E-01
I-132	3.64E-02	6.19E+00
I-133	1.90E-02	3.81E+00
I-134	9.27E-02	1.54E+01
I-135	2.23E-02	5.87E+00
Kr-83m	1.44E+01	2.93E+00
Kr-85m	2.54E+00	5.25E+00
Kr-85	7.94E-03	1.72E-02
Kr-87	8.73E+00	1.72E+01
Kr-88	8.73E+00	1.72E+01
Kr-89	5.40E+01	1.12E+02
Xe-131m	6.19E-03	1.29E-02
Xe-133m	1.19E-01	2.50E-01
Xe-133	3.33E+00	7.06E+00
Xe-135m	1.11E+01	2.24E+01
Xe-135	9.53E+00	1.89E+01
Xe-137	6.19E+01	1.29E+02
Xe-138	3.65E+01	7.67E+01

1. 1.35E-03 = 1.35×10^{-03}

Table 15E.2.4-2

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TABLE 15E.2.5-2

LOSS OF CONDENSER VACUUM
SEQUENCE OF EVENTS FOR FIGURE 15E.2.5-1

TIME, SEC	EVENT
-0.0 (est)	Initiate simulated loss of condenser vacuum at 2 inches of Hg per second.
0.0 (est)	Low condenser vacuum main turbine trip actuated.
0.0 (est)	Low condenser vacuum feedwater trip actuated.
-0.01 (est)	Main turbine trip initiates reactor scram.
0.01 (est)	Main turbine trip initiates recirculation pump trip (RPT)
.1 (est)	Turbine stop valve closes.
1.7	Group 1 relief valves set points actuated.
1.9	Group 2 relief valves set points actuated.
2.2	Group 3 relief valves set points actuated.
2.4	Group 4 relief valves set points actuated.
6.5	Low condenser vacuum initiates main steam line isolation valve closure.
6.5	Low condenser vacuum initiates bypass valve closure.
21.5	Group 1 relief valves close.
23.5	Vessel water level decreases to L2 vessel level set point.
53 (est)	HPCI/RCIC system flow enters vessel (not included in simulation).
90+	Relief valves cycle as required on pressure.

TABLE 15E.2.6-2

LOSS OF ALL GRID CONNECTIONS
SEQUENCE OF EVENTS FOR FIGURE 15E.2.6-2

TIME, SEC	EVENT
(-)0.015 (approx.)	Loss of Grid causes turbine-generator to detect a loss of electrical load.
0	Control valve fast closure.
0	Turbine-generator trip initiates main turbine bypass system operation.
0	Recirculation system pump motors are tripped.
0	Fast control valve closure (FCV) initiates a reactor scram trip.
0	Initiation of standby AC power systems.
0.1	Turbine bypass valves open.
0.15	Turbine control valves closed.
1.2	Group 1 relief valves actuated.
1.4	Group 2 relief valves actuated.
1.5	Group 3 relief valves actuated.
1.7	Group 4 relief valves actuated.
2.0	MSIV's start to closure.
4.0	Feedwater turbines tripped off.
18.7	Group 1 safety relief valves close.
37.2	Initiate Containment Isolation, HPCI and RCIC operation, (L2) (not simulated).

TABLE 15E.2.9-2

INPUT PARAMETERS FOR EVALUATION OF FAILURE
OF RHR SHUTDOWN COOLING

Core Thermal Power (MWt)	4031
Initial RPV Pressure (psia)	1050
Initial Vessel Temperature (°F)	550
Suppression Pool Temperature (°F)	90
Suppression Pool Liquid Volume (ft^3)	115,810
Service Water Temperature (°F)	97
RHR Heat Exchanger K-value (Btu/sec-°F)	317.5
RHR Pool Cooling Flow Rate (gpm)	9750
Core Spray (1 Loop) Flow Rate (gpm)	7900

TABLE 15E.3.1-2

SEQUENCE OF EVENTS FOR TRIP OF TWO RECIRCULATION PUMPS

TIME, SEC	EVENT
0	Trip of both recirculation pumps initiated.
4.0	Vessel water level (L8) trip initiates turbine trip.
4.0	Feedwater pumps are tripped off.
4.0	Turbine trip initiates bypass operation.
4.0	Turbine trip initiates reactor scram trip.
7.0	Group 1 pressure relief valves open.
12.0	Group 1 pressure relief valves closed.
46.0	L2 vessel level set point initiate HPCI and RCIC.
76	HPCI/RCIC flow enter vessel (not simulated).

TABLE 15E.1.4-3

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY RELEASED TO THE ENVIRONS (curies)

Isotope	Realistic	Design Basis
I-131	1.31E-05[1]	5.24E-03
I-132	3.35E-05	5.71E-03
I-133	1.46E-04	2.92E-02
I-134	1.52E-06	2.52E-04
I-135	9.80E-05	2.58E-02
Kr-83m	7.31E-01	1.48E-01
Kr-85m	7.18E-01	1.48E+00
Kr-85	7.94E-03	1.72E-02
Kr-87	1.10E-01	2.17E-01
Kr-88	1.20E+00	2.37E+00
Kr-89	1.65E-44	3.43E-44
Xe-131m	6.07E-03	1.27E-02
Xe-133m	1.08E-01	2.35E-01
Xe-133	3.20E+00	6.91E+00
Xe-135m	1.68E-03	4.42E-01
Xe-135m	5.37E+00	1.27E+01
Xe-137	4.13E-36	8.62E-36
Xe-138	1.12E-07	2.36E-07

1. 1.31E-05 = 1.31×10^{-05}

Table 15E.2.4-3

This Table Has Been Deleted

TABLE 15E.2.5-3

TRIPS SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

VACUUM (INCHES OF HG)	PROTECTIVE ACTION INITIATED
27 to 28	Normal Vacuum Range
21.7	Main Turbine Trip (Stop Valve Closure)
10.2	Main Steam Line Isolation Valve (MSIV) Closure
7	Mainsteam Turbine Bypass Valves Closure
17.4	Reactor Feed Pump Turbine Trip (Stop Valves Closure)

TABLE 15E.1.4-4

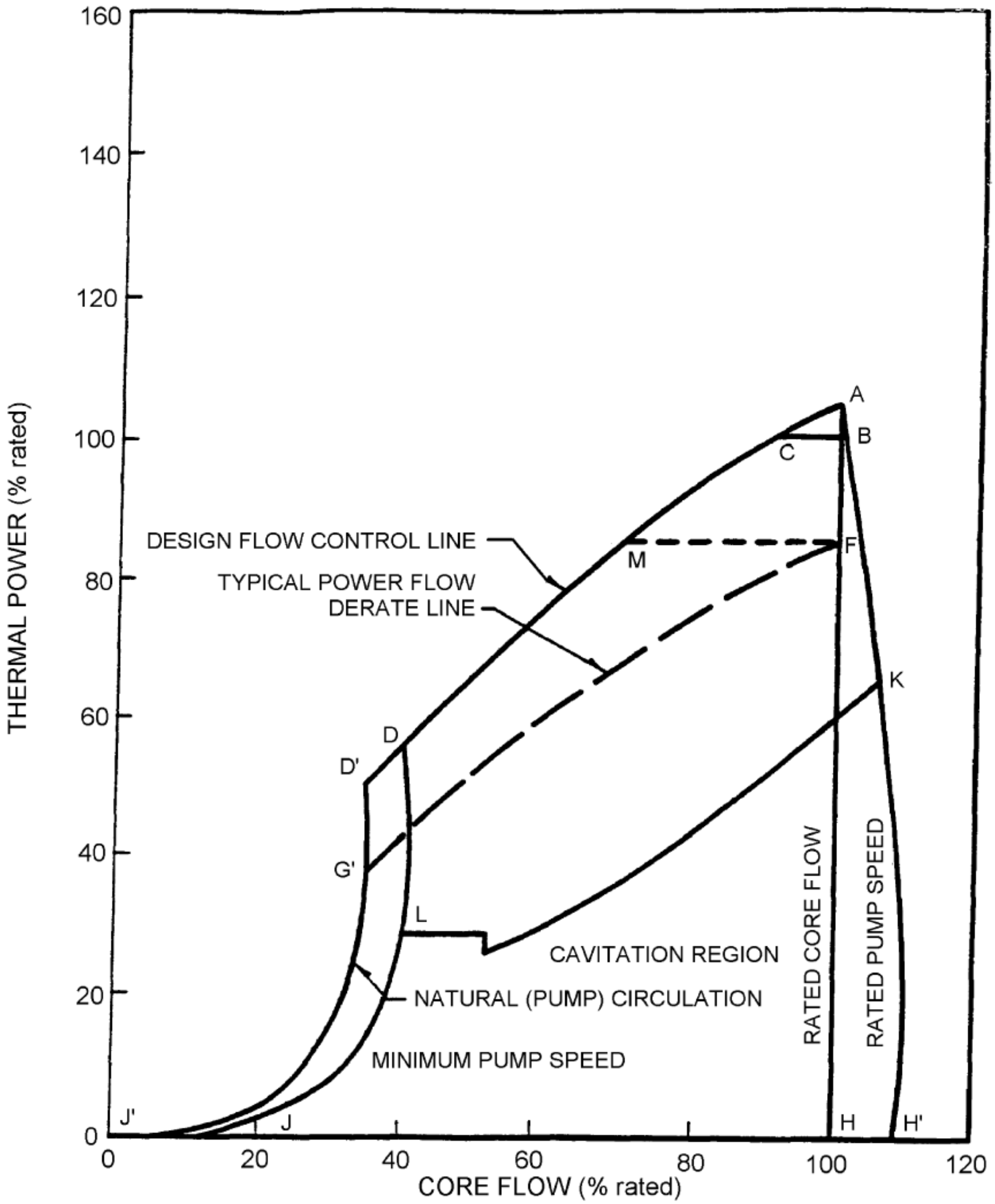
SAFETY RELIEF VALVE OPENING EVENT
OFFSITE RADIOLOGICAL DOSES (rems)

Source Terms	Total Whole Body Gamma	Thyroid Inhalation
Realistic	1.12E-06[(1)]	2.65E-08
Design Basis	2.43E-06	7.16E-06

1. $1.12E\text{-}06 = 1.12 \times 10^{-06}$

Table 15E.2.4-4

This Table Has Been Deleted



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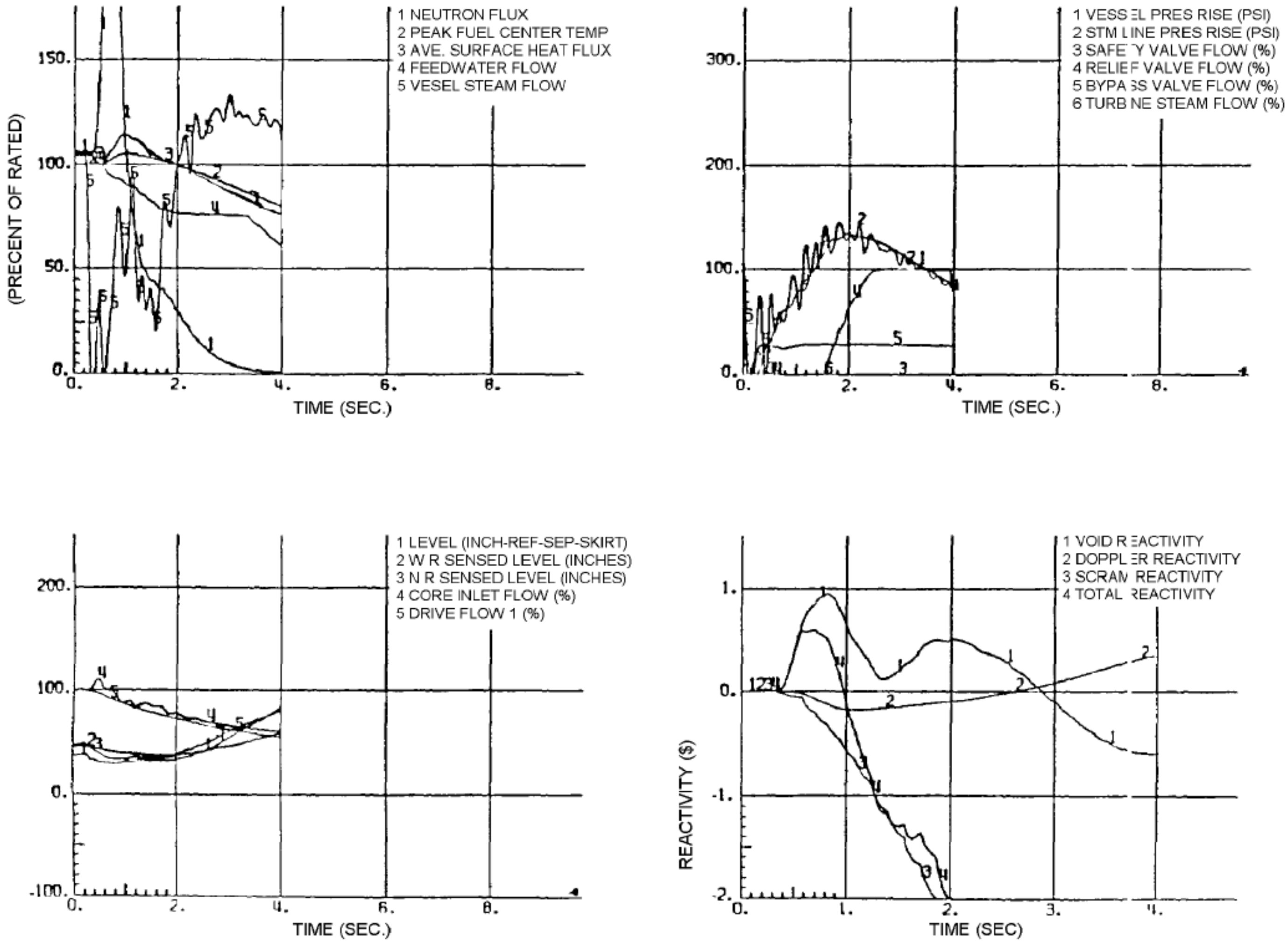
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL POWER/FLOW MAP

FIGURE 15E.0-1, Rev 54

AutoCAD: Figure Fsar 15E_0_1.dwg



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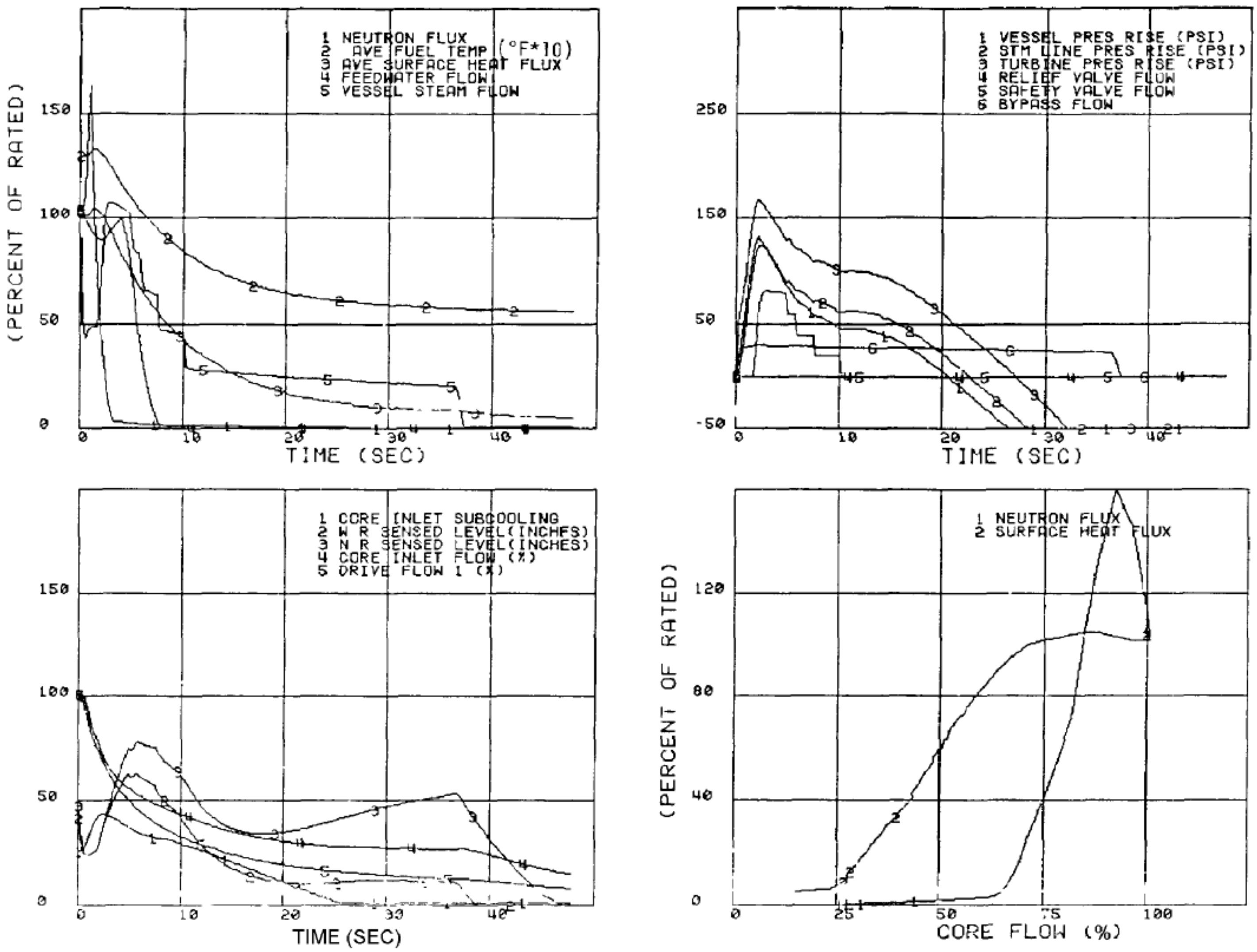
SUSQUEHANNA GENERATOR
LOAD REJECTION,
WITH BYPASS ON

FIGURE 15E.2.2-1, Rev 54

AutoCAD: Figure Fsar 15E_2_2_1.dwg



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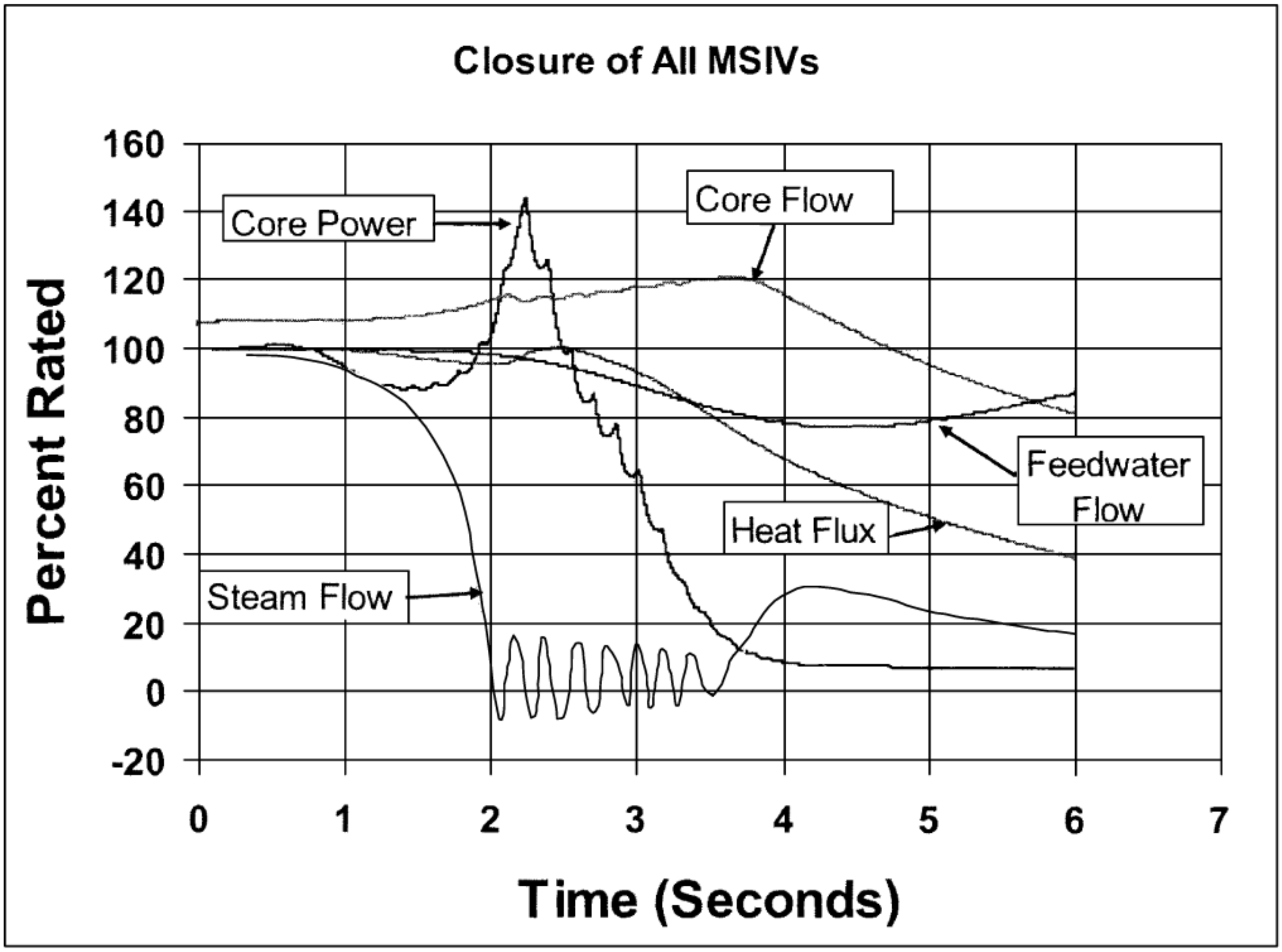
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TURBINE TRIP,
TRIP SCRAM,
BYPASS AND RPT-ON

FIGURE 15E.2.3-1, Rev 54

AutoCAD: Figure Fsar 15E_2_3_1.dwg



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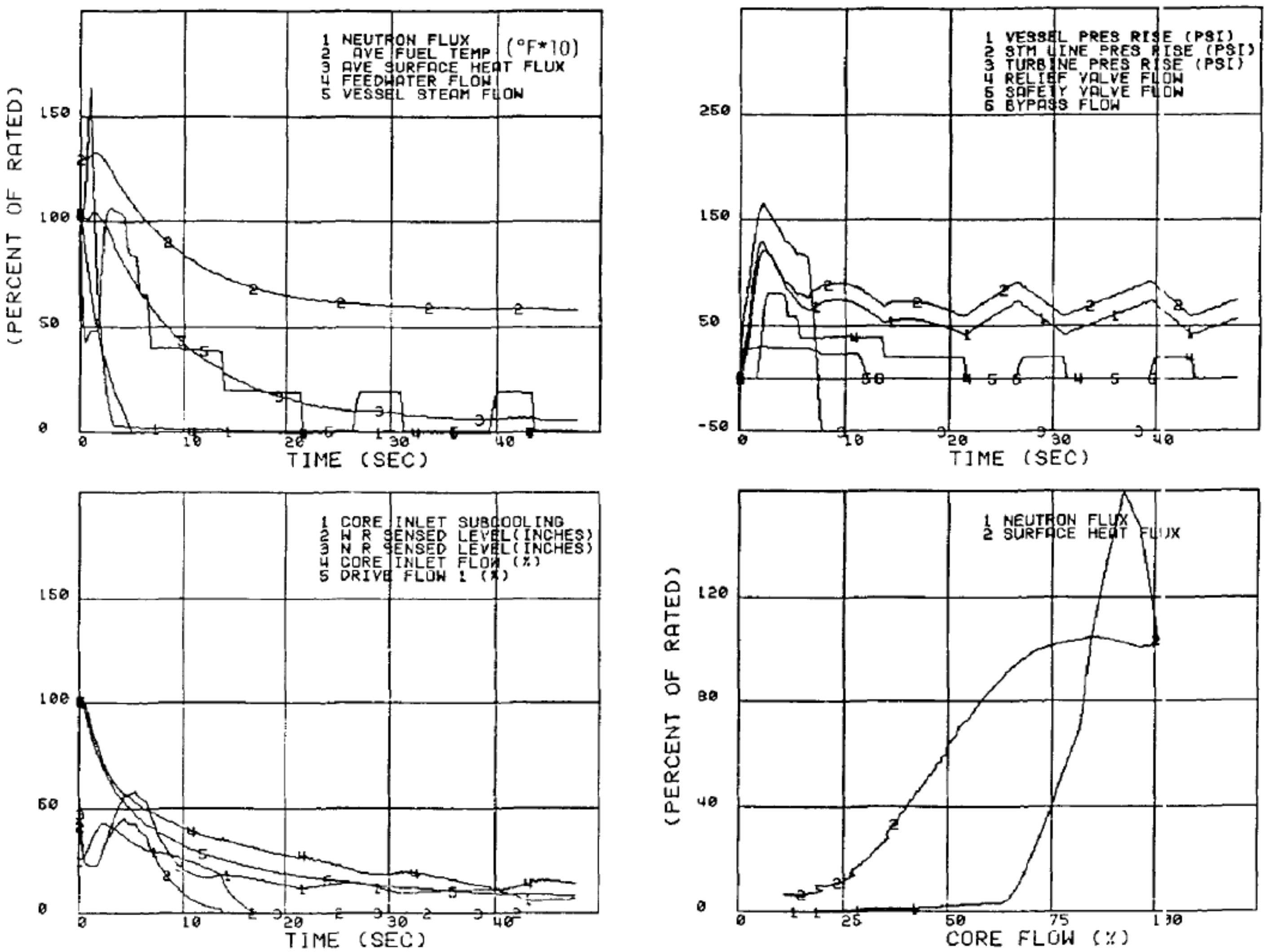
CLOSURE OF ALL MSIVs

FIGURE 15E.2.4-1, Rev 55

AutoCAD: Figure Fsar 15E_2_4_1.dwg



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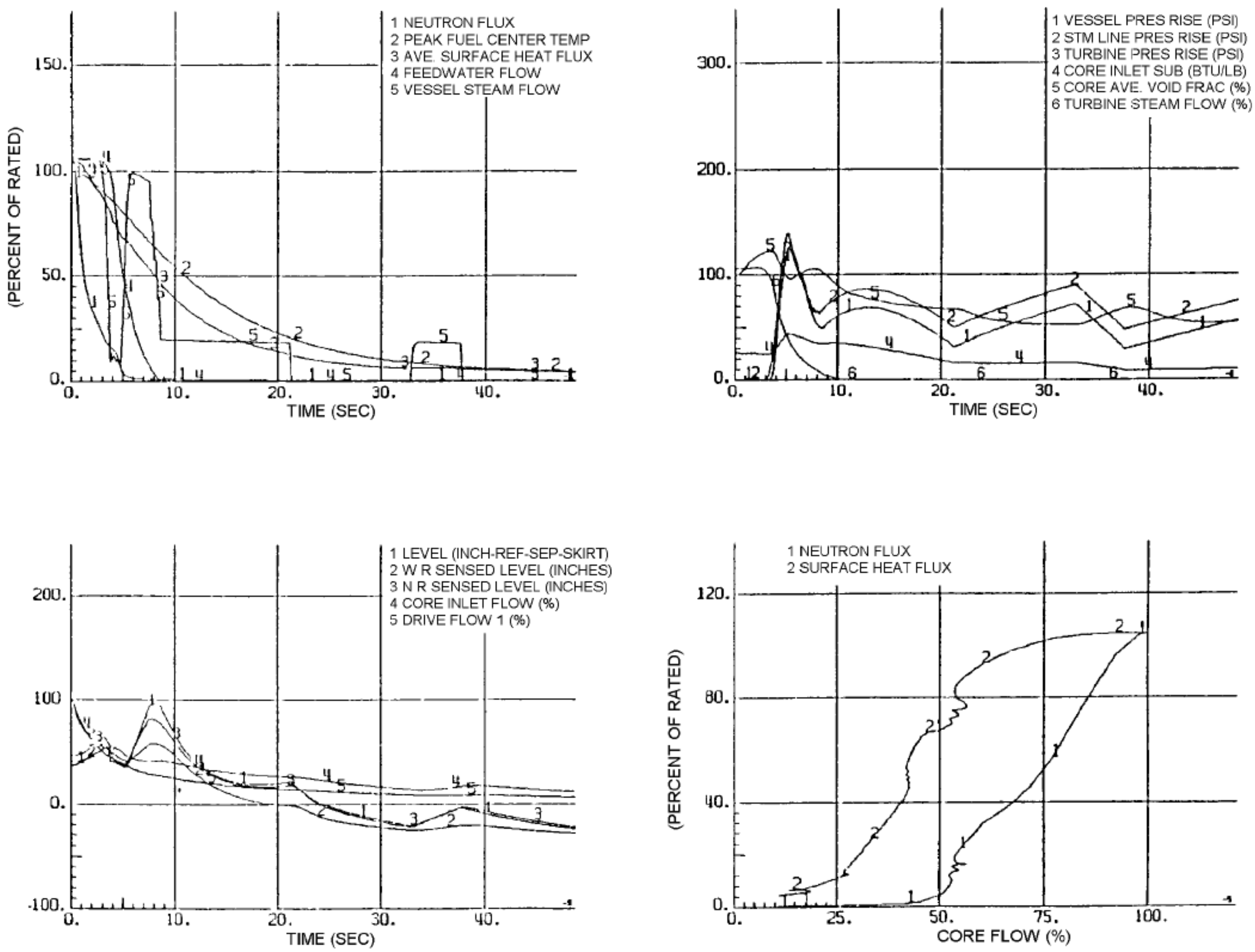
LOSS OF
CONDENSER VACUUM AT
2 INCHES PER SECOND

FIGURE 15E.2.5-1, Rev 54

AutoCAD: Figure Fsar 15E_2_5_1.dwg



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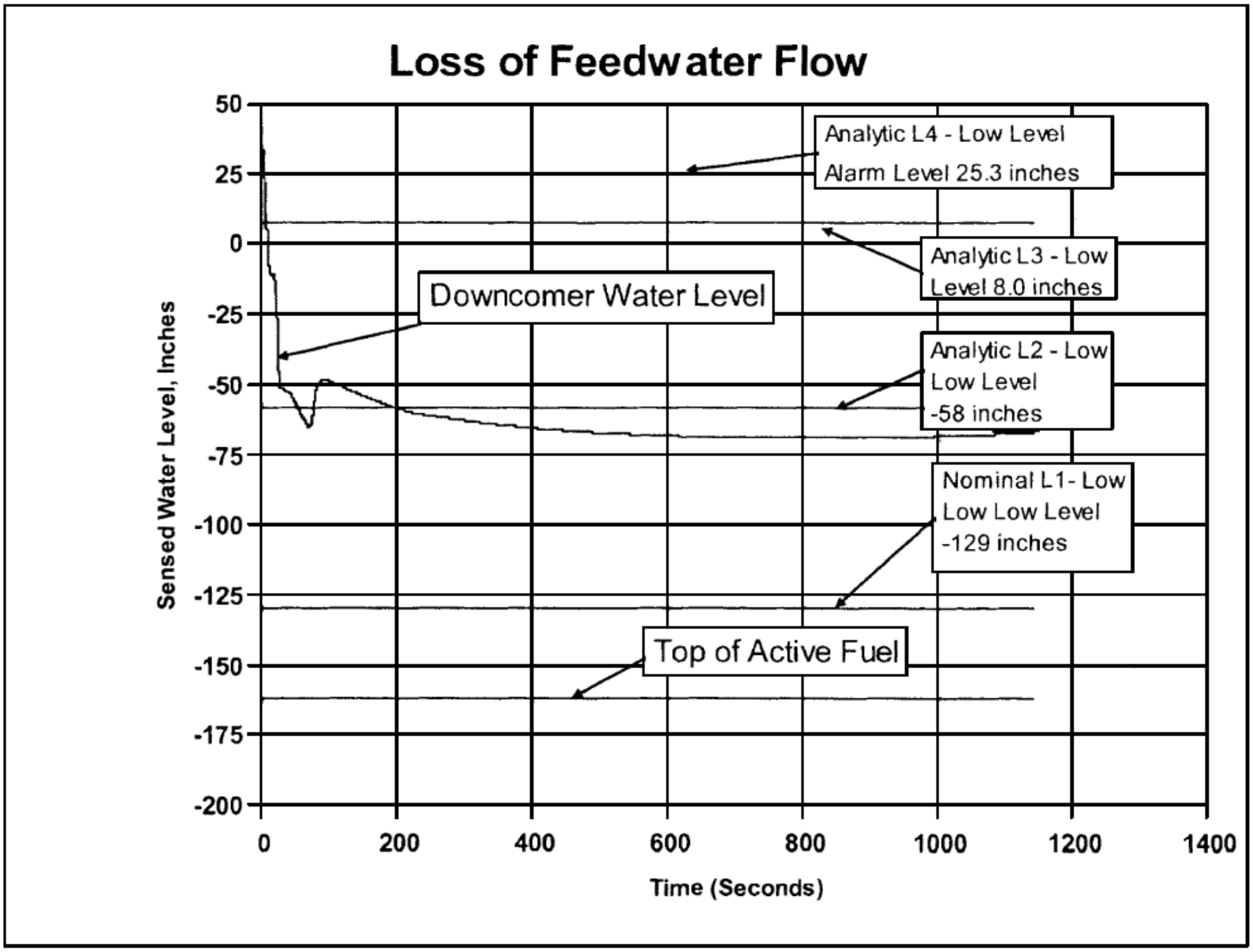
LOSS OF
AUXILIARY POWER
TRANSFORMER

FIGURE 15E.2.6-1, Rev 54

AutoCAD: Figure Fsar 15E_2_6_1.dwg



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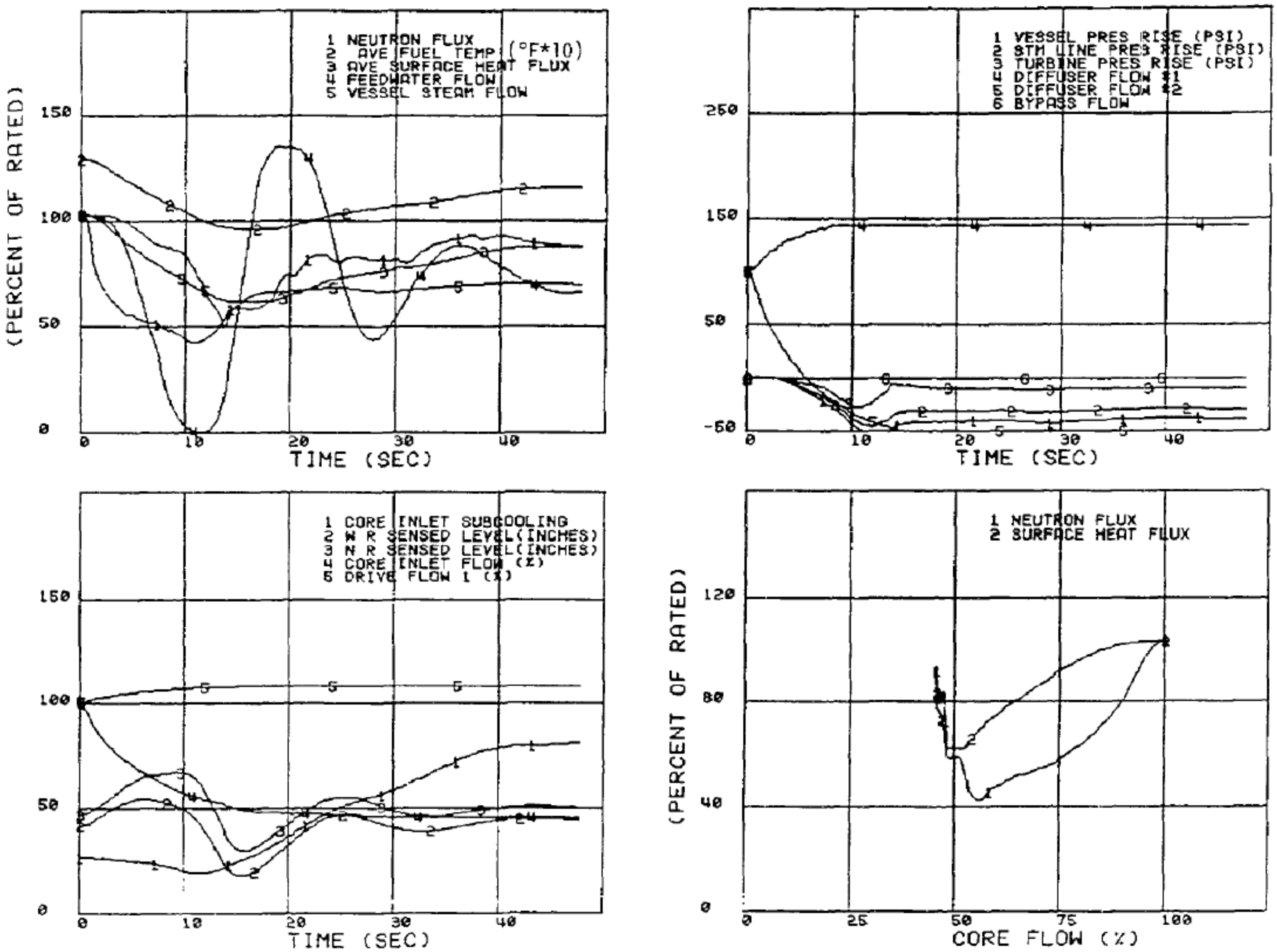
LOSS OF FEEDWATER
SENSED WATER LEVEL

FIGURE 15E.2.7-1, Rev 55

AutoCAD: Figure Fsar 15E_2_7_1.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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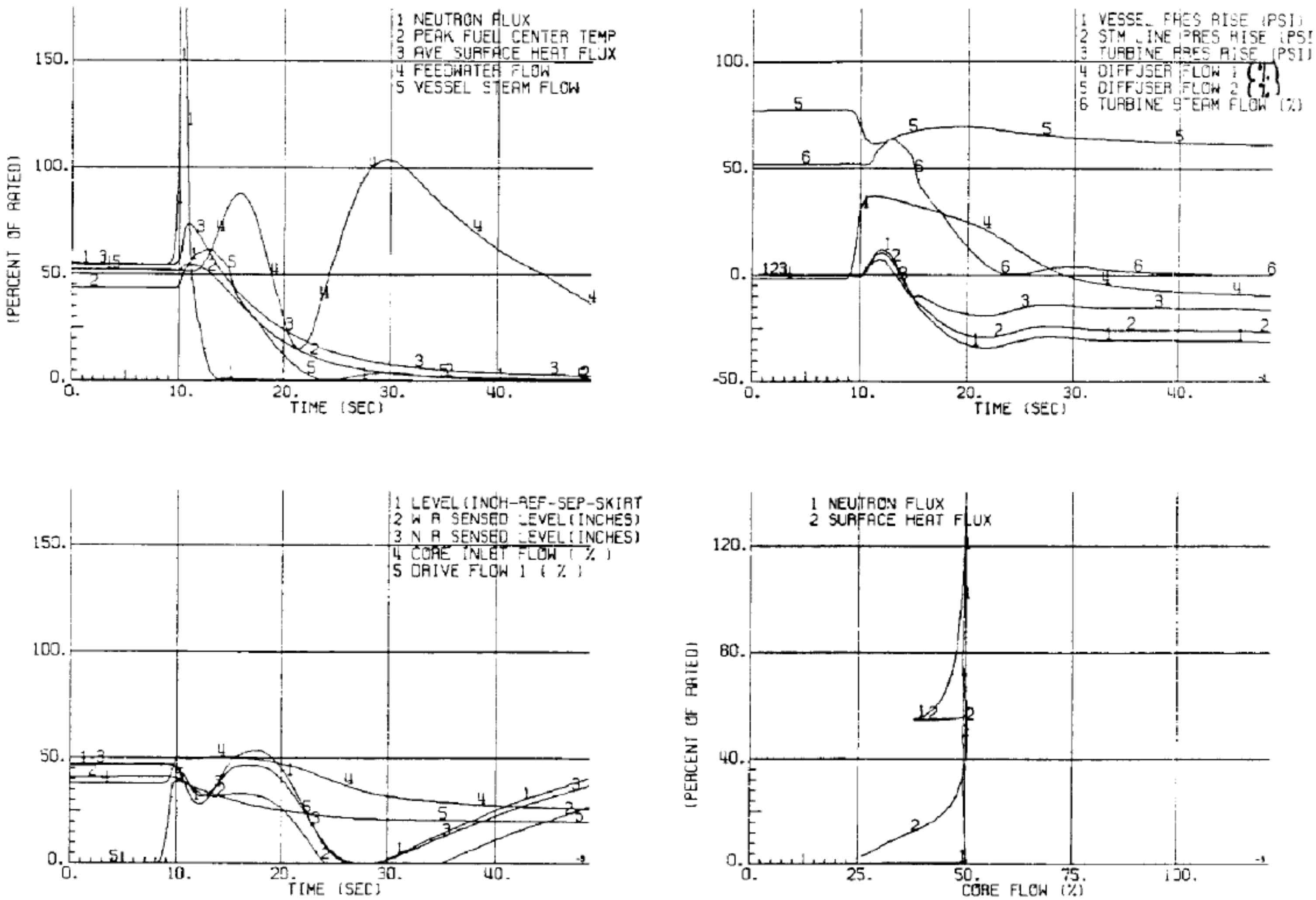
TRIP OF ONE
RECIRCULATION
PUMP MOTOR

FIGURE 15E.3.1-1, Rev 54

AutoCAD: Figure Fsar 15E_3_1_1.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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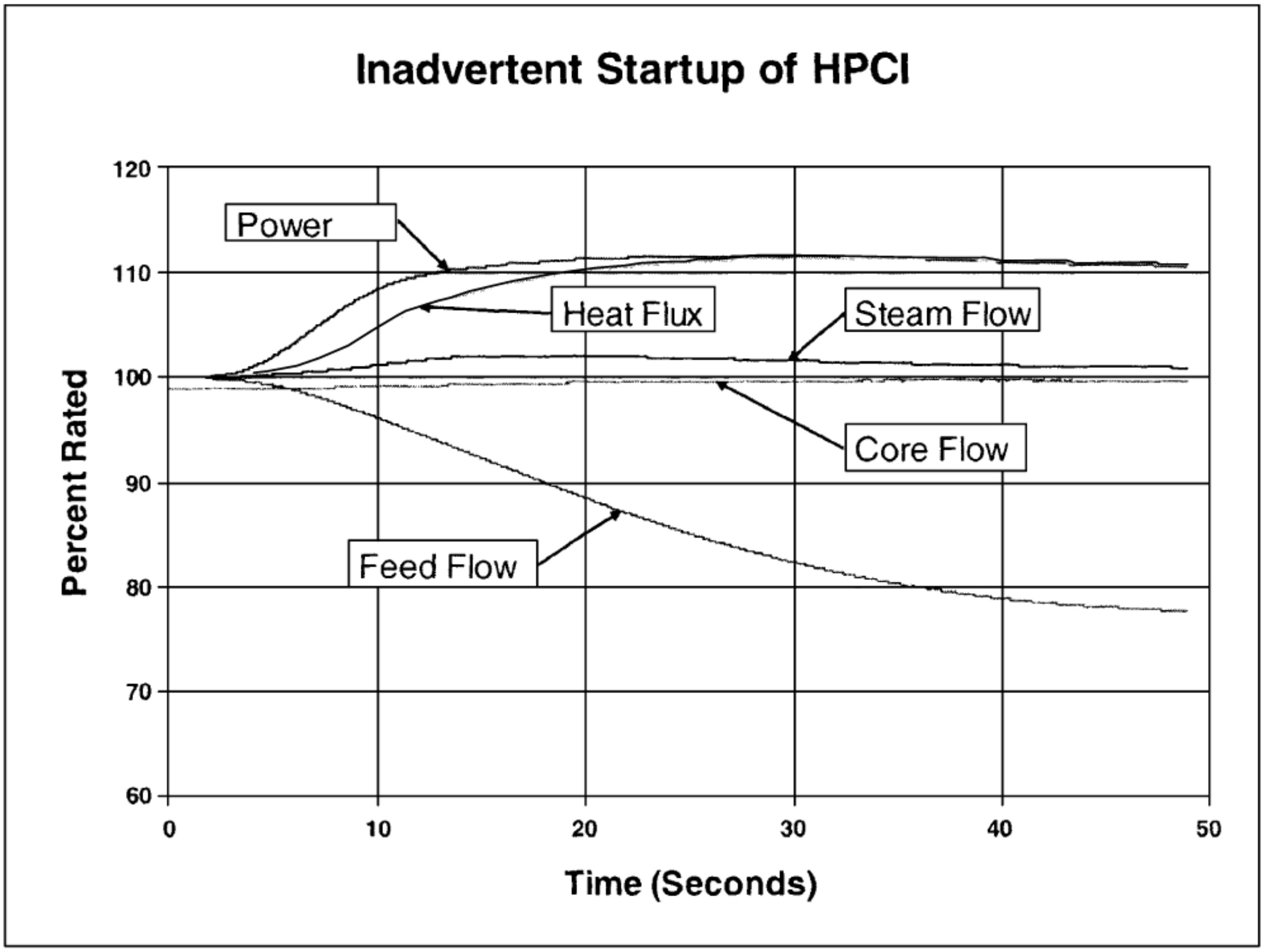
STARTUP OF IDLE
RECIRCULATION LOOP
PUMP

FIGURE 15E.4.4-1, Rev 54

AutoCAD: Figure Fsar 15E_4_4_1.dwg



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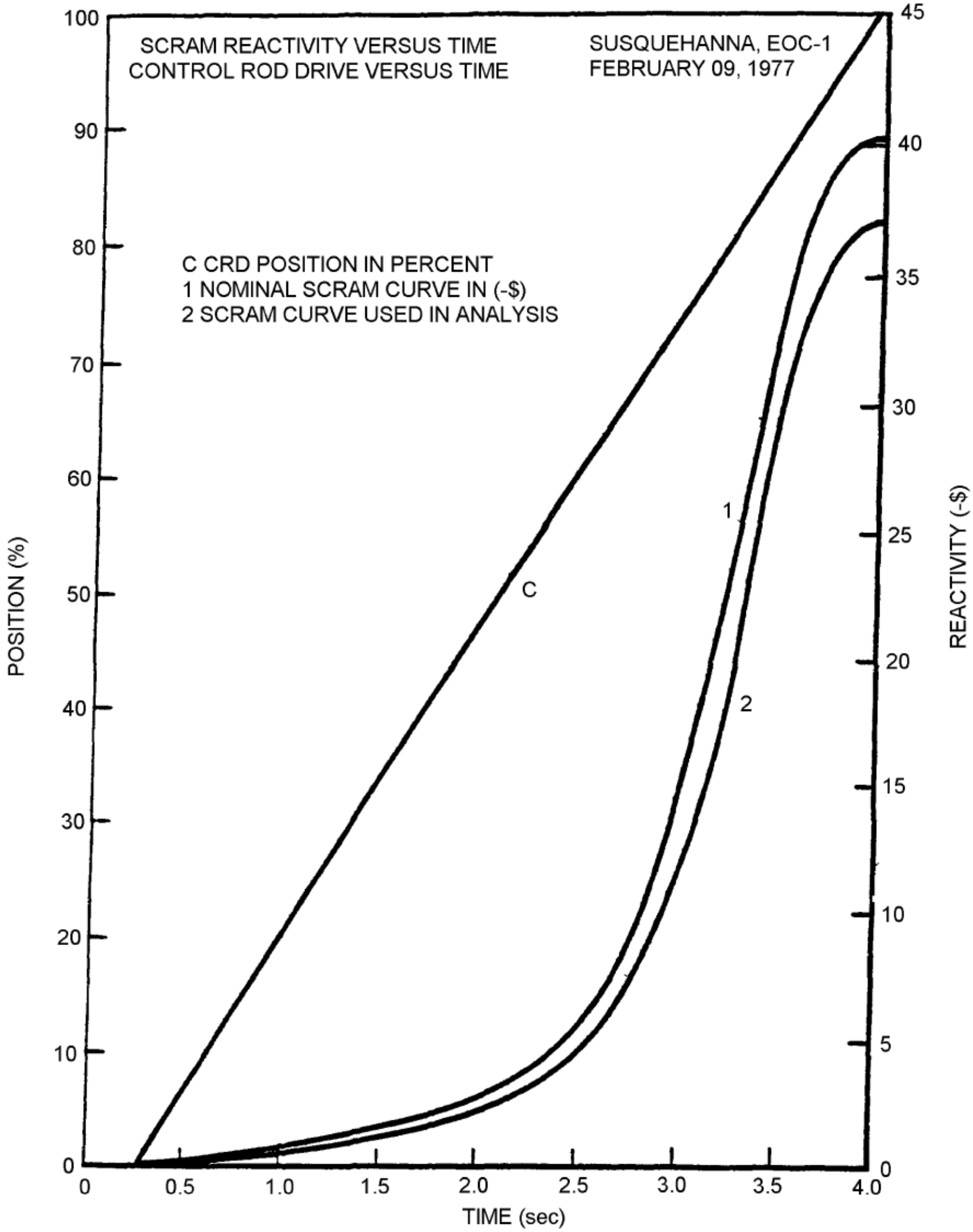
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
INADVERTENT HPCI PUMP START
FIGURE 15E.5.1-1, Rev 3

AutoCAD: Figure Fsar 15E_5_1_1 dwg



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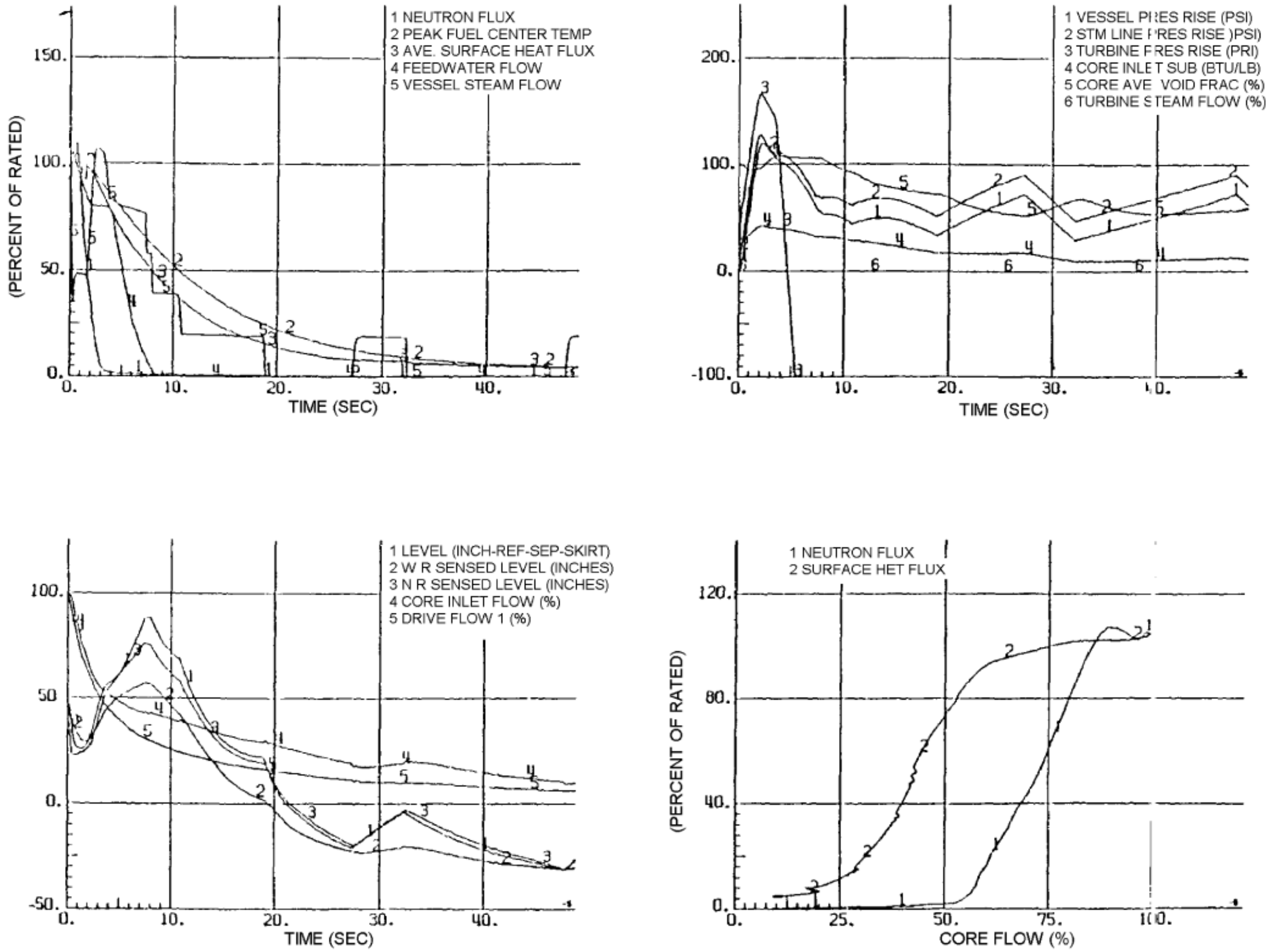
SCRAM POSITION
AND
REACTIVITY CHARACTERISTICS

FIGURE 15E.0-2, Rev 54

AutoCAD: Figure Fsar 15E_0_2.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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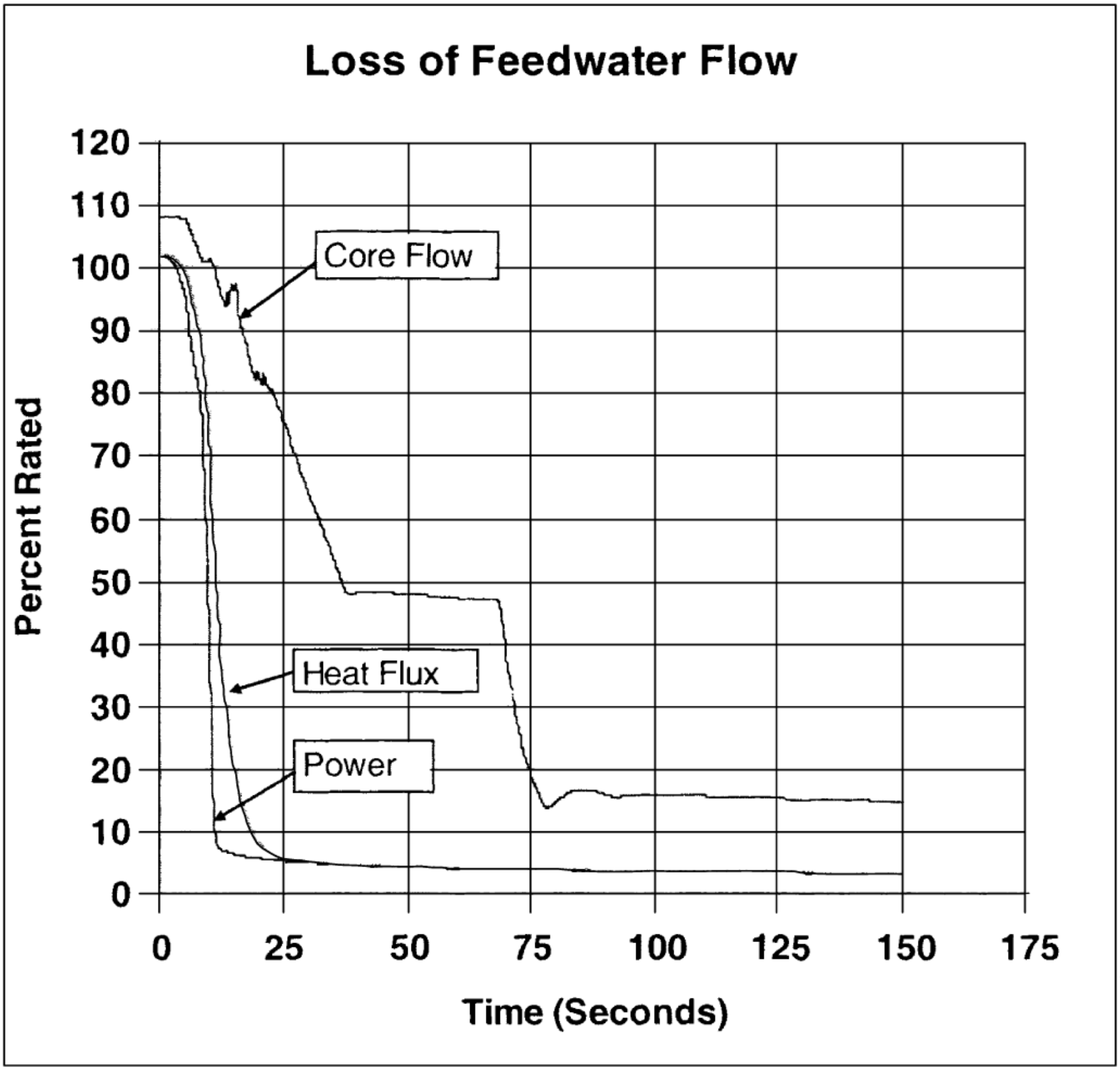
LOSS OF ALL GRID CONNECTIONS

FIGURE 15E.2.6-2, Rev 54

AutoCAD: Figure Fsar 15E_2_6_2.dwg



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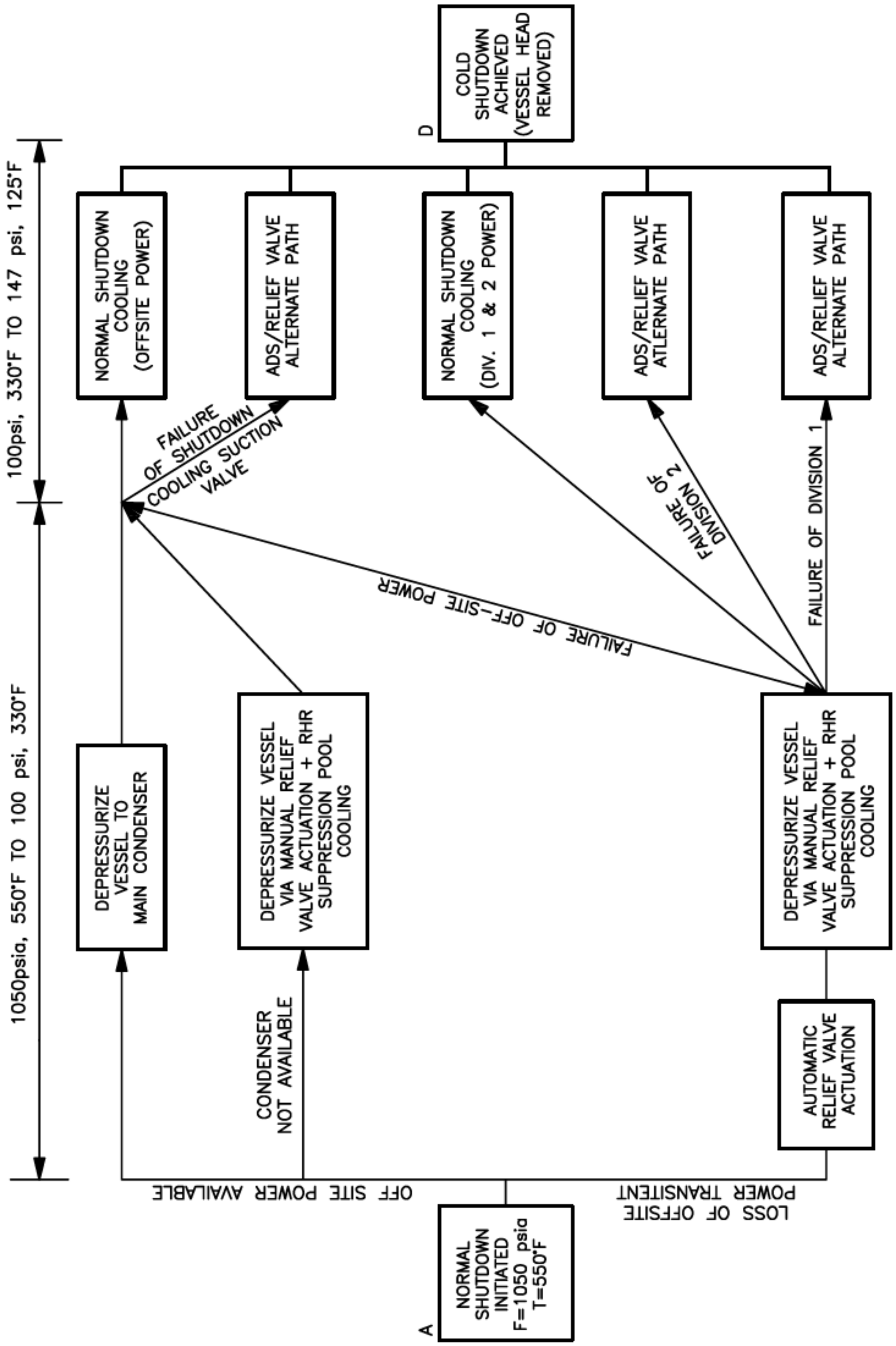
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LOSS OF FEEDWATER
CORE POWER, HEAT FLUX AND CORE FLOW

FIGURE 15E.2.7-2, Rev 1

AutoCAD: Figure Fsar 15E_2_7_2.dwg



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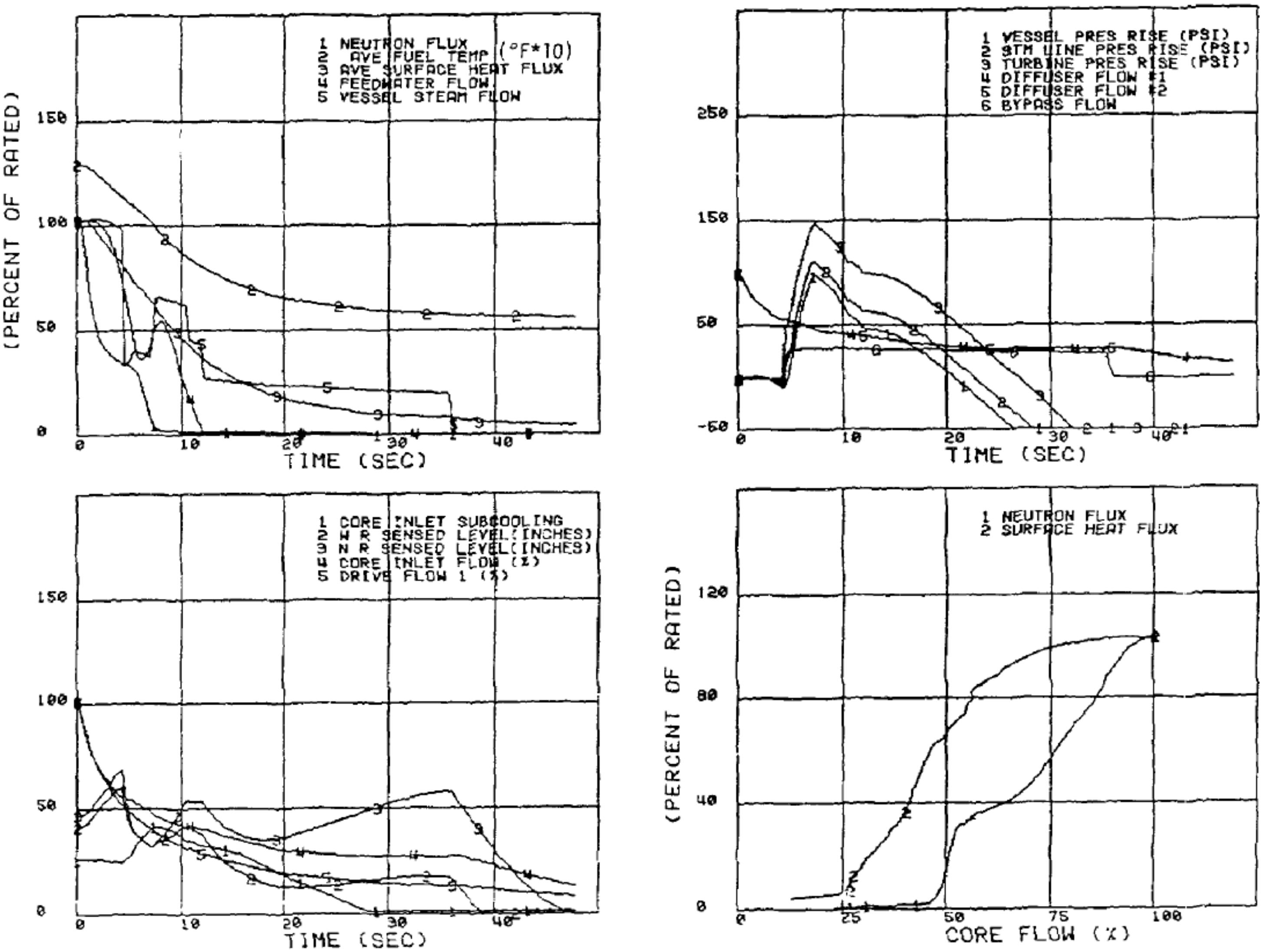
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SUMMARY OF PATHS
AVAILABLE TO ACHIEVE
COLD SHUT DOWN

FIGURE 15E.2.9-2, Rev 55

AutoCAD: Figure Fsar 15E_2_9_2.dwg



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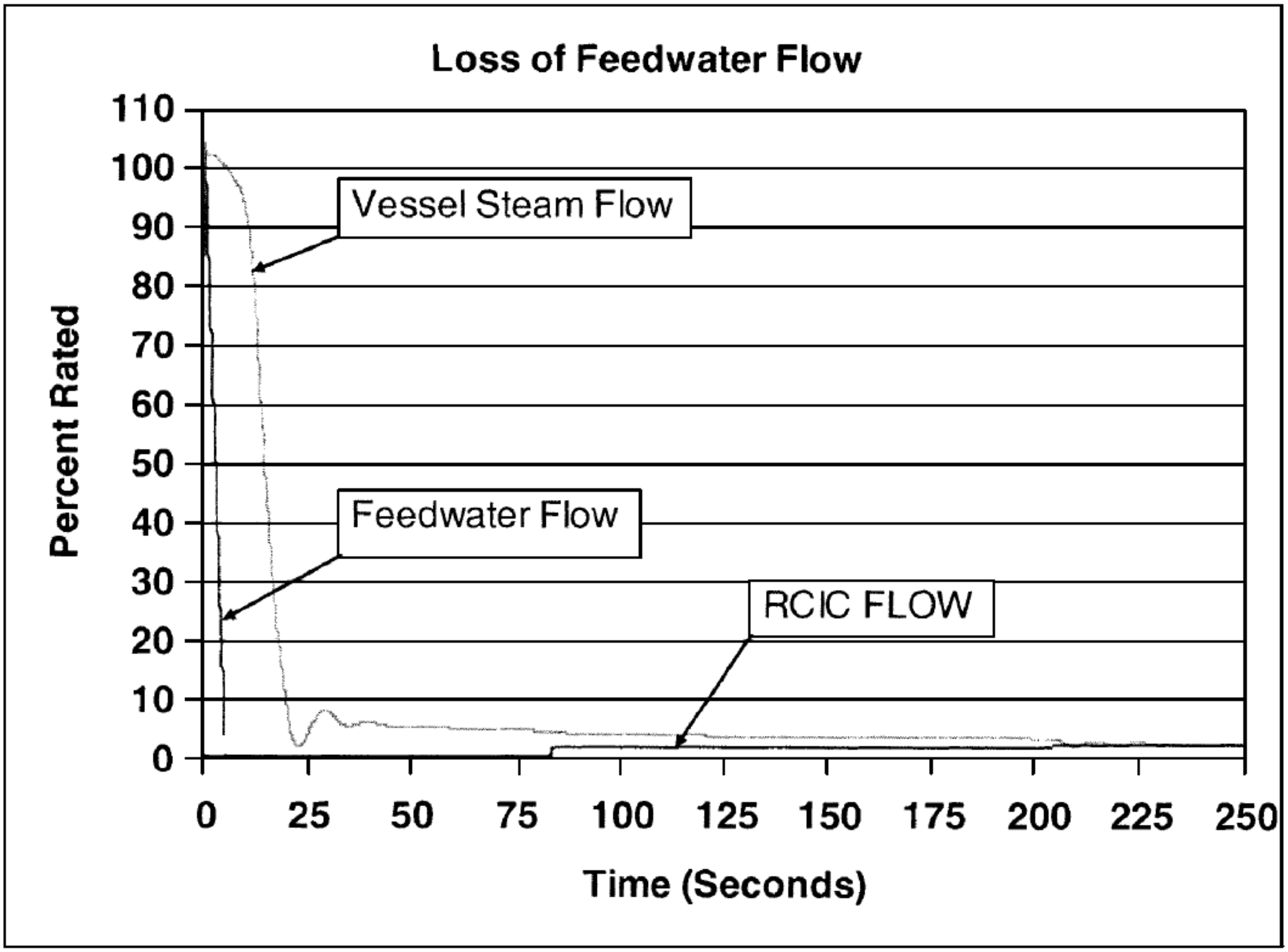
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TRIP OF BOTH
RECIRCULATION
PUMP MOTORS

FIGURE 15E.3.1-2, Rev 54

AutoCAD: Figure Fsar 15E_3_1_2.dwg



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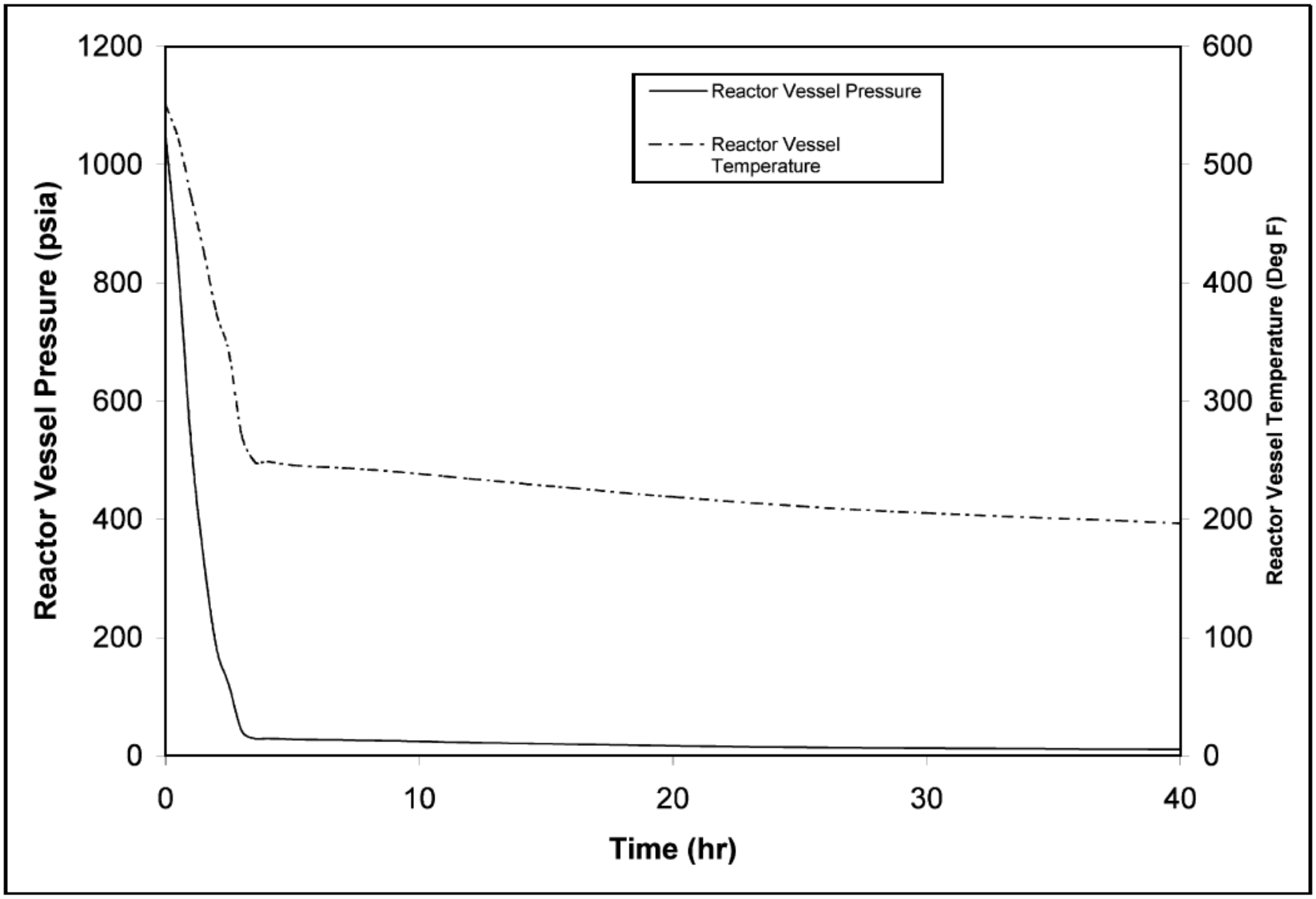
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LOSS OF FEEDWATER
FEEWATER FLOW, STEAM FLOW AND RCIC FLOW

FIGURE 15E.2.7-3, Rev 1

AutoCAD: Figure Fsar 15E_2_7_3.dwg



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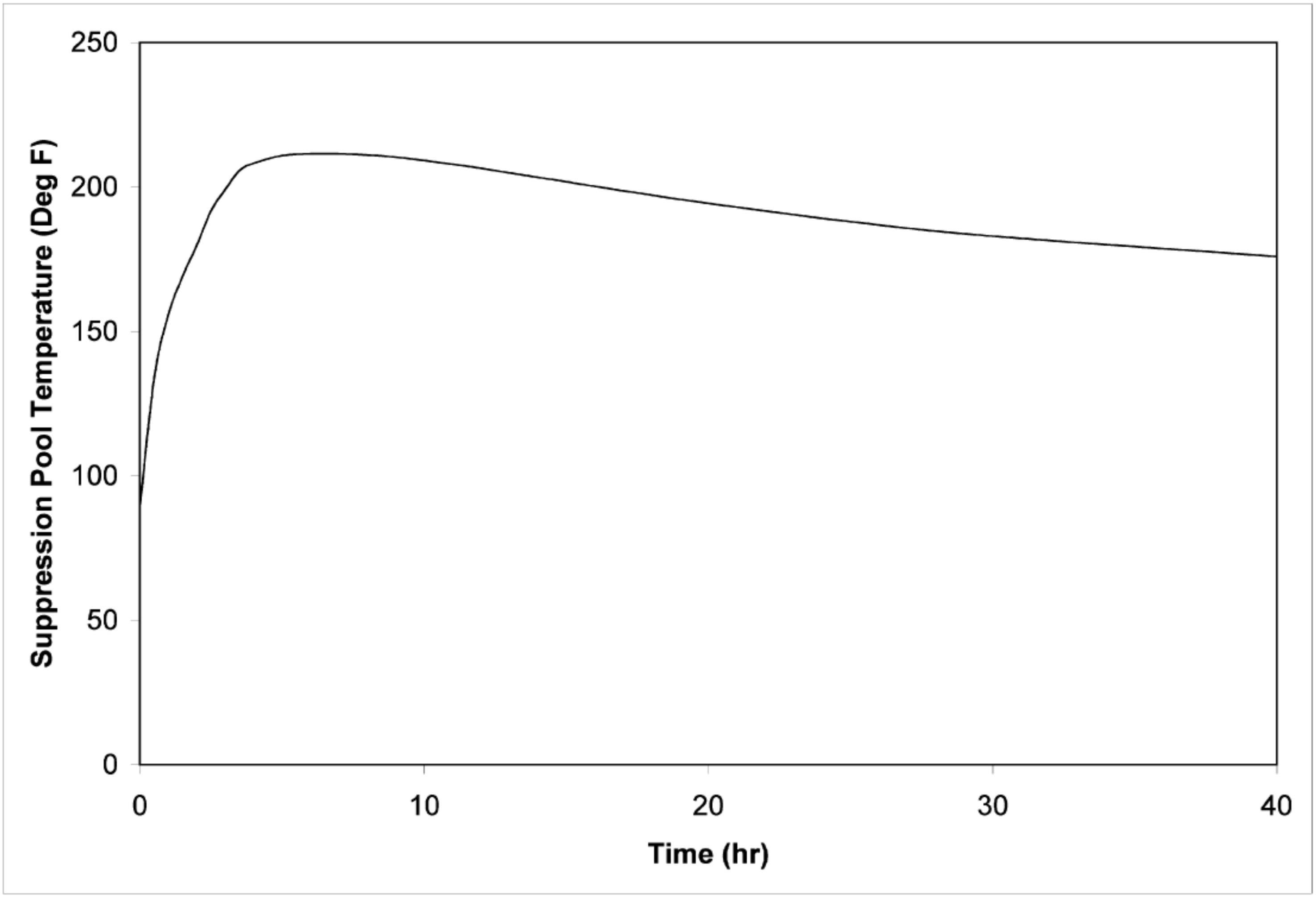
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

VESSEL TEMPERATURE AND
PRESSURE VERSUS TIME
(ACTIVITY C1 OR C2)

FIGURE 15E.2.9-3, Rev 55

AutoCAD: Figure Fsar 15E_2_9_3.dwg



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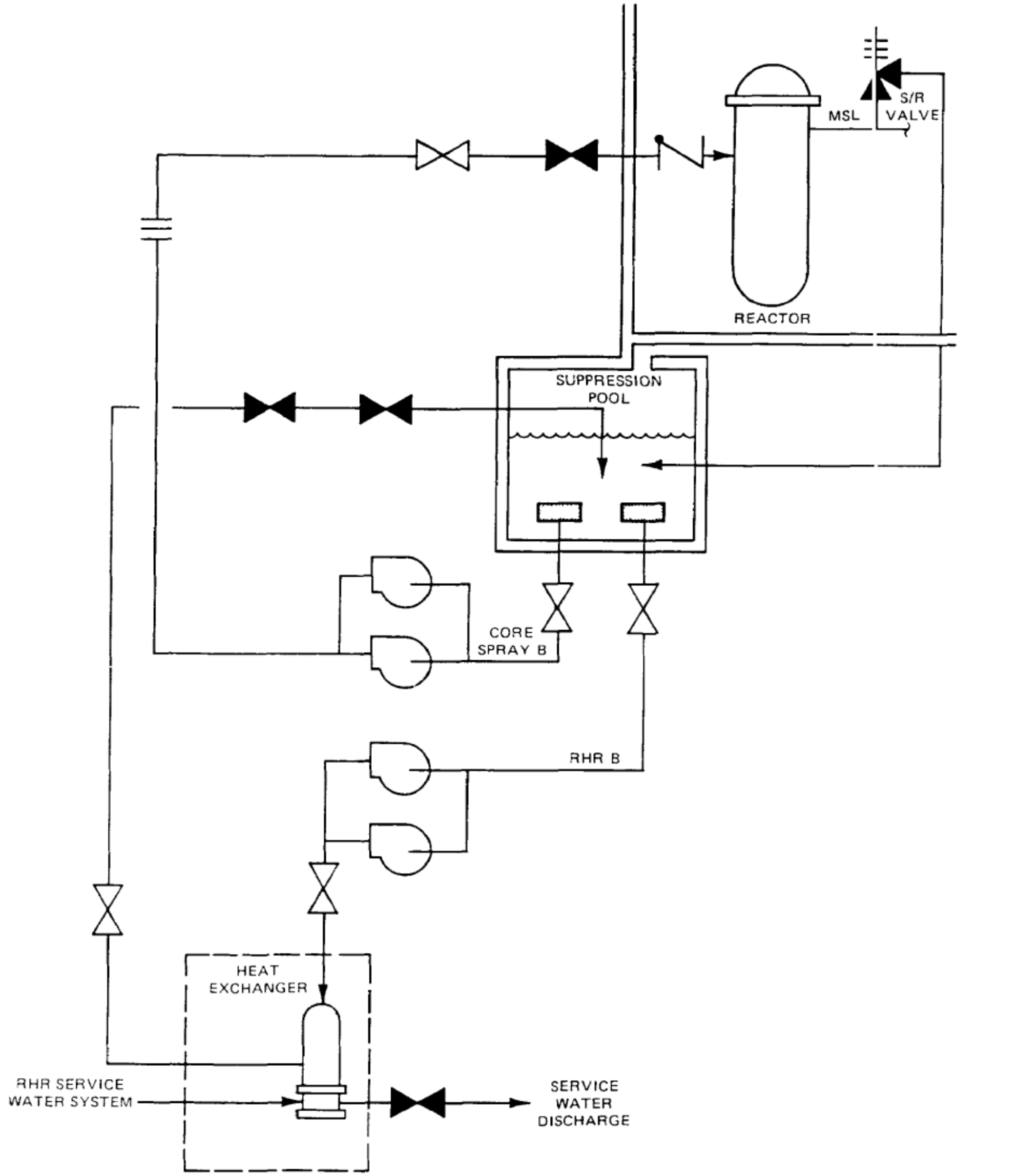
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION POOL TEMPERATURE
VERSUS TIME (90° SERVICE
WATER TEMPERATURE)
CAPACITY C1 OR C2

FIGURE 15E.2.9-4, Rev 55

AutoCAD: Figure Fsar 15E_2_9_4.dwg



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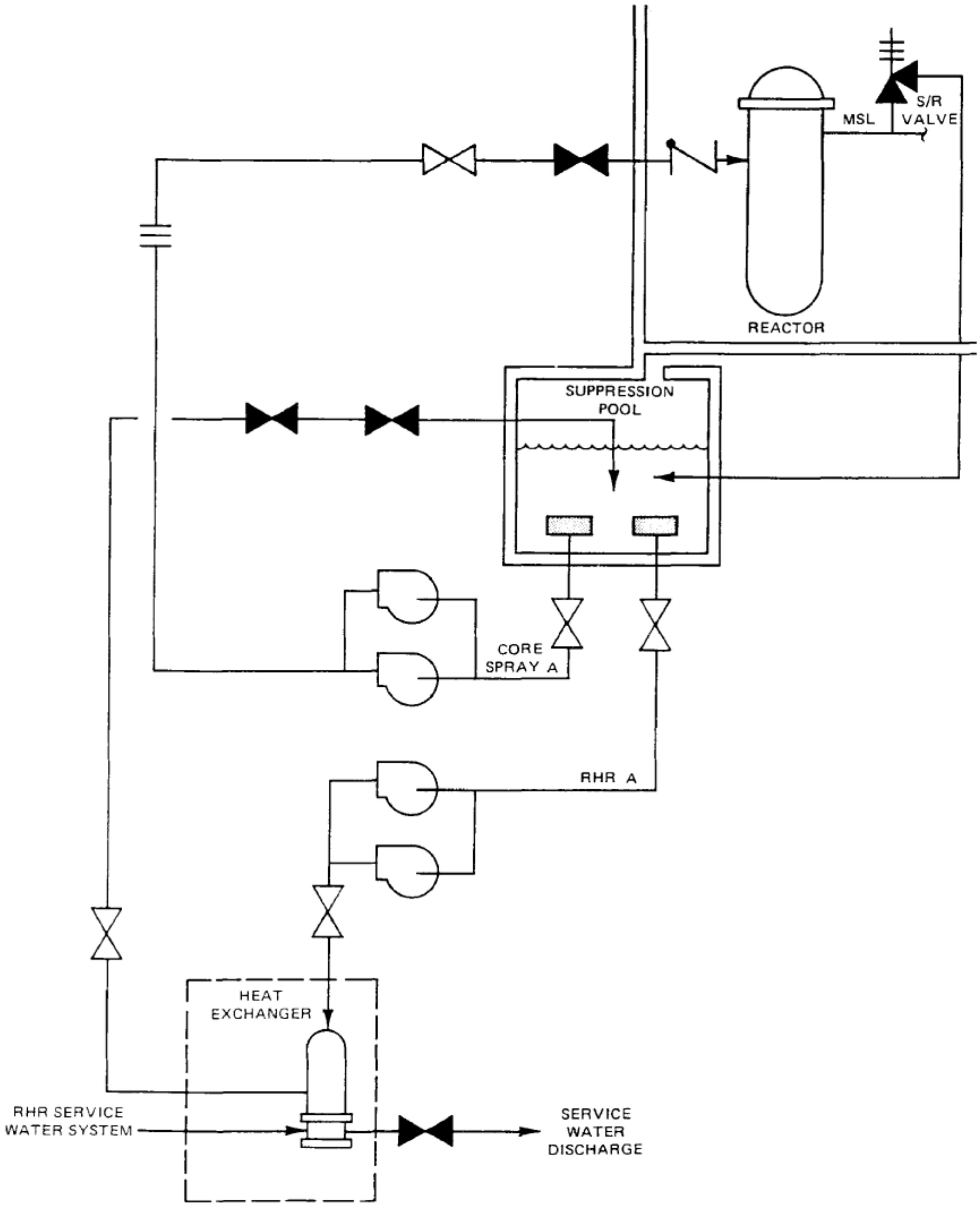
FINAL SAFETY ANALYSIS REPORT

ACTIVITY C1 ALTERNATE
SHUTDOWN COOLING PATH
UTILIZING RHR LOOP B

FIGURE 15E.2.9-5, Rev 54

AutoCAD: Figure Fsar 15E_2_9_5.dwg



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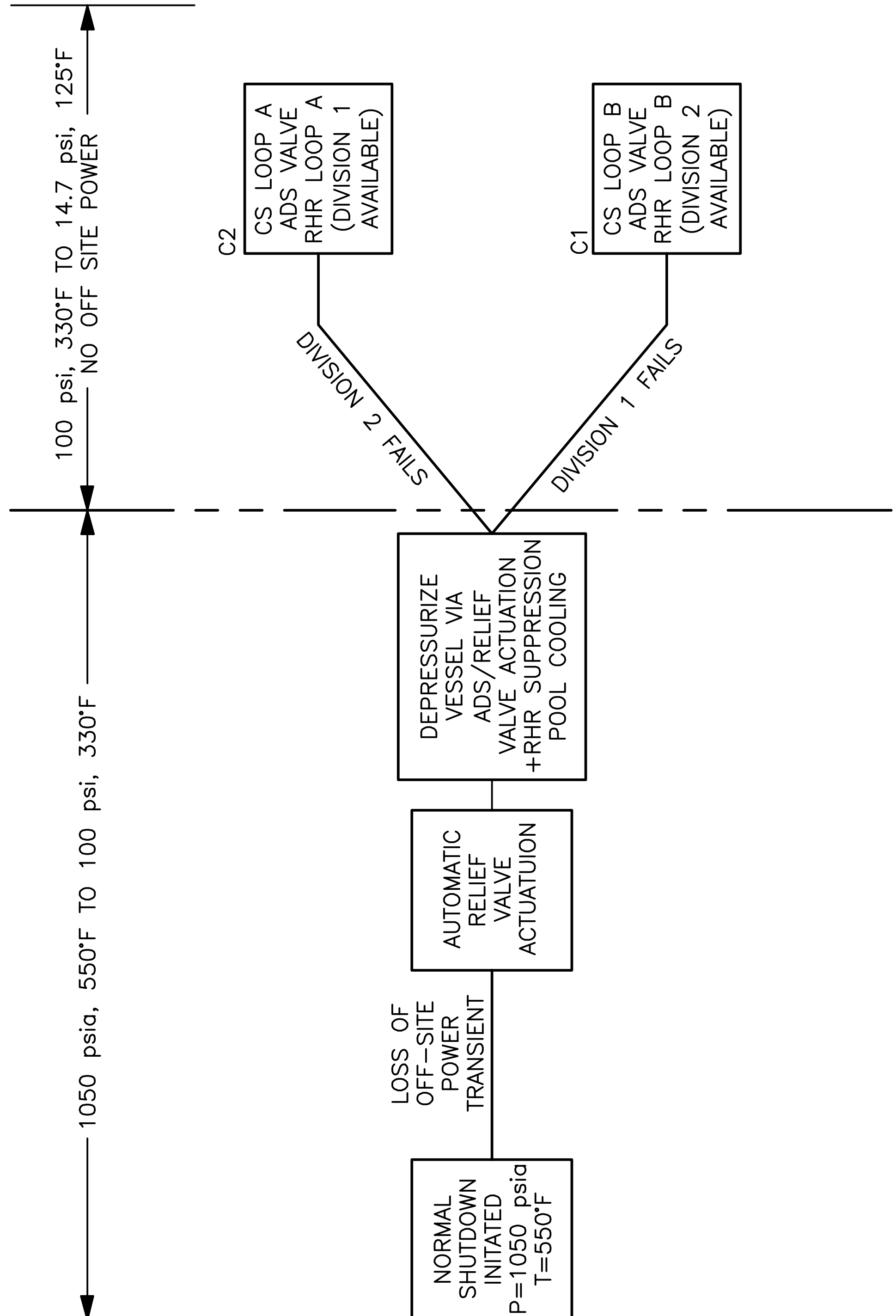
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ACTIVITY C2 ALTERNATE
SHUTDOWN COOLING PATH
UTILIZING RHR LOOP A

FIGURE 15E.2.9-6, Rev 54

AutoCAD: Figure Fsar 15E_2_9_6.dwg



See Figures 15E.2.9–1–2 & 15e.2.9–1–3 for notes.

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

ADS/RHR COOLING LOOPS

FIGURE 15E.2.9-1-1, Rev 55

AutoCAD: Figure Fsar 15E_2_9_1_1.dwg

NOTES FOR FIGURE 15E.2.9.1-1

ACTIVITY A

Initial pressure = 1050 psia
initial temperature = 550°F

For purposes of this analysis, the followinng worst-case conditions are assumed to exist.

(1) the reactor is assumed to be operating at 100% nuclear boiler rated steam flow;

(2) a loss of power transient occurs;

(3) a simultaneous loss of onsite power (Division 1 or Division 2), which eventually results in the operator not being able to open one of the RHR shutdown cooling line suction valves.

ACTIVITY B

Initial system pressure = 1050 psia
initial system temperature = 550°F

Operator Actions

During approximately the first 30 minutes, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCI and RCIC system automatic operation.

After approximately 10 minutes, it is assumed one RHR heat exchanger will be placed in the suppression pool cooling mode to remove decay heat. The operator will then initiate depressurization of the reactor vessel to control vessel pressure. Controlled depressurization procedures consisit of controlling vessel pressure and water level by using the ADS, RCIC and/or HPCI systems.

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode.

NOTES FOR FIGURE 15E.2.9.1–1

ACTIVITY C1

(Division 1 fails, Division 2 available) (Figure 15E.2.9-5)

System pressure - 100psi
System temperature - 330°F

Operator Actions

The operator establishes a closed cooling path as follows:

(1) One ADS valve (DC Division 2) is powered open;

(2) Water is pumped from the suppression pool into the reactor vessel. The cooled suppression pool water picks up decay heat and flows out of the vessel through the open ADS valves and back to the suppression pool as shown in Figure 15E.2.9-5. The RHR B loop is used to cool the suppression pool as required.

ACTIVITY C2

(Division 2 fails, Division 1available) (Figure 15E.2.9-6)

System pressure - 100psi
System temperature - 330°F

Operator Actions

The operator establishes a closed cooling path as follows:

(1) One ADS valve (DC Division 1) is powered open;

(2) Water is pumped from the suppression pool and into the reactor vessel as shown in Figure 15E.2.9-6. The cooled suppression pool water picks up decay heat and flows out of the vessel through the open ADS valves and back to the suppression pool. The RHR loop is used to cool the suppression pool as required. Cold shutdown (P=14.7psia. T RPV=200°F is reached in approxiamately 36 hours.)