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ABSTRACT

The automatic and manual operation of safety systems in a nuclear power plant (NPP) rely on cables and related equipment, such as electrical penetration assemblies (EPAs) and terminations, for power, control, and instrumentation signals. During an accident that creates a harsh environment, the cables need to perform reliably to operate the electrical equipment to meet the system's performance requirements. The accident performance capability of cables is addressed through environmental qualification (EQ) of electrical equipment. In probabilistic risk assessment (PRA) of NPPs**,** the cable reliability in a harsh environment is typically assumed to remain unaffected. In this report, we discuss a scoping study to assess the implications of cable failures under such conditions. Typically, in a PRA cables are not explicitly modeled**,** and the effect of their failure on hardware and operators' actions is not directly addressed. We use the existing PRA models to evaluate the impact of cable failures through the modeled component's and operators' actions. Several assumptions are made. Sensitivity analyses are conducted to assess the impact of failures of cables providing different functions inside the containment. The impact on core damage frequency (CDF), and the accident sequences that become dominant contributors when cables fail are studied. Relative CDF significance of cables providing different functions is obtained. A PWR plant, Surry Nuclear Power Station, and a BWR plant, Peach Bottom Atomic Station, are studied.

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1. INTRODUCTION

1.1 Background

Electrical equipment used to perform a safety function in a nuclear power plant must operate reliably under all service conditions, i.e., during normal operation, as well as under harsh environments caused by accidents which may occur during the equipment's installed life. The automatic and manual operation of safety systems rely on cables and related equipment, such as electrical penetration assemblies (EPAs) and terminations for power, control, and instrumentation signals. For brevity, the word "cable" will be used to include cable connectors, splices, and penetrations. The performance of electrical equipment on demand to meet the system performance requirements is addressed through environmental qualification (EQ) of electrical equipment.

To determine the performance capability of cables during an accident, NRC has conducted tests to assess the validity of the current qualification methods, among other objectives. This research was carried out at Brookhaven National Laboratory, and focused on low-voltage electrical cables used for instrumentation and control applications in nuclear power plants. The issues addressed are documented in NUREG/CR-6384 (Lofaro et al., April 1996), and NUREG/CR-6704, Vol. 1 and 2 (Lofaro et al., February 2001).

A risk analyses of the effects of cable failures provide an integrated perspective to related issues that can be addressed in a risk-informed decision-making process. The amount of cables in a nuclear power plant is substantial, and risk analyses of the cables can provide a risk-informed basis for NRC's decisions on cables for the operating life of a plant and also, for the renewal of licenses. The results of cable testing can be interpreted for their risk-significance. Specific accident sequences and the operators' actions affected by cable failures can be identified and ranked. Current and future needs for improved defenses can be evaluated and addressed. A scoping study was initiated to bring in risk perspective, and to identify the specific issues that can be addressed in a risk-informed decision-making process. This report presents the work completed under the scoping study.

1.2 Objective and Scope

The objective of this task is to carry out a scoping study of the effect of cable failures (including cables, connectors, splices, and penetrations) in harsh environment on plant risk for a pressurized- and a boiling-water-reactor (PWR and a BWR).

This objective of the risk analysis can be delineated into the tasks as follows:

1. Assess the risk impact of the cable failures due to a harsh environment using the PRA models of a PWR and a BWR,

2. Conduct sensitivity analyses to obtain a relative ranking of the different electrical cables in terms of their risk-significance,

3. Identify specific accident sequences that are affected by cable failures in a harsh environment,

4. Obtain risk-informed insights about the EQ of electrical cables.

The scope of the study is as follows:

1. The risk evaluation of the effect of cable failure is conducted using a Level 1 PRA model, i.e., using core damage frequency (CDF), as the measure. Accordingly, cable failures that impact release fractions and health risk but not the CDF are not considered in this study.

2. Available PRA models are used in the evaluation. Such models are limited in their inclusion of cables in a nuclear power plant. In fact, considering cable failures to be unlikely, they are either not included or considered to be included as part of the hardware failure of the component modeled in the PRA. No effort is made to change or improve the PRA model to incorporate cable failures. However, judgment is used to define surrogate components modeled in the PRA to represent them. In other words, current PRA models are used to the extent possible to approximate the effect of cable failure without changing the basic fault/event tree models therein.

3. Cable reliability in a harsh environment is difficult to estimate because of lack of data on their performance under such conditions. No effort is made to estimate cable reliability under a harsh environment but perspective of the experimental results on cables' performance (carried out under the NRC project at BNL) is used to define the sensitivity analyses conducted to obtain an understanding of the effect of cable failures.

4. Cable reliability, i.e., the cable response in an accident condition, also depends on the condition of the cable at the time of the accident considering the effect of the thermal environment on the cable. Because of the location of some of the cables, they may experience high temperature over the years and may be more vulnerable to failures compared to others experiencing less severe thermal environment, i.e., the effect of aging due to thermal environment can impact the cable's ability to function properly in an accident condition. Such evaluations are not conducted as part of this scoping study.

5. Since cables are not directly modeled in a PRA, the impact of a cable's failure is evaluated by assessing the components it may cause to fail and the operators' actions it may affect. In identifying them, only those that are modeled in the PRA are considered. Operators' actions that are not modeled in the PRA, but may potentially be affected are neither reviewed nor addressed.

6. The focus of the study being the cable performance under a harsh environment, this scoping study addressed accidents causing harsh environments inside the containment. Accidents outside the primary containment are not evaluated. Under this scope, the cables inside the containment are studied and not those outside.

7. Cables are associated with splices, penetrations, connections, and terminal boxes. Often, a failure of a cable may imply a failure within the cable system at one of these locations. Detailed drawings were not available to discriminate among these locations. The assumed failure of a cable implies failure at one of these locations within the respective cable system. Focus is on instrument and control cable failures since failure of power cables is considered less likely.

A PWR, Surry Nuclear Power Station Unit 1 and 2, and a BWR, Peach Bottom Atomic Station Unit 2, were studied to obtain a scoping evaluation of the effect of cable failures. The choice of these plants was based on the availability of the needed information and the availability of the PRA model in the SAPHIRE computer code, not because of any particular vulnerability to cable failures. The PRA developed under the Individual Plant Examination (IPE) program for the Surry plant and that developed under the NUREG 1150 program for the Peach Bottom plant, implemented in the SAPHIRE code, were used.

1.3 Organization of the Report

The report is organized in five chapters: Chapter 2 presents an analysis of vulnerability of plants to cable failures using the available IPE results. Chapter 3 discusses the cables that are expected to be exposed to harsh environments inside the containment during an accident. Chapter 4 presents the results of PRA-based evaluations. Chapter 5 summarizes the results and makes recommendations for further work to address the assumptions and issues. Appendix A provides the event trees to facilitate the reader's understanding of the accident sequences discussed in the main body of the report.

2. VULNERABILITY OF PLANTS TO CABLE FAILURES UNDER A HARSH ENVIRONMENT

Cables in a nuclear power plant may experience harsh environment both during routine daily operations and also in accident condition. Cables may experience harsh environment following an accident and the severity of the harsh condition may be such that it fails the cable. In addition, a cable may be in a location close to the vessel, e.g., cables associated with PORVs in a PWR or ADS in a BWR, and experience harsh thermal environment during power operation. These cables may be more vulnerable to failures when exposed to harsh environment.

A plant's vulnerability to cable failures due to harsh environment in a probabilistic risk assessment (PRA) may reveal in the following manner:

1. The contribution of accident sequences causing a harsh environment are high without consideration of cable failures and they could further increase when additional cable failures occur, and/or

2. The accident sequences causing a harsh environment have minimal contribution to the plant's core damage frequency, but can become a dominant contributors when cable failures under harsh environment are taken into consideration

To understand plant's vulnerable to cable failures, we conduct a scoping study to identify plants where the contribution of accident sequences causing a harsh environment are already dominant contributors. The frequency of these accident sequences can increase further due to the increased likelihood of cable failures under the harsh environment created by the accident. Additional plants may be identified where contribution of accident sequences affected by harsh environment is normally low, but will become dominant when cable failures due to harsh environment are taken into consideration. Such plants are not identified in the vulnerability assessment presented below in this chapter.

Accidents that create harsh environment can be broadly categorized into (a) in-containment, and (b) outside-containment ones. We focus on the in-containment accidents. Saltos (March, 1993) , Tzanos and Hanan (December, 1993), and Bustard et al. (January, 1989) have studied the accident sequences that are impacted by harsh environments. A review of their findings and a brief review of the PRA models for the plants being studied here were used to identify the accident sequences of concern.

For PWRs, the following are initiating events and the associated accident sequences that are of concern:

1. Large, medium, and small LOCAs inside the containment
2. Steam line break inside the containment
3. RCP seal failures causing LOCAs

4. Consequential LOCAs like RCP seal LOCA on station blackout, stuck-open PORVs, or stuck-open safety-relief valves.

Examples of accidents that cause a harsh environment outside the containment are interfacing system LOCA, and a steam line break outside containment.

For BWRs, the following are initiating events and accident sequences of concern:

1. Large, medium, and small LOCAs inside the containment
2. Small-small LOCA due to recirculation pump seal leaks
3. Transients involving loss of suppression pool cooling (when the power conversion system is not available).

Possible LOCAs within the mitigating systems are examples of accidents causing harsh environment outside the containment.

The cable failures under review are applicable for aged cables, 20 yrs or older. To identify plants with large CDF contributions for accident sequences causing harsh environment, we consider the following attributes:

1. The plant's age, i.e., plants which have been operating 20 or more years, and

2. Internal event core damage frequency (CDF) contribution of accidents causing a harsh environment, i.e., the plants where this contribution is greater than 1.0E-5 per year.

For PWRs, we consider the following initiating events: Large, medium, and small LOCAs inside the containment, and RCP-seal LOCAs (due to loss of CCW/SW and SBO-induced). For BWRs, we consider the large, medium, and small LOCAs inside the containment. Main steam line breaks and consequential LOCAs, like stuck-open PORVs or stuck-open relief valves in PWRs are usually small contributors to the plant's core damage frequency (CDF). In BWRs, transients involving loss of suppression pool cooling are relatively small contributors to the plant's CDF.

To assess the CDF contribution of accidents causing harsh environments inside the containment, we obtain the CDF contribution of the relevant accident sequences, as discussed above. The small contributors are not included in this analysis. The LOCA contributions are obtained from the IPE submittals using the IPE Database. The RCP-seal contributions are obtained from the NRC report on RCP seal LOCA modeling (USNRC, 1997). Extracting this contribution from some IPE submittals or the IPE Database can be difficult or time-consuming. Similarly, extracting the contribution of consequential LOCAs, like stuck-open PORVs for PWRs or of transients involving loss of suppression pool cooling for BWRs can be time-consuming. Tables 2.1 and 2.2 summarize the results for PWR and BWR plants respectively considering the CDF contribution of accidents causing harsh environment and the plant's age.

Among the 72 operating PWR units, 35 units are aged 20 or more years. Reviewing the CDF contribution of accidents causing harsh environment inside the containment, i.e., LOCAs including RCP-seal LOCAs, we note that for 29 of these units this contribution is greater than 1.0E-5/yr. Among these 28 units, 5 have a CDF contribution greater than 1.0E-4. In other words, about 39% of the operating PWRs have cables aged more than 20 yrs and have a sizable CDF contribution from accidents that cause harsh environment from cables. As discussed earlier, in estimating these CDF contributions, the cables are assumed to be qualified for LOCAs and their reliabilities in the accident conditions are assumed the same as those in normal operating conditions, i.e., the increased likelihood of failures under harsh conditions are not taken into consideration. If they were considered, these contribution are expected to be higher. Since age is a factor, with time, additional plant units will be added to the list.

Focusing on different types of PWR plants, the table provides the following breakdown where the LOCA CDF contribution (including RCP seal LOCAs) is greater than 1.0E-5:

- Five of the seven B&W units,
- Four of the six CE units,
- Six of the six W 2-LOOP units,
- Eight of the eight W 3-LOOP units,
- Four of the six W 4-LOOP units, and
- Two of the two W 4-LOOP Ice Condenser units.

Among the 34 operating BWR units, 19 units are 20 or more years old (2 BWR 2, 7 BWR 3, and 10 BWR 4). Operating BWR 5 and 6 units are less than 20 years old. The LOCA CDF contribution of all the 19 units is less than 1.0E-5/yr. However, as studied in Chapter 4 of this report, the CDF contribution of accident sequences causing harsh environment in a BWR can exceed 1.0E-05/yr when cable failures due to the harsh failures are taken into consideration.

In summary, to understand the impact of cable failures under harsh environment, we studied the contribution of LOCAs (including RCP-seal LOCAs) to the internal event CDF and plant age. Using a threshold value of 1.0E-5 for the CDF contribution of a plant to be judged vulnerable, we identify 28 units with ages 20 years or more. Additional plants can also have large CDF contributions due to cable failures if the impact of the harsh environment on cable reliability is very large, i.e., if because of the harsh environment impacting cable reliability many non-dominant contributors to the plant's CDF become a dominant contributor.

Table 2.1 CDF Contribution for Accidents Causing a Harsh Environment Inside Containment: PWR Plants (Age: 20 yrs or older)

Plant name	**Plant Age (yrs)**	**CDF (/yr)**	**LOCA CDF (/yr)**	**RCP Seal LOCA CDF (/yr)**	**LOCA (Incl. RCP Seal) CDF (/yr)**
B&W PWR 2 LOOP					
ANO 1	24	4.7E-5	1.6E-5 (34%)	4.6E-6 (10%)	2.1E-5 (44%)
Crystal River 3	21	1.5E-5	8.9E-6 (59%)	7.3E-7 (5%)	9.6E-6 (64%)
Davis Besse	28	6.6E-5	5.3E-6 (8%)	2.1E-5 (33%)	2.6E-5 (41%)
Oconee 1,2,3	25,24,24	2.3E-5	9.7E-6 (42%)	6.6E-6 (29%)	1.6E-5 (71%)
TMI 1	24	4.5E-5	1.6E-5 (35%)	9.8E-6 (22%)	2.6E-6 (57%)
CE PWR 2 LOOP					
Calvert Cliff 1,2	23,21	2.4E-4	6.7E-5 (28%)	2.6E-5 (11%)	9.3E-5 (39%)
Fort Calhoun	25	1.4E-5	1.1E-6 (8%)	2.2E-6 (16%)	3.3E-6 (24%)
St. Lucie 1	22	2.3E-5	1.2E-5 (53%)	1.3E-6 (5%)	1.3E-5 (58%)
Millstone 2	23	3.4E-5	6.1E-6 (18%)	Un-known	6.6E-6 (18%)

Plant name	**Plant Age (yrs)**	**CDF (/yr)**	**LOCA CDF (/yr)**	**RCP Seal LOCA CDF (/yr)**	**LOCA (Incl. RCP Seal) CDF (/yr)**
Palisades	27	5.1E-5	1.8E-5 (31%)	None (Screened out)	1.8E-5 (31%)
Westinghouse PWR 2-LOOP					
Ginna	28	8.7E-5	2.3E-5 (26%)	<5.1E-8 (<1%)	2.3E-5 (26%)
Kewaunee	24	6.7E-5	2.4E-5 (36%)	5.5E-6 (8%)	6.0E-5 (42%)
Point Beach 1,2	28,26	1.2E-4	4.0E-5 (33%)	9.2E-6 (8%)	4.9E-5 (41%)
Prairie Island 1,2	25,24	5.1E-5	1.2E-5 (24%)	9.5E-6 (19%)	2.1E-5 (43%)
Westinghouse PWR 3-LOOP					
Beaver Valley 1	22	2.1E-4	1.7E-5 (8%)	9.7E-5 (46%)	1.1E-4 (54%)
Farley 1	21	1.3E-4	2.5E-5 (19%)	6.3E-5 (48%)	8.8E-5 (67%)
Robinson 2	27	3.2E-4	7.4E-5 (23%)	6.7E-5 (21%)	1.4E-4 (44%)
North Anna 1	20	7.2E-5	2.1E-5 (29%)	7.7E-7 (1%)	2.2E-5 (30%)
Surry 1,2	26,25	1.3E-4	2.2E-5 (17%)	4.4E-6 (3%)	2.6E-5 (20%)
Turkey Point 3,4	26,25	3.7E-4	3.9E-5 (11%)	3.0E-4 (82%)	3.4E-4 (93%)

Plant name	Plant Age (yrs)	CDF (/yr)	LOCA CDF (/yr)	RCP Seal LOCA CDF (/yr)	LOCA (Incl. RCP Seal) CDF (/yr)
Westinghouse PWR 4 LOOP					
Haddam Neck	30	1.9E-4	6.1E-5 (32%)	5.3E-5 (28%)	1.1E-4 (60%)
Indian Point 2	24	3.1E-5	9.9E-6 (32%)	1.6E-6 (5%)	1.2E-5 (37%)
Indian Point 3	22	4.4E-5	8.8E-6 (20%)	2.7E-6 (6%)	1.2E-5 (26%)
Salem 1	21	5.2E-5	7.3E-6 (14%)	1.6E-5 (31%)	2.3E-5 (45%)
Zion 1,2	25,24	4.0E-6	1.8E-6 (45%)	~2.0E-7 (~5%)	2.0E-6 (50%)
Westinghouse PWR 4-LOOP Ice Condenser					
D.C.Cook 1,2	23,20	6.3E-5	3.5E-5 (56%)	1.2E-5 (19%)	4.7E-5 (75%)

Table 2.2 CDF Contribution for Accidents Causing Harsh Environment Inside Containment: BWR Plants (20 yrs or older)

Plant Name	**Plant Age (yrs)**	**CDF (/yr)**	**%LOCA Contribution**	**LOCA CDF (/yr)**
BWR 2				
Nine Mile Point 1	29	5.5E-6	12	6.6E-7
Oyster Creek	29	3.9E-6	6	2.5E-7
BWR 3				
Dresden 2,3	28,27	1.9E-5	8	1.4E-6
Millstone 1	27	1.1E-5	4	4.7E-7
Monticello	28	2.6E-5	5	1.2E-6
Pilgrim 1	26	5.8E-5	6	3.2E-6
Quad Cities 1,2	25,25	1.2E-6	18	2.1E-7
BWR 4				
Browns Ferry 2	23	4.8E-5	2	7.1E-7
Brunswick 1,2	21,23	2.7E-5	1	1.6E-7
Cooper	24	8.0E-5	10	7.8E-6
Duane Arnold	23	7.8E-6	2	1.9E-7
Fitzpatrick	23	1.9E-6	0.4	8.0E-9
Hatch 1	23	2.2E-5	13	3.0E-6
Peach Bottom 2,3	24,24	5.5E-6	11	6.3E-7
Vermont Yankee	26	4.8E-6	1	4.8E-8

3. CABLES AFFECTED BY A HARSH ENVIRONMENT

The concern about cables primarily relates to their reliability in a harsh environment, i.e., whether the cables will perform their intended function under such conditions. The cables that may be exposed to harsh environment are qualified for performance in a harsh environment, but with age and use, they may degrade increasing the likelihood of failures.

Cables can be exposed to harsh environments due to accidents that may occur in a plant. The accidents analyzed as part of the probabilistic risk assessment (PRA) models can be broadly categorized into two groups: a) in-containment accidents, and b) accidents outside the containment. The former include loss-of-coolant accidents (LOCAs), and main-steam-line break inside the containment; and high energy line break outside containment is an example of accidents outside the containment.

In this chapter, we identify the cables that may be exposed to harsh environment. Simply, for the in-containment accidents, the cables inside the containment can be exposed to harsh environment. Essentially, we focus on cables that are inside the containment. Since our interest is in conducting a PRA-based evaluation, we focus on cables that are included in the PRA model, i.e., the cables that support the actuation and operation of components modeled in the PRA. The operators' actions that depend on the instrument readings whose cables could be affected are also considered. A list of cables that may be exposed to harsh environment in an in-containment accident, and that are included in the PRA model is prepared for a PWR and a BWR plant.

3.1 Approach Used to Identify Cables Affected By Harsh Environment

The cables that may be affected by harsh environment are identified using the PRA models of a plant, the plant's system drawings, and the Regulatory Guide (RG) 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident. The RG lists the variables that are measured and indicated in the control room. In many cases, cables are likely used inside the containment in measuring the variable. Depending on the type of the accident, location of the break, and the location of the cables, the cables that are affected by the accident can be identified. All cables inside the containment are not necessarily exposed to harsh environment due to an accident. System drawings are used to judge the cables that are inside the containment and hence, may be exposed to harsh environment during an accident creating a harsh environment. Since our focus is on Level 1 PRA evaluations, we identify the cables that can affect theses evaluations. In other words, there are additional cables that are exposed to harsh environment and affect the risk of radioactive release and health consequences (Level 2 and 3 PRA evaluations), but they are not identified at this time.

The approach used to identify the cables that could be affected by harsh environment can be summarized as follows:

1. RG 1.97 is reviewed to identify the variables that are monitored and are potentially measured inside the containment. The variables that are incorporated in a typical Level 1

PRA analyses are selected. System and PRA models are reviewed, as described below, taking into consideration the variables identified.

2. The systems modeled in the PRA are reviewed to identify those portions that reside within the containment. Specifically, components within the system that are modeled in the PRA are reviewed to identify those that are within the containment. Within that list, those that depend on cables for operation or provide input for operation of other components are identified. Each of these components are associated with cables that may be affected by a harsh environment.

3. The instruments considered in the PRA model that provide actuation signals in accidents which create a harsh environment are considered to identify the associated cables that may be affected.

4. The operator errors modeled in the PRA are reviewed to identify those that depend on instrument readings. The cables that support the instrument readings and located within the containment are identified.

5. Since detailed cable drawings were not available, the drawings and information in the PRAs completed for the plant are used. For example, for the Surry plant, the information in the Surry Individual Plant Examination (VEPCO,1991), NUREG/CR-4550 study (Bertucio, 1990), and the WASH 1400 report (USNRC,1975) is used.

PRAs and system drawings within the PRA are not necessarily good information sources for identifying cables that can be affected by a harsh environment. Beyond the scoping study, cable drawings for the plants will be preferable for identifying cables that may be affected by a harsh environment.

3.2 Assumptions in Identifying the Cables Affected By Harsh Environment

As stated above, cable drawings or layouts in a plant are not used to identify the cables. Such drawings are currently not available and they are somewhat resource-consuming to review. The approach used involves assumptions and can be summarized as follows:

1. The cables that can be identified through the hardware and human errors modeled in Level 1 PRA are included; other cables are not reviewed. A detailed review of the cables in the containment might reveal some common links or interactions that may affect multiple components. However, such interactions can not be identified without detailed cable layout drawings.

2. PRA level 2 and 3 analyses will require including additional cables related to containment systems.

3. Judgments are used to identify the types of cables that are involved. Typically, the motor-operated pumps and valves are associated with power, control, and instrumentation cables. Solenoid valves are associated with control and instrumentation cables; pressure and level transmitters are associated with instrumentation cables. We primarily focus on the instrument and control cables, and not the power cables. Power cables are considered less vulnerable to harsh environment compared to instrument or control cables.

4. Judgments are made about the location of the components and the associated cables. Details of terminal boxes, penetrations, and splices were not available and are not provided.

5. Individual components are assumed to have separate cables for power, control, and instrumentation of the components. Redundancy of the safety-system components is not affected by failure of any single cable. In our evaluation, harsh environment may fail multiple cables failing redundant components.

6. Operators' errors modeled in the PRA are reviewed to identify the instruments needed by the operators to carry out the required actions. Some additional instruments and the associated cables are identified through this process. This implies that in the PRA model, failure of those cables will affect the risk measure, e.g., CDF, through its impact on the operators' actions.

7. Reactor trip function is assumed to take place very early before the onset of harsh environment. The instrumentation associated with the trip function was not focused upon.

8. Regulatory Guide 1.97 provides classification of the cables used in NPPs. No attempt was made to include in our list all the cables included in the guide. This is because of lack of information in relating the cables to the equipment modeled in the PRA. Also, the classification provided in RG 1.97 is not used to assess their vulnerability to harsh environment.

3.3 Identified Cables for a PWR and a BWR Plant

Tables 3.1 and 3.2 list the cables identified, respectively, for a PWR, Surry Nuclear Power Station, Unit 1 and a BWR, Peach Bottom Station, Unit 2. The specific cables are identified to the extent they can be determined from the sources reviewed. The identification of the type of cable, power, control, or instrumentation is based on the types of component with which it is associated. Comments are included on the impact of cable failure, its modeling in the PRA and other relevant information for conducting our scoping PRA evaluation.

For a PWR, a significant number and types of cables are inside the containment and may be exposed to a harsh environment following an accident. Many of these cables are involved in providing the automatic actuation signal for the safety systems. However, this list is plant-specific and there may be important variations from one plant to another. For example, the MOVs required to be open for hot leg recirculation are located outside the containment for the

Surry plant, but inside the containment for the Sequoyah plant, another W plant. For the Surry plant, the RHR pumps are located inside the containment, but that may not be the case for other plants. The location of many of the cables associated with pressure and instrumentation signals is expected to be similar for most plants.

For a BWR, the limited number of cables that can impact the Level 1 PRA analysis is inside the containment. Reactor water level, drywell pressure, reactor level, reactor pressure, suppression pool level are measured outside the containment and there are no inside cables associated with these measurements. (This is true for Peach Bottom, and may be valid for other BWRs, in general. However, other BWRs have not been reviewed at this time.) Additional cables that are inside the containment relate to sump leak detection system, loose part monitoring, temperature and vibration of reactor recirculation pumps, for example, which are not included in a Level 1 PRA analyses. Plant-specific differences may exist, but were not investigated.

Table 3.1 List of Equipment and Associated Cables Impacting Level 1 PRA Analyses That Can Be Affected by a Harsh Environment in a PWR (Surry Plant)

System/Equipment	Associated Cables	Comments
AFW System		
1. MOV 151 A,B,C,D,F,	Power, control, and instrument cable	Normally open; expected to fail in open position
Feedwater System		
1. Maintenance Isolation valve: MOV 154 A,B,C	Power, control, and instrument cable	Normally open: expected to fail in open position
2. Feedwater regulating valve: FCV 1478, 1488, 1498	Control and instrument cable	Receives input signal from SG Water Level Control; Control by MCR bench board
Containment Spray Recirculation System		
1. Pump RSP1A	Power, control, and instrument cable	
2. Pump RSP1B	Power, control, and instrument cable	
Consequence Limiting Control System		
1. Pressure Transmitter PTLM100A	Instrument Cable for Containment pressure Channel 1-LM-PT-100A	Cables for PT 100A and 100B use same penetration.
2. Pressure Transmitter PTLM100B	Instrument Cable for Containment Pressure Channel 1-LM-PT-100B	
3. Pressure Transmitter PTLM100C	Instrument Cable for Containment Pressure Channel 1-LM-PT-100C	Cables for PT100C and 100D use same penetration
4. Pressure Transmitter PTLM100D	Instrument Cable for Containment Pressure Channel 1-LM-PT-100D	
Accumulator		

System/Equipment	Associated Cables	Comments
1. MOV 1865A,B	Power, control, and instrument cable	normally open, expected to fail in open position
2. Pressure Transmitter PT 1927	Instrument cable	
3.Pressure Transmitter PT-1925	”	
4. Level Transmitter LT-1924	”	
5. Level Transmitter LT-1925	”	
Residual Heat Removal System		
1. Pump P1A	Power, control, and instrument cable	
2. Pump P1B	”	
3. MOV 1720 A, B	”	Normally open; expected to fail in open position
Pressurizer		
1. Pressure Channel 1-RC-PC-455	Instrument Cable	
2. Pressure Channel 1-RC-PC-456	”	
3. Pressure Channel 1-RC-PC-457	”	

System/Equipment	Associated Cables	Comments
Main Steam		
1. Pressure Channel 1-MS-PC-474A	Instrument Cable	
2. Pressure Channel 1-MS-PC-485A	”	
3. Pressure Channel 1-MS-PC-496A	”	
4. Differential Pressure Channel 1-MS-PC-474	”	Thermal degradation of the transmitter electronics
5. Differential Pressure Channel 1-MS-PC-475	”	”
6. Differential Pressure Channel 1-MS-PC-476	”	”
7.Differential Pressure Channel 1-MS-PC-484	”	”
8.Differential Pressure Channel 1-MS-PC-485	”	”
9.Differential Pressure Channel 1-MS-PC-486	”	”
10. Differential Pressure Channel 1-MS-PC-494	”	”
11.Differential Pressure Channel 1-MS-PC-495	”	”
12. Differential Pressure Channel 1-MS-PC-496	”	”
13. SG Train A Level Transmitter	Instrument Cable	Used in actuation signal and Op Action
14. SG Train B Level Transmitter	”	”
15. SG Train C Level Transmitter	”	”

System/Equipment	**Associated Cables**	**Comments**
PORV and Block Valves		
1. Block Valves 1535,1536	Power and control cables	dependent on the AC power buses; both fail open and fail closed conditions are applicable
2. PORV 1455C and 1456	Control cable	solenoid operated valve; dependent on DC power for control;
Reactor Coolant System		
1. Cold Leg Temperature Channel 412E	Instrument Cable	
2. Cold Leg temperature Channel 422E	”	
3. Cold Leg Temperature Channel 432E	”	
4. RCS Pressure	”	Identified through Op. Action
5. RCS Hot Leg Temperature	”	”
Instrumentation for Operator Actions		
1. Rod Bottom lights	Instrumentation cables	
2. Rod position indicators	”	
3. Reactor trip and bypass breakers	”	
4. Neutron flux	”	
5. Pressurizer level	”	

Table 3.2 List of Equipment and Associated Cables Impacting PRA level 1 Analyses That can Be Affected by Harsh Environment for a BWR (Peach Bottom Nuclear Power Plant)

System/Equipment	**Type of Cable**	**Comments**
HPCIS and LPCIS		
1. MOV 43A and B	Power, control, and instrument cable	normally open; expected to fail in open position
2. MOV 53A and B	”	”
3. MOV 65 A and B	”	normally closed; expected to fail in closed position
4. MOV 18	”	”
5. Pump 34A	”	
6. Pump 34B	”	
RCIC		
1. MOV 15	Power, instrumentation, and control cables	normally open; expected to fail in open position
2. MOV 29B	”	”
Safety Actuation Instrumentation		
1. Temperature Sensor 101 A,B,C,D	”	Steam Line Break (High temp.)
2. Temperature Sensor 102 A,B,C,D	”	”
3. Temperature Sensor 103 A,B,C,D	”	”
4. Temperature Sensor 104 A,B,C,D	”	”

System/Equipment	Type of Cable	Comments
ADS		
Relief Valves 71A and 72B (10 such pairs)	control cable	
Main Steam Isolation Valve	Control and instrumentation cables	
Suppression Pool		
Temperature wells	Instrumentation cables	

4. CDF AND ACCIDENT SEQUENCE IMPACT OF CABLE FAILURES IN A PWR AND A BWR

The impact of cable failures in a PWR and a BWR is assessed in terms of the impact on core damage frequency (CDF) and the affected accident sequences. Accident sequences are identified that may become dominant contributors to the plant's CDF when cables fail in a harsh environment. In this chapter, we summarize the approach used in the scoping evaluation, the assumptions involved, the results of the sensitivity evaluations, and the insights from the results.

4.1 Approach and Assumptions in the Scoping Evaluations

The scoping evaluation is conducted for the Surry Nuclear Power Station, Unit 1, a Westinghouse 3- Loop PWR, and Peach Bottom Station, Unit 2, a General Electric BWR-4. For the Surry plant, we use the latest version of the Individual Plant Examination (IPE) model implemented in the SAPHIRE code; for the Peach Bottom Station, the NUREG-1150 model implemented in the SAPHIRE code is used. The Level 1 internal event model is used to calculate the CDF contributions. The following summarizes the elements of the approach:

1. Data are not available to directly estimate the likelihood of cable failures under a harsh environment. For the scoping evaluation, the CDF impacts are obtained assuming that the cables fail in a harsh environment. No attempt was made to develop models that can estimate the likelihood of cable failures based on experimental evidence.

2. The cables identified to be affected by the harsh environment, based on their location inside the containment, are considered in defining the failed cables in the analysis. Details of splices, penetrations, and terminal boxes are not considered.

3. The failures of cables are assumed to cause the corresponding component to fail, e.g., the failure of cables associated with the containment pressure channel will provide an erroneous reading of the containment's pressure. The failure of a control cable for a pump inside the containment will result in the failure of the pump to perform its intended function. The impact of such failures are then assessed using the PRA model in determining the resultant CDF.

4. The failures of cables are not directly modeled in a PRA, i.e., in PRA terminology, they are not separate basic events in the model; they are assumed to be part of the corresponding component which is modeled. In practice, considering that cable failures are unlikely, such contributions to the component's failures have been neglected. In calculating the CDF impact of cable failures, the corresponding surrogate component was identified, and its failure was used to represent the failure of the cable. In representing the failures of multiple cables, corresponding multiple components and the common-cause failure events of the components are identified.

5. Cable failures also affect the operator’s actions in a PRA model. The operator’s actions modeled in the PRA are reviewed, and the probability of the failure of the operator to take the appropriate action is changed to reflect the impact of cable failures. Each of the operator’s actions modeled is reviewed to identify the corresponding indication/readings that are used in deciding/conducting the actions. In this approach, for each cable failure a set of corresponding operator’s actions are identified. In assessing the impact of the cable failures, the probability of this set of operator’s actions, along with the hardware failures are changed . For failure of multiple cables, similarly, a set of operator’s actions are identified. The impact of cable failures on the HEP is uncertain. In many cases, an operator action uses information from multiple indications which depend on different cables. Typically, information available to the operator becomes limited due to the cable failure. In this scoping study, no attempt is made to estimate the change in the human error probability associated with the operator action. Instead, sensitivity analyses are conducted by postulating that the probabilities are increased to 0.1.

6. Some operator’s actions that in a PRA are assumed to be executed very reliably may be affected by the loss of indications in the control room due to cable failures. No review of the operating procedures were undertaken to identify such actions, additional to those modeled in the PRA, that may be adversely affected by the cable failures.

7. The impact of cable failures are assessed by considering failures of different groups of cables. It is recognized that the zone of influence of the harsh environment caused by a particular accident is different from another, and the set of cables affected will vary depending on the harshness of the accident. Resources and information were not available to assess the set of cables that may be affected in a particular accident. Different groups of cable failures are chosen, based on judgment, to represent the impact of selective sets of cable failures. Also, groupings are made to represent failures of specific types of cables. Examples of groups of cable failure analyzed are cables associated with containment-pressure indication, cables associated with pressurizer pressure, and cables for recirculation spray pumps. Failures of combinations of cables involving control and instrumentation cables are possible, but are not analyzed at this time.

8. The progression in time of accidents causing a harsh environment can have an important influence on the CDF impact. These times are (a) the time needed for the harsh environment to develop in an accident, and (b) the time within which the components exposed to the harsh environment may need to operate, or the time needed to complete the operator actions that depend on instrument readings which may be affected by harsh environment. If the necessary actions can be completed before the harsh environment has developed enough to damage the cables, i.e., the information needed for the operator to act is unaffected, then the error probability remains the same as that in the PRA and the impact is negligible. In this scoping study, it is considered that a reactor trip will be unaffected. Otherwise, timing is not analyzed.

9. The analysis is focused on accidents causing a harsh environment inside the containment. Such accidents outside the containment are not considered. The accidents that cause a harsh environment inside the containment are defined based on engineering judgment and available thermal-hydraulic analyses (Bustard et al., 1989). No specific analyses are conducted to define these accidents. Judgment on some accident sequences is difficult, particularly about the time available before the harsh conditions exist.

4.2 CDF and Accident Sequence Impact of Cable Failures in a PWR

The CDF and accident sequence impact of cable failures are assessed for the Surry Nuclear Power Station, Unit 1 using the Surry IPE model implemented in the SAPHIRE code.

As discussed earlier, to obtain a scoping assessment of the impact of the cable failures in a harsh environment a bounding evaluation is made considering failure of different combinations of cables that are affected. The cable failures analyzed consist of the following:

1. Failure of instrument cables, and

2. Failure of control cables.

As stated earlier, in each case the cables involved are those inside the containment, i.e., those that can be affected by harsh environment and those that are modeled in some manner in the PRA.

In defining different combinations of cable failures, failures of cables associated with a particular type of measurement or a particular area where the cables may be located are focused upon. The selection of different cable failures is based on judgment, and not on any specific analysis of harsh environment following an accident. For the instrument cables, failures of containment pressure channels, pressurizer pressure channels, steam generator level transmitters, main steam pressure and differential pressure channels, and RCS cold leg temperature channels are analyzed. Each type of instrument cable is analyzed separately. For the control cables, those associated with PORV and block valves, residual heat removal (RHR) pumps, and recirculation spray (RS) pumps are analyzed.

For each case, two evaluations are made. First, only the hardware failure is addressed, assuming that in spite of the cable failures the operator's actions are unaffected, i.e., the operator will perform as reliably as assumed in the PRA where cable failures are assumed not likely. This scenario can be considered unlikely but it provides an insight for establishing a range on the impact of the cable failures. Second, the probability of the operator's actions being affected by the cable failures is increased to 0.1. Here, the impact on the operator actions is considered as limited only to those actions which require the instrument readings and control operations that are affected. The impact of cable failures on operator actions is uncertain. Here, the probability of operator's action is increased to 0.1 for a sensitivity analyses considering that other indications are available and the operator will rely on them. The error probability is increased because of the limitation on the available information due to the cable failure.

Accident Sequences Causing Harsh Environment

The accident sequences that are considered to cause a harsh environment inside the containment and evaluated in obtaining the impact on CDF are the following:

Large, intermediate, and small LOCA inside the containment
Transients with stuck-open power-operated relief valve (PORV)
RCP seal LOCA
Main steam line break inside the containment.

The contribution of anticipated transients without scram (ATWS) events are considered negligible. The CDF impact of cable failures is mainly dominated by small and intermediate LOCAs, as can be observed from the results presented below.

Analysis of the Results

Table 4.1 shows the results of the CDF impact of cable failures for different instrument cables inside the containment, with both the increased CDF when the failures are assumed and the increase in the CDF from the basecase. In each case, failure of all redundant cables is postulated. For example, failure of the containment's pressure channels signifies failure of all 4 of them. For each case, sensitivity analyses are presented for the probability of failure of the affected operator's actions. Table 4.2 lists the failures assumed in each case and the affected operator's actions.

These results show the largest impacts for the failure of the cables associated with the containment pressure channels. The CDF increases by a factor 2 when the operator's actions are unaffected and by approximately a factor 30 when the operator actions that depend on the containment pressure reading are increased from their basecase values to 0.1. The CDF of the failure of SG level transmitters is small. The impact of failure of the pressurizer pressure channels, main steam pressure channels, and RCS temperature measurements are negligible.

Since the impact of the failure of the containment pressure channels is larger than the others, this failure is analyzed further. We first discuss the role of these measurements in the Surry plant.

The containment pressure channels support the consequence limiting control system (CLCS) and initiates operation of equipment designed to control the containment environment when specific pressure levels are exceeded. The CLCS is made up of four measurement channels and four logic trains. The measurement channels each contain independent transducers to sense the containment pressure and the circuitry required to detect the two pressure levels. A rise in containment pressure to 1.5 psig produces signals which initiate the HI containment pressure phase of the CLCS. In this phase, the containment vacuum pumps are tripped, certain containment isolation valves are closed, and back-up signals are sent to the safety injection and control system (SICS). A further rise in the containment pressure to 10.3 psig produces a signal which initiates the HI HI containment pressure phase of the CLCS. The HI HI phase initiates the containment spray injection system and the containment spray recirculation system, closes the remaining containment isolation valves, and initiates operation of some other safety system components. Each of the four logic trains trips when three of the four measurement channels sense a trip pressure. The operator can also initiate either of the two sets of trains manually.

The Surry recirculation spray (RS) system is composed of four independent, 100% capacity Recirculation Spray trains. Two trains are located entirely inside containment while two trains have the pumps located outside the containment. The spray pumps draw water from the Containment Sump. Both inside and outside pumps draw from the same sump, although the sump is compartmentalized. As stated above, the system automatically starts on the a HI HI containment pressure signal from the CLCS.

To analyze further the impact of failure of cables associated with the containment pressure channels, we address different combinations of cable failures relating to containment pressure measurement (Table 4.3). The associated cables use two penetrations of the containment where in each the cables associated with two of the pressure transducers are routed. In addition to failure of all four channels, we analyze failure of two of the channels and a single channel. Since each logic train trips when three of the four channels sense a trip pressure, failure of two of the channels is sufficient to cause complete failure of the signals. Accordingly, we observe that the CDF impacts of the failure of two and four of the containment pressure channel measurements are the same. The impact of failure of only the cable associated with one of the pressure measurement channel is substantially smaller. Table 4.4 provides the breakdown of the contribution of different sizes of LOCAs for failure of 2 or more of the containment pressure channels.

Accident Sequences Impacted by Failure of Instrument Cables

To understand the accident sequences, we analyze those that are affected by the failure of the instrument cables. We focus on the failure of the containment pressure measurement cables as they dominate the CDF impact. A brief explanation is given below of the accident sequences that dominate this case and how they are affected by the cable failures. The event trees are presented in Appendix A.

S2-02 (S2Rs)

This is sequence 2 of the small LOCA (S2) event tree. This sequence is initiated by a small LOCA with all systems available except the recirculation sprays. When the containment pressure channels fail, the signal to start the containment recirculation spray pumps is not generated which causes the recirculation spray (RS) to fail. In the Surry IPE, the small LOCA frequency “S2" includes the very small LOCA frequency used in NUREG/CR- 4550. The small LOCA frequency considers LOCAs as small as 3/8". Hence, the frequency of the “S2" accident sequence is approximately 20 times higher than the frequency used in NUREG/CR-4550. In the IPE it was established that containment building heat removal is required after a small LOCA, as heat removal through the Steam Generator (SG) is not adequate when the temperature in the SG is below the maximum operating temperature of the Low-Head system. NUREG/CR-4550 modeled higher design temperatures for the Low-Head Safety Injection (LHSI) pumps, and thus postulated heat removal could be adequate through the SG in a post-LOCA environment, without any need for containment heat removal. The differences between IPE and NUREG/CR-4550 can be attributed to component modifications in the LHSI system and different thermal-hydraulic calculations for post-LOCA conditions (Surry IPE, pg. 1-8, 1991). The recirculation spray system provides for long-term containment pressure reduction and containment heat removal following an accident by drawing water from the containment sump and spraying the water into the containment atmosphere. Heat is removed from the sump water through the Service water cooled heat exchanger. The time available for the operator action is expected to be greater than 8 hours. Surry IPE calculated that for large LOCA the time available is greater than 1 hour.

S2-05 (S2ORs)

This sequence is similar to the previous one except that it includes the operator's failure to cooldown and depressurize (O). The successful completion of this operation before emptying the RWST enables the low head recirculation to be used instead of the high head recirculation. The probability of failure of operator action is 0.055. This makes this sequence frequency approximately a factor of 18 lower than the previous one. The time available for the operator action is approximately 2 hours.

S1-05 (S1ORs)

This sequence is similar to the previous sequence where the initiating event is an intermediate LOCA. In this case, the probability that the operator fails to cooldown and depressurize is assigned 1.0 (in the IPE) because of the short time available (approximately 1 hour).

A-04 (ARs) and A-08 (ACsRs)

These two sequences relate to the failure of containment spray (CS) and recirculation spray pumps in case of a large LOCA. Their failure is due to the loss of containment pressure signals resulting from the corresponding cables' failure. The time available for the operator action is approximately 1 hour.

ailure of the Control Cables

The failure of the control cables associated with the components inside the containment are presented in Table 4.4. The failure of the control cable is assumed to cause a loss of the control of the component. For the Surry plant, the control cables associated with the inside recirculation spray (RS) pumps, PORV and block valves, and residual heat removal (RHR) pumps are analyzed. In addition, failure of normally-closed MOVs in the hot leg recirculation to open due to cable failures is analyzed. In the Surry plant, these MOVs are outside the containment and are not expected to be exposed to harsh environment, but, in some plants, these valves can be inside the containment and may fail in a harsh environment.

For the RS and RHR pumps, failure of both redundant pumps is analyzed. The CDF of failure of RS pumps is higher compared to others; PORV failed closed affects the operator's ability to feed and bleed and the failure of the RHR pumps influences the Steam Generator Tube Rupture sequences which do not create a harsh environment inside the containment. Compared to the instrument cable failures, the impact of failure of the control cables on operator's actions is marginal.

The failure of the MOVs in the hot leg recirculation can have a large impact on the CDF. But, as noted earlier, in the Surry plant, they are not expected to be affected by a harsh environment. Plant-specific cases where these valves are inside the containment may need to be addressed to understand the specific implications.

4.3 Summary of Results From the Analysis of a PWR Plant

The scoping analysis of cable failures due to harsh environments for the Surry PWR plant is conducted assessing the CDF impact of the failure of cables located inside the containment. Cables providing different functions are separately analyzed. For situations where there are noticeable CDF impacts, specific affected accident sequences are identified to understand the otherwise non-dominant accident sequences that may become dominant when cables fail.

The analysis of the cable failures provides a perspective on the relative ranking of the cables in terms of their risk significance. The screening analysis, within its assumptions, shows the relative CDF-significance of instrument cables associated with the containment pressure measurement, and control cables associated with the inside containment recirculation pumps. The cables associated with many other measurements, e.g., pressurizer pressure, main steam pressure, SG level detection, RCS cold leg temperature, by themselves, are not observed to be risk-significant. The control cables associated with the RHR pumps also have no impact for accident sequences causing harsh environment inside the containment. In a general categorization, the results can be interpreted to show that the instrument cables, as defined in the report, have a higher risk significance than the control cables associated with the components inside the containment.

The scoping analysis also identified the accident sequences that become dominant contributors in cases of cable failures in a harsh environment. These sequences are primarily LOCAs (large, intermediate, small) followed by failure of the recirculation spray system. In case of small LOCAs, the time available for the operator action to initiate recirculation spray is longer compared to that in large LOCAs (greater than 8 hours compared to 1 hour) and the probability of the operator's action to fail to initiate recirculation spray will be lower. Such differences are not taken into consideration in this scoping study.

The results presented here are plant-specific. The components and cables that are located inside the containment and affected by harsh environment vary from plant to plant. Some aspects of the Surry design are unique, e.g., failure of recirculation spray system in case of a LOCA causing core damage, location of RHR pumps inside the containment, location of normally closed hot leg recirculation MOVs outside the containment, and have contributed to the results discussed here.

Table 4.1 Internal Event CDF Impact of Failure of Instrumentation Cables in a PWR Plant: Surry Nuclear Power Station

Case	CDF(/yr) and Increase in CDF (/yr)[2]			
	Operator Actions Unaffected		Affected Operator Action Error Probability =0.1	
	CDF(/yr)	CDF Increase (/yr)	CDF(/yr)	CDF Increase (/yr)
Basecase	7.3E-5			
Failure of containment pressure channels	1.4E-4	6.7E-5	2.4E-3[1]	2.3E-3[1]
Failure of SG level transmitters	7.3E-5	1.0E-7	7.4E-5	1.0E-6
Failure of pressurizer pressure channels	Un-changed		7.3E-5	1.0E-7
Failure of main steam pressure channels	Un-changed		Un-changed	
Failure of RCS Cold leg temperature measurement	Un-changed		Un-changed	

1 This result is dominated by small LOCA sequences where longer time is available for operator actions with limitations in the available information due to cable failures. Detailed analysis of the error probability may lower the estimate from that assumed in this sensitivity analysis for such situations reducing the CDF impact.

2 Cutset truncation at 1.0E-10.

Table 4.2 Operator Actions Affected By Cable Failures

Cable Failures	Affected Components	Affected Operator Actions and Basecase Error Probability
Failure of containment pressure channels	Containment pressure indication from pressure transmitters PT100A, PT100B, PT100C, PT100D	1. Rx Trip or SI Step 11 Check if MS should be isolated (HEP-1E0-11): 1.44E-3 2. Rx Trip or SI Step 12 Does no manual initiation CLS (HEP-1E0-12-NOCLS): 1.00E+0 3. Rx Trip or SI Step 12C Verify CS pumps are running (HEP-1E0-12C): 1.00E+0 4. Rx Trip or SI Step 12 D & E Check RS pumps running (HEP-1E0-12D-E): 1.44E-3 5. Rx trip or SI Attachment 1 CLS component Verification (HEP-1E0-ATTACH-1): 1.44E-3
Failure of PORV	Failure of PORVs PCV-1455C and PCV-1456	1. Rx Trip or SI Step 18 Check Przr PORVs or Spray Valves (HEP-1E0-21): 5.65E-2 2. Inadq. CC Step 10 Check RCS vent path (HEP-1FRC:1-10) 2.66E-3 3. Loss of 2nd heat sink Step 16 establish RCS bleed (HEP-1FRH:1-16): 2.66E-3
Failure of SG Level Control Cables	Failure of cables associated with SG train A, B, C level transmitters	1. Loss of 2nd heat sink Step 5 Check SG level (HEP-1FRH:1-5): 2.66E-3 2. Loss of 2nd heat sink Step 12 Verify RCS feed path (HEP-1FRH:1-12): 2.66E-3 3. Loss of 2nd heat sink Step 16 establish RCS bleed (HEP-1FRH:1-16): 2.66E-3

Cable Failures	Affected Components	Affected Operator Actions and Basecase Error Probability
Failure of cables for SG pressure measurements	Cables for pressure channels PC 474, 475, 476; PC 484, 485, 486: PC 494, 495, 496	1. Inadq CC Step 12 depressurize all intake SGS (HEP-1FRC:1-12-S1): 1.00E+0 2. Inadq CC Step 12 depressurize all intake SGS (HEP-1FRC:1-12-S2): 3.07E-1 3. Isolate Faulted SG (HEP-1E3-3): 2.26E-2
Failure of cables for pressurizer pressure	Cables for pressure channel PC 455, 456, 457	1. Rx Trip or SI Step 18 Check Przr PORVs and Spray valves(HEP-1E0-21): 5.65E-2 2. ATWS Step 4 Initiate emergency borate (HEP-1FRS:1-4): 1.44E-3
Failure of cables for RCS cold leg temperature measurement	Cables for temperature channel 412E, 422E, 432E	1. Post LOCA Cooldown and depressurization (HEP-1ES1:2-S1): 1.0E+0 2. Post LOCA Cooldown and depressurization (HEP-1ES1:2-S2): 5.33E-2 3. Transfer to Cold Leg recirculation (HEP-1ES1:3): 5.82E-2
Failure of cables for measuring RCS pressure		1. Rx trip or SI Step 13 Verify SI flow (HEP-1E0-13-S2): 1.55E-3 2. Rx trip or SI Step 13 verify SI flow (HEP-1E0-13-S2): 1.55E-3

Cable Failures	Affected Components	Affected Operator Actions and Basecase Error Probability
Failure of cables for recirculation spray pumps	Recirculation spray pumps RSP1A, RSP1B	1. Rx Trip or SI Step 12 Does no manual initiation CLS (HEP-1E0-12-NCOLS): 1.0E+0 2. Rx Trip or SI 12C Verify CS pumps running (HEP-1E0-12C): 1.0E+0 3. Rx trip or SI Step 12 D & E Check CS pumps running (HEP-1E0-12D-E): 1.44E-3
Failure of cables associated with RHR pumps	RHR Pumps P1A, P1B	1. RHR Step 5.2 Open TV-CC-109(HEP-10P14:1-5:2): 2.66E-3 2. RHR Step 5.10 Open MOV 1700 and 1701(HEP-10P14:1-5:10): 2.66E-3 3. Station Blackout Step 4, restore CCW (HEP-AP10.00-4): 2.66E-3

Table 4.3 CDF Impact of Failure of Containment Pressure Channels

Case	CDF(/yr) and Increase in CDF (/yr)			
	Operator Actions Unaffected		Affected Operator Action error Probability = 0.1	
	CDF(/yr)	CDF Increase (/yr)	CDF(/yr)	CDF Increase (/yr)
Failure of one pressure channel	7.4E-5	1.0E-6	1.5E-4	7.7E-5
Failure of 2 pressure channels	1.4E-4	6.7E-5	2.4E-3[1]	2.3E-3[1]
Failure of 4 pressure channels	1.4E-4	6.7E-5	2.4E-3[1]	2.3E-3[1]

[1] This result is dominated by small LOCA sequences where longer time is available for operator actions with limitations in the available information due to cable failures. Detailed analysis of the error probability may lower the estimate from that assumed in this sensitivity analysis for such situations reducing the CDF impact.

Table 4.4 Changes in LOCA Contributions to CDF Due to Failure of Containment Pressure Channels

LOCA Type	Basecase CDF(/yr)	CDF (/yr) given failure of (2 or 4) containment pressure channels	
		Operator Actions Unaffected	Affected Operator Action Error Probability=0.1
Large LOCA (A)	4.7E-6	8.8E-6	1.0E-4
Intermediate LOCA (S1)	5.3E-6	8.1E-6	1.0E-4
Small LOCA (S2)	1.3E-5	7.4E-5	2.1E-3[1]

[1] Detailed analysis of the error probability can lower the estimate from that assumed in this sensitivity analysis because of the longer time available reducing the CDF impact.

Table 4.5 Internal Event CDF Impact of Failures of Control Cables in a PWR Plant: Surry Nuclear Power Station

Case	CDF(/yr) and Increase in CDF (/yr)			
	Operator Action Unaffected		Affected Operator Action Error Probability=0.1	
	CDF(/yr)	Increase in CDF (/yr)	CDF (/yr)	Increase in CDF (/yr)
Basecase	7.3E-5			
Failure of control cables of RS pumps	1.1E-4	3.7E-5	1.1E-4	3.7E-5
Failure of control cables of PORVs	7.5E-5	2E-6	7.5E-5	2E-6
Failure of control cables of RHR pumps	Un-changed		Un-changed	
Failure of Control cables for MOVs* (hot leg recirculation)	1.5E-3	1.4E-3	Not Analyzed	

* Not applicable for Surry

4.4 CDF and Accident Sequence Impact of Cable Failures in a BWR

The CDF and accident sequence impact of cable failures in a BWR are assessed based on an analysis of the Peach Bottom Station, Unit 2. The NUREG 1150 model of the Peach Bottom Station (NUREG/CR-4550, August 1989) implemented in the SAPHIRE code was used.

Similar to the PWR analysis, the scoping assessment of cable failures in a harsh environment for BWRs was made considering failures of different combinations of cables inside the primary containment. Similar to the PWR, failures of instrument- and control-cables are analyzed.

As noted earlier in Chapter 3, the cables that can be affected by harsh environment in a BWR are minimal because very few components are located inside the primary containment and very few measurements are carried out there. The components that can be affected are motor-operated valve MV 18 of the shutdown cooling system, the solenoid-operated valves (SOVs) of the depressurization system, and the SOVs of the main steam isolation valves (MSIVs). The failure of the control cables of these valves are analyzed. The instrumentation measuring the reactor and drywell parameters which are used to actuate safety system components are located outside the primary containment including the associated electrical cables. These parameters include: reactor vessel water level and pressure, drywell pressure, suppression pool water level. The suppression pool temperature is measured inside the containment and the cables can be affected by harsh environment. The failure of the suppression pool temperature reading can affect some of the operator's actions: failure to vent the containment, failure to align the RHR, and failure to operate the second pump and open valves in the control rod drive system. Failure of the instrument cable associated with the suppression pool temperature measurement is analyzed through these operator errors. The instrument cables associated with suppression pool temperature measurement are expected to be in a metal conduit providing protection in a harsh environment. This may limit the likelihood of failure of the suppression pool temperature measurement.

Accident Sequences Causing Harsh Environment

The accident sequences affected by harsh environment in the Peach Bottom plant are as follows:

1. Large, intermediate, small, and small-small LOCAs. The contribution of large LOCA is relatively small.

2. Transients with failure of the suppression pool cooling, i.e., the sequences of the following transients that involve loss of suppression pool cooling: loss of offsite power transients (T1), transients with the power conversion system unavailable (T2), transients with power conversion system initially available (T3A) but subsequent loss of this system, and transients due to an inadvertent open relief valve in the primary system (T3C). These sequences are sometimes referred to as TW sequences. Transients due to an inadvertent open relief valve without the loss of suppression pool cooling is assumed not to cause a

harsh environment; it is assumed that suppression pools are designed to handle such situations.

3. ATWS sequence of concern is the one involving failure of reactor depressurization due to failure of safety relief valves. Such a sequence involves an ATWS followed by successful opening of safety relief valves, operation of standby liquid control system, and inhibition of ADS by the operator. But, high pressure coolant injection fails and low pressure cooling can't be accomplished due to failure of the safety relief valves (from failure of cables due to harsh environment). The contribution of this sequence, assuming failure of safety relief valves, remains around 1.0E-7and has negligible impact on our results.

Analysis of Results

The impact of the cable failures is analyzed in terms of the impact on the accident sequences and the CDF. Since relatively few cables can be affected by harsh environments in a BWR plant, the results presented in the tables address the following:

1. Control cable failures associated with failures of the ADS and non-ADS relief valves; in cases of failure of the ADS valves, the relevant operator's action is to depressurize the reactor manually using the relief valves. The failure of the cables in the relief valves will make action by the operator ineffective. No other action is expected to be affected.

2. For analyzing the failure of the instrument cables, only the cable associated with suppression pool temperature can be relevant. Its failure can affect the operator's actions mentioned above: failure to vent the containment, failure to align the RHR, and failure to operate the second pump in the control rod drive system. Two of the errors, failure to vent the containment, and failure to operate the second pump, are assigned a probability of 0.5 in the NUREG-1150 model, i.e., in our basecase. The failure to align the RHR is assigned a basecase probability of 1.0E-5. However, the cables for measuring the suppression pool temperature is in a metal conduit outside the suppression pool. The likelihood of cable failure is considered small.

Table 4.6 presents the CDF results of the BWR analyses. The internal event CDF is obtained by considering the impact of the cable failure on the sequences affected by the harsh environment. The remaining sequences are unaffected, but included to obtain the revised CDF. The increase in CDF is obtained by subtracting the basecase CDF from the revised CDF.

The results show the risk-significance of the safety relief valves: ADS and the non-ADS relief valves. The CDF increases by about a factor of 7. The CDF impact of the failure of only the automatically actuated ADS valves is about a factor of six lower. The CDF impact of the MOV MV 18 is a factor of 1.7 from the basecase. The failure of the suppression pool temperature measurement influences the CDF through the operator's actions. Specifically, the operator's action that contributes to the changed CDF is the failure to align the RHR system. Since the impact on the CDF is the largest for the cables associated with the relief valves, we further analyze the failure of the ADS and non-ADS safety relief valves.

For the Peach Bottom plant, there are five ADS safety relief valves, and six non-ADS relief valves. Each valve discharges via a tailpipe line through a downcomer to the suppression pool. The ADS is automatically actuated, but the operator may manually initiate the ADS or may depressurize the reactor vessel using the six relief valves that are not connected to the ADS logic.

The success criterion for the ADS is three of five valves opening to depressurize the reactor, i.e., failure of three of the valves causes the automatic depressurization system to fail. The success criterion of the non-ADS relief valves is three of the six valves opening, i.e., failure of four of the six valves causes failure of depressurization through the relief valves.

The ADS and the non-ADS valves are located inside the containment. The ADS valves are designed to operate under accident conditions. However, should the containment pressure be excessively high (~85 psig or greater), the valves could not be kept open since the air/nitrogen supply pressure is limited to ~85 psig (NUREG/CR-4550, Peach Bottom plant; based on discussions with Philadelphia Electric Company, PECO, personnel).

The ADS logic consists of two divisions. Each division can actuate all ADS valves. Power dependencies for each division are the 125 VDC/A bus as a primary source, and the 125 VDC/B as a backup source.

In our analysis, failure of the ADS and non-ADS valves implies failure of three of the five ADS and four of the six non-ADS valves. Also, cable failures which provide the actuation signals to the valves will result in failure of the valves.

Accident Sequences Impacted by Cable Failures

To analyze the accident sequences impacted, we study the failure of the ADS and non-ADS valves. Two previously non-dominant accident sequences become a dominant contributor, given the failure of these valves. These two sequences (S1-51 and S2-21) are discussed below; event trees are given in Appendix A.

S1-51 is sequence 51 of an intermediate LOCA scenario. The reactor scram is successful and the offsite power is available but HPCI fails to initiate and depressurization of the primary system is unsuccessful (because of the cable failures), disabling the low pressure coolant system, leading to core damage. The contribution of this sequence increases from negligible to 2.83E-5/yr for the failure of the relief valves.

S2-21 is a small LOCA where reactor scram is successful and offsite power is maintained. The power conversion system (PCS) fails to remove heat from the core, and both HPCI and RCIC fail to operate. Depressurization of the reactor is unsuccessful because of the cable failures. Core damage ensues because no other system is available for coolant makeup. The contribution of this sequence increases from negligible to 1.45E-5/yr.

The analysis of the failure of suppression pool temperature measurement shows that the change in the CDF contribution is dominated by the operator's error probability in aligning the RHR system. The basecase error probability is 1.0E-5 which was changed to 0.1 given lack of suppression pool temperature measurement. This increase in operator's error probability increases the likelihood of losing suppression pool cooling, making the contribution of loss of offsite power sequences coupled with loss of suppression pool cooling become large.

4.5 Summary of Results from BWR Analyses

The analysis for the Peach Bottom, BWR, plant was conducted in a manner similar to that for the PWR plant. The failures of cables inside the primary containment are studied in terms of their impact on CDF and in identifying otherwise less dominant accident sequences that may become dominant contributors.

As stated earlier, relatively few cables that impact the safety analyses of a BWR are inside the primary containment. The primary concern is the failure of the relief valves that are used to depressurize the reactor. The failure of cables associated with these valves can increase the CDF to approximately 6E-5/yr (the increase in CDF is approximately 5E-5/yr). It is important to note that the increase in the CDF is from accident sequences that become dominant contributors due to the loss of the safety valves; otherwise, these sequences had negligible contribution to the basecase CDF. As also noted earlier, the safety relief valves may degrade with age because of their location which exposes them to adverse thermal environment during routine operation and may make them more vulnerable to failure when exposed to harsh environment.

Many measurements monitoring the progression of an accident and that influence the operator actions modeled in a PRA are carried out outside the primary containment. Accordingly, the cable failures due to harsh environment is unlikely and consequently, have a limited impact on operator's actions. The cables relating to suppression pool temperature measurement are placed in a metal conduit outside the torus and are considered protected against harsh environment. The likelihood of failure of these cables due to harsh environment is also considered small.

The results from this scoping analysis of a BWR plant show the impact on CDF for cable failures due to harsh environment; these results have features that are expected to be common for BWR plants. The relief valves used to depressurize the reactor are similar, although the number may vary. The impact is due to failure of the relief valves and has minimal influence of the operator action error probability where the impact of the cable failures is uncertain. In that regard, generalization of the BWR results may be easier than for a PWR.

Table 4.6 Internal Event CDF Impact of Failures of Instrument and Control Cables in a BWR Plant: Peach Bottom Station

Case	CDF(/yr) and Increase in CDF (/yr)[3]			
	Operator Actions Unaffected		Affected Operator Action Error Probability = 0.1	
	CDF (/yr)	Increase in CDF (/yr)	CDF(/yr)	Increase in CDF(/yr)
Basecase	7.2E-6			
Failure of Control cables of ADS Valves	9.2E-6	2.0E-6	NA[1]	
Failure of Control Cables of ADS and non-ADS Relief Valves	5.8E-5	5.1E-5	NA[1]	
Failure of Control Cables for MOV MV18	1.2E-5	4.8E-6	NA[1]	
Failure of Instrument Cables[2] (failure of suppression pool temperature measurement)	Unchanged		NA[2]	NA[2]

[1] NA: Not Applicable

[2] The cables providing the suppression pool temperature are placed inside metal conduits outside the suppression pool. The likelihood of their failure due to harsh environment is considered small. Assuming operator action error probability of 0.1 is considered very conservative.

[3] Cut set truncation at 1.0E-9

5. SUMMARY AND RECOMMENDATIONS

In this report, a scoping study is presented of the impact of cable failures due to harsh environment in a nuclear power plant. The study used the existing PRA models to assess the core damage frequency (CDF) due to cable failures in a harsh environment created by accidents. The current PRA models do not explicitly include cable failure, but rather implicitly assume that cables will perform reliably in an accident environment (similar to their performance in a normal environment). In this approach used in the PRA models, the impact of reduced instrument readings in the control panel on the operators' actions also is not addressed. In this study, the available PRA models are used to simulate the cable failures considering the components and operators' actions that may be affected by cable failures.

The study involves several assumptions as it uses the available PRA models, and conducts sensitivity studies to obtain insights on the impact of cable failures. The major assumptions which should be considered in interpreting the results can be summarized as follows:

- a harsh environment is assumed to cause cable failure which results in loss of actuation signals failing involved component(s) and instrumentation signals,

- cable failures are expected to occur in splices, connections, and terminal boxes, but their specific locations in a cable are not identified nor evaluated,

- differences in the harshness of the environment in different types of accidents (e.g., large vs small LOCAs) are not analyzed; in both cases, the cables in the containment are assumed to be affected and failed. In a small LOCA, the area within the containment affected by harsh environment may be limited and some of the cables may remain unaffected. Also, in a small LOCA accident, time for some of the operator actions (e.g., operation action for aligning high pressure recirculation or aligning the recirculation spray pumps) is long and it is possible that the limitation due to loss of some information due to cable failures may be overcome and the impact on the HEP for such actions will be limited. In this scoping study, such differences are not taken into consideration.

- the location of different cables with respect to each other, which may influence the likelihood of multiple cable failures, are only considered in defining cable failures for the sensitivity analyses,

- the likelihood of cable failures due to age and a harsh environment is not estimated because of the very limited experimental data on the performance of cables under such conditions,

- the impact of cable failures on operator's actions is considered only for those actions that are modeled in a PRA; there is no detailed review of operator procedures that will be used in such situations nor other operator actions that are considered very reliable in a PRA. Sensitivity analyses are conducted by changing the failure probability of operator’s action to 0.1.

- accidents causing a harsh environment inside the containment are addressed; harsh environment elsewhere in the plant and its effects are not addressed.

- the time needed for the harsh environment to develop was not analyzed nor were the times at which the components and operator’s actions (affected by cable failures) needed during the progression of an accident. However, both can influence the success/failure of equipment and operator’s actions.

A PWR and a BWR plant are studied. This study does not include analyses of plants of different vendor design with differences in location of equipment and cables inside the containment.

The results of PWR analyses can be summarized as follows:

The failure at the Surry plant of some combinations of cables resulting in failure of the containment pressure indications, or failure of the inside recirculation spray pumps can cause a sizable increase in the CDF. The failure of many other cables by themselves have negligible impacts on the CDF.

The scoping analysis of the cable’s failures provide a perspective on the relative ranking of the cables in terms of their risk significance. The analysis, within its assumptions, shows the relative importance of instrument cables associated with the containment pressure measurement, and the control cables associated with the inside containment recirculation pumps. The cables associated with many other measurements, e.g., pressurizer pressure, main steam pressure, SG level detection, RCS cold leg temperature, by themselves, are not observed to be risk-significant. The cables associated with the RHR pumps also have no impact on accident sequences causing a harsh environment inside the containment. In a general categorization, the results can be interpreted as showing that the instrument cables, as defined in the report, have a higher risk significance than control cables associated with the components inside the containment.

The scoping analysis also identified the accident sequences that become dominant contributors in case of cable failures in a harsh environment. These sequences are primarily LOCAs (large, intermediate, small) followed by failure of the recirculation spray system. In case of small LOCAs, the time available for the operator action to initiate recirculation spray is longer compared to that in large LOCAs (greater than 8 hours compared to 1 hour) and the probability of the operator’s action to fail to initiate recirculation spray will be lower. Such differences are not taken into consideration in this scoping study.

The results obtained for the Surry plant are plant-specific. The components and cables that are located inside the containment and affected by harsh environments vary from plant to plant. Some aspects of the Surry design, e.g., failure of recirculation spray system in a LOCA causing core damage, location of RHR pumps inside the containment, location of normally-closed hot leg recirculation MOVs outside the containment, are unique and contributed to the results presented here.

The results of the BWR analyses can be summarized as follows:

The analysis for the Peach Bottom, BWR, plant shows that the CDF impact of cable failure is sizable when ADS and non-ADS safety relief valves fail, or if the operator fails to align the RHR system to cool the suppression pool due to loss of its temperature measurement.

As stated earlier, relatively few of the cables that impact the safety analyses of a BWR are inside the primary containment. The primary concern is the failure of the relief valves that depressurize the reactor. The failure of cables associated with these valves can increase the CDF to approximately 6E-5/yr (the increase in CDF is approximately 5E-5/yr). It is important to note that the increase in the CDF is from accident sequences that become dominant contributors due to the loss of the safety valves; otherwise, these sequences had negligible contribution to the basecase CDF. As also noted earlier, the safety relief valves may degrade with age because of their location which exposes them to adverse thermal environment during routine operation and may make them more vulnerable to failure when exposed to harsh environment.

Many measurements monitoring the progression of an accident and that influence the operator's actions modeled in a PRA are made outside the primary containment. Accordingly, the cable failures have a limited impact on operators' actions. The failure of cables related to suppression-pool temperature measurements can affect some of the operator's actions. These cables are placed in a metal conduit while inside the torus and are partially protected against harsh environment. The action that dominates the CDF contribution in this scenario is the failure to align the RHR system to cool the suppression pool. The effect on this action may not be large.

The results from this scoping analysis of a BWR plant show the impact on CDF for cable failures due to harsh environment; these results have features that are expected to be common for BWR plants. The relief valves used to depressurize the reactor are similar, although the number may vary. The impact is due to failure of the relief valves and has minimal influence of the operator action error probability where the impact of the cable failures is uncertain. In that regard, generalization of the BWR results may be easier than for a PWR.

Recommendation for Additional Research

The scoping analysis presented in this report provides useful insights on the effect of cable failures in nuclear power plants. As noted, several assumptions are made in conducting the scoping evaluation. The results identify selected cable failures for which the increase in CDF impact may need additional attention. The results of PWR analyses show the plant-specific features of the impact of cable failures and their relative significance.

Further activity in this area can be focused to address the following:

1. Conducting detailed analyses to address the assumptions in the scoping evaluation.

The scoping study identifies the impact on CDF for a PWR and a BWR plant due to harsh environment resulting in cable failures. It identifies through sensitivity analysis the types of cable failures that dominates the CDF impact. The scoping study made a number of assumptions which can be further addressed to obtain a more realistic estimate.
This detailed evaluation can include:

a) consideration of severity and zone of the harshness of environment following a LOCA (e.g., differences in cables that may be affected in a small vs large LOCA): considering the zone of influence of the harsh environment and the location of the cables within the containment an assessment can be made of the cables that are expected to be affected in such accidents. This will address the assumption in the scoping study that in different LOCAs the cables within the containment are affected equally;

b) the likelihood of failing multiple cables providing redundant signals: failure of multiple cables providing the redundant signals depend on their locations and the conduits through which they are routed. The likelihood of losing multiple signals can be better addressed when such factors are taken into consideration;

c) improved human reliability analysis considering limitations in the available information and the time available in different accident sequences: in a situation involving cable failures, operators may lose some of the information and may fail to take the appropriate action. However, for many actions alternate information may be available. A detailed analysis may be performed which includes consideration of the time available to carry out the action in a specific accident sequence to estimate the human error probabilities to be used to obtain more realistic assessment of the impact of cable failures replacing the sensitivity analysis presented in this scoping study;

d) consideration of aging degradation of the cables due to thermal environment in routine operations: cables exposed to adverse thermal environment over a long period of operation may have increased vulnerability to harsh environment. For BWR plants where failure of the relief valves due to cable failures is of concerns, such degradations are relevant in assessing the impact of harsh environment.

2. Conducting scoping analyses of plants of other vendor designs.

Additional PWR plants may be analyzed to understand whether different designs can have different risk implications and if plant-specific differences can be of significance in identifying any additional significant accident sequences that may need attention. As noted in the scoping study, plants-specific differences exist in the location of the relevant systems and components. Understanding of vulnerability of plants because of the location of systems/components is useful considering the differences that are expected to be observed in the operating NPPs. A Combustion Engineering (CE) and a Babcock-Wilcox (B&W) plant can be studied. A BWR of different design can also be analyzed to compare and complement the analyses of the Peach Bottom Station.

3. Analysis of accidents outside the containment. Some accidents outside the containment can adversely affect multiple safety system components. A screening evaluation can be conducted to understand the risk-implications of such accidents.

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APPENDIX A

APPLICABLE EVEN TREES FOR THE SURRY AND PEACH BOTTOM PLANTS

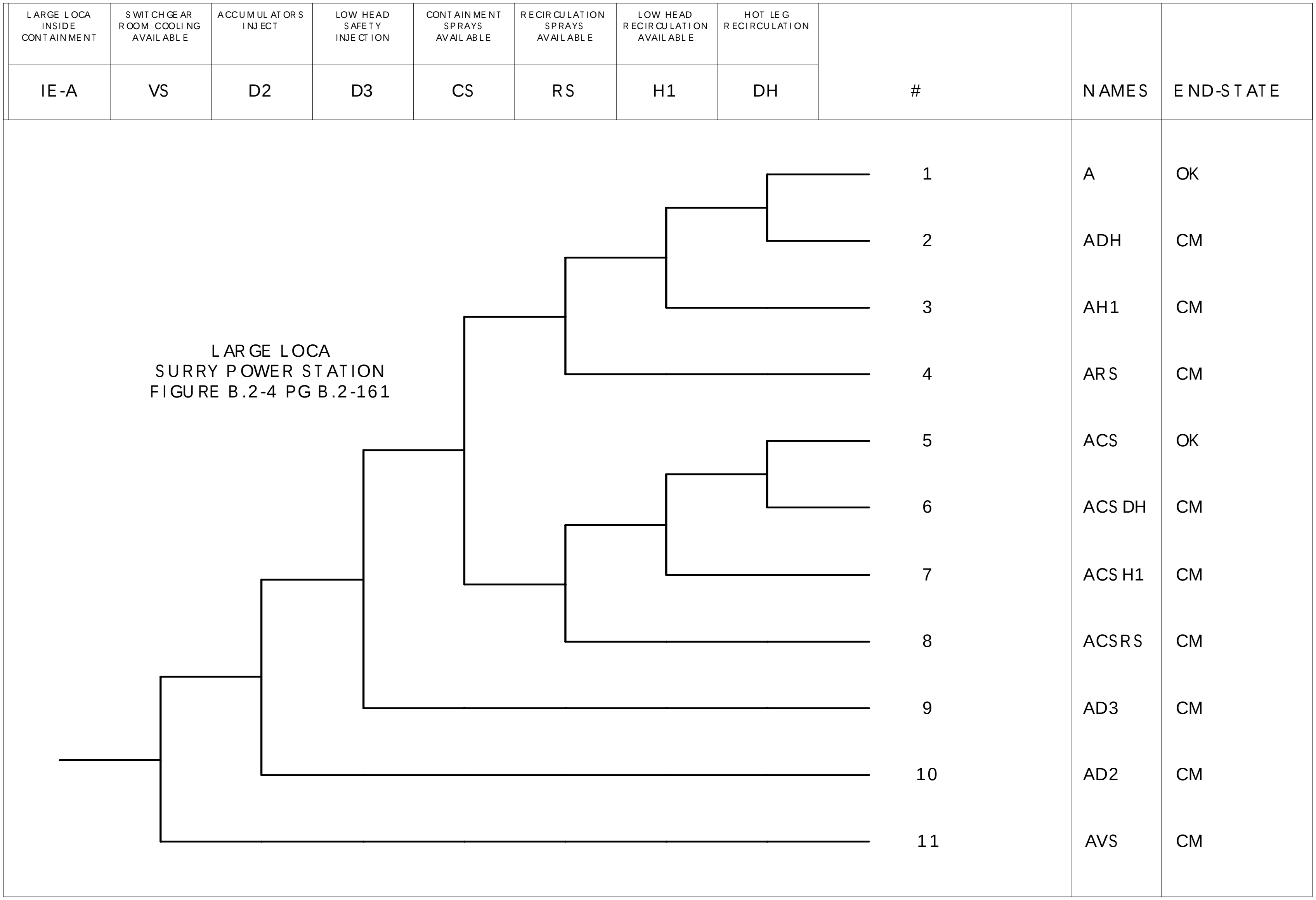
This appendix presents selected event trees from the Surry and the Peach Bottom PRA models. These event trees provide an understanding of the relevant accident sequences discussed in the main body of the report. The reader is referred to the respective PRAs for the detailed discussion of the event trees.

Surry’s Event Trees

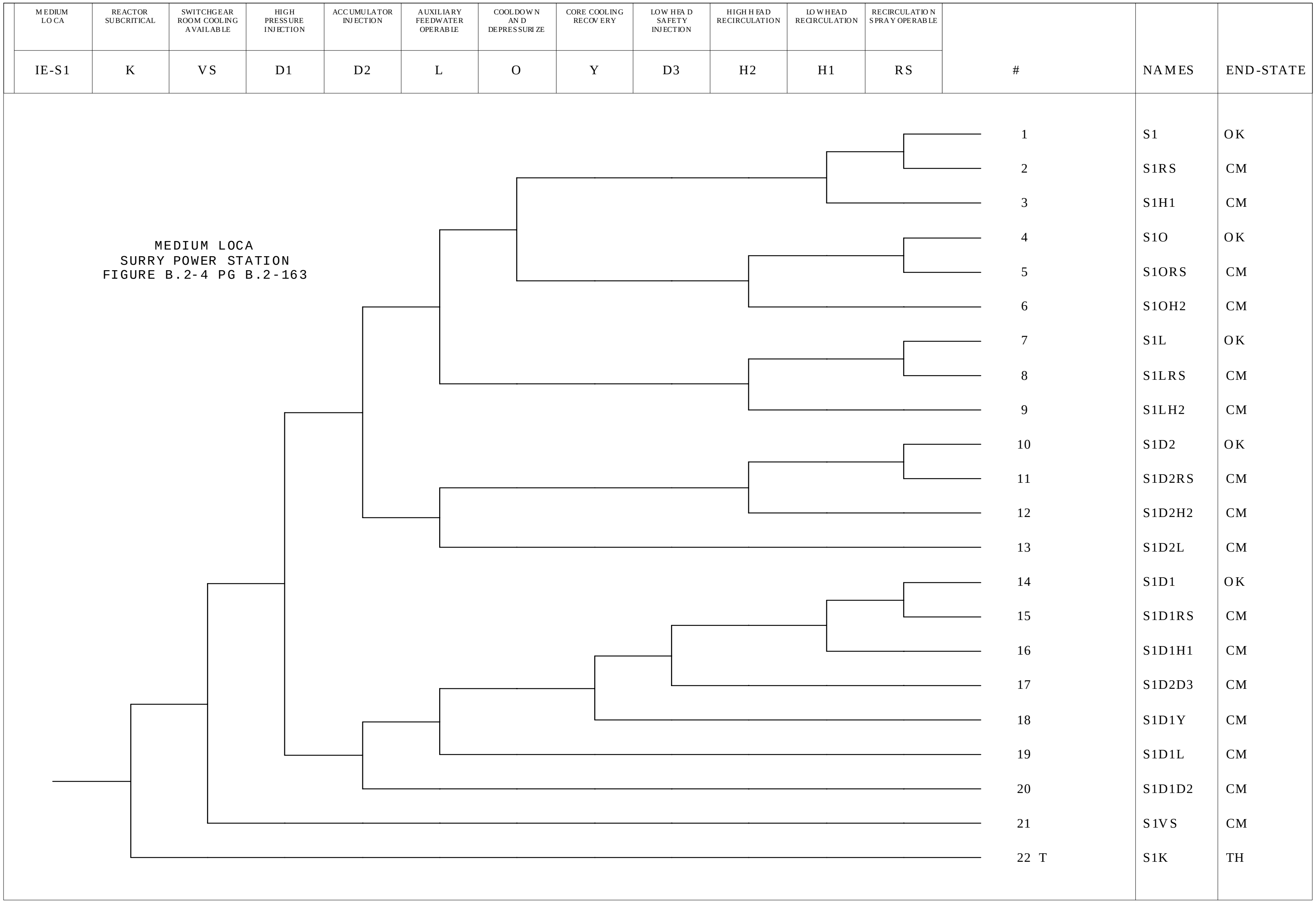
Event trees are presented in alphabetical order:

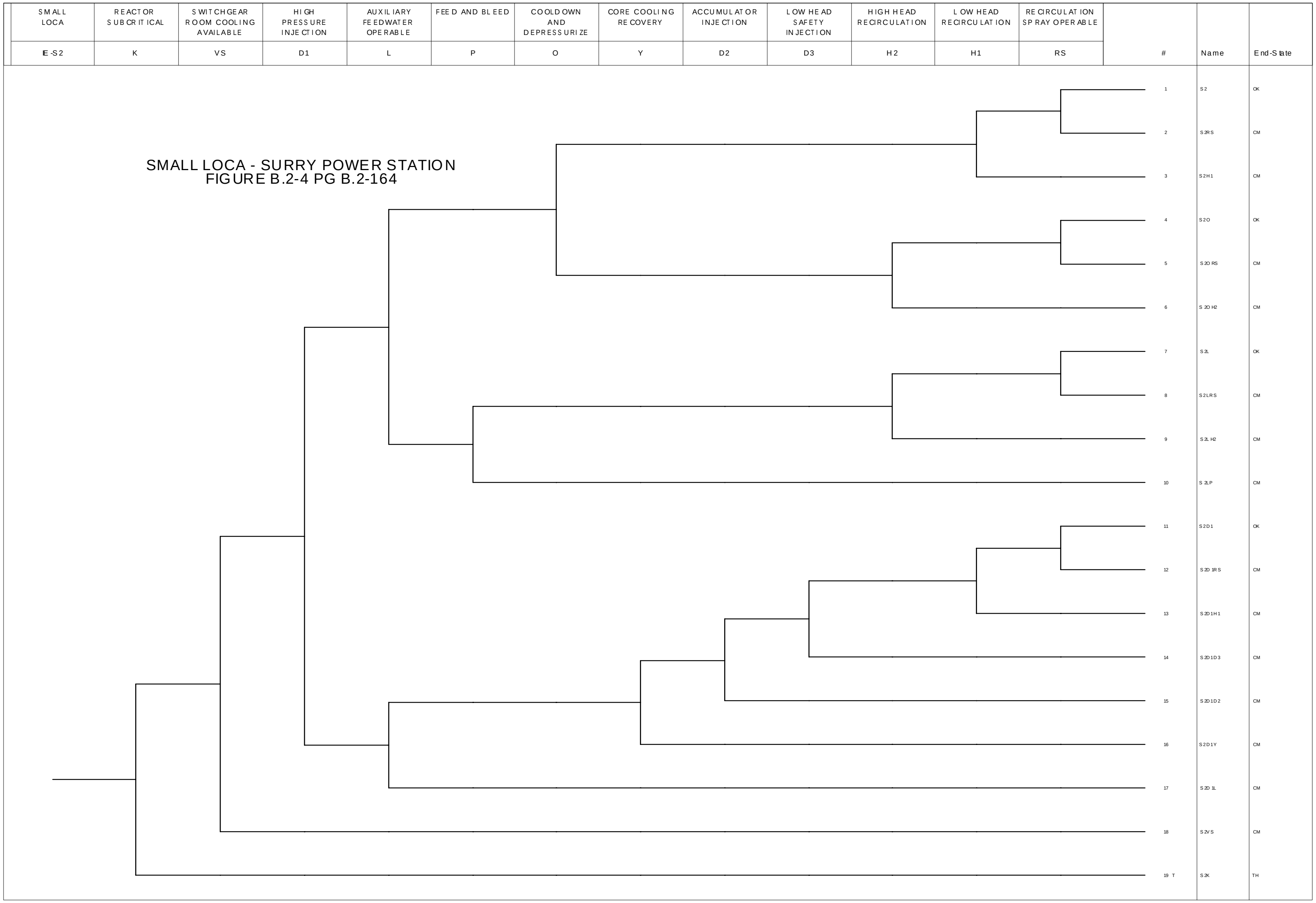
14. A
15. S1
16. S2
17. T1
18. T1A
19. T1B
20. T7
21. T8

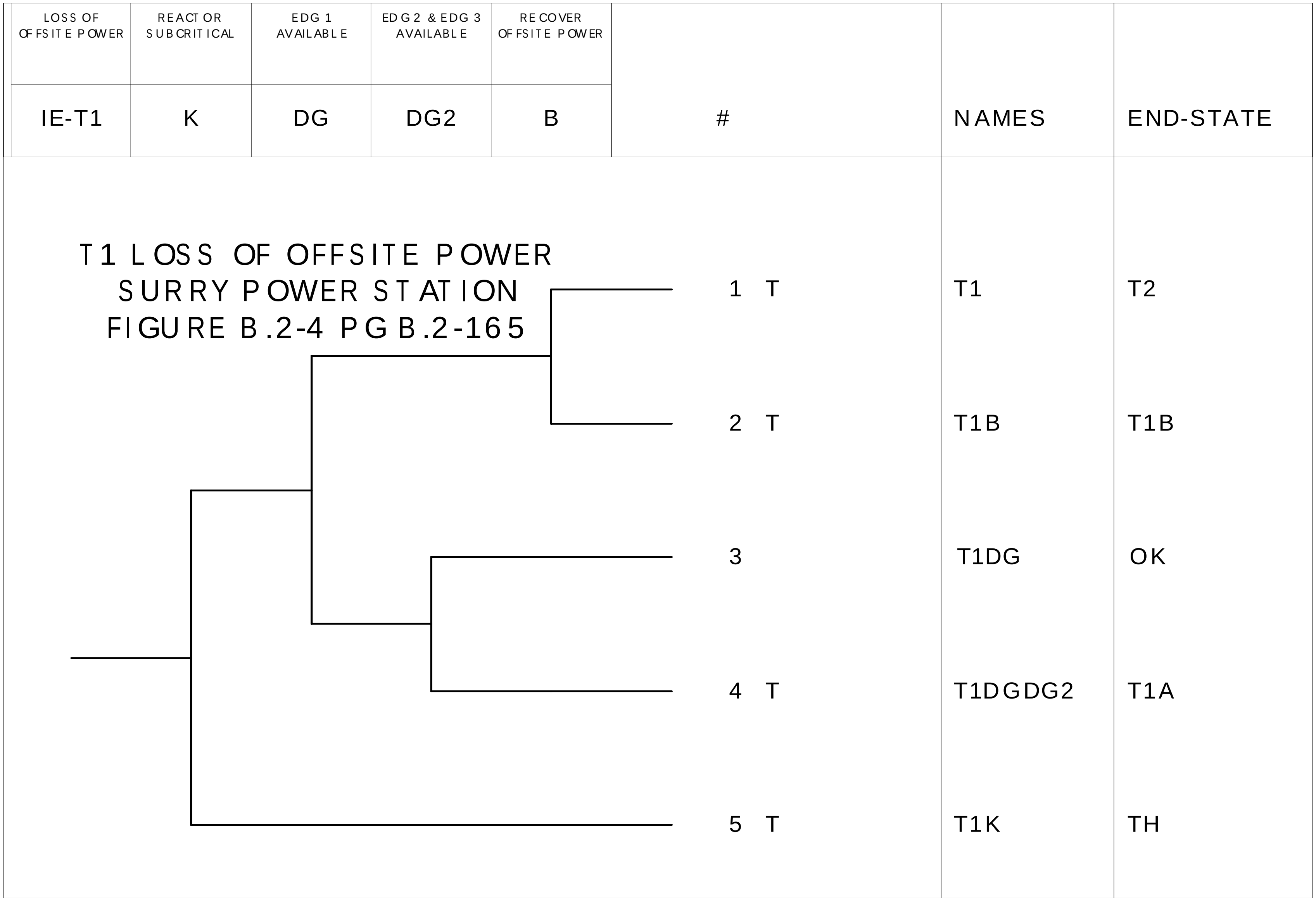
The legend in the figures refers to the figure number in the Surry’s IPE.

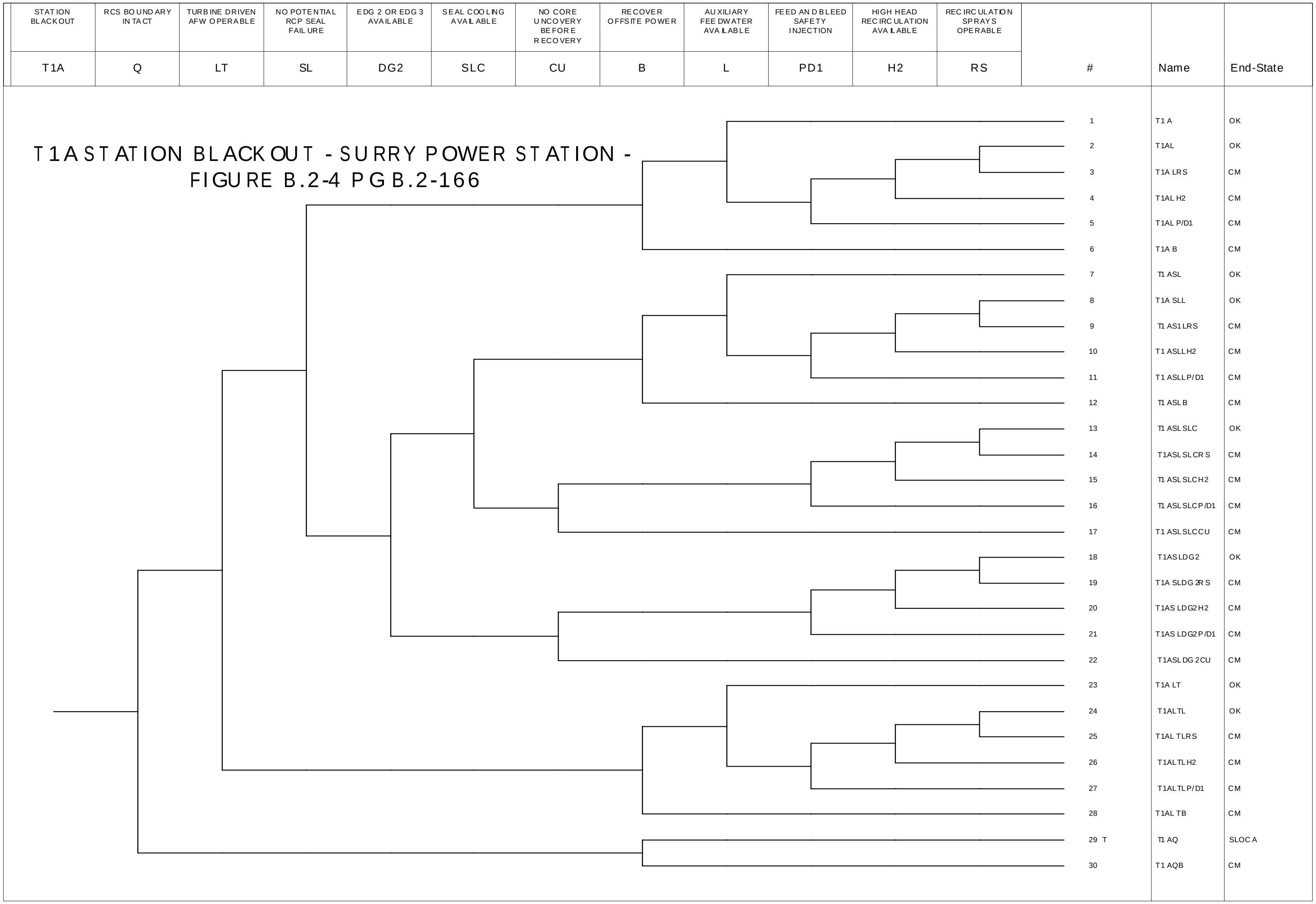


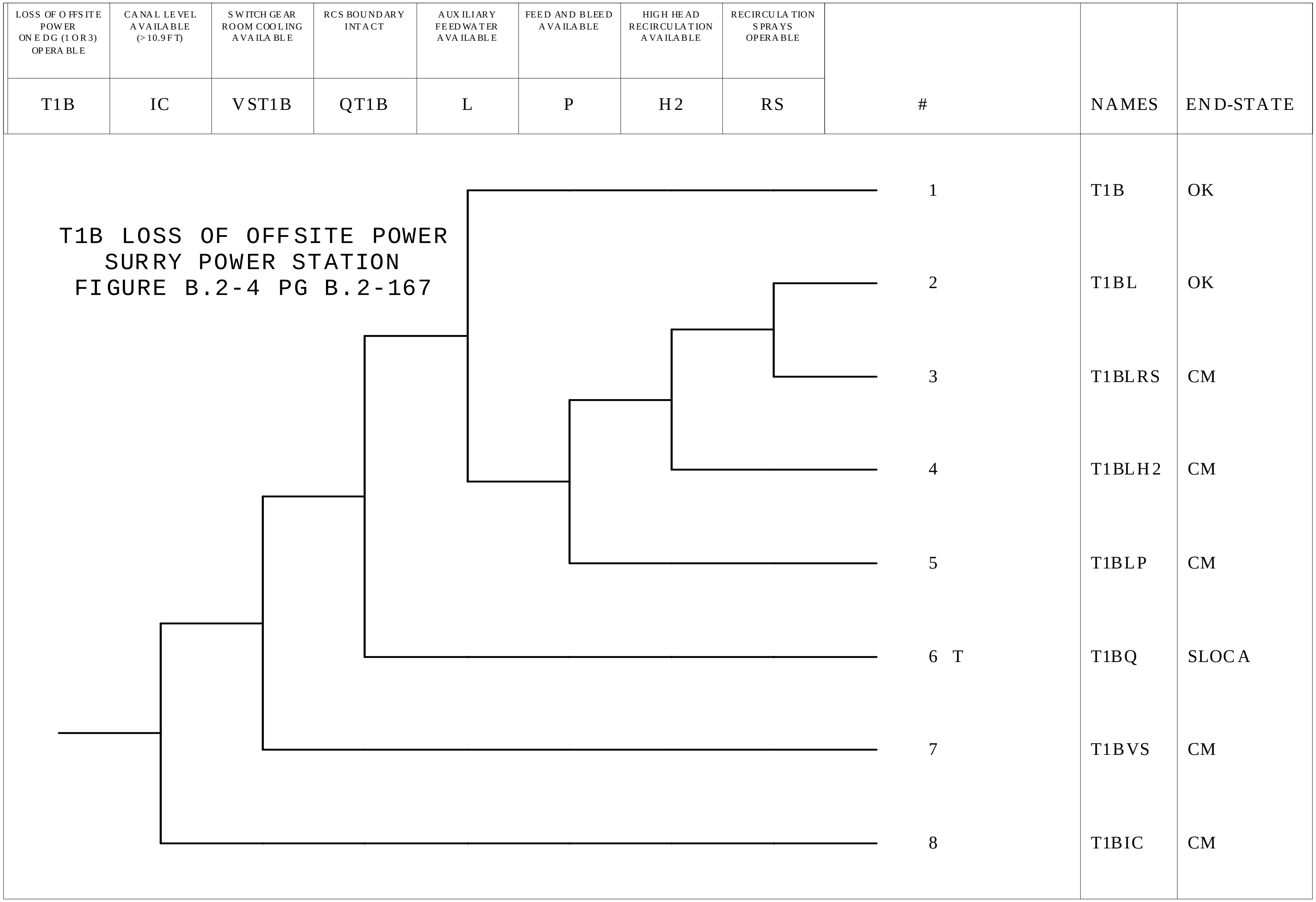
LARGE LOCA INSIDE CONTAINMENT
IE-A
SWITCHGEAR ROOM COOLING AVAILABLE
VS
ACCUMULATORS INJECT
D2
LOW HEAD SAFETY INJECTION
D3
CONTAINMENT SPRAYS AVAILABLE
CS
RECIRCULATION SPRAYS AVAILABLE
RS
LOW HEAD RECIRCULATION AVAILABLE
H1
HOT LEG RECIRCULATION
DH
#
NAMES
END-STATE
LARGE LOCA
SURRY POWER STATION
FIGURE B.2-4 PG B.2-161
1 A OK
2 ADH CM
3 AH1 CM
4 ARS CM
5 ACS OK
6 ACS DH CM
7 ACS H1 CM
8 ACSRS CM
9 AD3 CM
10 AD2 CM
11 AVS CM

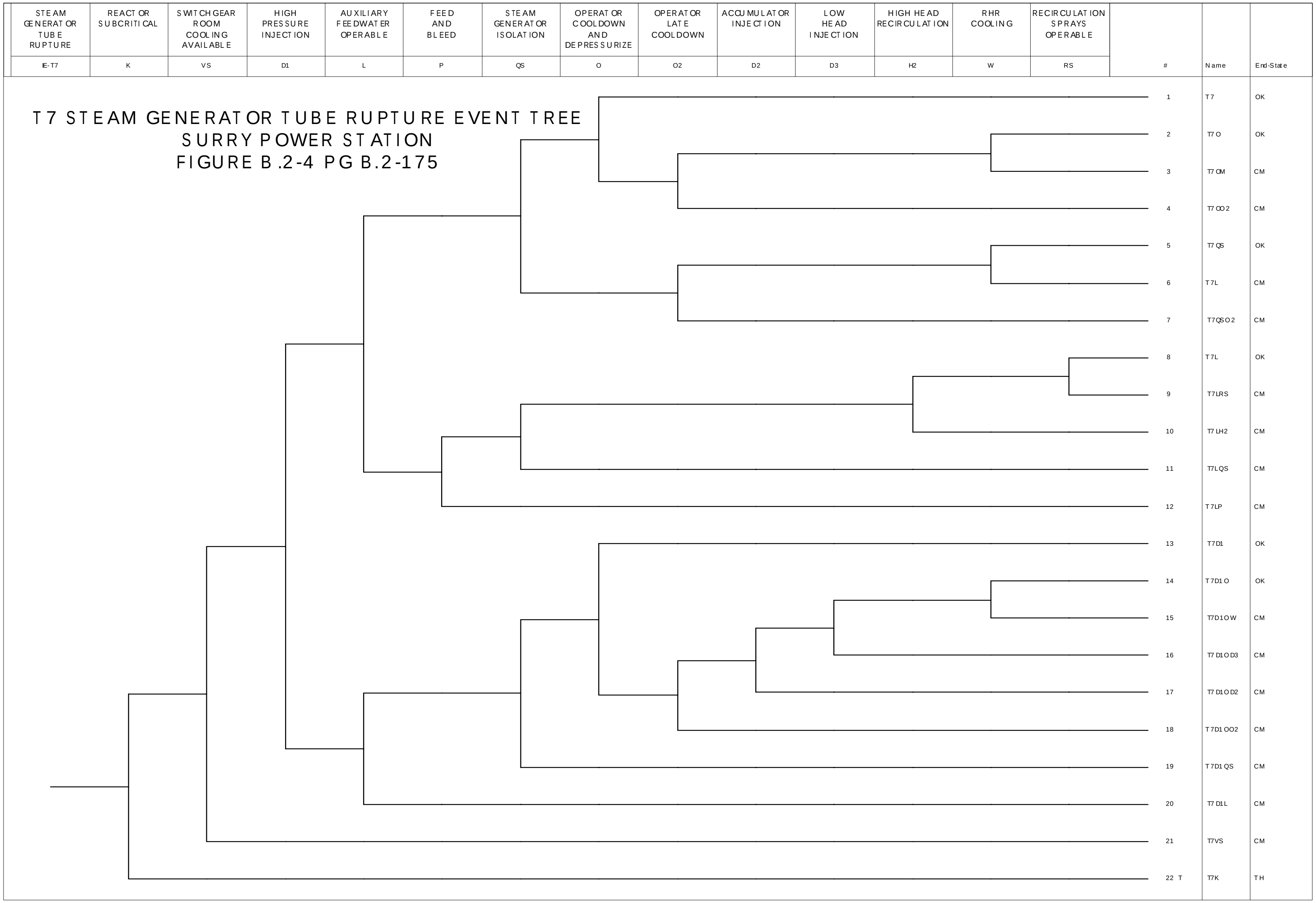

MEDIUM LOCA
IE-S1
REACTOR SUBCRITICAL
K
SWITCHGEAR ROOM COOLING AVAILABLE
VS
HIGH PRESSURE INJECTION
D1
ACCUMULATOR INJECTION
D2
AUXILIARY FEEDWATER OPERABLE
L
COOLDOWN AND DEPRESSURIZE
O
CORE COOLING RECOVERY
Y
LOW HEAD SAFETY INJECTION
D3
HIGH HEAD RECIRCULATION
H2
LOW HEAD RECIRCULATION
H1
RECIRCULATION SPRAY OPERABLE
RS
#
NAMES
END-STATE
MEDIUM LOCA
SURRY POWER STATION
FIGURE B.2-4 PG B.2-163
1 S1 OK
2 S1RS CM
3 S1H1 CM
4 S1O OK
5 S1ORS CM
6 S1OH2 CM
7 S1L OK
8 S1LRS CM
9 S1LH2 CM
10 S1D2 OK
11 S1D2RS CM
12 S1D2H2 CM
13 S1D2L CM
14 S1D1 OK
15 S1D1RS CM
16 S1D1H1 CM
17 S1D2D3 CM
18 S1D1Y CM
19 S1D1L CM
20 S1D1D2 CM
21 S1VS CM
22 T S1K TH


SMALL LOCA
E-S2
REACTOR SUBCRITICAL
K
SWITCHGEAR ROOM COOLING AVAILABLE
VS
HIGH PRESSURE INJECTION
D1
AUXILIARY FEEDWATER OPERABLE
L
FEED AND BLEED
P
COOLDOWN AND DEPRESSURIZE
O
CORE COOLING RECOVERY
Y
ACCUMULATOR INJECTION
D2
LOW HEAD SAFETY INJECTION
D3
HIGH HEAD RECIRCULATION
H2
LOW HEAD RECIRCULATION
H1
RECIRCULATION SPRAY OPERABLE
RS
#
Name
End-State
SMALL LOCA - SURRY POWER STATION
FIGURE B.2-4 PG B.2-164
1 S2 OK
2 S2RS CM
3 S2H1 CM
4 S2O OK
5 S2ORS CM
6 S2OH2 CM
7 S2L OK
8 S2LRS CM
9 S2LH2 CM
10 S2LP CM
11 S2D1 OK
12 S2D1RS CM
13 S2D1H1 CM
14 S2D1D3 CM
15 S2D1D2 CM
16 S2D1Y CM
17 S2D1L CM
18 S2VS CM
19 T S2K TH


LOSS OF OFFSITE POWER
REACTOR SUBCRITICAL
EDG 1 AVAILABLE
EDG2 & EDG 3 AVAILABLE
RECOVER OFFSITE POWER
IE-T1
K
DG
DG2
B
#
NAMES
END-STATE
T1 LOSS OF OFFSITE POWER
SURRY POWER STATION
FIGURE B.2-4 PG B.2-165
1 T
T1
T2
2 T
T1B
T1B
3
T1DG
OK
4 T
T1DGDG2
T1A
5 T
T1K
TH

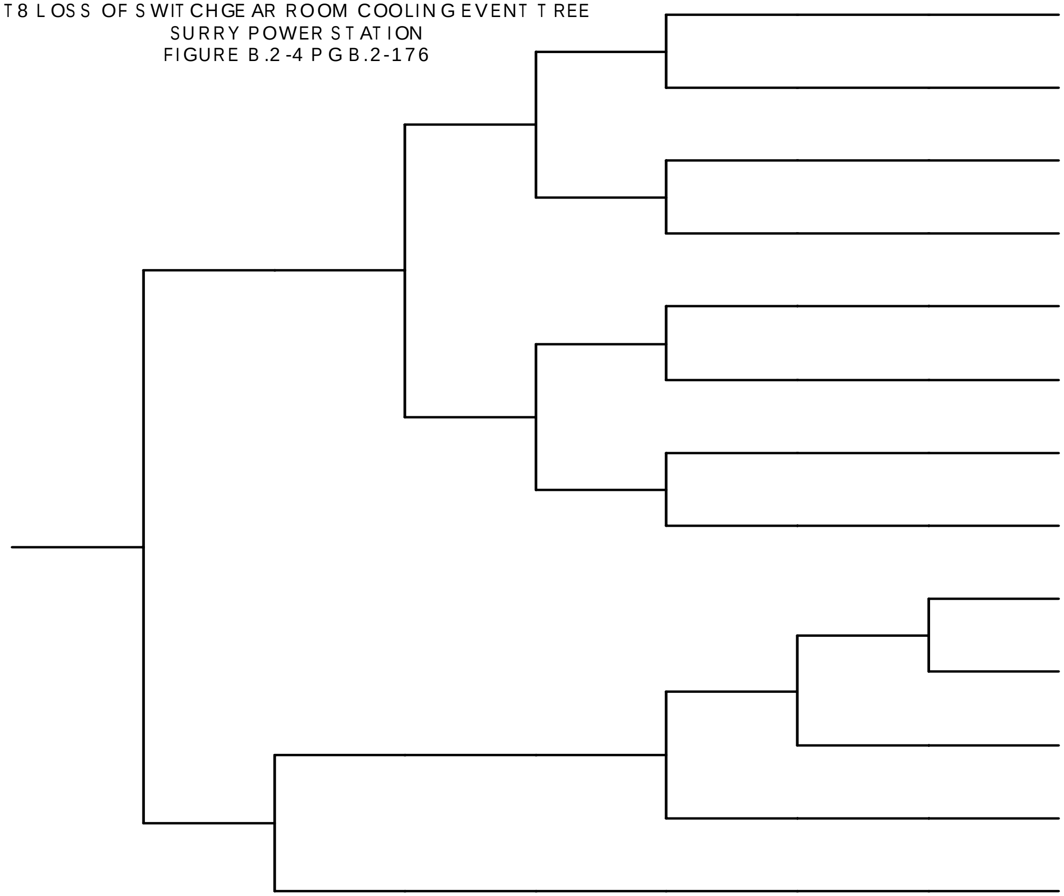

STATION BLACKOUT
T1A
RCS BOUNDARY INTACT
Q
TURBINE DRIVEN AFW OPERABLE
LT
NO POTENTIAL RCP SEAL FAILURE
SL
EDG 2 OR EDG 3 AVAILABLE
DG2
SEAL COOLING AVAILABLE
SLC
NO CORE UNCOVERY BEFORE RECOVERY
CU
RECOVER OFFSITE POWER
B
AUXILIARY FEEDWATER AVAILABLE
L
FEED AND BLEED SAFETY INJECTION
PD1
HIGH HEAD RECIRCULATION AVAILABLE
H2
RECIRCULATION SPRAYS OPERABLE
RS
#
Name
End-State
T1A STATION BLACKOUT - SURRY POWER STATION -
FIGURE B.2-4 PG B.2-166
1 T1A OK
2 T1AL OK
3 T1A LRS CM
4 T1AL H2 CM
5 T1AL P/D1 CM
6 T1A B CM
7 T1 ASL OK
8 T1A SLL OK
9 T1 AS1 LRS CM
10 T1 ASLL H2 CM
11 T1 ASLL P/D1 CM
12 T1 ASL B CM
13 T1 ASL SLC OK
14 T1ASL SLC RS CM
15 T1 ASL SLC H2 CM
16 T1 ASL SLC P/D1 CM
17 T1 ASL SLC CU CM
18 T1AS LDG2 OK
19 T1A SLDG 2RS CM
20 T1AS LDG2 H2 CM
21 T1AS LDG2 P/D1 CM
22 T1ASL DG 2CU CM
23 T1A LT OK
24 T1ALTL OK
25 T1AL TLRS CM
26 T1ALTL H2 CM
27 T1ALTL P/D1 CM
28 T1AL TB CM
29 T T1 AQ SLOCA
30 T1 AQB CM


LOSS OF OFFSITE POWER ONE DG (1 OR 3) OPERABLE
CANAL LEVEL AVAILABLE (> 10.9 FT)
SWITCHGEAR ROOM COOLING AVAILABLE
RCS BOUNDARY INTACT
AUXILIARY FEEDWATER AVAILABLE
FEED AND BLEED AVAILABLE
HIGH HEAD RECIRCULATION AVAILABLE
RECIRCULATION SPRAYS OPERABLE
T1B
IC
VST1B
QT1B
L
P
H2
RS
#
NAMES
END-STATE
T1B LOSS OF OFFSITE POWER
SURRY POWER STATION
FIGURE B.2-4 PG B.2-167
1 T1B OK
2 T1BL OK
3 T1BLRS CM
4 T1BLH2 CM
5 T1BLP CM
6 T T1BQ SLOCA
7 T1BVS CM
8 T1BIC CM


STEAM GENERATOR TUBE RUPTURE
IE-T7
REACTOR SUBCRITICAL
K
SWITCHGEAR ROOM COOLING AVAILABLE
VS
HIGH PRESSURE INJECTION
D1
AUXILIARY FEEDWATER OPERABLE
L
FEED AND BLEED
P
STEAM GENERATOR ISOLATION
QS
OPERATOR COOLDOWN AND DEPRESSURIZE
O
OPERATOR LATE COOLDOWN
O2
ACCUMULATOR INJECTION
D2
LOW HEAD INJECTION
D3
HIGH HEAD RECIRCULATION
H2
RHR COOLING
W
RECIRCULATION SPRAYS OPERABLE
RS
#
Name
End-State
T7 STEAM GENERATOR TUBE RUPTURE EVENT TREE
SURRY POWER STATION
FIGURE B.2-4 PG B.2-175
1 T7 OK
2 T7O OK
3 T7OM CM
4 T7OO2 CM
5 T7QS OK
6 T7L CM
7 T7QSO2 CM
8 T7L OK
9 T7LRS CM
10 T7LH2 CM
11 T7LQS CM
12 T7LP CM
13 T7D1 OK
14 T7D1O OK
15 T7D1OW CM
16 T7D1OD3 CM
17 T7D1OD2 CM
18 T7D1OO2 CM
19 T7D1QS CM
20 T7D1L CM
21 T7VS CM
22 T T7K TH

LOSS OF SWITCHGEAR ROOM COOLING	AUXILIARY FEEDWATER OPERABLE	FEED AND BLEED OPERABLE	TURBINE DRIVEN AFW OPERABLE	RCS BOUNDARY INTACT	ROOM COOLING RESTORED	HIGH HEAD RECIRCULATION AVAILABLE	RECIRCULATION SPRAYS OPERABLE			
IE-T8	LT8	P	LT	Q	RC	H2	RS	#	NAMES	END-STATE

T8 LOSS OF SWITCHGEAR ROOM COOLING EVENT TREE
SURRY POWER STATION
FIGURE B.2-4 PG B.2-176



#		NAMES	END-STATE
1		T8	OK
2		T8RC	CM
3	T	T8Q	S2
4		T8QRC	CM
5		T8LT	OK
6		T8LTRC	CM
7	T	T8LTQ	S2
8		T8LTQRC	CM
9		T8L	OK
10		T8LRS	CM
11		T8LH2	CM
12		T8LRC	CM
13		T8LP	CM

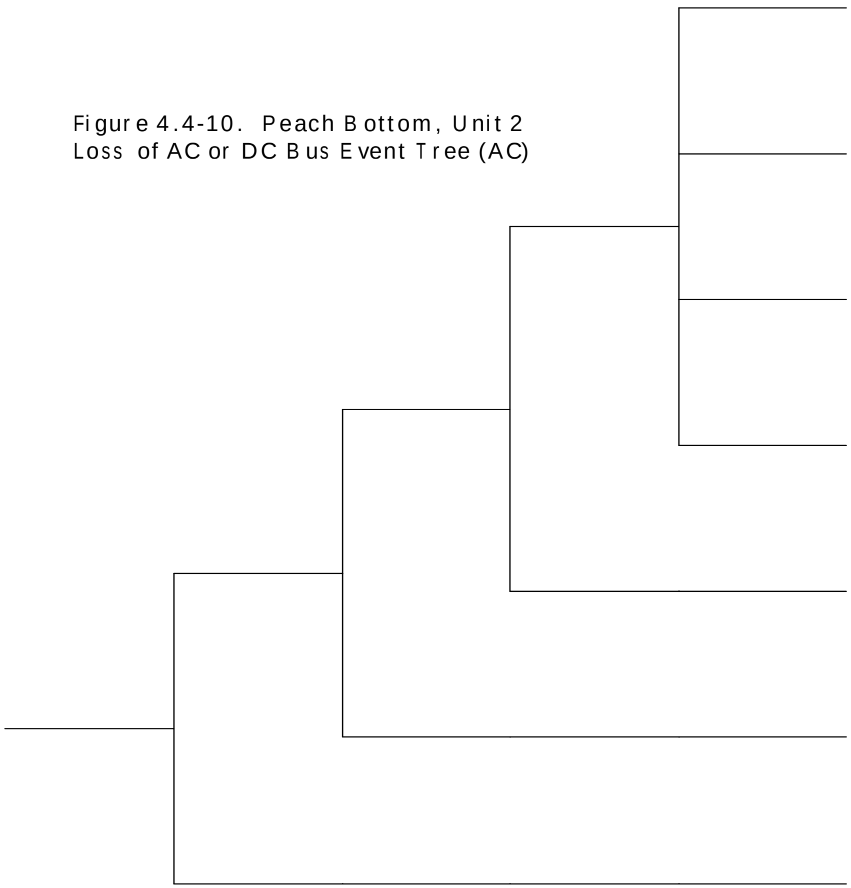
Peach Bottom, Unit 2's Event Trees

Event trees are presented in alphabetical order:

22. AC
23. AC1
24. AC8
25. ATWST3A
26. S1
27. S2
28. S3
29. T1
30. T11
31. T13
32. T15
33. T21
34. T3A
35. T3D
36. W1T3A

The legend in the figures refers to the figure number in NUREG/CR-4550, Vol. 4, Rev. 1, Pt. 1.

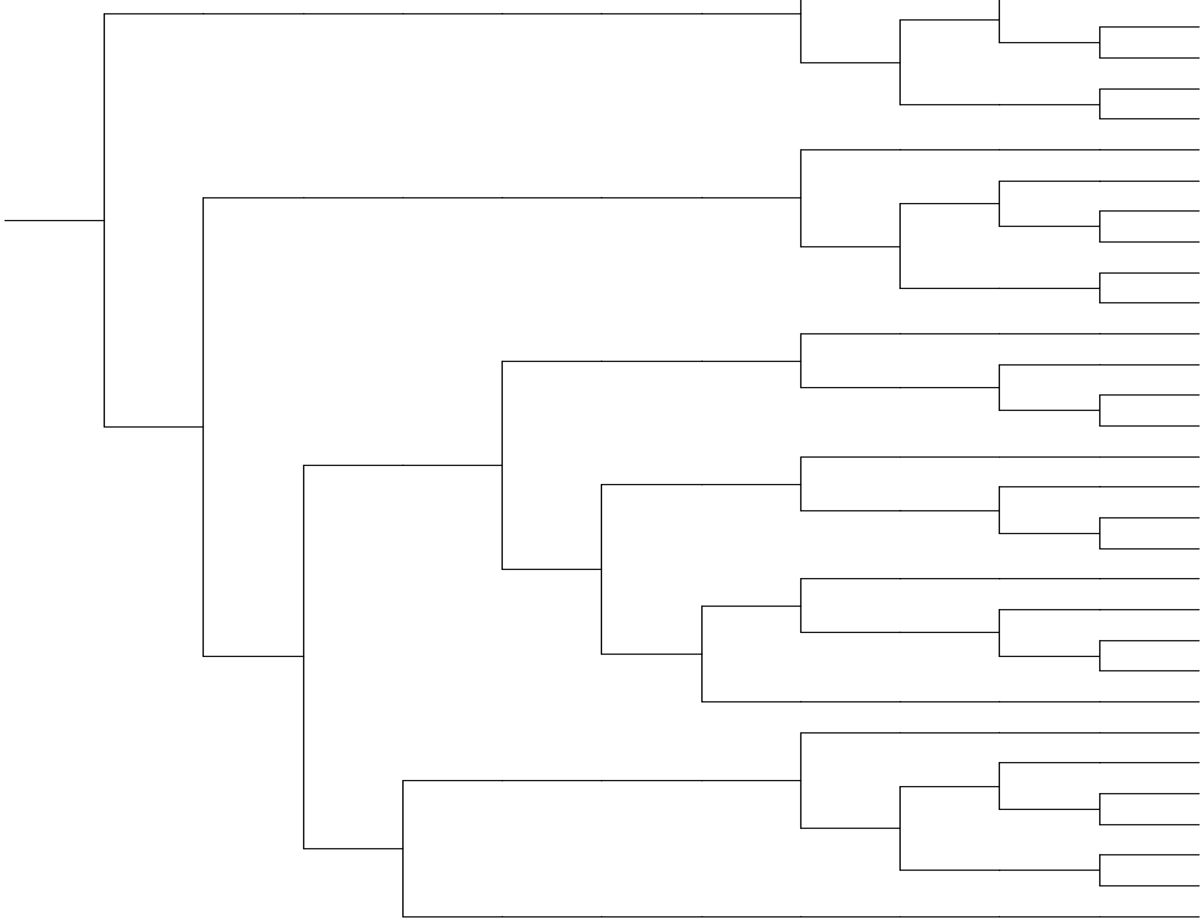
Figure 4.4-10. Peach Bottom, Unit 2
Loss of AC or DC Bus Event Tree (AC)



LOSS OF AC OR DC BUS	REACTOR PROTECTION SYSTEM	OFFSITE POWER MAINTAINED	SRVS OPEN	SRVS CLOSE				
IE-TAC/DC	C	LOSP	M	P	#	NAMES	END-STATE	TRANSFER
					1 T	AC	AC1	GO-TO-AC1
					2 T	AC-33	S2C	GO-TO-S2C
					3 T	AC-34	S1A	GO-TO-S1A
					4 T	AC-35	A1	GO-TO-A1
					5	AC-36	OK	NOT-DEVELOPED
					6 T	AC-37	T1A	GO-TO-T1A
					7 T	AC-38	ATWSAC	GO-TO-ATWSAC

SRVS CLOSE	HIGH PRESSURE COOLANT INJECTION	REACTOR CORE ISOLATION COOLING	REACTOR DEPRESS FOR CORE COOLING	CRD ENHANCED 2 PUMPS	LOW PRESSURE CORE SPRAY	LOW PRESSURE COOLANT INJECTION	HIGH PRESSURE SERVICE WATER	RESIDUAL HEAT REMOVAL: SPC MODE	REACTOR DEPRESS FOR RHR-SDC	RESIDUAL HEAT REMOVAL: SDC MODE	RESIDUAL HEAT REMOVAL: CS MODE
P	U1	U2	X1	U3	V2	V3	V4	W1	X2	W2	W3

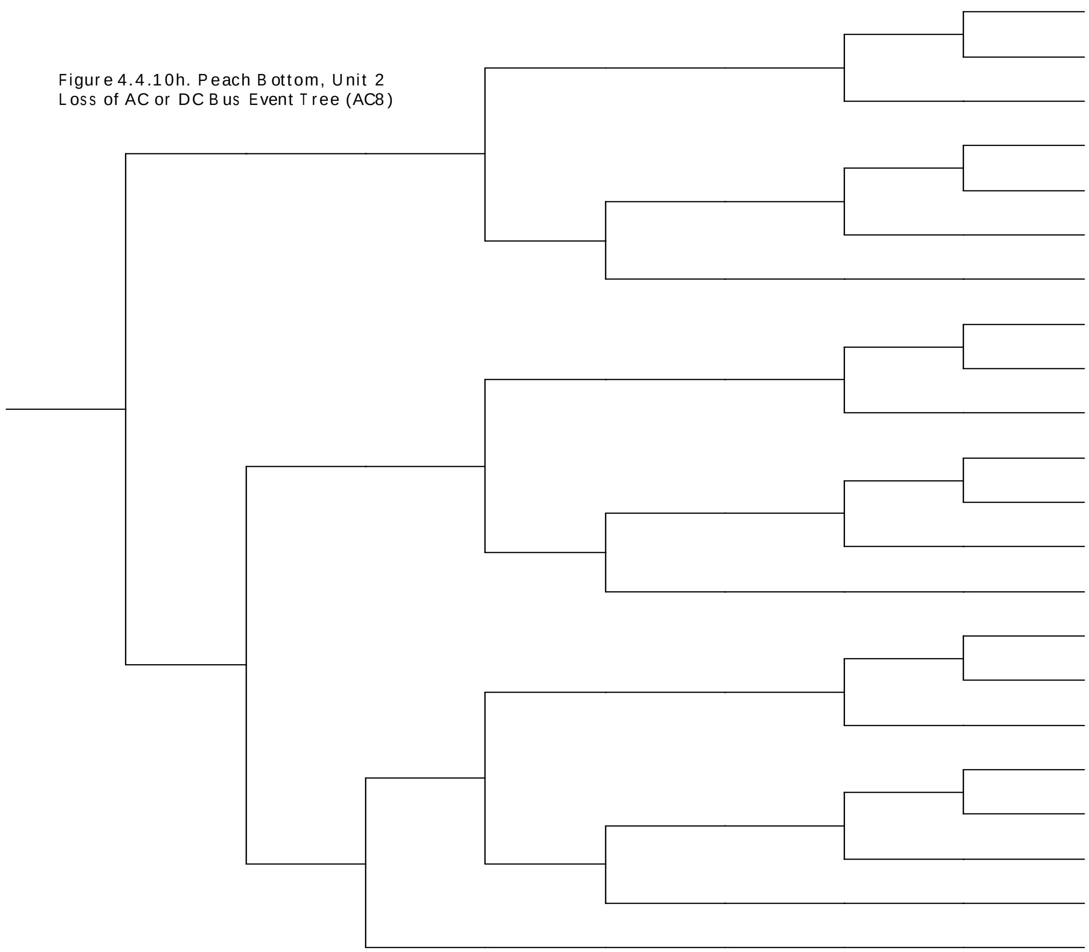
Figure 4.4-10a. Peach Bottom, Unit 2
Loss of AC or DC Bus Event Tree (AC1)



#		NAMES	END-STATE	TRANSFER
1		-1	OK	CORE+CV-OK
2		-2	OK	CORE+CV-OK
3	T	-3	AC2	GO-TO-AC2
4	T	-4	AC3	TO-TO-AC3
5	T	-5	AC4	GO-TO-AC4
6	T	-6	AC5	GO-TO-AC5
7		-7	OK	CORE+CV-OK
8		-8	OK	CORE+CV-OK
9	T	-9	AC2	GO-TO-AC2
10	T	-10	AC3	GO-TO-AC3
11	T	-11	AC4	GO-TO-AC4
12	T	-12	AC5	GO-TO-AC5
13		-13	OK	CORE+CV-OK
14		-14	OK	CORE+CV-OK
15		-15	OK	CORE+CV-OK
16	T	-16	AC6	GO-TO-AC6
17		-17	OK	CORE+CV-OK
18		-18	OK	CORE+CV-OK
19		-19	OK	CORE+CV-OK
20	T	-20	AC6	GO-TO-AC6
21		-21	OK	CORE+CV-OK
22		-22	OK	CORE+CV-OK
23		-23	OK	CORE+CV-OK
24	T	-24	AC6	GO-TO-AC6
25		-25	CD	CORE-DAM/CV-VULN
26		-26	OK	CORE+CV-OK
27	T	-27	AC7	GO-TO-AC7
28	T	-28	AC7	GO-TO-AC7
29	T	-29	AC8	GO-TO-AC8
30		-30	OK	CORE+CV-OK
31	T	-31	AC9	GO-TO-AC9
32		-32	CD	CORE-DAM/CV-VULN

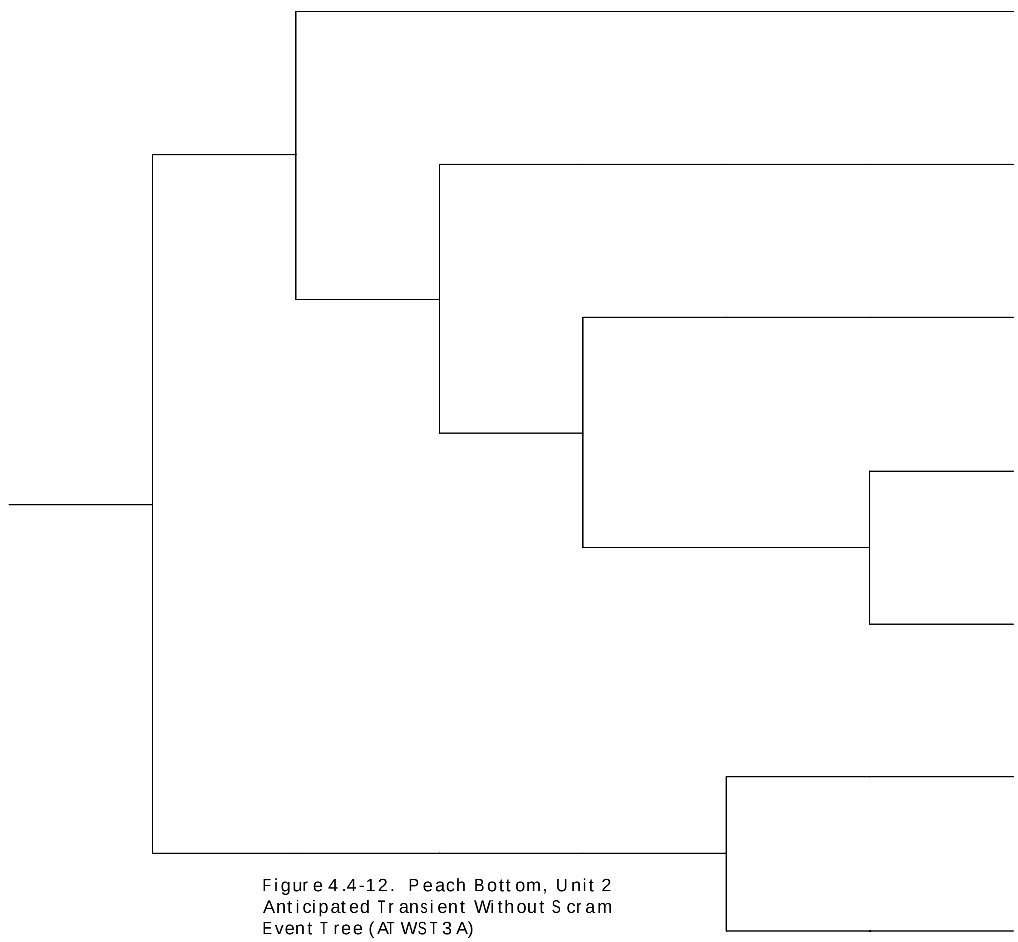
RESIDUAL HEAT REMOVAL: CS MODE	LOW PRESSURE CORE SPRAY	LOW PRESSURE COOLANT INJECTION	HIGH PRESSURE SERVICE WATER	CONTAINMENT VENTING	CONTAINMENT RUPTURES BEFORE CORE DAMAGE	CRD 1 PUMP	REACTOR DEPRESS RE-OCCURS	HIGH PRESSURE SERVICE WATER
W3	V2	V3	V4A	Y	R	U4B	X3	V4B

Figure 4.4.10h. Peach Bottom, Unit 2
Loss of AC or DC Bus Event Tree (AC8)



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	-1	OK	CV-VENT / CORE-OK
2	-2	CD	CV-VENT / CORE-DAM
3	-3	CD	CV-VENT / CORE-DAM
4	-4	OK	CV-RUPT / CORE-OK
5	-5	CD	CV-RUPT / CORE-DAM
6	-6	CD	CV-RUPT / CORE-DAM
7	-7	CD	CORE-DAM / CV-VULN
8	-8	OK	CV-VENT / CORE-OK
9	-9	CD	CV-VENT / CORE-DAM
10	-10	CD	CV-VENT / CORE-DAM
11	-11	OK	CV-RUPT / CORE-OK
12	-12	CD	CV-RUPT / CORE-DAM
13	-13	CD	CV-RUPT / CORE-DAM
14	-14	CD	CORE-DAM / CV-VULN
15	-15	OK	CV-VENT / CORE-OK
16	-16	CD	CV-VENT / CORE-DAM
17	-17	CD	CV-VENT / CORE-DAM
18	-18	OK	CV-RUPT / CORE-OK
19	-19	CD	CV-RUPT / CORE-DAM
20	-20	CD	CV-RUPT / CORE-DAM
21	-21	CD	CORE-DAM / CV-VULN
22	-22	CD	CORE-DAM / CV-VULN

REACTOR PROTECTION SYSTEM	REACTOR PROTECTION SYSTEM - MECHANICAL	REACTOR PROTECTION SYSTEM - ELECTRICAL	ALTERNATE ROD INSERTION	MANUAL SCRAM	RECIRCULATION PUMP TRIP	MANUAL ROD INSERTION
C	RPSM	RPSE	ARI	SCRM	RPT	ROD

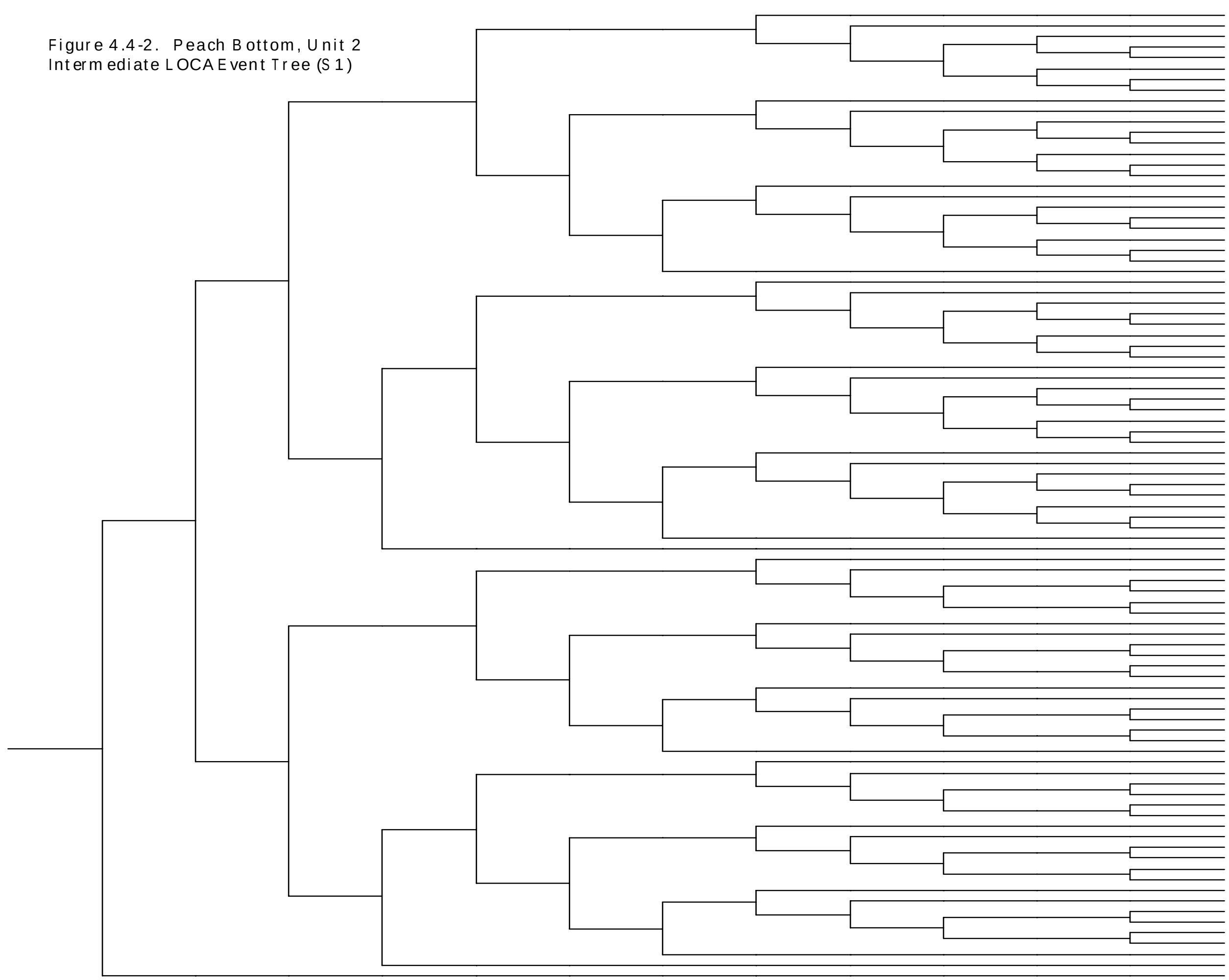


#	NAMES	END-STATE	SEQ-DESCRIPTION
1	-1	OK	TO-OTHER-TRANS
2	-2	OK	TO-OTHER-TRANS
3	-3	OK	TO-OTHER-TRANS
4	-4	OK	TO-OTHER-TRANS
5	-5	OK	NOT-DEVELOPED
6 T	-	W1T3A	GO-TO-W1T3A
7	-17	OK	NOT-DEVELOPED

Figure 4.4-12. Peach Bottom, Unit 2
Anticipated Transient Without Scram
Event Tree (ATWST3A)
Transfer from T3A

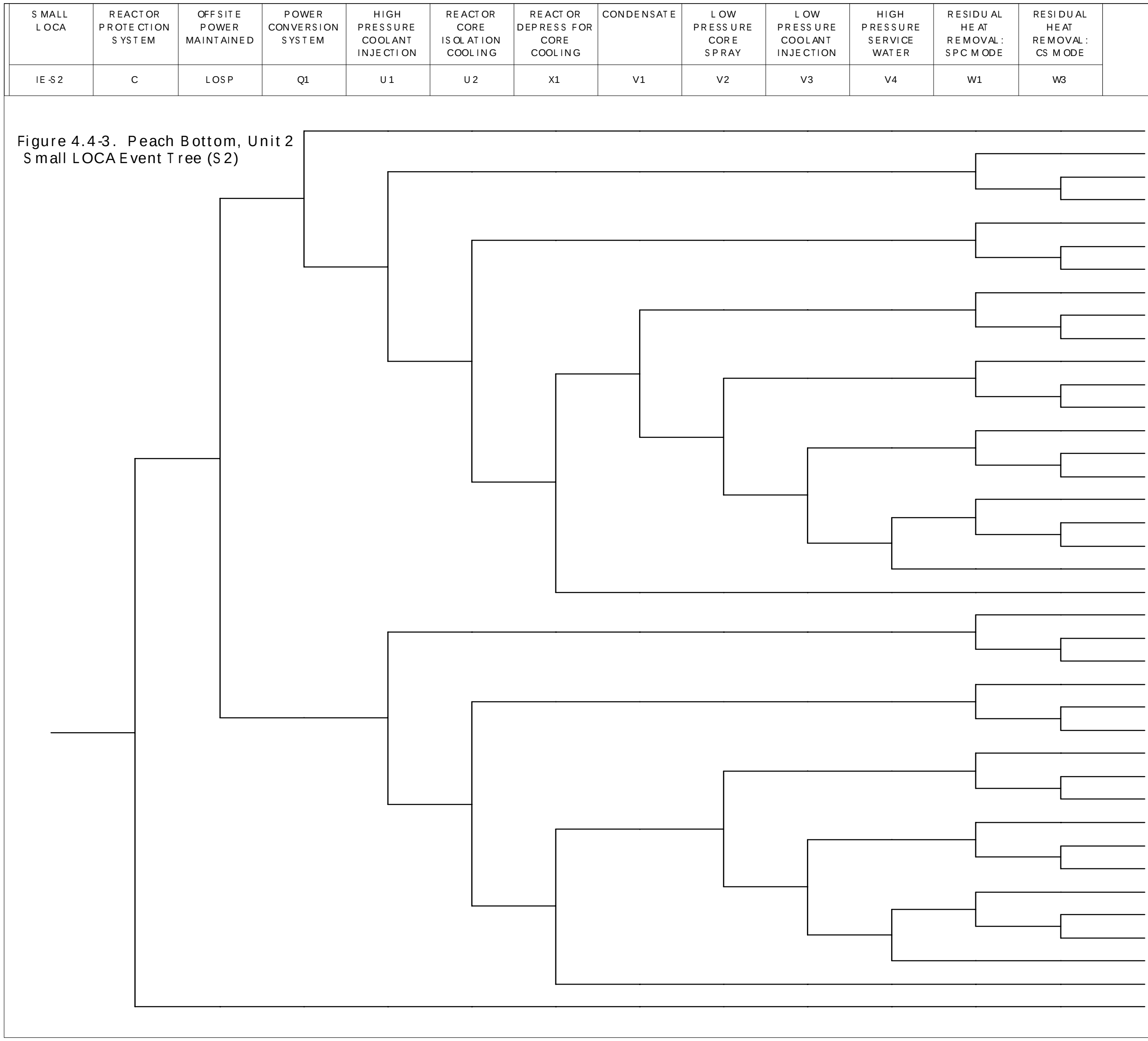
INTERMED LOCA	REACTOR PROTECTION SYSTEM	OFFSITE POWER MAINTAINED	HIGH PRESSURE COOLANT INJECTION	REACTOR DEPRESS FOR CORE COOLING	LOW PRESSURE CORE SPRAY	LOW PRESSURE COOLANT INJECTION	HIGH PRESSURE SERVICE WATER	RESIDUAL HEAT REMOVAL: SPC MODE	RESIDUAL HEAT REMOVAL: CS MODE	CONTAINMENT VENTING	CONDENSATE	HIGH PRESSURE SERVICE WATER
IE-S1	C	LOSP	U1	X1	V2	V3	V4	W1	W3	Y	V1B	V4B

Figure 4.4-2. Peach Bottom, Unit 2
Intermediate LOCA Event Tree (S1)



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	S1-1	OK	CORE+CV-OK
2	S1-2	OK	CORE+CV-OK
3	S1-3	OK	CV-VENT / CORE-OK
4	S1-4	OK	CV-VENT / CORE-OK
5	S1-5	@ CD	CV-VENT / CORE-DAM
6	S1-6	OK	CV-FAIL / CORE-OK
7	S1-7	OK	CV-FAIL / CORE-OK
8	S1-8	@ CD	CV-FAIL / CORE-DAM
9	S1-9	OK	CORE+CV-OK
10	S1-10	OK	CORE+CV-OK
11	S1-11	OK	CV-VENT / CORE-OK
12	S1-12	OK	CV-VENT / CORE-OK
13	S1-13	@ CD	CV-VENT / CORE-DAM
14	S1-14	OK	CV-FAIL / CORE-OK
15	S1-15	OK	CV-FAIL / CORE-OK
16	S1-16	@ CD	CV-FAIL / CORE-DAM
17	S1-17	OK	CORE+CV-OK
18	S1-18	OK	CORE+CV-OK
19	S1-19	OK	CV-VENT / CORE-OK
20	S1-20	OK	CV-VENT / CORE-OK
21	S1-21	@ CD	CV-VENT / CORE-DAM
22	S1-22	OK	CV-FAIL / CORE-OK
23	S1-23	OK	CV-FAIL / CORE-OK
24	S1-24	@ CD	CV-FAIL / CORE-DAM
25	S1-25	S1-V2V3V4NU11	CORE-DAM / CV-VULN
26	S1-26	OK	CORE+CV-OK
27	S1-27	OK	CORE+CV-OK
28	S1-28	OK	CV-VENT / CORE-OK
29	S1-29	OK	CV-VENT / CORE-OK
30	S1-30	@ CD	CV-VENT / CORE-DAM
31	S1-31	OK	CV-FAIL / CORE-OK
32	S1-32	OK	CV-FAIL / CORE-OK
33	S1-33	@ CD	CV-FAIL / CORE-DAM
34	S1-34	OK	CORE+CV-OK
35	S1-35	OK	CORE+CV-OK
36	S1-36	OK	CV-VENT / CORE-OK
37	S1-37	OK	CV-VENT / CORE-OK
38	S1-38	@ CD	CV-VENT / CORE-DAM
39	S1-39	OK	CV-FAIL / CORE-OK
40	S1-40	OK	CV-FAIL / CORE-OK
41	S1-41	@ CD	CV-FAIL / CORE-DAM
42	S1-42	OK	CORE+CV-OK
43	S1-43	OK	CORE+CV-OK
44	S1-44	OK	CV-VENT / CORE-OK
45	S1-45	OK	CV-VENT / CORE-OK
46	S1-46	@ CD	CV-VENT / CORE-DAM
47	S1-47	OK	CF-FAIL / CORE-OK
48	S1-48	OK	CV-FAIL / CORE-OK
49	S1-49	@ CD	CV-FAIL / CORE-DAM
50	S1-50	@ CD	CORE-DAM / CV-VULN
51	S1-51	@ CD	CORE-DAM / CV-VULN
52	S1-52	OK	CORE+CV-OK
53	S1-53	OK	CORE+CV-OK
54	S1-54	OK	CV-VENT / CORE-OK
55	S1-55	@ CD	CV-VENT / CORE-DAM
56	S1-56	OK	CV-FAIL / CORE-OK
57	S1-57	@ CD	CV-FAIL / CORE-DAM
58	S1-58	OK	CORE+CV-OK
59	S1-59	OK	CORE+CV-OK
60	S1-60	OK	CV-VENT / CORE-OK
61	S1-61	@ CD	CV-VENT / CORE-DAM
62	S1-62	OK	CV-FAIL / CORE-OK
63	S1-63	@ CD	CV-FAIL / CORE-DAM
64	S1-64	OK	CORE+CV-OK
65	S1-65	OK	CORE+CV-OK
66	S1-66	OK	CV-VENT / CORE-OK
67	S1-67	@ CD	CV-VENT / CORE-DAM
68	S1-68	OK	CV-FAIL / CORE-OK
69	S1-69	@ CD	CV-FAIL / CORE-DAM
70	S1-70	CD	CORE-DAM / CV-VULN
71	S1-71	OK	CORE+CV-OK
72	S1-72	OK	CORE+CV-OK
73	S1-73	OK	CV-VENT / CORE-OK
74	S1-74	@ CD	CV-VENT / CORE-DAM
75	S1-75	OK	CV-FAIL / CORE-OK
76	S1-76	@ CD	CV-FAIL / CORE-DAM
77	S1-77	OK	CORE+CV-OK
78	S1-78	OK	CORE+CV-OK
79	S1-79	OK	CV-VENT / CORE-OK
80	S1-80	@ CD	CV-VENT / CORE-DAM
81	S1-81	OK	CV-FAIL / CORE-OK
82	S1-82	@ CD	CV-FAIL / CORE-DAM
83	S1-83	OK	CORE+CV-OK
84	S1-84	OK	CORE+CV-OK
85	S1-85	OK	CV-VENT / CORE-OK
86	S1-86	@ CD	CV-VENT / CORE-DAM
87	S1-87	OK	CV-FAIL / CORE-OK
88	S1-88	@ CD	CV-FAIL / CORE-DAM
89	S1-89	@ CD	CORE-DAM / CV-VULN
90	S1-90	CD	CORE-DAM / CV-VULN
91	S1-91	OK	NOT-DEVELOPED

Figure 4.4-3. Peach Bottom, Unit 2 Small LOCA Event Tree (S2)



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	S2-1	OK	CORE+CV-OK
2	S2-2	OK	CORE+CV-OK
3 T	S2-3	@S21	GO-TO-S2-1
4 T	S2-4	S22	GO-TO-S2-2
5	S2-5	OK	CORE+CV-OK
6 T	S2-6	@S21	GO-TO-S2-1
7 T	S2-7	@S22	GO-TO-S2-2
8	S2-8	OK	CORE+CV-OK
9	S2-9	OK	CORE+CV-OK
10 T	S2-10	@S23	GO-TO-S2-3
11	S2-11	OK	CORE+CV-OK
12	S2-12	OK	CORE+CV-OK
13 T	S2-13	@S24	GO-TO-S2-4
14	S2-14	OK	CORE+CV-OK
15	S2-15	OK	CORE+CV-OK
16 T	S2-16	@S24	G-TO-S2-4
17	S2-17	OK	CORE+CV-OK
18	S2-18	OK	CORE+CV-OK
19 T	S2-19	@S24	GO-TO-S2-4
20	S2-20	@CD	CORE-DAM/CV-VULN
21	S2-21	@CD	CORE-DAM/CV-VULN
22	S2-22	OK	CORE+CV-OK
23 T	S2-23	@S21	GO-TO-S2-1
24 T	S2-24	@S22	GO-TO-S2-2
25	S2-25	OK	CORE+CV-OK
26 T	S2-26	@S21	GO-TO-S2-1
27 T	S2-27	@S22	GO-TO-S2-2
28	S2-28	OK	CORE+CV-OK
29	S2-29	OK	CORE+CV-OK
30 T	S2-30	@S23	GO-TO-S2-3
31	S2-31	OK	CORE+CV-OK
32	S2-32	OK	CORE+CV-OK
33 T	S2-33	@S24	GO-TO-S2-4
34	S2-34	OK	CORE+CV-OK
35	S2-35	OK	CORE+CV-OK
36 T	S2-36	@S24	GO-TO-S2-4
37	S2-37	@CD	CORE-DAM/CV-VULN
38	S2-38	CD	CORE-DAM/CV-VULN
39	S2-39	OK	NOT-DEVELOPED

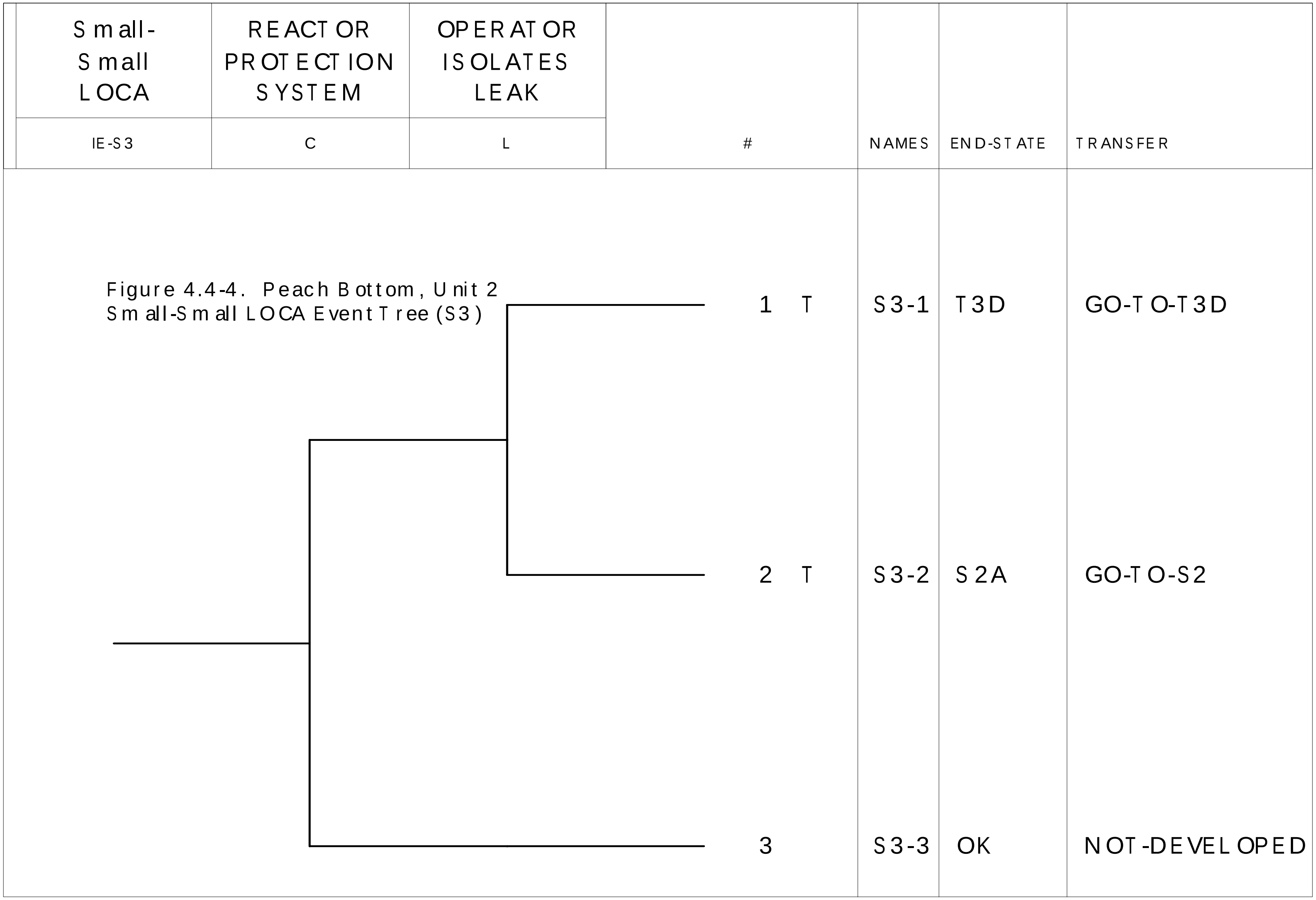
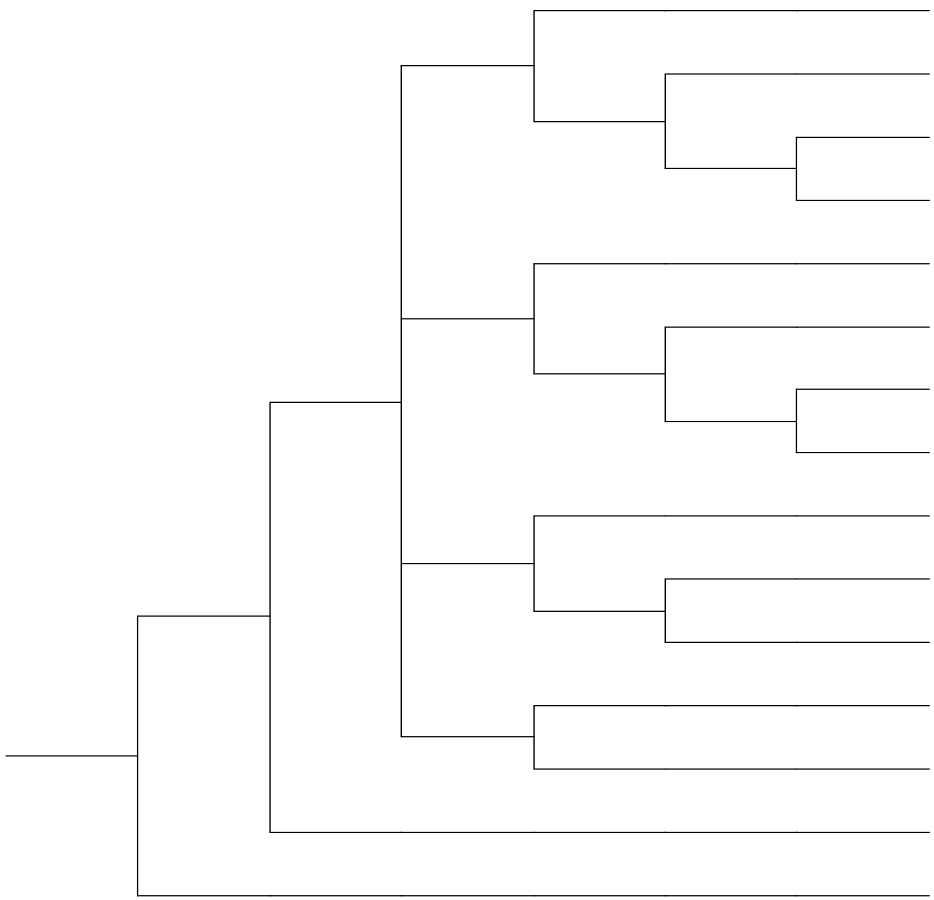


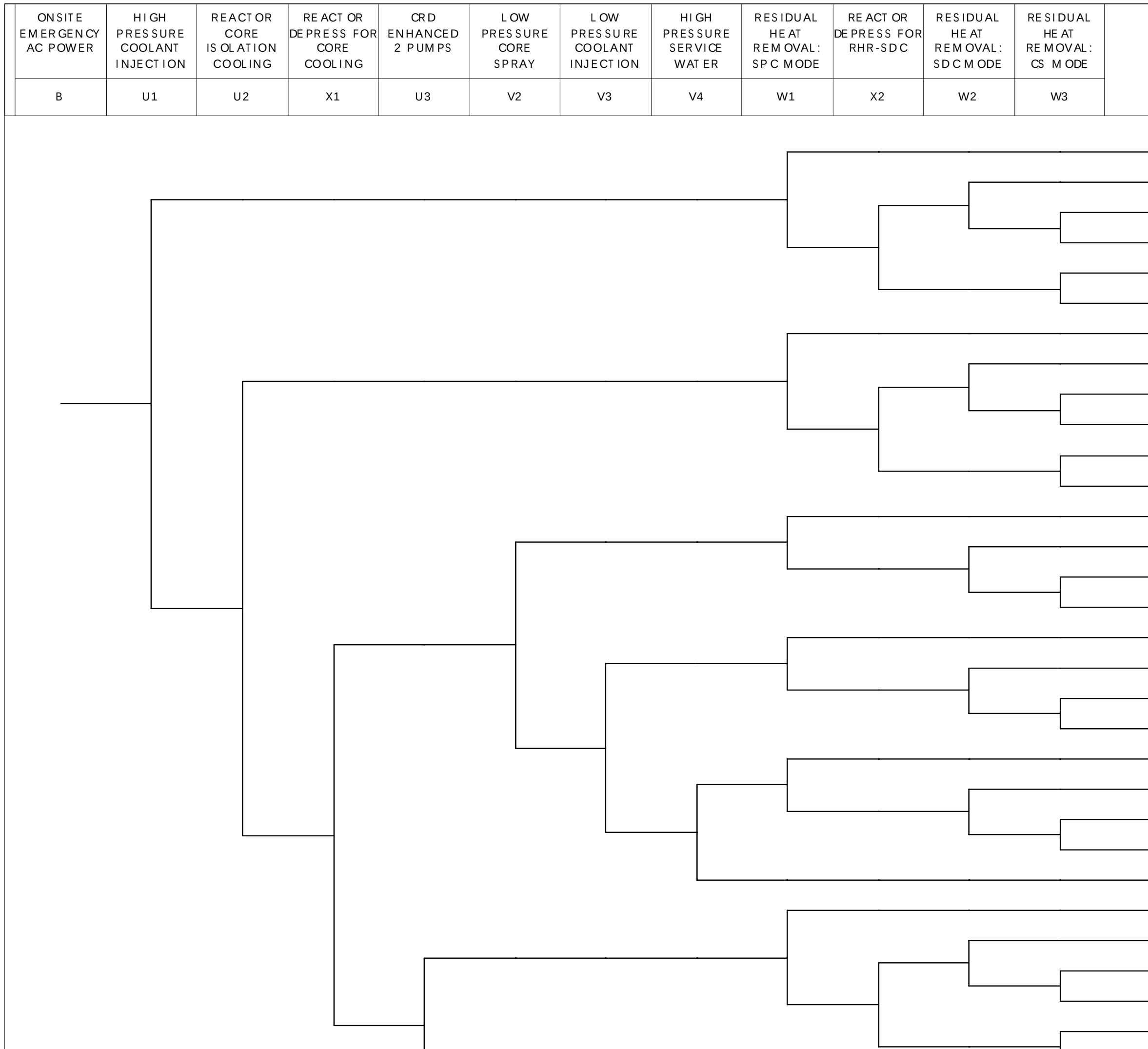
Figure 4.4-4. Peach Bottom, Unit 2 Small-Small LOCA Event Tree (S3)

LOSS OF OFFSITE POWER	REACTOR PROTECTION SYSTEM	SRVS OPEN	SRVS CLOSE	ONSITE EMERGENCY AC POWER	HIGH PRESSURE COOLANT INJECTION	REACTOR CORE ISOLATION COOLING
IE-T1	C	M	P	B	U1	U2



#	NAMES	END-STATE	SEQ-DESCRIPTION
1 T	T1	T11	GO-TO-T1-1
2	T1-33	T1-BNU11	CORE-DAM/CV-VULN
3	T1-34	T1-BU11NU21	CORE-DAM/CV-VULN
4	T1-35	T1-BU11U21	CORE-DAM/CV-VULN
5 T	T1-36	S2B	GO-TO-S2B
6	T1-37	T1-P1BNU11	CORE-DAM/CV-VULN
7	T1-38	CD	CORE-DAM/CV-VULN
8	T1-39	T1-P1BU11U21	CORE-DAM/CV-VULN
9 T	T1-40	S1B	GO-TO-S1B
10	T1-41	CD	CORE-DAM/CV-VULN
11	T1-42	CD	CORE-DAM/CV-VULN
12 T	T1-43	A2	GO-TO-A2
13	T1-44	CD	CORE-DAM/CV-VULN
14	T1-45	OK	NOT-DEVELOPED
15 T	T1-46	T1ATWS	GO-TO-T1ATWS

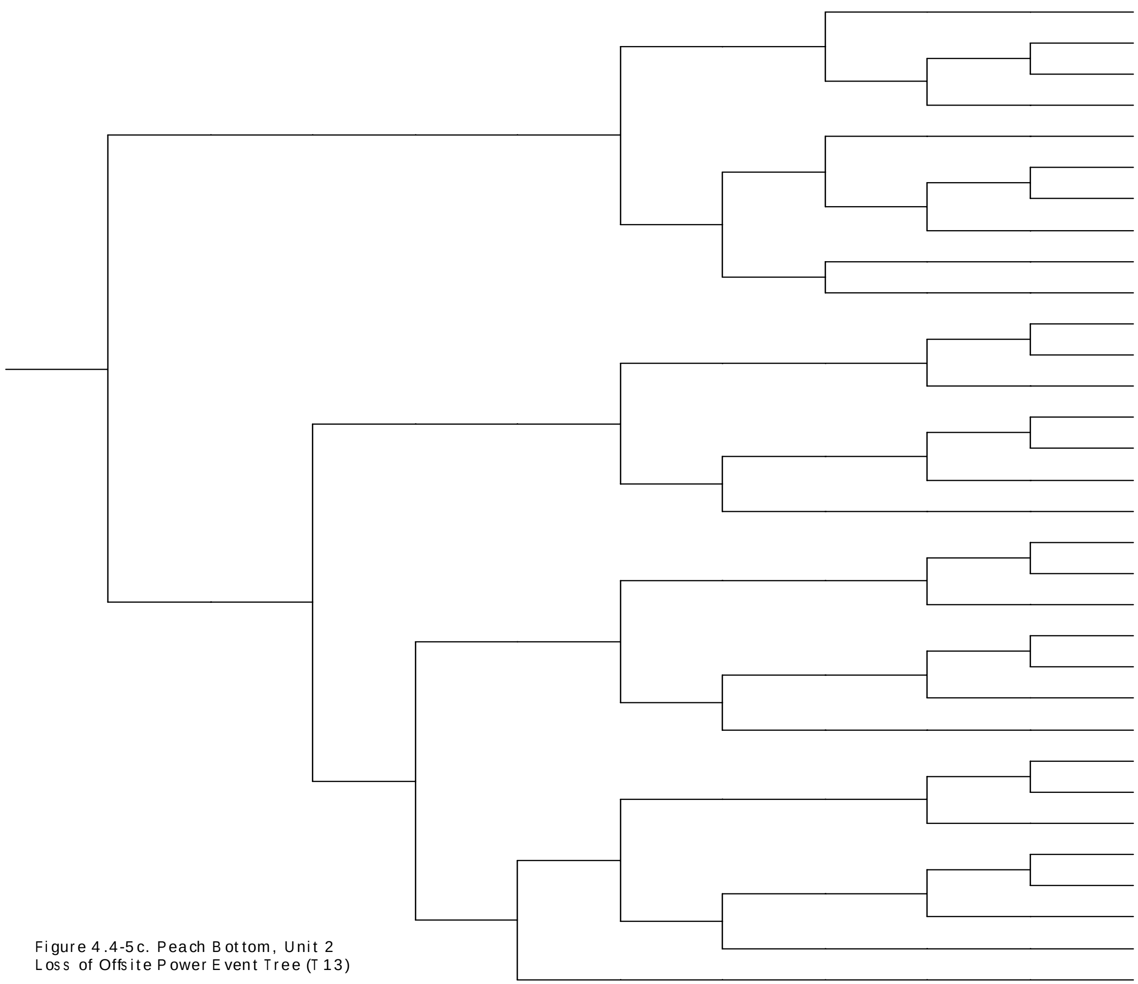
Figure 4.4-5. Peach Bottom, Unit 2
Loss of Offsite Power Event Tree (T1)



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	-1	OK	CORE+CV-OK
2	-2	OK	CORE+CV-OK
3 T	-3	@T12	CORE-VUL/TO-T1-2
4 T	-4	T13	CORE-VUL/TO-T1-3
5 T	-5	@T14	CORE-VUL/TO-T1-4
6 T	-6	@T15	CORE-VUL/TO-T1-5
7	-7	OK	CORE+CV-OK
8	-8	OK	CORE+CV-OK
9 T	-9	@T12	CORE-VUL/TO-T1-2
10 T	-10	@T13	CORE-VUL/TO-T1-3
11 T	-11	@T14	CORE-VUL/TO-T1-4
12 T	-12	@T15	CORE-VUL/TO-T1-5
13	-13	OK	CORE+CV-OK
14	-14	OK	CORE+CV-OK
15	-15	OK	CORE+CV-OK
16 T	-16	@T16	CORE-VUL/TO-T1-6
17	-17	OK	CORE+CV-OK
18	-18	OK	CORE+CV-OK
19	-19	OK	CORE+CV-OK
20 T	-20	@T16	CORE-VUL/TO-T1-6
21	-21	OK	CORE+CV-OK
22	-22	OK	CORE+CV-OK
23	-23	OK	CORE+CV-OK
24 T	-24	@T16	CORE-VUL/TO-T1-6
25	-25	@CD	CORE-DAM/CV-VULN
26	-26	OK	CORE+CV-OK
27 T	-27	@T17	CORE-VUL/TO-T1-7
28 T	-28	@T17	CORE-VUL/TO-T1-7
29 T	-29	@T18	CORE-VUL/TO-T1-8
30	-30	OK	CORE+CV-OK
31 T	-31	@T19	CORE-VUL/TO-T1-9
32	-32	@CD	CORE-DAM/CV-VULN

Figure 4.4-5a. Peach Bottom, Unit 2
Loss of Offsite Power Event Tree (T11)

RESIDUAL HEAT REMOVAL: CS MODE	CRD 1 PUMP	REACTOR DEPRESS FOR CORE COOLING	LOW PRESSURE CORE SPRAY	LOW PRESSURE COOLANT INJECTION	HIGH PRESSURE SERVICE WATER	CONTAINMENT VENTING	CONTAINMENT RUPTURES BEFORE CORE DAMAGE	CRD 1 PUMP	REACTOR DEPRESS RE-OCCURS	HIGH PRESSURE SERVICE WATER
W3	U4	X1	V2	V3	V4A	Y	R	U4B	X3	V4B



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	-1	OK	CV-VENT / CORE-OK
2	-2	OK	CV-VENT / CORE-OK
3	-3	@ CD	CV-VENT / CORE-DAM
4	-4	@ CD	CV-VENT / CORE-DAM
5	-5	OK	CV-RUPT / CORE-OK
6	-6	OK	CV-RUPT / CORE-OK
7	-7	@ CD	CV-RUPT / CORE-DAM
8	-8	@ CD	CV-RUPT / CORE-DAM
9	-9	OK	CV-LEAK / CORE-OK
10	-10	@ CD	CORE-DAM / CV-VULN
11	-11	OK	CV-VENT / CORE-OK
12	-12	@ CD	CV-VENT / CORE-DAM
13	-13	@ CD	CV-VENT / CORE-DAM
14	-14	CD	CV-RUPT / CORE-OK
15	-15	@ CD	CV-RUPT / CORE-DAM
16	-16	@ CD	CV-RUPT / CORE-DAM
17	-17	@ CD	CORE-DAM / CV-VULN
18	-18	OK	CV-VENT / CORE-OK
19	-19	@ CD	CV-VENT / CORE-DAM
20	-20	@ CD	CV-VENT / CORE-DAM
21	-21	CD	CV-RUPT / CORE-OK
22	-22	@ CD	CV-RUPT / CORE-DAM
23	-23	@ CD	CV-RUPT / CORE-DAM
24	-24	@ CD	CORE-DAM / CV-VULN
25	-25	OK	CV-VENT / CORE-OK
26	-26	@ CD	CV-VENT / CORE-DAM
27	-27	@ CD	CV-VENT / CORE-DAM
28	-28	OK	CV-RUPT / CORE-OK
29	-29	@ CD	CV-RUPT / CORE-DAM
30	-30	@ CD	CV-RUPT / CORE-DAM
31	-31	@ CD	CORE-DAM / CV-VULN
32	-32	@ CD	CORE-DAM / CV-VULN

Figure 4.4-5c. Peach Bottom, Unit 2
Loss of Offsite Power Event Tree (T13)

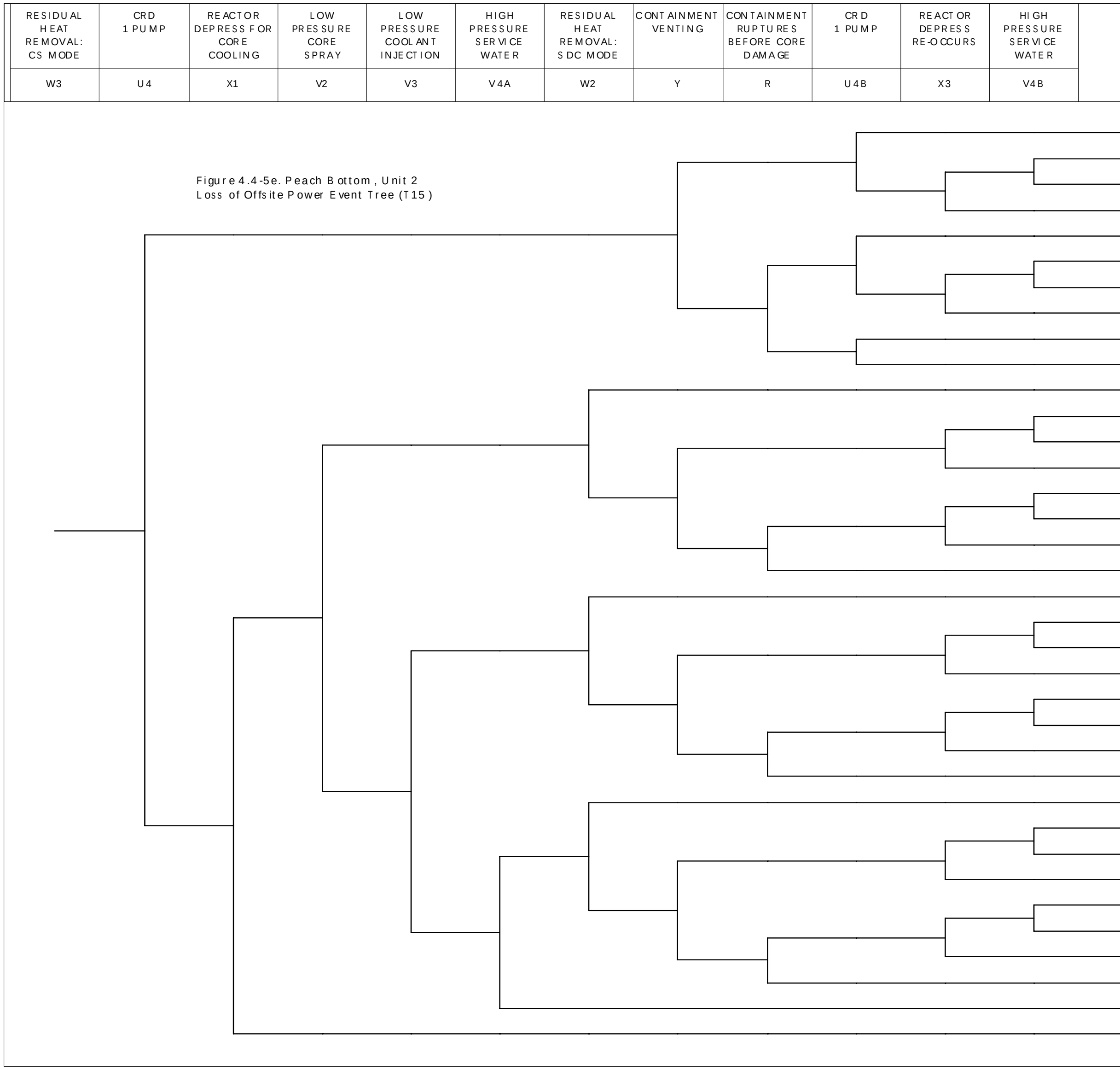


Figure 4.4-5e. Peach Bottom, Unit 2
Loss of Offsite Power Event Tree (T15)

#	NAMES	END-STATE	SEQ-DESCRIPTION
1	-1	OK	CV-VENT / CORE-OK
2	-2	OK	CV-VENT / CORE-OK
3	-3	CD	CV-VENT / CORE-DAM
4	-4	CD	CV-VENT / CORE-DAM
5	-5	OK	CV-RUPT / CORE-OK
6	-6	OK	CV-RUPT / CORE-OK
7	-7	CD	CV-RUPT / CORE-DAM
8	-8	CD	CV-RUPT / CORE-DAM
9	-9	OK	CV-LEAK / CORE-OK
10	-10	CD	CORE-DAM / CV-VULN
11	-11	OK	CORE+CV-OK
12	-12	OK	CV-VENT / CORE-OK
13	-13	CD	CV-VENT / CORE-DAM
14	-14	CD	CV-VENT / CORE-DAM
15	-15	OK	CV-RUPT / CORE-OK
16	-16	CD	CV-RUPT / CORE-DAM
17	-17	CD	CV-RUPT / CORE-DAM
18	-18	CD	CORE-DAM / CV-VULN
19	-19	OK	CORE+CV-OK
20	-20	OK	CV-VENT / CORE-OK
21	-21	CD	CV-VENT / CORE-DAM
22	-22	CD	CV-VENT / CORE-DAM
23	-23	OK	CV-RUPT / CORE-OK
24	-24	CD	CV-RUPT / CORE-DAM
25	-25	CD	CV-RUPT / CORE-DAM
26	-26	CD	CORE-DAM / CV-VULN
27	-27	OK	CORE+CV-OK
28	-28	OK	CV-VENT / CORE-OK
29	-29	CD	CV-VENT / CORE-DAM
30	-30	CD	CV-VENT / CORE-DAM
31	-31	OK	CV-RUPT / CORE-OK
32	-32	CD	CV-RUPT / CORE-DAM
33	-33	CD	CV-RUPT / CORE-DAM
34	-34	CD	CORE-DAM / CV-VULN
35	-35	CD	CORE-DAM / CV-VULN
36	-36	CD	CORE-DAM / CV-VULN

SRVS CLOSE	HIGH PRESSURE COOLANT INJECTION	REACTOR CORE ISOLATION COOLING	REACTOR DEPRESS FOR CORE COOLING	CONDENSATE	CRD ENHANCED 2 PUMPS	LOW PRESSURE CORE SPRAY	LOW PRESSURE COOLANT INJECTION	HIGH PRESSURE SERVICE WATER	RESIDUAL HEAT REMOVAL: SPC MODE	REACTOR DEPRESS FOR RHR-SDC	RESIDUAL HEAT REMOVAL: SDC MODE	RESIDUAL HEAT REMOVAL: CS MODE
P	U1	U2	X1	V1	U3	V2	V3	V4	W1	X2	W2	W3

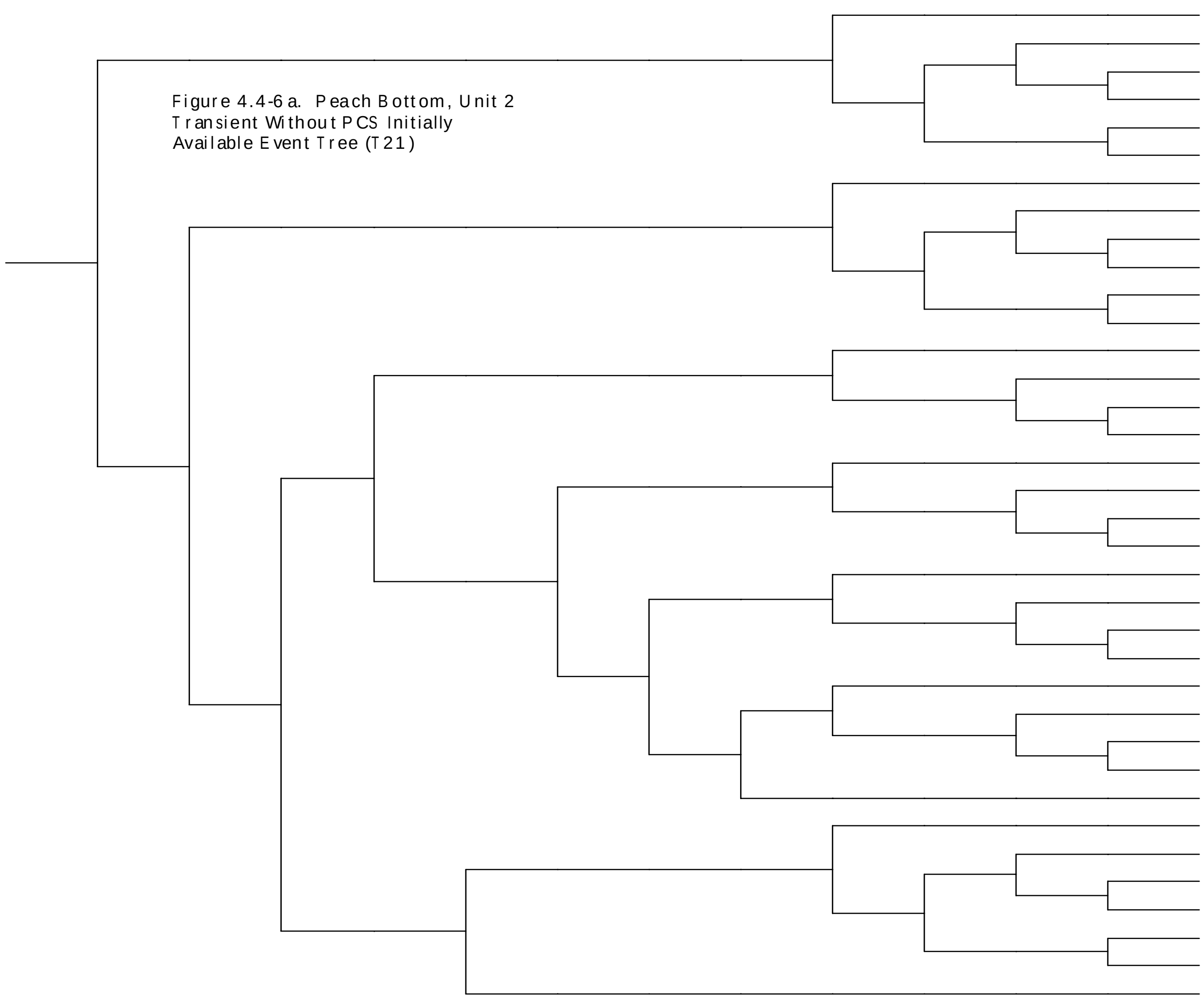
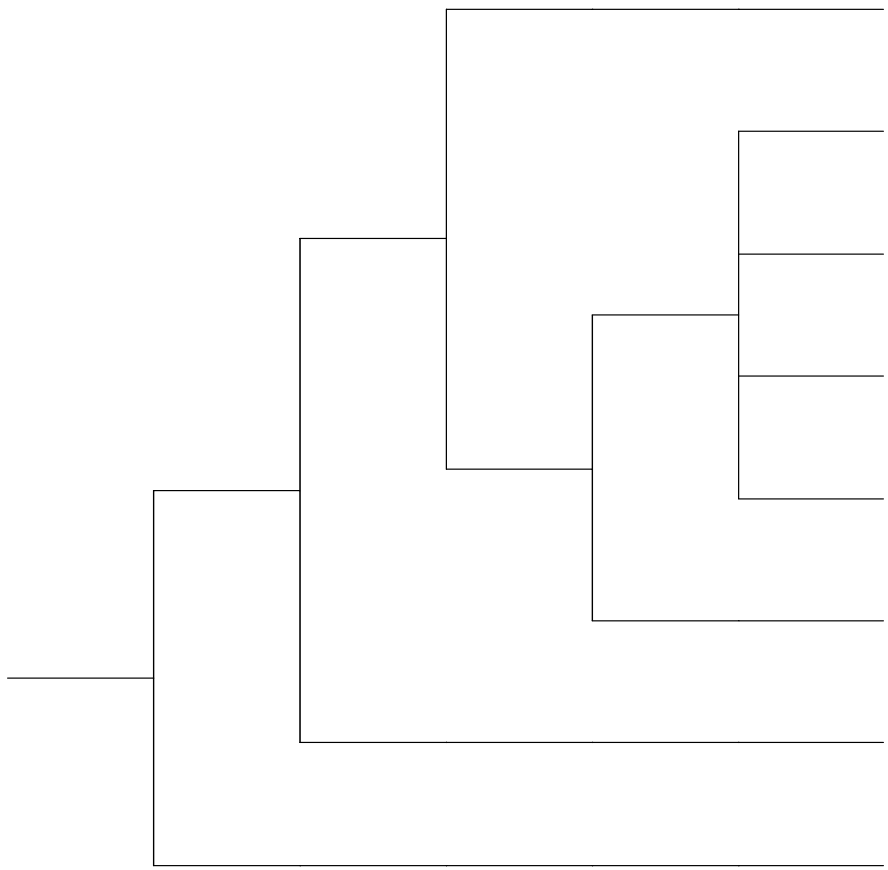


Figure 4.4-6a. Peach Bottom, Unit 2 Transient Without PCS Initially Available Event Tree (T21)

#		NAMES	END-STATE	SEQ-DESCRIPTION
1		-1	OK	CORE+CV-OK
2		-2	OK	CORE+CV-OK
3	T	-3	T22	CORE-VUL/TO-T22
4	T	-4	T23	CORE-VUL/TO-T23
5	T	-5	T24	COR-VUL/TO-T24
6	T	-6	T25	CORE-VUL/TO-T25
7		-7	OK	CORE+CV-OK
8		-8	OK	CORE+CV-OK
9	T	-9	T22	CORE-VUL/TO-T22
10	T	-10	T23	CORE-VUL/TO-T23
11	T	-11	T24	CORE-VUL/TO-T24
12	T	-12	T25	CORE-VUL/TO-T25
13		-13	OK	CORE+CV-OK
14		-14	OK	CORE+CV-OK
15		-15	OK	CORE+CV-OK
16	T	-16	T26	CORE-VUL/TO-T26
17		-17	OK	CORE+CV-OK
18		-18	OK	CORE+CV-OK
19		-19	OK	CORE+CV-OK
20	T	-20	T27	CORE-VUL/TO-T27
21		-21	OK	CORE+CV-OK
22		-22	OK	CORE+CV-OK
23		-23	OK	CORE+CV-OK
24	T	-24	T27	CORE-VUL/TO-T27
25		-25	OK	CORE+CV-OK
26		-26	OK	CORE+CV-OK
27		-27	OK	CORE+CV-OK
28	T	-28	T27	CORE-VUL/TO-T27
29		-29	CD	CORE-DAM/CV-VULN
30		-30	OK	CORE+CV-OK
31	T	-31	T28	CORE-VUL/TO-T28
32	T	-32	T28	CORE-VUL/TO-T28
33	T	-33	T29	CORE-VUL/TO-T29
34		-34	OK	CORE+CV-OK
35	T	-35	T210	CORE-VU/TO-T210
36		-36	CD	CORE-DAM/CV-VULN

Transient With PCS Initially Available	REACTOR PROTECTION SYSTEM	OFFSITE POWER MAINTAINED	POWER CONVERSION SYSTEM	SRVS OPEN	SRVS CLOSE
IE-T3A	C	LOSP	Q	M	P



#		NAMES	END-STATE	SEQ-DESCRIPTION
1		T3A-37	OK	CORE+CV-OK
2	T	T3A	T21	GO-TO-T21
3	T	T3A-38	S2C	GO-TO-S2C
4	T	T3A-39	S1A3A	GO-TO-S1A3A
5	T	T3A-40	A1	GO-TO-A1
6		T3A-41	OK	NOT-DEVELOPED
7	T	T3A-42	T1A	GO-TO-T1A
8	T	T3A-43	ATWST3A	GO-TO-ATWST3A

Figure 4.4-7. Peach Bottom, Unit 2
Transient With PCS Initially Available Event Tree (T3A)

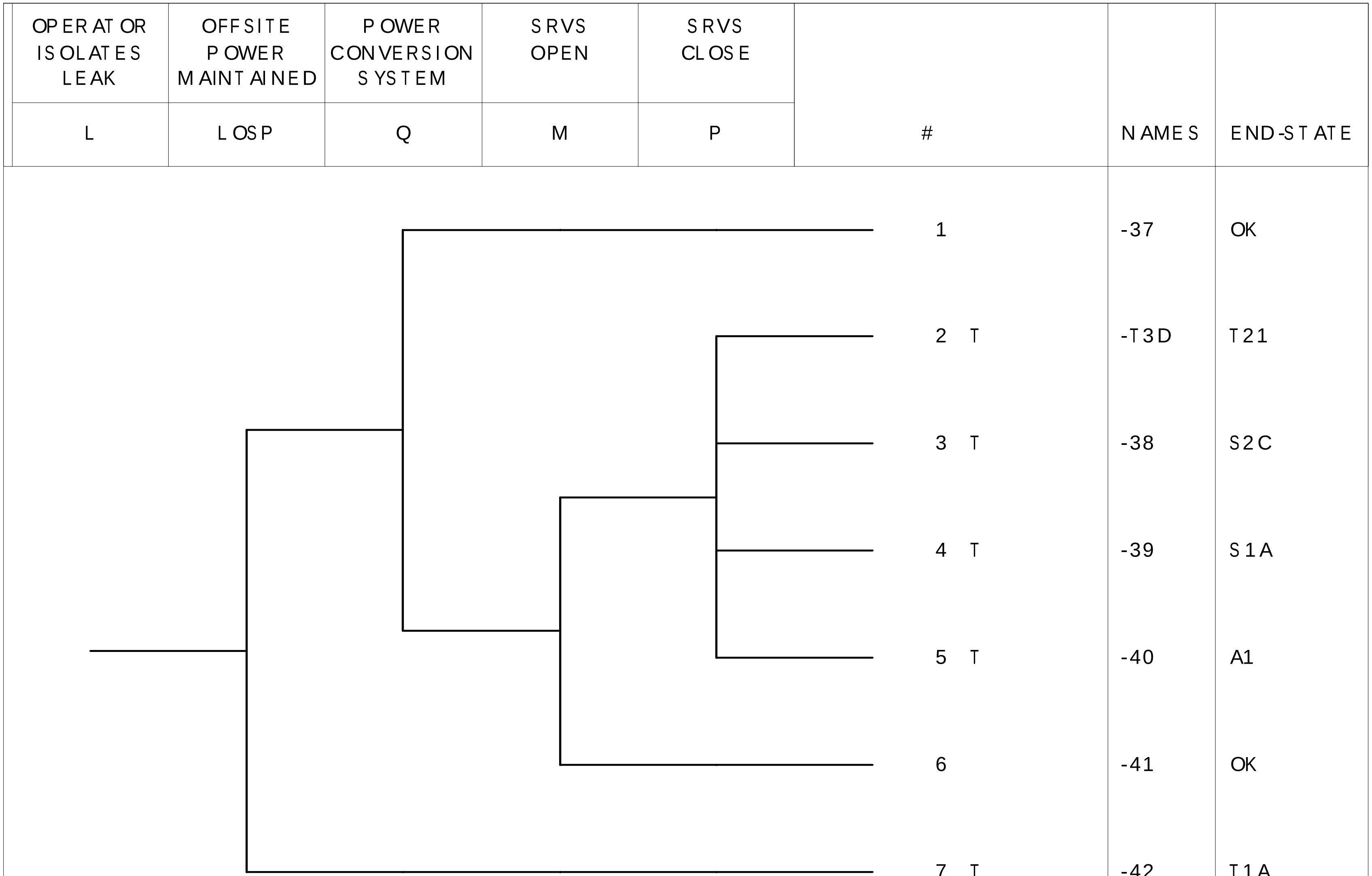
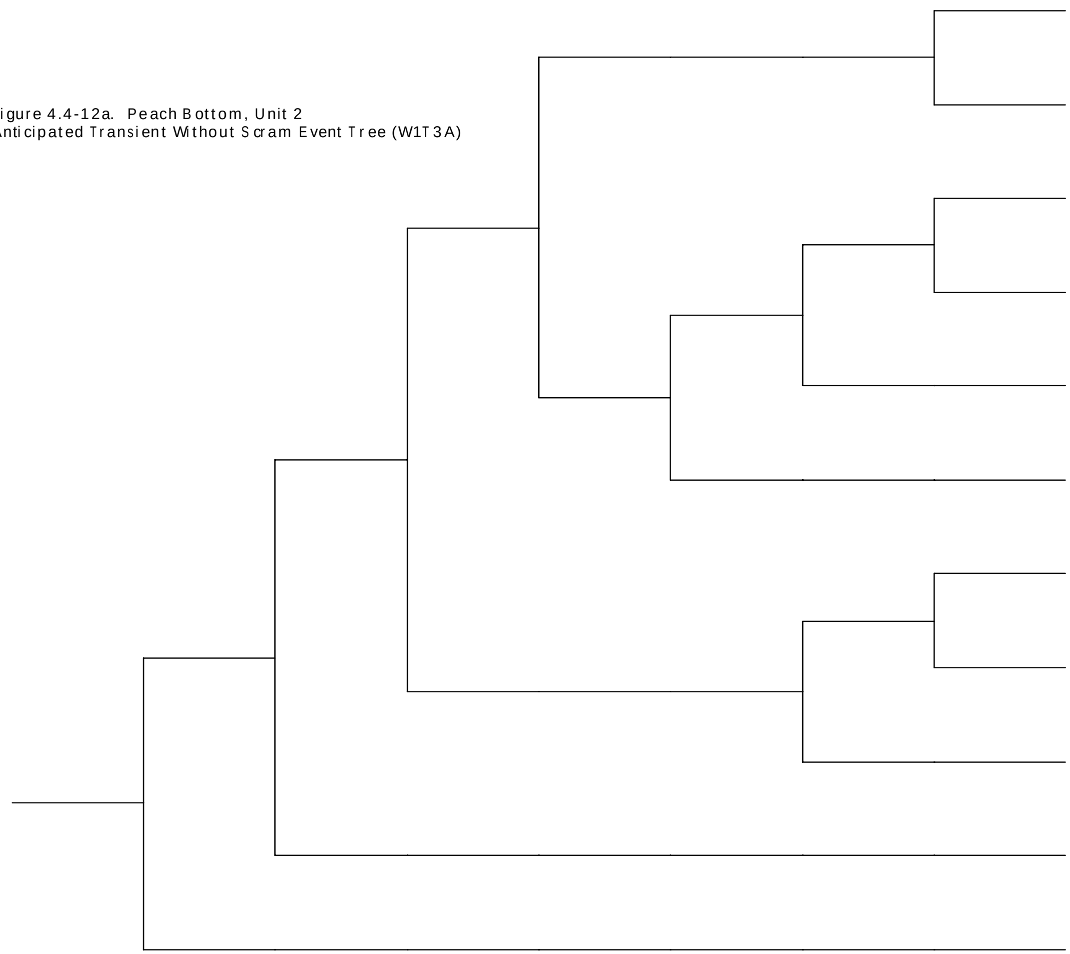


Figure 4.4-7a. Peach Bottom, Unit 2
Transient With PCS Initially Available Event Tree (T3D)
Transfer from S3 Event Tree

Figure 4.4-12a. Peach Bottom, Unit 2
Anticipated Transient Without Scram Event Tree (W1T3A)

RECIRCULATION PUMP TRIP	SRVS OPEN	STANDBY LIQUID CONTROL	ADS INHIBITED	HIGH PRESSURE COOLANT INJECTION	REACTOR DEPRESS FOR CORE COOLING	LOW PRESSURE CORE COOLING	RHR (SPC OR CS)
RPT	M	SLC	I	U1	X1	V2	W1



#	NAMES	END-STATE	SEQ-DESCRIPTION
1	6	OK	CORE+CV-OK
2	7	CD	CORE-VULNERABLE
3	8	OK	CORE+CV-OK
4	9	CD	CORE-VULNERABLE
5	10	CD	CORE-DAM/CV-VULN
6	11	T3A-CU11X	CORE-DAM/CV-VULN
7	12	OK	CORE+CV-OK
8	13	CD	CORE-VULNERABLE
9	14	CD	CORE-DAM/CV-VULN
10	15	T3A-C-SLC	CORE-DAMAGE
11	16	OK	NOT-DEVELOPED