DCP/NRC1548
Docket No. 52-006

February 6, 2003

Attachment 3

APP-GW-GL-022, Revision 1

"AP1000 Probabilistic Risk Assessment"

AP1000 Probabilistic Risk Assessment

Instructions for Inserting Revision 1 Change Pages and Tabs

(January 2003)

Volume 5

An additional volume – Volume 5, complete with binder inserts and title page – has been provided due to an increase in the PRA material from Revision 1. Before adding the new Revision 1 pages, please follow the steps below for shifting the PRA material among the volumes:

Step 1. Remove all of the contents from Volume 4 except the Volume 4 Title Page (this includes the entire Table of Contents, and Appendixes A through F with tabs), and insert the Volume 4 contents into Volume 5 behind the Volume 5 Title Page.

Step 2. Remove Chapter 49 through Chapter 59 (including tabs) from Volume 3 and insert the Chapters into Volume 4 after the Volume 4 Title Page.

Step 3. Insert the Revision 1 pages as described in the following paragraphs.

Revision 1 Pages

Volume 1

Remove the Revision No. 0 "AP1000 Document Cover Sheet" from the front of the volume, and insert the Revision No. 1 "AP1000 Document Cover Sheet."

Then after the AP1000 Volume 1 Title Page (color), remove the following pages and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
i (Rev. 0)/ii (Rev. 0) through li (Rev. 0)/Blank	i (Rev. 1)/ii (Rev. 1) through ciii (Rev. 1)/civ (Rev. 1)

Volume 1 (Cont.)

After Tab 1 "Introduction," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
1-1 (Rev. 0)/1-2 (Rev. 0) through 1-5 (Rev. 0)/1-6 (Rev. 0)	1-1 (Rev. 1)/1-2 (Rev. 1) through 1-5 (Rev. 1)/1-6 (Rev. 1)

After Tab 3 "Modeling of Special Initiators," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
3-1 (Rev. 0)/3-2 (Rev. 0) through 3-15 (Rev. 0)/Blank	3-1 (Rev. 1)/3-2 (Rev. 1) through 3-15 (Rev. 1)/Blank

After Tab 4 "Event Tree Models," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
4-1 (Rev. 0)/4-2 (Rev. 0) through 4-271 (Rev. 0)/Blank	4-1 (Rev. 1)/4-2 (Rev. 1) through 4B-109 (Rev. 3)/4B-110 (Rev. 3)

After Tab 6, "Success Criteria Analysis," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
6-1 (Rev. 0)/6-2 (Rev. 0) through 6-71 (Rev. 0)/Blank	6-1 (Rev. 1)/6-2 (Rev. 1) through 6-71 (Rev. 1)/Blank

After Tab 8 "PXS – Passive Residual Heat Removal," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
8-1 (Rev. 0)/8-2 (Rev. 0) through 8-25 (Rev. 0)/Blank	8-1 (Rev. 1)/8-2 (Rev. 1) through 8-25 (Rev. 1)/Blank

Volume 1 (Cont.)

After Tab 12, "PXS – In-Containment Refueling Water Storage Tank," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
12-1 (Rev. 0)/12-2 (Rev. 0) through 12-41 (Rev. 0)/12-42 (Rev. 0)	12-1 (Rev. 1)/12-2 (Rev. 1) through 12-41 (Rev. 1)/12-42 (Rev. 1)

Volume 2

After the AP1000 Volume 2 Title Page (color), remove the following pages and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
i (Rev. 0)/ii (Rev. 0) through li (Rev. 0)/Blank	i (Rev. 1)/ii (Rev. 1) through ciii (Rev. 1)/civ (Rev. 1)

After Tab 26 "Protection and Safety Monitoring System," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
26-1 (Rev. 0)/26-2 (Rev. 0) through 26-229 (Rev. 0)/Blank	26-1 (Rev. 1)/26-2 (Rev. 1) through 26-229 (Rev. 1)/Blank

After Tab 28 "Plant Control System," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
28-1 (Rev.0)/28-2 (Rev. 0) through 28-133 (Rev. 0)/28-134 (Rev. 0)	28-1 (Rev. 1)/28-2 (Rev. 1) through 28-133 (Rev. 1)/28-134 (Rev. 1)

Volume 2 (Cont.)

After Tab 29 "Common Cause Analysis," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
29-1(Rev. 0)/29-2 (Rev. 0) through 29A-13 (Rev. 0)/Blank	29-1 (Rev. 1)/29-2 (Rev. 1) through 29A-13 (Rev. 1)/Blank

After Tab 30 "Human Reliability Analysis," remove all of the pages within the tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
30-1 (Rev. 0)/30-2 (Rev. 0) through 30A-17 (Rev. 0)/30A-18 (Rev. 0)	30-1 (Rev. 1)/30-2 (Rev. 1) through 30A-17 (Rev. 1)/30A-18 (Rev. 1)

Volume 3

After the AP1000 Volume 3 Title Page (color), remove the following pages and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
i (Rev. 0)/ii (Rev. 0) through li (Rev. 0)/Blank	i (Rev. 1)/ii (Rev. 1) through ciii (Rev. 1)/civ (Rev. 1)

After Tab 34 "Severe Accident Phenomena Treatment," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
34-1 (Rev. 0)/34-2 (Rev. 0) through 34-7 (Rev. 0)/34-8 (Rev. 0)	34-1 (Rev. 1)/34-2 (Rev. 1) through 34-243 (Rev. 1)/Blank

Volume 3 (Cont.)

After Tab 35 "Containment Event Tree Analysis," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
35-1 (Rev. 0)/35-2 (Rev. 0) through 35-29 (Rev. 0)/Blank	35-1 (Rev. 1)/35-2 (Rev. 1) through 35-29 (Rev. 1)/Blank

After Tab 39 "In-Vessel Retention of Molten Core Debris," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
39-1 (Rev. 0)/39-2 (Rev. 0) through 39-41 (Rev. 0)/Blank	39-1 (Rev. 1)/39-2 (Rev. 1) through 39A-39 (Rev. 1)/39A-40 (Rev. 1)

After Tab 41 "Hydrogen Mixing and Combustion Analysis," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
41-1 (Rev. 0)/41-2 (Rev. 0) through 41-57 (Rev. 0)	41-1 (Rev. 1)/41-2 (Rev. 1) through 41B-11 (Rev. 1)/(Blank)

After Tab 43 "Release Frequency Quantification," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
43-1 (Rev. 0)/43-2 (Rev. 0) through 43C-1 (Rev. 0)/43C-2 (Rev. 0)	43-1 (Rev. 1)/43-2 (Rev. 1) through 43C-1 (Rev. 1)/43C-2 (Rev. 1)

After Tab 44 "MAAP4 Code Description and AP1000 Modeling," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
44-1 (Rev. 0)/44-2 (Rev. 0) through 44-13 (Rev. 0)/Blank	44-1 (Rev. 1)/44-2 (Rev. 1) through 44-13 (Rev. 1)

Volume 3 (Cont.)

After Tab 45 "Fission-Product Source Terms," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
45-1 (Rev. 0)/45-2 (Rev. 0) through 45-5 (Rev. 0)/45-6 (Rev. 0)	45-1 (Rev. 1)/45-2 (Rev. 1) through 45-47 (Rev. 1)/45-48 (Rev. 1)

Volume 4

After the AP1000 Volume 4 Title Page (color), remove the following pages and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
i (Rev. 0)/ii (Rev. 0) through li (Rev. 0)/Blank	i (Rev. 1)/ii (Rev. 1) through ciii (Rev. 1)/civ (Rev. 1)

After Tab 49 "Offsite Dose Risk Quantification," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
49-1 (Rev. 0)/49-2 (Rev. 0) through 49-49 (Rev. 0)/Blank	49-1 (Rev. 1)/49-2 (Rev. 1) through 49-49 (Rev. 1)/Blank

After Tab 50 "Importance and Sensitivity Analysis," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
50-1 (Rev. 0)/50-2 (Rev. 0) through 50-119 (Rev. 0)/Blank	50-1 (Rev. 1)/50-2 (Rev. 1) through 50-119 (Rev. 1)/Blank

Volume 4 (Cont.)

After Tab 57 "Fire Risk Assessment," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
57-1 (Rev. 0)/57-2 (Rev. 0)	57-1 (Rev. 1)/57-2 (Rev. 1) through 57D-7 (Rev. 1)/Blank

After Tab 59 "PRA Results and Insights," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
59-1 (Rev. 0)/59-2 (Rev. 0) through 59-73 (Rev. 0)/59-74 (Rev. 0)	59-1 (Rev. 1)/59-2 (Rev. 1) through 59-99 (Rev. 1)/59-100 (Rev. 1)

Volume 5

After the AP1000 Volume 5 Title Page (color), remove the following pages and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
i (Rev. 0)/ii (Rev. 0) through li (Rev. 0)/Blank	i (Rev. 1)/ii (Rev. 1) through ciii (Rev. 1)/civ (Rev. 1)

After Tab "Appendix A," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
A-1 (Rev. 0)/A-2 (Rev. 0) through A-103 (Rev. 0)/A-104 (Rev. 0)	A-1 (Rev. 1)/A-2 (Rev. 1) through A-165 (Rev. 1)/A-166 (Rev. 1)

Volume 5 (Cont.)

After Tab "Appendix B," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
B-1 (Rev. 0)/B-2 (Rev. 0) through B-25 (Rev. 0)/Blank	B-1 (Rev. 1)/B-2 (Rev. 1) through B-25 (Rev. 1)/Blank

After Tab "Appendix D," remove all of the pages within this tab and insert the Revision 1 pages as follows:

Remove Pages	Insert Pages
D-1 (Rev. 0)/D-2 (Rev. 0) through D-35 (Rev. 0)/D-36 (Rev. 0)	D-1 (Rev. 1)/D-2 (Rev. 1) through D-57 (Rev. 1)/D-58 (Rev. 1)

AP1000 DOCUMENT COVER SHEET

TDC. _____ Permanent File _____ S _____

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AP1000 DOCUMENT NO.	REVISION NO		ASSIGNED TO
APP-GW-GL-022	1	Page 1 of 1	W – R. P. Vijuk

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #:

ORIGINATING ORGANIZATION. Westinghouse Electric Company LLC

TITLE: **AP1000 Probabilistic Risk Assessment**

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE: N/A	Class 3 revision to correct input to latest available (DCP APP-GW-GEE-00), to expand documentation per RAI requests, and to add AP1000 specific analyses

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION

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Probabilistic Risk Assessment, Revision 1

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57D	57D-1 through 57D-7	Letter DCP/NRC1515
59	59-20 and 59-21	Editorial
59	59-23 through 59-25	Letter DCP/NRC1515
59	59-26	720.056
59	59-28	Editorial
59	59-29 through 59-32	720.038
59	59-34 through 59-38	720.038
59	59-72	Editorial
59	59-73	Technical
59	59-83	Editorial
59	59-75 through 59-96	720.038
59	59-97	720.038 720.066
59	59-98	720.038
59	59-100	Technical
Appendix A	A-1 through A-166	720.007 720.010 720.011 720.012 720.013 720.015 720.016

REVISION 1 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change[1]
Appendix B	B-6	720.058
Appendix B	B-13	720.076
Appendix D	D-14 and D-15	720.078
Appendix D	D-31	720.078
Appendix D	D-35 and D-36	Technical
Appendix D	D-37 through D-58	720.078

1. Changes incorporated as a result of Westinghouse responses to NRC Request for Additional Information (RAI) identified by RAI number. RAI number in parenthesis contains a reference to RAI response listed above.

CHAPTER 1

INTRODUCTION

1.1 Introduction

Part 52 of the 10 Code of Federal Regulations requires that a probabilistic risk assessment (PRA) be submitted as a part of an application for design certification. The PRA provides an evaluation of the design, including plant, containment, and typical site analyses that consider both internal and external events.

The AP1000 design process includes a risk assessment of the design prior to being finalized to optimize the plant with respect to safety. Westinghouse accomplishes this by committing to the early application of probabilistic analysis techniques in the AP1000 design process. This work resulted in information used in the selection of design alternatives, with a goal that the overall level of safety of the completed design exceeds design objectives.

1.1.1 Background and Overview

The AP1000 PRA was developed to support the application for Design Certification of the AP1000 nuclear plant. The AP1000 design is based extensively on the AP600 standard nuclear plant that received Design Certification in December 1999. The AP600 PRA, which was reviewed by the US NRC in detail during the seven-year review of the AP600, is used as the starting point for the AP1000 PRA. Since the configuration of the AP1000 reactor and safety systems is the same as the AP600, the AP600 PRA is used as the basis of the AP1000 PRA with relevant changes implemented in the model to reflect the AP1000 design changes. AP1000 plant-specific T&H analyses are performed in order to determine the system success criteria. The core damage frequency and large release frequency are calculated for internal events. The external events and shutdown models are also assessed to derive plant insights and plant risk conclusions.

The purpose of the PRA is to provide inputs to the optimization of the AP1000 design and to verify that the US NRC PRA safety goals have been satisfied. As in the AP600, the PRA is being performed interactively with the design, analysis and operating procedures. The PRA results show that there are only minor impacts on the PRA results compared to AP600, and that the very low risk of the AP600 has been maintained in the AP1000; the AP1000 PRA meets the US NRC safety goals with significant margin. Insights from the analysis are provided discussing the effect on the PRA of differences between the AP600 and the AP1000 designs.

1.1.2 Objectives

The objectives of the AP1000 PRA are to:

- Provide an integrated view of the AP1000 behavior in response to transients and accidents, including severe accidents

- Satisfy the NRC regulatory requirements that a design-specific PRA be conducted as part of the application for design certification (10 CFR 52.47(a)(i)(v))

- Demonstrate that the design meets the proposed safety goals for core damage frequency and large fission product releases

- Construct a PRA Level 1 (core damage frequency), Level 2 (large release frequency), and Level 3 (offsite dose) model that is consistent with the AP1000 design configuration and operation requirements and the ALWR URD requirements on PRA methodology (Reference 1.1-1)

- Demonstrate the low vulnerability and insensitivity of the AP1000 design to human interaction

- Provide input to the design process (that is, provide a tool to investigate detailed design solutions and operational strategies to optimize AP1000 safety)

- Demonstrate compliance with the hydrogen control criteria set forth in 10 CFR 50.34(f)(2)(ix)

- Serve as a basis for an accident management program

1.1.3 Technical Scope

The technical tasks for the AP1000 PRA are defined in the following categories:

- Level 1 Analysis for Internal Events
- Level 2 Analysis for Internal Events
- Level 3 Analysis for Internal Events
- Sensitivity, Importance, and Uncertainty Analyses for Internal Events
- Shutdown Risk Assessment
- External Events Risk Assessment

The ALWR URD document serves as the base document to define the source of data.

The Level 1 analysis includes:

- Internal initiating events evaluation
- Event tree and success criteria analyses
- Plant systems analysis using fault tree models
- Common cause failure and human reliability analyses
- Data analysis
- Fault tree and event tree quantification to calculate the core damage frequency

The Level 2 analysis includes:

- An evaluation of severe accident phenomena and fission product source terms
- Modeling of the containment event tree and associated success criteria
- Analysis of hydrogen burning and mixing

The Level 3 analysis is an offsite dose evaluation.

The low power and shutdown analysis includes Level 1 shutdown assessment.

External events analyses include:

- Internal fire assessment
- Internal flooding assessment
- Seismic margin assessment

1.1.4 Project Methodology Overview

Guidelines have been developed for the major tasks. These guidelines provide homogeneity among similar tasks that are performed by different analysts (such as fault tree construction) and to standardize the methodology for selected tasks (such as human reliability or common cause failure analysis).

The major activities performed during this study include:

- Initiating event and event tree analysis – Evaluations are performed to identify a comprehensive set of initiating events. This evaluation includes review of pressurized water reactor (PWR) operating experience, past PRAs, and consideration of AP1000-specific features. For each initiating event category, an event tree is constructed to model the accident sequences that may result.

- Success criteria – Extensive analyses are performed with MAAP4 (Reference 1.1-2), NOTRUMP, and other codes to determine the success criteria for system mitigation following initiating events.

- Analysis of individual systems – Qualitative analysis and fault tree construction are performed for safety-related and nonsafety-related front-line systems and supporting systems that contribute to prevention or mitigation of severe accident events. The analysis identifies the importance of each component for each system.

- Human reliability analysis – A detailed human reliability analysis is performed, with emphasis on the evaluation of the effect of single operator decisions on more than one system.

- Common cause failure analysis – An analysis is performed to identify and model the dependencies (common cause failures), both internal to individual systems and among systems, that use similar components exposed to similar environments.

- Severe accident analysis – Analyses are performed with the MAAP4 code to study the progression of severe accident sequences and to define the radionuclide source terms.
- Dose evaluation – The dose at the plant site boundary for the various fission product release categories are calculated.
- Hydrogen control analysis – Analyses to demonstrate the effectiveness of the hydrogen igniters are carried out using the MAAP4 code.
- Shutdown assessment – The frequency of core damage is assessed for low power and shutdown conditions.
- Fire and flood assessment – Internal fire and internal flood risk assessment evaluate potential vulnerabilities within the plant.
- Seismic margin assessment – Seismic margin methodology is used to identify potential seismic vulnerabilities and to assess the margin beyond the design-level safe shutdown earthquake.
- Assembly of results – The frequency of the dose at the site boundary exceeding a certain level is obtained by combining the results of the core damage analysis, severe accident analysis, and dose analysis.

1.1.5 Results

The AP1000 PRA is an integrated view of the AP1000 behavior in response to transients and accidents, including severe accidents.

The AP1000 core damage frequency for internal events from at-power conditions is extremely low. The core damage frequency calculated for internal events at shutdown conditions is also very low. The combined core damage frequency from internal events at power and at shutdown conditions meets the NRC and URD safety goals with substantial margin.

The AP1000 large release frequency of the dose at the site boundary exceeding 1 rem effective dose equivalent in 24 hours after core damage for internal events from at-power conditions is very low. Like the core damage frequency, the combined large release frequency from internal events at power and at shutdown conditions meets the NRC safety goals with substantial margin.

In the AP600 licensing process, an initial set of sensitivity analyses were made to assess the importance of non-safety related systems. Later on, this exercise grew into a full-fledged PRA model which was named the focused PRA. The focused PRA was performed to assess the importance of the nonsafety-related systems. The results of the focused PRA (Reference 1.1-3) demonstrated that the AP600 passive plant design was able to meet the NRC safety goals crediting only safety-related equipment, with no credit for any of the nonsafety-related systems. To resolve the regulatory treatment of nonsafety-related systems,

Westinghouse and the NRC agreed to availability controls of selected nonsafety-related systems for the purposes of providing defense-in-depth as well as investment protection.

The AP1000 PRA demonstrates a very similar low risk profile for the AP1000 as for the AP600. Sensitivity studies performed for the AP1000 demonstrates that no nonsafety-related system is of high risk importance. The same nonsafety-related system availability controls adopted for the AP600 will be applied to the AP1000 for the purpose of providing defense-in-depth and investment protection and are discussed in Design Control Document (DCD) Section 16.3.

There are no critical operator actions in the AP1000 PRA analyses. The core damage frequency remains relatively small even if all operator actions are assumed to fail. Only a small improvement in the core damage frequency can be realized by improving the reliability of the plant operators.

The AP1000 containment is capable of providing an effective barrier to the release of fission-products to the environment and includes effective hydrogen control measures. The AP1000 design meets the criteria in 10 CFR 50.34(f)(2)(ix).

These results demonstrate that the AP1000 meets and exceeds the design goals specified in Section 1.1.2.

Insights regarding the AP1000, derived from or verified by this PRA, include:

- Passive safety-related systems eliminates the dependence of safety-related system operation on ac electric power and compressed air. This significantly reduces the core damage frequency resulting from a loss of offsite power or station blackout event.

- Reactor coolant pump seal loss-of-coolant accidents are eliminated because of the use of canned motor reactor coolant pumps.

- Simplified passive safety-related systems reduce the need for, and importance of, operator action.

- The analysis shows that many of the events, which in the past, were leading contributors to the risk of nuclear power plants, are not as significant for the AP1000. The contribution of interfacing systems loss-of-coolant accidents, which are typically the highest risk severe accident sequences, is made insignificant by the design of the AP1000.

- The ability to flood the reactor cavity is an important contributor to maintaining a low release frequency for AP1000. This feature and the design of the reactor insulation that provides for cooling of the reactor vessel keeps a damaged core inside the reactor vessel. This reduces the potential for ex-vessel severe accident events.

- The AP1000 design provides a passive means of maintaining the containment integrity by removing decay heat from the containment with water on the containment shell or

through air cooling. This cooling ability reduces the potential of containment failure due to overpressurization after severe accident.

- The AP1000 containment design enhances the deposition of aerosols before they are released to the environment and reduces the potential environmental effects of a severe accident that has failed the containment.

1.1.6 Plant Definition

1.1.6.1 General Description

See AP1000 DCD Chapter 1.

1.1.6.2 AP1000 Design Improvement as a Result of Probabilistic Risk Assessment Studies

Design improvements were incorporated in the AP600 design based on the results of the AP600 PRA and other design analyses and are discussed in Reference 1.1-3. These improvements have been retained in the AP1000 design. Additional design changes have been incorporated in the AP1000 as a result of the AP1000 PRA. The most significant design changes prompted by the AP1000 PRA are:

- Two recirculation lines, each containing a motor-operated valve and a squib valve or a check valve and a squib valve in series, are used to provide recirculation flow from containment sump to the core through direct vessel injection line. Diversity is provided in the actuation by using diverse squib valves. The motor-operated valve is designed so that it remains open in case of failure.

- Three parallel supply lines allow water flow from PCCWST to the containment shell. Diversity is provided in the actuation by using motor-operated valves for one path.

1.1.7 References

1.1-1 Advanced Light Water Reactor Requirements Document, Volume III, Appendix A to Chapter 1, "PRA Key Assumptions and Groundrules," Revisions 5 and 6, December 1993.

1.1-2 EPRI MAAP 4.0 Users Manual.

1.1-3 AP600 PRA Report GW-GL-022.

CHAPTER 3

MODELING OF SPECIAL INITIATORS

3.1 Introduction

This chapter provides the models used to quantify the "special" initiating event frequencies. The treatment of special initiating events in the AP1000 PRA is the same as the treatment of initiating events in the AP600 PRA. Special initiating events are those events that represent a system failure which result in a reactor trip and affect the performance of the front-line systems. The special initiators evaluated for AP1000, as defined in Chapter 2, include:

- Loss of component cooling water system (CCS)/service water system (SWS)
- Loss of compressed and instrument air system (CAS)
- Spurious actuation of automatic depressurization system (ADS)

The evaluation of these special initiators is provided in this chapter.

3.2 Ground Rules for Special Initiating Event Analysis

Fault tree models are used to evaluate the yearly frequency of occurrence for these special initiating events. Fault tree models represent system failure during normal plant operation. In modeling the special initiator fault trees, the following ground rules are used:

- In the system fault tree modeling normal plant operation, the mission time adopted for the operating train is 8760 hours (1 year), while the mission time of the standby train is 24 hours. Train A is assumed to be the normally operating train during the year and is substituted, in case of failure, by train B for 24 hours. Twenty-four hours is the time period estimated as necessary to repair the failed train.

- Common cause failures of components are based on a 24-hour mission time. This approach for common cause failures reflects the actual length of common operating time or the common number of demands for components in the same common cause group.

- The special initiator fault trees are quantified using the fault tree linking process. The initiating event frequencies calculated during quantification are reported in this chapter.

- Subtrees that relate to the instrumentation and control systems are provided in Chapters 26 and 28.

The method for fault tree construction and quantification of each special initiator is presented in the subsections that follow.

3.3 Loss of Component Cooling Water System/Service Water System Initiating Event

The loss of component cooling water system/service water system special initiating event consists of the complete loss of the component cooling water system or service water system during normal plant operations due to either random failure or operator error.

Loss of the component cooling water system leads to an initiating event, since it causes the loss of the reactor coolant pumps, which results in a reactor trip. Loss of the service water system leads to an initiating event because it causes loss of the component cooling system and thus, leads to a reactor trip. Since the plant response to the loss of service water system or the component cooling water system is the same, the two initiating events are combined.

The fault tree model for the component cooling water system/service water system initiating event is labeled CSWF. The CSWF fault tree consists of two branches under an "OR" gate. One branch models the failure of the component cooling water system, and the other models the failure of the service water system. The branch of the CSWF fault tree representing failure of the component cooling water system during normal plant operation is the CCN fault tree presented in Chapter 18. The branch of the CSWF fault tree representing failure of the service water system during normal plant operation is the SWN fault tree presented in Chapter 19. The assumptions associated with the CCN and SWN fault trees are discussed in Chapters 18 and 19, respectively.

The success criteria of the CSWF fault tree is summarized in Table 3-1. The top logic event of the CSWF fault tree representing the loss of component cooling water/service water systems special initiating event can be found in Appendix E.

The yearly frequency of the component cooling water system/service water system initiating event was quantified during the fault tree linking stage of quantification. The calculated initiating event frequency is 1.44E-01 per year.

3.4 Loss of Compressed and Instrument Air System Initiating Event

Complete loss of the compressed and instrument air system could result in loss of the main feedwater system due to the closure of the main feedwater flow control valves. Main feedwater flow control valve closure will initiate a reactor and turbine trip due to a low steam generator level. Therefore, loss of the compressed and instrument air system during normal plant operation is considered a special initiator.

The fault tree model for the compressed and instrument air system initiating event is labeled CASF. The CASF fault tree discussed in this chapter is the same as the CASF fault tree described in Chapter 25. The assumptions associated with the fault tree are also discussed in Chapter 25.

The yearly frequency of the compressed and instrument air system initiating event was quantified during the fault tree linking stage of quantification. The calculated initiating event frequency is 3.48E-02 per year.

3.5 Spurious Actuation of Automatic Depressurization System

Depending on the number of automatic depressurization system lines spuriously opening, the frequency of automatic depressurization system spurious actuation contributes to either the medium or large loss-of-coolant accident (LOCA) initiating event frequency. Table 3-2 presents a list of different automatic depressurization system spurious line opening combinations and

which loss-of-coolant accident category these failure scenarios belong. The evaluation of the automatic depressurization system spurious actuation is provided in the following subsections.

The automatic depressurization system is described and modeled for system mitigation functions following the occurrence of an initiating event in Chapter 11.

3.5.1 Evaluation of Automatic Depressurization System Spurious Actuation

Spurious actuation of the automatic depressurization system can occur because of catastrophic hardware failure of the automatic depressurization system valves, leakage of the automatic depressurization system valves, spurious actuation by the protection and safety monitoring system (PMS) and by the diverse actuation system, or an operator error that causes the system to actuate. Each of these types of spurious actuations is discussed below.

3.5.1.1 Catastrophic Hardware Failure of Automatic Depressurization System Valves

- Catastrophic Hardware Failure of Automatic Depressurization System Valves During Plant Operation

  The automatic depressurization system consists of four stages of valves that open sequentially during the depressurization process. Stages 1, 2, and 3 are arranged into two identical groups. Each group has a common header from the top of the pressurizer and a common discharge line to one of the spargers in the in-containment refueling water storage tank (IRWST). Each of these first three stages consists of two lines, each with two motor-operated valves in series that are both normally closed. The first-stage valves are 4-in. valves. The second- and third-stage valves are 8-in. valves. A failure of both valves in a line must occur to initiate a spurious automatic depressurization system event. Spurious reactor coolant system depressurization caused by failure of the automatic depressurization system motor-operated valves is postulated to occur if the gate or disc of the valve ruptures, thereby allowing unrestricted fluid flow. Failures of the stage 1, 2, and 3 valves due to catastrophic hardware failure during normal plant operation are included in the frequency analysis.

  Stage 4 of the automatic depressurization system is arranged into two identical groups. Each group has a common header connected to one of the hot legs and discharges directly to containment. Stage 4 consists of four lines. Each line is arranged with a 14-in. normally open motor-operated isolation valve in series with a 14-in. normally closed explosively actuated (squib) valve. The squib valve is designed such that a solid metal cap closes off the flow path. An explosive charge is provided that, when actuated, can rupture the metal cap. The valve is actuated by the protection and safety monitoring system (automatically and manually) and by the diverse actuation system (manually). Failure of the metal cap is similar to failure of a pipe segment; thus, failure of a fourth-stage squib valve is like a pipe rupture. Pipe ruptures of the automatic depressurization system, including the pipe up to the squib valve, are considered in the calculation of the primary system pipe break frequency, which is discussed in Section 2.3. Therefore, catastrophic failure of the automatic depressurization system fourth-stage lines is limited to either operator error or spurious actuation via the protection and safety monitoring system and diverse actuation system.

- Catastrophic Hardware Failure of Automatic Depressurization System Valves During Inservice Testing

The twelve motor-operated valves of the first three stages of the automatic depressurization system are stroke tested every cold shutdown. This failure probability is conservatively modeled as an at-power special initiator. During the test, one of the control room operators activates the automatic depressurization system in-service soft control mode, which sets up an interlock so that two motor-operated valves in series along an automatic depressurization system line cannot be simultaneously opened by manual controls. If one of the motor-operated valves has a catastrophic failure during the interval between tests, such as breaking of the valve stem or a disk rupture, then an automatic depressurization system spurious actuation is initiated when the other motor-operated valve in series with the failed valve is opened for the test. Catastrophic valve failures are those that lead to a reactor trip and a safety injection signal before the valve is re-closed. Lesser failures, termed as leaks, are defined as those motor-operated valve failures with flow rates low enough that a reactor trip will not be generated during the short time interval (on the order of two minutes) that a valve is cycled open and close. Note that a single motor-operated valve leak is terminable within two minutes and can also be diagnosed during the test by the high-temperature alarm downstream of the automatic depressurization system line. Leaks are not further analyzed since they are not a source of reactor trip.

For catastrophic failures of one of the first-stage, second stage or third stage valves during in-service, a medium loss-of-coolant accident event is initiated. The event can be terminated by re-closing the valve opened for the test. However, it is expected that a reactor trip and a safety injection signal would be generated by the time the operator closes the tested valve. There is high confidence that the operator will re-close the tested valve quickly for the following reasons:

1. Since it is a stroke test, the operator already intends to re-close the valve.

2. There are temperature sensors downstream of the automatic depressurization system valves. If the tested motor-operated valve is opening, the leakage would be detected by these temperature sensors and alarmed on the monitor used for in-service testing. Thus, the operator will have an almost instantaneous knowledge of the leak.

3. The operator is aware of the sensitive nature of opening automatic depressurization system valves for test purposes. If the high-temperature alarm downstream of the automatic depressurization system valves is on, or the reactor trip and safety injection signals are generated after the operator opens an automatic depressurization system valve, the operator would restore the valve to its normal closed position before getting out of the in-service test mode and going into the emergency procedures (if necessary).

In that case, the event will proceed as a loss of main feed water transient with a safety injection signal.

Note that it is estimated that it takes about 25 seconds for stage 1 motor-operated valves to fully open or close; this time is about 1 minute for the stage 2 and 3 valves. Once a motor-operated valve is actuated to open, its motion cannot be reversed until it completes its opening. Thus, for stage 2 or 3 motor-operated valves, the full minimum cycle time to open and close a motor-operated valve is estimated to be about two minutes.

The above model can be illustrated and summarized in terms of the following event tree.

Operator Opens a Motor-Operated Valve for Test	Status of the second motor-operated valve in the same line	High-temperature alarm downstream works	Reactor Trips SI-signal is Generated	Operator Recloses Test Valve		
	OK		(No)		1	OK
	LEAKS	(Yes)	(No)		2	OK
	FAILS	(Yes)	(Yes)	YES	3	LMFW with SI-signal
	P (valves)			NO		
				P(ADS-MANTEST)	4	ADS-LOCA

For this case, the contribution of loss of main feedwater with safety injection signal to the transient without main feedwater initiating event frequency is on the order of E-05 or less, which is negligible compared with the frequency related to the other generating causes.

Frequency of the automatic depressurization system spurious actuation per motor-operated valve per year (ADS-LOCA) is given by the equation.

$$F_{(ADS\text{-}LOCA)} = N * [P_{VALVE} * P_{(ADS\text{-}MANTEST)}]$$

where,

N = Number tests per year.

P_{VALVE} = Catastrophic failure probability of a motor-operated valve for a surveillance interval of 6 months.

$P_{(ADS\text{-}MANTEST)}$ = Unavailability of the operator to reclose the tested valve, given high-temperature alarm and/or reactor trip with safety injection signal.

The expression is evaluated for the combinations of automatic depressurization system valves in Table 3-3.

3.5.1.2 Spurious Actuation by Protection and Safety Monitoring System and Diverse Actuation System

The automatic depressurization valves are designed to automatically open when actuated and to remain open for the duration of an automatic depressurization event.

Valve stages 1 and 4 actuate at discrete core makeup tank levels as the level in either tank decreases during injection or from spilling out of a broken direct vessel injection line. The valve actuation logic is based on two-out-of-four level detectors in either core makeup tank for automatic depressurization system stages 1 and 4. For events that rapidly empty the core makeup tanks, there is a short time delay before the fourth stage is actuated.

Valve stages 2 and 3 actuate based on a time delay after actuation of the preceding stage. This opening sequence provides a controlled depressurization of the reactor coolant system.

The modeling of spurious actuation of the automatic depressurization system is based on the following. The instrumentation and control system hardware must fail such that a pair of controllers in either protection and safety monitoring system or diverse actuation system must actuate causing both valves to open in a stage 1, 2, or 3 line. For stage 4, the squib valves are each actuated by a redundant pair of controllers, both of which must actuate to cause detonation. Redundant controllers are used for the stage 4 lines since the opening of any one squib valve provides a depressurization path.

The potential for spurious actuation of the automatic depressurization system due to failure of the protection and safety monitoring system and diverse actuation system is further discussed in Chapter 26.

3.5.1.3 Operator Errors

- Inadvertent Automatic Depressurization System by Operator Action

  AP1000 provides a special control design for actuating the automatic depressurization system for reactor coolant system depressurization. The control mechanism is special because it consists of two controls that must be activated for automatic depressurization system actuation. This configuration of two controls with interlocks makes it necessary for two operators to manipulate both controls, simultaneously, for actuation of the automatic depressurization system. Prior to actuating the automatic depressurization system, the decision would have to be made by the operating crew to perform the action. Thus, inadvertent or accidental actuation of the automatic depressurization system is extremely unlikely.

- Spurious Actuation of Automatic Depressurization System During In-Service Test due to Operator Errors

  The automatic depressurization system motor-operated valves are tested during cold shutdown. This failure probability is conservatively modeled as an at-power special initiator. As discussed below, a spurious automatic depressurization system actuation

requires at least two failures, whose joint failure probability is estimated to be small (e.g., on the order of E-06 or less), as discussed below.

For automatic depressurization system in-service motor-operated valve testing, the operator uses the applicable procedure, which places the in-service test mode soft controls on the monitor. This mode also actuates the interlock that prevents inadvertent manual opening of two motor-operated valves in series. Two operator failures may be postulated:

1. The operator performs the test without going into the in-service test mode (OP1). In this case the interlock is not established.

2. The operator attempts to open both automatic depressurization system motor-operated valves in the same line (OP2).

Neither failure alone is sufficient to cause spurious automatic depressurization system actuation. However, OP2 together with OP1 or OP2 together with failure of the interlock would initiate a spurious actuation of the automatic depressurization system event. Both of these double failures are unlikely, given that the test conditions are normal (e.g., not emergency); performed in the control room; operators are aware of the implications of opening automatic depressurization system valves (e.g., causing a reactor trip and shutting the plant down); and supervision and other operators are present in the control room to observe the test (e.g., the operator would not likely skip the step of bringing up the in-service test soft controls). Additionally, the hardware failure of the interlock is small (passive system with a failure probability of 1E-3 or less).

Failure of each operator action is independent and is estimated to be on the order of 1E-03 (similar to omission or commission error with no stress factor, from THERP). Considering a recovery factor of 0.1 for the supervisor for the first error, the total expected error is estimated to be in the range of 1E-06 to 1E-07 (1E-03 * 1E-03 * 1E-01).

Based on the above discussion, the failure modes leading to the above scenarios are not further evaluated quantitatively.

Results of Spurious Actuation of Automatic Depressurization System

The yearly frequencies of the spurious actuation of the automatic depressurization system categories are calculated as shown in Table 3-3. The calculated initiating event frequencies are as follows:

- Spurious automatic depressurization system classified as medium loss-of-coolant accident 2.17E-06/yr

- Spurious automatic depressurization system classified as large loss-of-coolant accident 5.4E-05/yr

Table 3-1

LOSS OF CCS/SWS INITIATING EVENT FAULT TREE CSWF SUCCESS CRITERIA SUMMARY

Event Description	Component cooling water system and service water system fail to remove heat during normal plant operation
Success Configuration	One-out-of-two CCS trains to remove heat from the loads (CCN fault tree) AND One-out-of two SWS trains to remove heat from the CCS heat exchangers (SWN fault tree)
System Initial Status	CCS train A operating and train B in standby SWS train A operating and train B in standby
Mission Time	8760 hours for train A, 24 hours for train B when called into operation (applicable for both CCS and SWS)
Components Required to Change Status	Standby pump operates if required
Initiating Signals	For CCS CCS standby train B automatic actuation on a loss of operating train A (low discharge pressure) For SWS SWS standby train B automatic actuation on a loss of operating train A (low discharge pressure) Strainer backwash cycle automatic actuation on high differential pressure across the strainer
Operator Actions	For CCS CCB-MAN01N - recognize the need and start standby component cooling water pump B if automatic actuation fails during normal plant operation (This basic event is modeled in subtree SUB-CCS-IC1.) CCN-MAN02 - exclude the operating component cooling water heat exchanger in the case of excessive leakage and align the standby heat exchanger during normal plant operation For SWS SWN-MAN01N - recognize the need and open air-operated valve on the motorized strainer line if automatic actuation fails during normal plant operation (This basic event is modeled in subtree SUB-SWS-IC1.) SWB-MAN02N - recognize the need and start standby service water pump B if automatic actuation fails during normal plant operation (This basic event is modeled in subtree SUB-SWS-IC2.)

Table 3-2

SPURIOUS ACTUATION OF AUTOMATIC DEPRESSURIZATION FAILURE CRITERIA SUMMARY[1]

Spurious Automatic Depressurization System Scenarios	LOCA Category
One line of the ADS first-stage OR one line of the second-stage OR one line of the third-stage opens	Medium LOCA
Two lines of the ADS fourth-stage or one line of the automatic depressurization system fourth-stage with all lines of the automatic depressurization system first-, second-, and third-stage open	Large LOCA

Note:

1. Based on information from Appendix A of the AP1000 PRA

Table 3-3 (Sheet 1 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

In general, the model used to quantify the frequency of the spurious actuation of the automatic depressurization system is given by the equation:

$$F_I = F_{Ai} + F_{Bi} + F_{Ci}$$

where;

F_i = Frequency of automatic depressurization system spurious actuation that is classified as either medium or large loss-of-coolant accident (i = 1 - medium LOCA, i = 2 - large LOCA)

F_{Ai} = Frequency of two valves in series fail to remain close due to internal rupture

F_{Bi} = Frequency of one valve rupture (given the other open due to testing) and the operator fails to re-close the tested valve

F_{Ci} = Spurious actuation of automatic depressurization system due to instrumentation and control system

The frequency of each automatic depressurization system spurious actuation category is evaluated as follows.

1. Automatic Depressurization System Spurious Actuation Classified As Medium LOCA

The frequency of the automatic depressurization system spurious actuation that is classified as a medium loss-of-coolant accident event is calculated as follows:

$$\begin{aligned} F_M &= F_1 + F_2 \\ &= (7.16E\text{-}07 + 1.45E\text{-}07)/yr \\ &= 2.17E\text{-}06/yr \end{aligned}$$

The evaluation of F_1 and F_2 is provided below.

$$\begin{aligned} F_1 &= F_{A1} + F_{B1} + F_{C1} \\ &= (1.64E\text{-}07 + 5.50E\text{-}07 + 1.83E\text{-}09)/yr \\ &= 7.16E\text{-}07/yr \end{aligned}$$

The evaluation of F_{A1}, F_{B1}, and F_{C1} is provided below.

- Failure Criteria:

  From Table 3-2, the failure criteria for this event is defined as either line of the ADS first stage fails to remain closed causing a medium loss-of-coolant accident event.

Table 3-3 (Sheet 2 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

- ADS First-stage Lines Fail Open Due to Hardware Failure During Normal Plant Operation (F_{A1}):

  There are two lines of the ADS first-stage where two motor-operated valves in each line must rupture for the spurious actuation of automatic depressurization system, resulting in a medium LOCA, to occur. Thus, the total yearly frequency for this type of failure can be expressed as:

  $$F_{A1} = \{2 * [(P_{VA} * P_{VB/VA}) + (P_{VB} * P_{VA/VB})]\} + CCF$$

  where;

2	=	Two automatic depressurization system first-stage lines
P_{VA}	=	Probability per year (8760 hrs) of valve 1 rupture
$P_{VB/VA}$	=	Probability of valve 2 rupturing given the rupture of valve 1
P_{VB}	=	Probability per year (8760 hrs) of valve 2 rupture
$P_{VA/BA}$	=	Probability of valve 1 rupturing given the rupture of valve 2
CCF	=	Common cause failure of two motor-operated valves due to rupture

  It is assumed that the second valve ruptures at the mid-point of 8760 hrs. The values of P_{VA} (or P_{VB}) and $P_{VB/VA}$ (or $P_{VA/VB}$) are calculated as:

  $$P_{VA} \text{ (or } P_{VB}) = \lambda \text{ (valve rupture failure rate/hour)} * 8760 \text{ hrs,}$$

  and

  $$P_{VB/VA} \text{ (or } P_{VA/VB}) = \lambda \text{ (valve rupture failure rate/hour)} * 4380 \text{ hrs.}$$

  From Chapter 32, the failure rate of motor-operated valve rupture (λ) is 5.0E-09/hr. From Chapter 29, the yearly probability of motor-operated valves failure due to common cause for this scenario is 1.6E-07. The total frequency per year for this type of failure is:

  $$\begin{aligned} F_{A1} &= \{2 * [(5E\text{-}09 * 8760) * (5E\text{-}09 * 4380)]) + \\ &\quad [(5E\text{-}09 * 8760) * (5E\text{-}09 * 4380)] + 1.6E\text{-}07 \\ &= 1.64E\text{-}07/yr \end{aligned}$$

Table 3-3 (Sheet 3 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

- ADS First-stage Lines Fail Open Due to Hardware Failure During In-service Testing (F_{B1}):

As discussed in Subsection 3.5.1, if the failed motor-operated valve is one of the four valves of the automatic depressurization system first-stage during in-service testing, a medium loss-of-coolant accident event would be initiated. The event can be terminated by re-closing the tested valve. There are two tests per year. Thus, from Subsection 3.5.1.1 the total frequency for this type of failure can be expressed as:

$$F_{B1} = 2 * 4 [P_{VA \text{ or } VB} * P_{(ADS\text{-}MANTEST)}]$$

where;

2	=	Two tests per year
4	=	Four motor-operated valves in two lines of ADS first-stage
$P_{VA \text{ or } VB}$	=	Probability that the valve not being tested has ruptured since the last test, given other valve open due to testing (every 6 months = 4380 hrs)
$P_{(ADS\text{-}MANTEST)}$	=	Operator fails to reclose the tested valve

From Chapter 30 the unavailability of the operator failing to reclose the tested valve is 3.14E-03. The total frequency per year for this type of failure is:

$$F_{B1} = 2 * 4 *[(5E\text{-}09 * 4380) * (3.14E\text{-}03)]$$
$$= 5.5E\text{-}07 /yr$$

- ADS First-stage Lines Fail Open Due Instrumentation and Control System (F_{C1}):

From Chapter 26, the frequency of PMS and DAS causing spurious actuation of the automatic depressurization system, that is classified as a medium loss-of-coolant accident event, is 1.83E-09/yr

$$F_2 = F_{A2} + F_{B2} + F_{C2}$$
$$= (3.38E\text{-}07 + 1.10E\text{-}06 + 1.10E\text{-}08) /yr$$
$$= 1.45E\text{-}06/yr$$

Table 3-3 (Sheet 4 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

The evaluation of F_{A2}, F_{B2}, and F_{C2} is provided below.

- Failure Criteria:

  From Table 3-2 the failure criteria for this event is defined as one line of the ADS second-, third-, or fourth-stage that fails to remain closed causing a medium loss-of-coolant accident event.

- ADS second- and third-stage lines fail open due to hardware failure during normal plant operation (F_{A2}):

  As discussed in Subsection 3.5.1, failure of the fourth-stage automatic depressurization system squib valves is limited to either operator error or spurious actuation via the PMS or DAS. The failure probability of ADS second- and third-stage motor-operated valves due to hardware failure is calculated below.

  There are four lines of the automatic depressurization system second- and third-stage where two motor-operated valves in each line must rupture for the spurious actuation of automatic depressurization system, resulting in a medium LOCA, to occur. Thus, the total yearly frequency for this type of failure can be expressed as:

  $$F_{A1} = \{4 * [(P_{VA} * P_{VB/VA}) + (P_{VB} * P_{VA/VB})]\} + CCF$$

  where;

4	=	Four automatic depressurization system second- and third-stage lines
F_{VA}	=	Probability per year (8760 hrs) of valve 1 rupture
$P_{VB/VA}$	=	Probability of valve 2 rupturing given the rupture of valve 1
F_{VB}	=	Probability per year (8760 hrs) of valve 2 rupture
$P_{VA/VB}$	=	Probability of valve 1 rupturing given the rupture of valve 2
CCF	=	Common cause failure of two motor-operated valves due to rupture

Table 3-3 (Sheet 5 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

From Chapter 32, the failure rate of motor-operated valve rupture is 5.0E-09/hr. From Chapter 29, the yearly probability of motor-operated valves failure due to common cause for this scenario is 3.3E-07. Using the same assumptions as were used for the evaluation of ADS spurious actuation classified as medium LOCA, the total frequency per year for this type of failure is:

$$F_{A2} = \{4 * 2[(5E\text{-}09 * 8760) * (5E\text{-}09 * 4380)] + 3.3E\text{-}07$$
$$= 3.38E\text{-}07 /yr$$

- ADS Second- and Third-stage Lines Fail Open Due to Hardware Failure During In-service Testing (F_{B2}):

As discussed in Subsection 3.5.1 the failed motor-operated valve is one of the eight valves of the automatic depressurization system second- and third-stage during in-service testing, a medium loss-of-coolant accident event would be initiated. The event can be terminated by reclosing the tested valve. There are two tests per year. Thus, the total frequency for this type of failure can be expressed as:

$$F_{B2} = 2 * 8 [P_{VA \text{ or } VB} * P_{(ADS\text{-}MANTEST)}]$$

where;

2	=	Two tests per year
8	=	8 motor-operated valves in four lines of automatic depressurization system second- and third-stage
$P_{VA \text{ or } VB}$	=	Probability that the valve not being tested has ruptured since the last test given other valve open due to testing (every 6 months - 4380 hrs)
$P_{(ADS\text{-}MANTEST)}$	=	Operator fails to reclose the tested valve

From Chapter 30, the unavailability of operator fails to reclose the tested valve is 3.14E-03. The total frequency per year for this type of failure is:

$$F_{B2} = 2 * 8 *[(5E\text{-}09 * 4380) * (3.14E\text{-}03)]$$
$$= 1.10E\text{-}06/yr$$

Table 3-3 (Sheet 6 of 6)

EVALUATION OF SPURIOUS AUTOMATIC DEPRESSURIZATION SYSTEM FREQUENCIES

- ADS First-stage Lines Fail Open Due Instrumentation and Control System (F_{C2}):

  From Chapter 26, the frequency of PMS and DAS causing spurious actuation of the automatic depressurization, that is classified as a medium loss-of-coolant accident event, is 1.10E-08/yr.

2. Automatic Depressurization System Spurious Actuation Classified As Large LOCA

From Table 3-2, the failure criteria for this event is defined as two lines of the ADS fourth-stage or one line of the ADS fourth-stage with all lines of the ADS first-, second-, and third-stage fail to remain closed causing a large loss-of-coolant accident event.

The frequency of one ADS fourth-stage line and all lines of ADS stage 1, 2, and 3 fail open is much less than the frequency of two ADS fourth-stage lines fail open. Therefore, only the frequency of two ADS fourth-stage lines fail open is calculated.

As discussed in Subsection 3.5.1, failure of the fourth-stage ADS is limited to either operator error or spurious actuation via the PMS or DAS. However, the fourth-stage squib valves do not have in-service testing. Therefore, the frequency of the automatic depressurization system spurious actuation is dominated by the failure of PMS and DAS. From Chapter 26, the frequency of PMS and DAS causing spurious actuation of the automatic depressurization, that is classified as a large loss-of-coolant accident event, is 5.4E-05/yr.

CHAPTER 4

EVENT TREE MODELS

4.1 Introduction

This chapter presents the event models for internal initiating events, with the purpose of defining the core damage sequences, and plant damage sequences for the AP1000 PRA study. For each initiating event category defined in Chapter 2, the accident sequences that may stem from the initiating event are modeled in the form of an event tree. The event trees are constructed in terms of non-safety and safety systems that include support systems and operator actions that either respond to the initiating events or mitigate failure of other systems. The event trees result in definition of core damage event sequences, which are categorized in terms of core damage categories.

Dominant core damage sequences are collected in core damage categories, which are used as input to the containment event tree.

This chapter provides the event tree models for core damage for each internal initiating event category. The success criteria for event tree nodes (e.g., fault trees, operator actions, etc.) are defined in Chapter 6.

Attachment 4A contains further information about event tree modeling, as event tree guidelines for core damage and plant damage state event trees. The core damage categories are defined in the same Attachment.

All event trees are shown in Attachment 4B.

In the discussion of event tree models below, the reader should refer to the following chapters for additional information:

Chapter 2	For initiating event definition and frequency quantification
Attachment 4A	For event tree guidelines; core damage categories
Attachment 4B	For event tree pictures
Chapter 6	For event tree core damage and event tree node success criteria

4.2 Large LOCA Event Tree Model

4.2.1 Event Description

The large loss-of-coolant accident (LOCA) event includes breaks in the reactor coolant system with sizes sufficient to produce a depressurization of the reactor coolant system (RCS) that allows gravity injection from the in-containment refueling water storage tank (IRWST). For AP1000 design, this category includes breaks in the reactor coolant system with sizes greater than 9 inches in diameter (excluding reactor vessel rupture event).

The blowdown of the reactor coolant system occurs within a very short period of time (from a few seconds to a few minutes). The resulting reactor trip and safety injection (SI) signal (S-signal) causes reactor coolant pumps (RCPs) to trip, core makeup tanks (CMTs) actuation, and containment isolation. When the reactor coolant system pressure decreases below the accumulators' normal operating pressure (approximately 700 psia), the borated water in the accumulators is forced into the reactor vessel to reflood the core.

Following the complete depressurization of the reactor coolant system and actuation of CMTs in order to actuate the IRWST injection line squib valves, the IRWST begins draining into the reactor coolant system and, consequently, out of the break into the reactor cavity. The steam produced from the time the initiating event occurs until the long-term phase begins is condensed on the containment steel shell. It is returned back into the reactor coolant system via the gravity injection lines or via the recirculation lines. Containment cooling is accomplished by the ultimate heat sink.

Core damage is avoided if the accumulator injection, IRWST gravity injection, and subsequent recirculation of water from the containment sump into the reactor vessel are successful. Reactor trip is expected to be successful, but is not necessary for success due to initial voiding and subsequent boration. Also expected but not necessary is the success of passive containment cooling for heat removal from the containment. Outside air cooling of the containment is sufficient for heat removal in the time frame involved.

Containment isolation is not necessary to avoid core damage for larger break sizes. However, at the lower end of the large LOCA break size (e.g., closer to 9 inches equivalent diameter), the RCS will depressurize more slowly. The RCS must depressurize for IRWST gravity injection to succeed. The interface between the upper end of medium LOCA sizes and the lower end of large LOCA sizes introduces success criteria issues. To cover these issues, success of the automatic depressurzation system (ADS) is stipulated for success in a large LOCA event. See Chapter 6 for a further discussion on this subject.

4.2.2 Event Tree Model and its Nodes

The large LOCA event tree is shown in Figure 4B-1 in Attachment 4B. Event tree node descriptions are given below.

LLOCA – Large LOCA Event Occurs

This node represents the occurrence of a large LOCA initiating event. A large LOCA due to spurious actuation of ADS is modeled separately.

ACC – Accumulators Inject

Actuation of the accumulators provides a means for injecting cooling water into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel (RPV) through the safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig. The injection occurs even if the reactor coolant pumps are not

tripped. The accumulators are the primary means to rapidly reflood the core, because the core makeup tank injection is insufficient to mitigate the large LOCA event.

Failure of this function is the failure of one-of-two accumulators to inject water into the reactor pressure vessel, given successful RCS depressurization to below the accumulator nitrogen pressure.

ADS-F – Full Reactor Coolant System Depressurization by ADS Occurs

Upon failure of containment isolation for break sizes in the lower end of the break range, ADS is needed to reduce the delta P (difference in pressure) between the RCS and IRWST. Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection from the IRWST can be initiated. Full depressurization success criteria for this event are defined in Chapter 6. Actuation is initiated automatically when core makeup tank low water level is obtained.

Failure of this function is the inability to actuate the minimum configuration of ADS valves required for full reactor depressurization.

IRWST – RCS Refill from IRWST by Gravity Injection Occurs

After full depressurization of the reactor coolant system with accumulator injection, IRWST gravity injection can be established for long-term core cooling. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. Failure of this function is the inability of both gravity injection lines to open. Successful CMT actuation is needed for automatic actuation of IRWST injection line squib valves.

CIS – Containment Isolation

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criterion for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to the Reactor Pressure Vessel from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criteria of two-of-four lines to the recirculation paths are required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in containment pressure and temperature such as a LOCA and main steam line break (MSLB) accident inside containment. If the RECIRC is successful, one-of-three possible paths of

water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

4.2.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-1 and are also shown in the event tree in Figure 4B-1 in Attachment 4B.

4.2.4 Operator Actions

Credit for operator actions following a large LOCA event is limited due to the rapid development of the event.

However, in the RECIRC node, credit is given for manual opening of recirculation valves (REN-MAN02) if automatic actuation fails. Credit is also given for manual containment isolation.

4.2.5 Transfers and Consequential Events

No end states for transfers to other events or consequential events are modeled in this event tree.

4.3 Large Spurious ADS Event Tree Model

4.3.1 Event Description

The ADS consists of four different valve stages that open sequentially to reduce RCS pressure sufficiently so that long-term core cooling can be provided from the passive core cooling system (PXS).

The spurious ADS actuation event due to actuation logic failure or mechanical common cause failure of the motor-operated valves of stages 1, 2, 3 or 4 (i.e., spurious opening of the two valves in series of any stage line) results in a LOCA. Those spurious ADS events that are large LOCAs are included in this category.

4.3.2 Event Tree Model and its Nodes

The large spurious ADS actuation event tree is shown in Figure 4B-2 in Attachment 4B. Event tree node descriptions are given below.

SPADS – Large Spurious ADS Actuation Event Occurs

This node represents the occurrence of a spurious ADS actuation initiating event.

ACC – Accumulators Inject

Actuation of the accumulators provides a means for injecting cooling water into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel through the

safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig. The injection occurs even if the reactor coolant pumps are not tripped. The accumulators are the primary means to rapidly reflood the core, because the core makeup tank injection is insufficient to mitigate the large LOCA event.

Failure of this function is the failure of two-of-two accumulators to inject water into the reactor pressure vessel, given successful RCS depressurization to below the accumulator nitrogen pressure.

ADS-F – Full RCS Depressurization by ADS Occurs

This event tree node addresses the remaining ADS valves to open to satisfy the success criteria for full ADS depressurization.

When the reactor cannot be cooled at high pressure, the depressurization system should be actuated to permit injection either by normal residual heat removal (RHR) or by gravity.

Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection can be initiated. The success criteria for full depressurization for small LOCA are discussed in Chapter 6. ADS actuation can be obtained either manually by the operator or automatically on core makeup tank low water level.

Failure of this function is the failure to actuate the minimum configuration of valves required for full reactor depressurization.

IRWST – RCS Refill from IRWST by Gravity Injection Occurs

After full depressurization of the reactor coolant system and either core makeup tank or accumulator injection, gravity injection can be established. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. Drop in CMT level automatically actuates the IRWST injection.

CIS – Containment Isolation Occurs

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criterion for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to RPV from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criteria of two-of-four lines to the recirculation paths are required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in containment pressure and temperature such as a LOCA and main steam line break accident inside containment. If the RECIRC is successful, one-of-three possible paths of water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

4.3.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-2 and are also shown in the event tree in Figure 4B-2 in Attachment 4B.

4.3.4 Operator Actions

Credit for operator actions following a large SPADS event is limited due to the rapid development of the event.

However, in the RECIRC node, credit is given for manual opening of recirculation valves (REN-MAN02) if automatic actuation fails. Credit is also given for manual containment isolation.

4.3.5 Transfers and Consequential Events

No end states for transfers to other events or consequential events are modeled in this event tree.

4.4 Medium LOCA Event Tree Model

4.4.1 Event Description

Medium LOCA events are characterized by breaks in the reactor coolant system with sizes between approximately 2 and 9 inches diameter. In addition, the following events are special cases of Medium LOCA, and are studied in their own initiating event categories: the safety injection line break, the core makeup tank line break, and consequential medium LOCAs resulting from stuck open pressurizer safety valves. The medium LOCA event is defined as all RCS ruptures with break sizes insufficient to depressurize the RCS to the normal residual heat removal system operating pressure without operation of the ADS, but sufficient to allow the automatic actuation of the ADS fourth stage without the operation of the ADS stages 1, 2, and 3. Stuck-open pressurizer safety valves and ADS spurious actuation events, which result in a consequential medium LOCA event, are included in this category.

The medium LOCA event results in a reactor trip and safety injection signal (S-signal) causing core makeup tank actuation and reactor coolant pump trip, and containment isolation. When the reactor coolant system pressure decreases below the accumulators' normal operating pressure (approximately 700 psia), the borated water in the accumulators is forced into the reactor vessel. Injection of both core makeup tanks and accumulators is not required

to meet the success criteria. Success criteria analysis (per Chapter 6) shows that water from only one source is sufficient for short-term reactor coolant system makeup.

Following the complete depressurization of the reactor coolant system, the IRWST begins draining into the reactor coolant system and, consequently, out of the break into the reactor cavity. The steam produced from the time the initiating event occurs until the long-term phase begins is condensed on the containment steel shell. It is returned back into the reactor coolant system via the gravity injection lines or via the recirculation lines. Containment cooling is accomplished by the ultimate heat sink.

4.4.2 Event Tree Model and its Nodes

The medium LOCA event tree in Figure 4B-3 in Attachment 4B. Event tree node descriptions are given below.

MLOCA – Medium LOCA Event Occurs

This node represents the occurrence of a medium LOCA initiating event.

RTRIP – Reactor Trip Occurs

This node is used to indicate that the reactor trip function has succeeded and the reactivity control is achieved. The failure of this function is separately modeled in the anticipated transient without scram (ATWS) modeling (ATW-S category).

CMT – Reactor Coolant System Pumps Trip and Core Makeup Tanks

Core makeup tank injection requires that the reactor coolant pumps are tripped to reduce the pressure in the reactor coolant system cold leg. If the pumps are operating, the reactor coolant system pressure differences are insufficient to provide injection without cold leg voiding. The core makeup tanks are designed to actuate on an S-signal, low-2 pressurizer level, or high hot leg temperature with low steam generator level.

The pumps are automatically tripped on the CMT actuation signal. Failure of this function is conservatively assumed to be the failure of any of the pumps to trip. Following successful reactor coolant pump trip, the core makeup flow can be initiated.

Failure of this top event is the failure to actuate at least one core makeup tank, either automatically or manually, or failure to trip the reactor coolant pumps as described above.

ADS-F – Full Reactor Coolant System Depressurization

When the reactor cannot be cooled at high pressure, the depressurization system should be actuated to permit injection either by normal RHR or by gravity.

Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection can be initiated. The success criteria for full depressurization for

small LOCA events are discussed in Chapter 6. ADS actuation can be obtained either manually by the operator or automatically on core makeup tank low water level.

Failure of this function is the failure to actuate the minimum configuration of valves required for full reactor depressurization.

ADS-P – Partial Depressurization

Partial depressurization is defined as the reduction of the reactor coolant system pressure to a value so that injection into the reactor coolant system can be achieved with the normal residual heat removal pumps, but not by gravity. The success criteria for partial depressurization for this event is discussed in Chapter 6.

ACC – Accumulators

Actuation of the accumulators is a diverse means for providing cooling water injection into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel through the safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig.

Failure of this function is the failure of both accumulators to inject water into the reactor pressure vessel, given successful depressurization to below the accumulator nitrogen pressure.

NRHR – Normal Residual Heat Removal in Injection Mode

The normal RHR system is configured to take suction from the cask loading pit or the IRWST and discharge to the safety injection line.

The normal residual heat removal system, aligned in this mode with one out of two pumps operating, can also accomplish long-term core cooling if the recirculation function is successful. The recirculation function success is necessary to allow core cooling after the IRWST water is depleted and the reactor cavity is flooded. This is obtained by opening one of the two recirculation lines, from which the operating normal residual heat removal takes suction.

Failure of this function is the failure to actuate at least one out of the two normal residual heat removal pumps after the RCS depressurizes or failure of the system to operate for the PRA's mission time. Failure is also assumed if the recirculation function fails.

IRWST – Gravity Injection

After full depressurization of the reactor coolant system and either core makeup tank or accumulator injection, gravity injection can be established. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. A drop in the CMT level automatically actuates the IRWST injection.

Failure of this function is the failure of both gravity injection lines to open.

CIS – Containment Isolation

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criteria for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to the Reactor Pressure Vessel from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criteria of two-of-four lines to the recirculation paths are required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in containment pressure and temperature such as a LOCA and main steam line break accident inside containment. If the RECIRC is successful, one-of-three possible paths of water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

If normal RHR, component cooling water (CCS) and service water system (SWS) are available, decay heat removal from the containment by the CCS heat exchangers is also credited.

4.4.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-3 and are also shown in the event tree in Figure 4B-3 in Attachment 4B.

4.4.4 Operator Actions

Operator actions are needed for the following functions:

NRHR manual actuation only
ADS-F manual actuation only if CMT fails

The core damage sequences containing failure of ADS-F and NRHR (following CMT failure) contain two operator actions in series.

Credit is taken for manual actuation of the following functions, only if automatic actuation fails:

- CMT
- ADS-F
- RECIRC

4.4.5 Transfers and Consequential Events

The RTRIP node is placed in the event tree to designate that reactor trip is successful in this model; the failure of reactor trip is modeled in the ATW-S event tree.

No other end states for transfers to other events, or consequential events are modeled in this event tree.

4.5 Core Makeup Tank Line Break Event Tree Model

4.5.1 Event Description

This event consists of reactor coolant system breaks occurring in one of the core makeup tank lines. It impairs only one of the core makeup tanks with no other systems being impacted. It is assumed that all core makeup tank line breaks are medium breaks. The core makeup tank line break (CMTLB) event tree (Figure 4B-4) is similar to the medium LOCA event tree, except that the core makeup tank function is quantified with degraded capability for core makeup tank operation constituting success.

CMTLB event results in a reactor trip and safety injection signal (S-signal) causing core makeup tank actuation and reactor coolant pump trip, and containment isolation. When the reactor coolant system pressure decreases below the accumulators' normal operating pressure (approximately 700 psia), the borated water in the accumulators is forced into the reactor vessel. Injection of both core makeup tanks and accumulators is not required to meet the success criteria. Success criteria analysis (per Chapter 6) shows that water from only one source is sufficient for short-term reactor coolant system makeup.

Following the complete depressurization of the reactor coolant system, the IRWST begins draining into the reactor coolant system and, consequently, out of the break into the reactor cavity. The steam produced from the time the initiating event occurs until the long-term phase begins is condensed on the containment steel shell. It is returned back into the reactor coolant system via the gravity injection lines or via the recirculation lines. Containment cooling is accomplished by the ultimate heat sink.

Reactor trip is expected to be successful, but is not necessary for success due to initial voiding and subsequent boration. Also expected but not necessary is the success of passive containment cooling for decay heat removal from the containment. Outside air cooling of the containment is sufficient for decay heat removal in the time frame involved.

4.5.2 Event Tree Model and its Nodes

The CMTLB event tree is given in Figure 4B-4 in Attachment 4B. Event tree node descriptions are given below.

CMTLB – Core Makeup Tank Line Break Event Occurs

This node represents the occurrence of a CMTLB initiating event.

RTRIP – Reactor Trip Occurs

This node is used to indicate that the reactor trip function has succeeded and the reactivity control is achieved. The failure of this function is separately modeled in the ATWS modeling (ATW-S category).

CMT – Reactor Coolant System Pumps Trip and Core Makeup Tanks

One of the core makeup tanks is assumed to spill due to the nature of the initiating event; thus, only one CMT is available. Core makeup tank injection requires that the reactor coolant pumps are tripped to reduce the pressure in the reactor coolant system cold leg. If the pumps are operating, the reactor coolant system pressure differences are insufficient to provide injection without cold leg voiding. The core makeup tanks are designed to actuate on an S-signal, low-2 pressurizer level, or high hot leg temperature with low steam generator level.

The pumps are automatically tripped on the CMT actuation signal. Failure of this function is conservatively assumed to be the failure of any of the pumps to trip. Following successful reactor coolant pump trip, the core makeup flow can be initiated.

Failure of this top event is the failure to provide injection from the intact core makeup tank, either automatically or manually, or failure to trip the reactor coolant pumps as described above.

ADS-F – Full Reactor Coolant System Depressurization

When the reactor cannot be cooled at high pressure, the depressurization system should be actuated to permit injection either by normal RHR or by gravity.

Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection can be initiated. The success criteria for full depressurization for small LOCA events are discussed in Chapter 6. ADS actuation can be obtained either manually by the operator or automatically on core makeup tank low water level.

Failure of this function is the failure to actuate the minimum configuration of valves required for full reactor depressurization.

ADS-P – Partial Depressurization

Partial depressurization is defined as the reduction of the reactor coolant system pressure to a value so that injection into the reactor coolant system can be achieved with the normal residual heat removal pumps, but not by gravity. The success criteria for partial depressurization for this event is discussed in Chapter 6.

ACC – Accumulators

Actuation of the accumulators is a diverse means for providing cooling water injection into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel

through the safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig.

Failure of this function is the failure of both accumulators to inject water into the reactor pressure vessel, given successful depressurization to below the accumulator nitrogen pressure.

NRHR – Normal Residual Heat Removal in Injection Mode

The normal RHR system is configured to take suction from the cask loading pit or the IRWST and discharge it to the safety injection line.

The normal RHR system, aligned in this mode with one out of two pumps operating, can also accomplish long-term core cooling if the recirculation function is successful. The recirculation function success is necessary to allow core cooling after the IRWST water is depleted and the reactor cavity is flooded. This is obtained by opening one of the two recirculation lines, from which the operating normal residual heat removal takes suction.

Failure of this function is the failure to actuate at least one out of the two normal residual heat removal pumps after the RCS depressurizes or failure of the system to operate for the PRA's mission time. Failure is also assumed if the recirculation function fails.

IRWST – Gravity Injection

After full depressurization of the reactor coolant system and either core makeup tank or accumulator injection, gravity injection can be established. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. A drop in the CMT level automatically actuates the IRWST injection.

Failure of this function is the failure of both gravity injection lines to open.

CIS – Containment Isolation

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criteria for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to the Reactor Pressure Vessel from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criteria of two-of-four lines to the recirculation paths are required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in

containment pressure and temperature such as a LOCA and main steam line break accident inside containment. If the RECIRC is successful, one-of-three possible paths of water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

If normal RHR, CCS, and SWS are available, decay heat removal from the containment by the CCS heat exchangers is also credited.

4.5.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-4 and are also shown in the event tree in Figure 4B-4 in Attachment 4B.

4.5.4 Operator Actions

Operator actions are needed for the following functions:

NRHR manual actuation only
ADS-F manual actuation only if CMT fails

The core damage sequences containing failure of ADS-F and NRHR (following CMT failure) contain two operator actions in series.

Credit is taken for manual actuation of the following functions, only if automatic actuation fails:

- CMT
- ADS-F
- RECIRC

4.5.5 Transfers and Consequential Events

The RTRIP node is placed in the event tree to designate that reactor trip is successful in this model; the failure of reactor trip is modeled in the ATW-S event tree.

No other end states for transfers to other events, or consequential events, are modeled in this event tree.

4.6 Safety Injection Line Break Event Tree Model

4.6.1 Event Description

Given the size of the line (8 inches), the safety injection (direct vessel injection) line break (SI-LB) event is classified as a medium LOCA event. There is a 4-inch flow restrictor in the line near the reactor vessel, which limits the effective break size for most breaks. The difference between the medium LOCA event tree and here is the unavailability of one core makeup tank, one accumulator, one gravity injection line, and one recirculation line. For this event, it is pessimistically assumed that the break occurs in the line where the NRHR is

connected; thus NRHR system injection into both safety injection lines is assumed to have failed.

The SI-LB event results in a reactor trip and safety injection signal (S-signal) causing core makeup tank actuation, reactor coolant pump trip, and containment isolation. When the reactor coolant system pressure decreases below the accumulators' normal operating pressure (approximately 700 psia), the borated water in the accumulators is forced into the reactor vessel. Injection of both core makeup tanks and accumulators is not required to meet the success criteria. Success criteria analysis (per Chapter 6) shows that water from only one source is sufficient for short-term reactor coolant system makeup.

Following the complete depressurization of the reactor coolant system, the IRWST begins draining into the reactor coolant system and, consequently, out of the break into the reactor cavity. The steam produced from the time the initiating event occurs until the long-term phase begins is condensed on the containment steel shell. It is returned back into the reactor coolant system via the gravity injection lines or via the recirculation lines. Containment cooling is accomplished by the ultimate heat sink.

Reactor trip is expected to be successful, but is not necessary for success due to initial voiding and subsequent boration. Also expected but not necessary is the success of passive containment cooling for heat removal from the containment. Outside air cooling of the containment is sufficient for heat removal in the time frame involved.

4.6.2 Event Tree Model and its Nodes

The SI-LB event tree is given in Figure 4B-5 in Attachment 4B. Event tree node descriptions are given below.

SI-LB – Safety Injection Line Break Event Occurs

This node represents the occurrence of a SI-LB initiating event.

RTRIP – Reactor Trip Occurs

This node is used to indicate that the reactor trip function has succeeded and the reactivity control is achieved. The failure of this function is separately modeled in the ATWS modeling (ATW-S category).

CMT – Reactor Coolant System Pumps Trip and Core Makeup Tanks

Core makeup tank injection requires that the reactor coolant pumps are tripped to reduce the pressure in the reactor coolant system cold leg. If the pumps are operating, the reactor coolant system pressure differences are insufficient to provide injection without cold leg voiding. The core makeup tanks are designed to actuate on an S-signal, low-2 pressurizer level, or high hot leg temperature with low steam generator level.

The pumps are automatically tripped on the CMT actuation signal. Failure of this function is conservatively assumed to be the failure of any of the pumps to trip. Following successful reactor coolant pump trip, the core makeup flow can be initiated.

Failure of this top event is the failure to actuate at least one core makeup tank, either automatically or manually, or failure to trip the reactor coolant pumps as described above.

ADS-F – Full Reactor Coolant System Depressurization

When the reactor cannot be cooled at high pressure, the depressurization system should be actuated to permit injection either by normal RHR or by gravity.

Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection can be initiated. The success criteria for full depressurization for small LOCA events are discussed in Chapter 6. ADS actuation can be obtained either manually by the operator or automatically on core makeup tank low water level.

Failure of this function is the failure to actuate the minimum configuration of valves required for full reactor depressurization.

ADS-P – Partial Depressurization

Partial depressurization is defined as the reduction of the reactor coolant system pressure to a value so that injection into the reactor coolant system can be achieved with the normal residual heat removal pumps, but not by gravity. The success criteria for partial depressurization for this event is discussed in Chapter 6.

ACC – Accumulators

Actuation of the accumulators is a diverse means for providing cooling water injection into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel through the safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig.

Failure of this function is the failure of both accumulators to inject water into the reactor pressure vessel, given successful depressurization to below the accumulator nitrogen pressure.

IRWST – Gravity Injection

After full depressurization of the reactor coolant system and either core makeup tank or accumulator injection, gravity injection can be established. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. A drop in the CMT level automatically actuates the IRWST injection.

Failure of this function is the failure of both gravity injection lines to open.

CIS – Containment Isolation

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criterion for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to the Reactor Pressure Vessel from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criteria of two-of-four lines to the recirculation paths are required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in containment pressure and temperature such as a LOCA and main steam line break accident inside containment. If the RECIRC is successful, one-of-three possible paths of water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

If normal RHR, CCS, and SWS are available, decay heat removal from the containment by the CCS heat exchangers is also credited.

4.6.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-5 and are also shown in the event tree picture in Figure 4B-5 in Attachment 4B.

4.6.4 Operator Actions

Operator actions are needed for the following functions:

NRHR manual actuation only
ADS-F manual actuation only if CMT fails

The core damage sequences containing failure of ADS-F and NRHR (following CMT failure) contain two operator actions in series.

Credit is taken for manual actuation of the following functions, only if automatic actuation fails:

- CMT
- ADS-F
- RECIRC

4.6.5 Transfers and Consequential Events

The RTRIP node is placed in the event tree to designate that reactor trip is successful in this model; the failure of reactor trip is modeled in the ATW-S event tree.

No other end states for transfers to other events, or consequential events are modeled in this event tree.

4.7 Small LOCA Event Tree Model

4.7.1 Event Description

Small LOCA (SLOCA) events are characterized by breaks in the reactor coolant system with sizes between 3/8 and 2 inches diameter. These events consist of reactor coolant system breaks having a size greater than those producing leakage that can be made up by the chemical and volume control system and less than that required to depressurize sufficiently that PRHR or a stage 1, 2, 3 ADS valve is not needed before a stage 4 ADS valve can automatically open. The small LOCA event results in a reactor trip and safety injection signal (S-signal) causing core makeup tank actuation and reactor coolant pumps trip, passive RHR actuation, and containment isolation. The operation of the passive RHR system removes decay heat and also changes the ADS success criterion. If passive RHR fails, for some break sizes, then the fourth stage ADS system requires the opening of at least one other valve in ADS stages 1, 2 or 3 in order to reduce RCS pressure to the 4th stage pressure interlock setpoint. When the reactor coolant system pressure decreases below the accumulators' normal operating pressure (approximately 700 psia), the borated water in the accumulators is forced into the reactor vessel. Injection of both core makeup tanks and accumulators is not required to meet the success criteria. Success criteria analysis shows that water from only one source is sufficient for short-term reactor coolant system makeup (see Chapter 6). Once the normal RHR system operating pressure is reached, operation of normal RHR in injection mode provides continued core cooling and makeup.

If complete depressurization of the reactor coolant system occurs, the IRWST begins draining into the reactor coolant system and, consequently, out of the break and the ADS valves into the reactor cavity. The steam produced from the time the initiating event occurs until the long-term phase begins is condensed on the containment steel shell. It is returned back into the reactor coolant system via the gravity injection lines or via the recirculation lines.

Expected but not necessary is the success of passive containment cooling for heat removal from the containment. Outside air cooling of the containment is sufficient to maintain the containment pressure well below the containment failure pressure.

4.7.2 Event Tree Model and its Nodes

The small LOCA event tree is given in Figure 4B-6 in Attachment 4B. Event tree node descriptions are given below.

SLOCA – Small LOCA Event Occurs

This node represents the occurrence of a small LOCA initiating event.

RTRIP – Reactor Trip Occurs

This node is used to indicate that the reactor trip function has succeeded and the reactivity control is achieved. The failure of this function is separately modeled in the ATWS modeling (ATW-S category).

CMT – Reactor Coolant System Pumps Trip and Core Makeup Tanks

Core makeup tank injection requires that the reactor coolant pumps are tripped to reduce the pressure in the reactor coolant system cold leg. If the pumps are operating, the reactor coolant system pressure differences are insufficient to provide injection without cold leg voiding. The core makeup tanks are designed to actuate on an S-signal, low-2 pressurizer level, or high hot leg temperature with low steam generator level.

The pumps are automatically tripped on the CMT actuation signal. Failure of this function is conservatively assumed to be the failure of any of the pumps to trip. Following successful reactor coolant pump trip, the core makeup flow can be initiated.

Failure of this top event is the failure to actuate at least one core makeup tank, either automatically or manually, or failure to trip the reactor coolant pumps as described above.

PRHR – Passive Residual Heat Removal System

The passive residual heat removal system consists a heat exchanger and inlet and outlet lines connected to the reactor coolant system.

Operation of the passive residual heat removal system is initiated (valves opened) upon receipt of any of the following signals: low steam generator level narrow range with low startup feedwater flow to any steam generator, low steam generator level wide range, high hot leg temperature, and also CMT or ADS actuation.

The passive RHR actuation signal automatically opens the normally closed valves in the line from the passive RHR heat exchanger to the cold leg. The passive RHR heat exchanger transfers heat from the reactor coolant system to the water contained in the IRWST. This is accomplished by the natural circulation of reactor coolant system fluid from the reactor coolant system hot leg to the reactor coolant system cold legs. The water in the IRWST provides a large heat sink that acts to delay direct steaming to the containment for about 2 hours. It provides sufficient water inventory for passive residual heat removal cooling for 72 hours.

Failure of this function is the failure of the passive RHR system to remove decay heat from the reactor coolant system.

ADS-F – Full Reactor Coolant System Depressurization

When the reactor cannot be cooled at high pressure, the depressurization system should be actuated to permit injection either by normal RHR or by gravity.

Full reactor coolant system depressurization is the reduction of the system pressure to a value such that gravity injection can be initiated. The success criteria for full depressurization for small LOCA events are discussed in Chapter 6. ADS actuation can be obtained either manually by the operator or automatically on core makeup tank low water level.

Failure of this function is the failure to actuate the minimum configuration of valves required for full reactor depressurization.

ADS-P – Partial Depressurization

Partial depressurization is defined as the reduction of the reactor coolant system pressure to a value so that injection into the reactor coolant system can be achieved with the normal residual heat removal pumps, but not by gravity. The success criteria for partial depressurization for this event is discussed in Chapter 6.

ACC – Accumulators

Actuation of the accumulators is a diverse means for providing cooling water injection into the reactor coolant system. The driving force for injection is pressurized nitrogen in the accumulators. The water contained in the two accumulators enters the reactor pressure vessel through the safety injection lines whenever the reactor coolant system pressure goes below the nitrogen pressure of 700 psig.

Failure of this function is the failure of both accumulators to inject water into the reactor pressure vessel, given successful depressurization to below the accumulator nitrogen pressure.

NRHR – Normal Residual Heat Removal in Injection Mode

The normal RHR system is configured to take suction from the cask loading pit or the IRWST and discharge to the safety injection line.

The normal RHR system, aligned in this mode with one out of two pumps operating, can also accomplish long-term core cooling if the recirculation function is successful. The recirculation function success is necessary to allow core cooling after the IRWST water is depleted and the reactor cavity is flooded. This is obtained by opening one of the two recirculation lines, from which the operating normal RHR takes suction.

Failure of this function is the failure to actuate at least one out of the two normal residual heat removal pumps after the RCS depressurizes or failure of the system to operate for the PRA's mission time. Failure is also assumed if the recirculation function fails.

IRWST – Gravity Injection

After full depressurization of the reactor coolant system and either core makeup tank or accumulator injection, gravity injection can be established. This is achieved by permitting the water from the IRWST to flow through one of the two gravity injection lines into the safety injection lines. Drop in CMT level automatically actuates the IRWST injection.

Failure of this function is the failure of both gravity injection lines to open.

CIS – Containment Isolation Occurs

Containment isolation is modeled to distinguish the success criteria for the recirculation event tree node with or without CIS. A more stringent success criteria for recirculation exists if the containment isolation is not established.

RECIRC – Water Recirculation to RPV from the Sump Occurs

The recirculation of water from the containment sump to the reactor pressure vessel is necessary for long-term core cooling. If the CIS is successful, one-of-four lines to the two recirculation paths is sufficient for success. If the CIS is not successful, a more stringent success criterion of two-of-four lines to the recirculation paths is required.

CHR – Containment Cooling is Established

The removal of thermal energy from the containment atmosphere to the environment via the steel containment vessel is necessary following events that cause a significant increase in containment pressure and temperature such as a LOCA and main steam line break accident inside containment. If the RECIRC is successful, one-of-three possible paths of water supply lines will be activated in response to a high-2 containment pressure signal or a high containment temperature signal.

If normal RHR, CCS, and SWS are available, decay heat removal from the containment by the CCS heat exchangers is also credited.

4.7.3 Event Tree Success Criteria and Core Damage Categories

Event tree success criteria and core damage category assignments are summarized in Table 4-6 and are also shown in the event tree in Figure 4B-6 in Attachment 4B.

4.7.4 Operator Actions

Operator actions are needed for the following functions:

NRHR manual actuation only
ADS-F manual actuation only if CMT fails

The core damage sequences containing failure of ADS-F and NRHR (following CMT failure) contain two operator actions in series.

Credit is taken for manual actuation of the following functions, only if automatic actuation fails:

- PRHR
- CMT
- ADS-F
- RECIRC

4.7.5 Transfers and Consequential Events

The RTRIP node is placed in the event tree to designate that reactor trip is successful in this model; the failure of reactor trip is modeled in the ATW-S event tree.

No other end states for transfers to other events, or consequential events, are modeled in this event tree.

4.8 RCS Leak Event Tree Model

4.8.1 Event Description

The RCS leakage event includes leak rates from 1 to 100 gpm, or less than the flow from approximately a 3/8" break. The chemical and volume control (CVS) system is designed to make up RCS leaks of 100 gpm or less. As long as the RCS inventory is replenished by the CVS, the plant will remain at power, in a safe state. If the CVS system fails, the pressurizer level will drop and the event will proceed as a small LOCA. If the leak is very small, the CVS will continue to make up the RCS inventory. However, if the leak is on the high side (e.g., 100 gpm), the CVS pumps will run continuously, and the CVS inventory will start to be depleted. CVS can provide water for 100 gpm leaks for approximately 6 hours; then it will run out of inventory. During this period, the plant operators are expected to follow off-normal operating procedures and start bringing the plant to a controlled safe shutdown, which will reduce or eliminate the RCS leak and allow repairs. If this operator action is not performed in time, and the CVS runs out of water, the event will proceed as a small LOCA.

It should be noted that even if this event becomes a small LOCA, it will still be at the lower end of the SLOCA spectrum and CMT injection is expected to last a very long time, before the ADS is actuated.

4.8.2 Event Tree Model and its Nodes

The RCS leak event tree is given in Figure 4B-7 in Attachment 4B. Event tree node descriptions are given below.

RCSLK – RCS Leak Event Occurs

This node represents the occurrence of a RCS leak initiating event.

CVS – Chemical and Volume Control System

This event models the failure of the chemical and volume control system to provide makeup water following a LOCA break at a rate less than the capacity of the CVS makeup pumps. Automatic actuation signals modeled are: low pressurizer water level relative to programmed level. Although the CVS can provide makeup at a rate sufficient to prevent a reactor trip, it cannot do so indefinitely without operator action to refill the boric acid tank. The event trees take no credit for this action, so that it is assumed that even if CVS is successful, a reactor trip will eventually occur, unless the operators successfully perform a controlled shutdown.

SDMAN – Controlled Manual Shutdown

This node models the operator action (RTN-MAN01) of a controlled shutdown of the plant from a full power operation, following an RCS leak event. It is expected that the operators will diagnose the event through continuous or intermittent running of the CVS pumps, and depletion of boric acid tank. At that point, the operators are expected to follow off-normal operating procedures. Even with the 100 gpm leak, the operators will have sufficient time to perform this manual shutdown. If this action is not completed before the CVS inventory is depleted, the plant will eventually trip due to low pressurizer level, and will respond to the event in a similar manner to the SLOCA event.

4.8.3 Event Tree Success Criteria

Event tree success criteria, as modeled in Figure 4B-7 is given below:

RLK-OK1 The most likely progression of the event leading to success is that the CVS and SDMAN are successful.

For those sequences that transfer to SLOCA, the SLOCA event success criteria apply.

Table 4-7 summarizes the success criteria and other relevant information about the event tree model.

4.8.4 Operator Actions

Operator action is needed for the following function:

CVS Operators align spent fuel pool to the CVS to extend CVS running time, when the RCS leak is high (e.g., 100 gpm)

SDMAN operator manually shuts down the plant

The remaining operator actions are the same as in the SLOCA event tree, into which this event tree transfers.

4.8.5 Transfers and Consequential Events

This event transfers to the SLOCA event tree if CVS or SDMAN nodes fail.

4.8.6 Core Damage Categories

There are no explicitly assigned core damage categories in this event tree. However, if the event transfers to the SLOCA event tree, SLOCA core damage categories apply.

4.9 PRHR Tube Rupture Event Tree Model

4.9.1 Event Description

This event includes events in which a passive residual heat removal heat exchanger complete circumferential tube rupture occurs. As soon as the event initiates, the inlet pipe temperature increases up to a value at which an alarm in the main control room is actuated. Also, when the pressurizer water level decreases to a low set point value due to the loss of coolant, the chemical and volume control system is actuated. The combination of the temperature alarm and continuous operation of the chemical and volume control system informs the operator of the event occurrence, and the operator can take steps to isolate the break. This action is performed from the main control room and it results in isolation of both of the passive residual heat removal system trains.

Small PRHR tube leaks that can be isolated locally (inside the containment) by isolating the affected heat exchanger are not modeled here.

Sequences where passive residual heat removal heat exchanger isolation is timely and successful are transferred to the loss of main feedwater (without PRHR) event tree.

It is possible to mitigate this event with startup feedwater and auxiliary pressurizer spray for the cases in which the isolation is not successfully performed. However, these methods, conservatively, are not taken into account.

If CVS or passive RHR isolation fail, the event progresses as a small LOCA. To simplify the event tree modeling, no credit is taken for PRHR during this SLOCA event. This is due to the existence of different failure modes that may lead to failure to isolate the break flow from the PRHR but yet isolate the flow path through the PRHR heat exchanger (e.g. failure to isolate a downstream valve, but successful isolation of the inlet valve).

4.9.2 Event Tree Model and its Nodes

The PRHR tube rupture event tree is given in Figure 4B-8 in Attachment 4B.

Event tree node descriptions are given below.

PRSTR – PRHR Tube Rupture Event Occurs

This node represents the occurrence of a PRHR tube rupture initiating event.

CVS – Chemical and Volume Control System

This event models the failure of the chemical and volume control system to provide makeup water following a LOCA break at a rate less than the capacity of the CVS makeup pumps.

The automatic actuation signals modeled are: low pressurizer water level relative to programmed level. Although the CVS can provide makeup at a rate sufficient to prevent a reactor trip, it cannot do so indefinitely without alignment to the spent fuel pool or operator action to refill the boric acid tank. The event trees take no credit for this action. It is assumed that even if CVS is successful, a reactor trip will eventually occur, and no credit is taken for the operators to successfully perform a controlled shutdown.

PRISO – Passive Residual Heat Removal System Isolation

This top event addresses the failure to isolate the faulted passive RHR heat exchanger before the level in the CMTs reaches the automatic depressurization actuation setpoint. Successful and timely passive RHR heat exchanger isolation is transferred to the loss of main feedwater transient event tree, with the condition that PRHR is unavailable.

4.9.3 Event Tree Success Criteria

Event tree sequences transfer either to LMFW or SLOCA events.

Table 4-8 summarizes the success criteria and other relevant information about the event tree model.

4.9.4 Operator Actions

Operator action is needed for the following function:

PRISO operator manually isolates the PRHR inlet and outlet valves

The remaining operator actions are the same as in LMFW or SLOCA event trees, into which this event tree transfers.

4.9.5 Transfers and Consequential Events

This event transfers to two different event trees:

- If PRHR isolation is successful, the event transfers to LMFW with the condition that PRHR is unavailable.
- If PRISO fails, the event transfers to the SLOCA event tree. PRHR is assumed to be unavailable, to account for the possibility of partial flow isolation failure modes, as discussed previously.

4.9.6 Core Damage Categories

There are no explicitly assigned core damage categories in this event tree. However, if the event transfers to the SLOCA event tree, SLOCA core damage categories apply. If the event transfers to the LMFW event tree, LMFW core damage categories apply.