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WOG Risk-Informed ATWS Assessment and Licensing Implementation Process

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

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WOG Risk-Informed ATWS Assessment and Licensing Implementation Process

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TABLE OF CONTENTS

LIST OF TABLES xiii

LIST OF FIGURES xix

LIST OF ACRONYMS xxi

EXECUTIVE SUMMARY xxiii

1 INTRODUCTION 1-1

2 BACKGROUND 2-1
- 2.1 ATWS Rule 2-1
- 2.2 WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process" 2-2
- 2.3 Byron/Braidwood PMTC License Amendment 2-3
- 2.4 NRC/WOG Meetings 2-4
  - 2.4.1 NRC/WOG December 17, 1998 Meeting 2-5
  - 2.4.2 NRC/WOG August 23, 2000 Meeting 2-5
  - 2.4.3 NRC/WOG January 24, 2001 Meeting 2-6
- 2.5 NRC Letter, "Westinghouse Owners Group Risk-Informed Anticipated Transient without Scram Approach" 2-7

3 NEED FOR THE CHANGE 3-1

4 DETERMINISTIC ANALYSIS 4-1
- 4.1 Critical Power Trajectory Calculations 4-1
- 4.2 Unfavorable Exposure Time Calculations 4-4
- 4.3 Peak RCS Pressure Calculations 4-15
- 4.4 Part-Power ATWS Analyses 4-18
- 4.5 Byron/Braidwood Analysis 4-23

5 PROBABILISTIC RISK ANALYSIS 5-1
- 5.1 ATWS Core Damage Frequency Analysis for the Low, High, and Bounding Reactivity Cores 5-2
  - 5.1.1 ATWS State 3/4: Plant At-Power Operation and Plant Shutdown, Power Level ≥40% 5-2
  - 5.1.2 ATWS State 2: Plant Startup, Power Level ≥40% 5-23
  - 5.1.3 ATWS State 1: Plant Startup, Power Level <40% 5-32
  - 5.1.4 ATWS State 5: Plant Shutdown, Power Level <40% 5-38
  - 5.1.5 Summary of ATWS Core Damage Frequency Results 5-43
  - 5.1.6 ATWS Core Damage Frequency Analysis Sensitivity Studies 5-46
  - 5.1.7 Incremental Conditional Core Damage Probability 5-50

TABLE OF CONTENTS (cont.)

5.2 ATWS Large Early Release Frequency Analysis 5-51
5.2.1 ATWS Large Early Release Frequency for the Low and Bounding Reactivity Cores 5-52
5.2.2 Incremental Conditional Large Early Release Probability 5-53
5.3 Loss of Offsite Power ATWS Core Damage Frequency Analysis 5-58
5.4 Summary of Results from the Probabilistic Risk Analysis 5-63

6 IMPACT ON DEFENSE-IN-DEPTH AND SAFETY MARGINS 6-1
6.1 Impact on Defense-in-Depth 6-1
6.2 Impact on Safety Margins 6-5

7 CONFIGURATION MANAGEMENT PROGRAM 7-1

8 WOG ATWS APPROACH AND MODEL 8-1
8.1 Accident Progression 8-1
8.2 ATWS Event Tree Model 8-2
8.2.1 IEV: Initiating Event Frequency 8-4
8.2.2 RT: Reactor Trip Signal from the RPS 8-4
8.2.3 OAMG: Operator Action to Trip the Reactor via the MG Sets 8-5
8.2.4 CRI: Action to Drive the Control Rods into the Core 8-5
8.2.5 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor 8-6
8.2.6 ESFAS: Turbine Trip and AFW Pump Start by the ESFAS 8-7
8.2.7 AMSAC: ATWS Mitigation System Actuation Circuitry 8-8
8.2.8 AF100: AFW System Provides 100% Flow 8-8
8.2.9 AF50: AFW System Provides 50% Flow 8-8
8.2.10 PR: Availability of Primary Pressure Relief 8-8
8.2.11 LTS: Long Term Shutdown 8-9
8.2.12 Event Tree Sequence Endstates 8-10
8.3 PRA Model Quantifications and Application of Regulatory Guide 1.174 8-10

9 BRAIDWOOD LEAD PLANT EVALUATION 9-1
9.1 Braidwood ATWS PRA Model 9-1
9.2 Braidwood ATWS Core Damage Frequency Quantifications 9-3
9.3 Incremental Conditional Core Damage Probability 9-4
9.4 Configuration Management Program 9-5
9.5 Conclusions from the Lead Plant Evaluation 9-6

10 RELOAD IMPLEMENTATION PROCESS 10-1

11 CONCLUSIONS 11-1

12 REFERENCES 12-1

TABLE OF CONTENTS (cont.)

Appendix A Issues Identified by the NRC at the NRC/WOG December 17, 1998 Meeting A-1

Appendix B Issues and Additional Information Needs Identified by the NRC in the Summary for the NRC/WOG August 23, 2000 Meeting B-1

Appendix C Issues Identified in NRC letter "Westinghouse Owners Group Risk-Informed Anticipated Transient without Scram Approach" (Reference 10) C-1

Appendix D Fault Trees for Primary Pressure Relief with Power Level ≥40% used for CDF Analysis D-1

Appendix E Fault Trees for Primary Pressure Relief with Power Level <40% used for CDF Analysis E-1

Appendix F Fault Trees for Primary Pressure Relief Used for LERF Analysis F-1

LIST OF TABLES

Table 4-1 Loss of Normal Feedwater ATWS Critical Power Trajectory Data 4-3

Table 4-2 Loss of Load ATWS Critical Power Trajectory Data 4-3

Table 4-3 UETs for Low Reactivity Model with Equilibrium Xenon, No Control Rod Insertion 4-6

Table 4-4 UETs for Low Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-6

Table 4-5 UETs for Low Reactivity Model with No Xenon, No Control Rod Insertion 4-6

Table 4-6 UETs for Low Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-7

Table 4-7 UETs for High Reactivity Model with Equilibrium Xenon, No Control Rod Insertion 4-7

Table 4-8 UETs for High Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-7

Table 4-9 UETs for High Reactivity Model with No Xenon, No Control Rod Insertion 4-8

Table 4-10 UETs for High Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-8

Table 4-11 UETs for Bounding Reactivity Model with Equilibrium Xenon, No Control Rod Insertion 4-8

Table 4-12 UETs for Bounding Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-9

Table 4-13 UETs for Bounding Reactivity Model with No Xenon, No Control Rod Insertion 4-9

Table 4-14 UETs for Bounding Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps) 4-9

Table 4-15 HFP and HZP Moderator Temperature Coefficients for the Low Reactivity Core Model 4-10

LIST OF TABLES (cont.)

Table 4-16 HFP and HZP Moderator Temperature Coefficients for the High Reactivity Core Model 4-11

Table 4-17 HFP and HZP Moderator Temperature Coefficients for the Bounding Reactivity Core Model 4-12

Table 4-18 Differential and Integral Rod Worths for the Bounding Reactivity Core at 2000 MWD/MTU 4-13

Table 4-19 Differential and Integral Rod Worths for the Low Reactivity Core at 2000 MWD/MTU 4-14

Table 4-20 Loss of Load ATWS, Bounding Reactivity Core Model HZP MTC of ~+7 pcm/°F (HFP MTC = -2.63 pcm/°F) 4-16

Table 4-21 Loss of Load ATWS, Low Reactivity Core Model HZP MTC of +3.5 pcm/°F (HFP MTC = -7.42 pcm/°F) 4-17

Table 4-22 Loss of Normal Feedwater ATWS, Critical Power Trajectory Data 4-20

Table 4-23 Loss of Load ATWS Critical Power Trajectory Data 4-20

Table 4-24 UETs for Low Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition 4-20

Table 4-25 UETs for Low Reactivity Model with No Xenon, 40% Power Initial Condition 4-21

Table 4-26 UETs for High Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition 4-21

Table 4-27 UETs for High Reactivity Model with No Xenon, 40% Power Initial Condition 4-21

Table 4-28 UETs for Bounding Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition 4-21

Table 4-29 UETs for Bounding Reactivity Model with No Xenon, 40% Power Initial Condition 4-22

Table 4-30 Loss of Normal Feedwater ATWS Critical Power Trajectory Data for Byron 1/Braidwood 1 with BWI RSGs 4-25

LIST OF TABLES (cont.)

Table 4-31 Loss of Load ATWS Critical Power Trajectory Data for Byron 1/Braidwood 1 with BWI RSGs 4-26

Table 4-32 Loss of Normal Feedwater ATWS Critical Power Trajectory Data for Byron 2/Braidwood 2 with W D5 SGs 4-27

Table 4-33 Loss of Load ATWS Critical Power Trajectory Data for Byron 2/Braidwood 2 with W D5 SGs 4-28

Table 4-34 UETs for Current Byron/Braidwood Core Designs, No Control Rod Insertion 4-29

Table 4-35 UETs for Current Byron/Braidwood Core Designs, 1 Minute of Control Rod Insertion (72 Steps) 4-29

Table 4-36 UETs for Future Byron/Braidwood Core Designs, No Control Rod Insertion 4-29

Table 4-37 UETs for Future Byron/Braidwood Core Designs, 1 Minute of Control Rod Insertion (72 Steps) 4-30

Table 5-1 Plant Operating States for the ATWS Event 5-14

Table 5-2 Number of Transient Events and Frequency by ATWS State 5-14

Table 5-3 Transient Events Occurring in 30-Day Intervals Relative to Cycle Start 5-15

Table 5-4 Weighted UET Values for a Low Reactivity Core, Equilibrium Xenon, Power Level ≥40% 5-16

Table 5-5 Weighted UET Values for a High Reactivity Core, Equilibrium Xenon, Power Level ≥40% 5-16

Table 5-6 Weighted UET Values for a Bounding Reactivity Core, Equilibrium Xenon, Power Level ≥40% 5-16

Table 5-7 Summary of Pressure Relief Intervals, Equilibrium Xenon 5-17

Table 5-8 Yearly Core Damage Frequency Summary: ATWS State 3/4 5-18

Table 5-9 Distribution of Plant Startups Across the Cycle 5-28

LIST OF TABLES (cont.)

Table 5-10 Weighted UET Values for a Low Reactivity Core, No Equilibrium Xenon, Power Level ≥40% 5-30

Table 5-11 Weighted UET Values for a High Reactivity Core, No Equilibrium Xenon, Power Level ≥40% 5-30

Table 5-12 Weighted UET Values for a Bounding Reactivity Core, No Equilibrium Xenon, Power Level ≥40% 5-30

Table 5-13 Summary of Pressure Relief Intervals, No Equilibrium Xenon 5-31

Table 5-14 Yearly Core Damage Frequency Summary: ATWS State 2 5-31

Table 5-15 Weighted UET Values for a Low Reactivity Core, No Equilibrium Xenon, Power Level <40% 5-35

Table 5-16 Weighted UET Values for a High Reactivity Core, No Equilibrium Xenon, Power Level <40% 5-36

Table 5-17 Weighted UET Values for a Bounding Reactivity Core, No Equilibrium Xenon, Power Level <40% 5-36

Table 5-18 Summary of Pressure Relief Intervals, No Equilibrium Xenon, Power Level <40% 5-36

Table 5-19 Yearly Core Damage Frequency Summary: ATWS State 1 5-36

Table 5-20 Weighted UET Values for a Low Reactivity Core, HFP Equilibrium Xenon, Power Level <40% 5-41

Table 5-21 Weighted UET Values for a High Reactivity Core, HFP Equilibrium Xenon, Power Level <40% 5-41

Table 5-22 Weighted UET Values for a Bounding Reactivity Core, HFP Equilibrium Xenon, Power Level <40% 5-41

Table 5-23 Summary of Pressure Relief Intervals, HFP Equilibrium Xenon, Power Level <40% 5-41

Table 5-24 Yearly Core Damage Frequency Summary: ATWS State 5 5-42

Table 5-25 Summary of ATWS Core Damage Frequency by ATWS State 5-44

LIST OF TABLES (cont.)

Table 5-26 Summary of Important Modeling Differences Between ATWS States 5-45

Table 5-27 Sensitivity Studies: Core Damage Frequency Summary for All Cores, Standard Probabilities for Blocked PORVs 5-48

Table 5-28 Sensitivity Studies: Core Damage Frequency Summary for Low Reactivity Core, Standard Probabilities for Blocked PORVs 5-48

Table 5-29 Sensitivity Studies: Core Damage Frequency Summary for High Reactivity Core, Standard Probabilities for Blocked PORVs 5-48

Table 5-30 Sensitivity Studies: Core Damage Frequency Summary for High Reactivity Core, Worst Time in Cycle 5-49

Table 5-31 Sensitivity Studies: Core Damage Frequency Summary for Bounding Reactivity Core 5-49

Table 5-32 Plant Conditions that Result in Large Early Releases, Low Reactivity Core 5-55

Table 5-33 Plant Conditions that Result in Large Early Releases, Bounding Reactivity Core 5-56

Table 5-34 Summary of Large Release Frequencies 5-57

Table 6-1 Summary of the Capability of Operator Actions to Trip the Reactor for Various RPS Failures 6-6

Table 6-2 Summary of the Capability of Automatic Signals to Actuate Auxiliary Feedwater and Trip the Turbine for Various RPS Failures 6-6

Table 6-3 Plant Configuration Probabilities, Plant Configuration Management Scheme 1 6-7

Table 6-4 Plant Configuration Probabilities, Plant Configuration Management Scheme 2 6-7

Table 7-1 Configuration Management Approach for the High Reactivity Core 7-5

Table 9-1 Braidwood Weighted UET Values, Current Core Design with the 5% UET Restriction 9-8

Table 9-2 Braidwood Weighted UET Values, New Core Design without the 5% UET Restriction 9-8

Table 9-3 Braidwood: Summary of Pressure Relief Intervals 9-8

LIST OF TABLES (cont.)

Table 9-4 Summary of Comparison of WOG and Braidwood ATWS PRA Models 9-9

Table 9-5 Core Damage Frequency Summary, Current and Future Cores 9-11

Table 9-6 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Time in Cycle 9-11

Table 9-7 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Blocked PORVs 9-11

Table 9-8 Configuration Management Approach for the Future Braidwood Core 9-12

Table A-1 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi A-7

Table A-2 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi and Deadweight Load Stresses of 5000 psi on Pipe with 2/3 of Original Thickness A-8

Table A-3 Letdown System Temperature Distribution A-12

Table A-4 CRDM Stress Summary A-19

Table A-5 RCP Model 93A-1 Pressure Boundary Components A-21

Table A-6 RCP Stress Summary A-22

Table A-7 Low Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario A-31

Table A-8 High Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario A-32

Table A-9 Bounding Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario A-33

Table B-1 Core Damage Frequency Results Summary for Low and Bounding Reactivity Cores, Standard Blocked PORV Probabilities, Control Rod Insertion Failure Probability = 0.5 .... B-7

LIST OF FIGURES

Figure 3-1 HFP Moderator Temperature Coefficient versus Cycle Burnup for the Low, High, and Bounding Reactivity Core Designs 3-3

Figure 5-1 ATWS State 3/4 Event Tree 5-19

Figure 5-2 ATWS State 1 Event Tree 5-37

Figure 5-3 LOSP ATWS Event Tree 5-62

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% 8-11

Figure 9-1 Braidwood ATWS Event Tree 9-14

Figure A-1 Critical Powers for the Bounding, High, and Low Reactivity Core Designs for the Reference ATWS Scenario A-34

Figure B-1 HFP Moderator Temperature Coefficient versus Cycle Burnup B-13

Figure B-2 HZP Moderator Temperature Coefficient versus Cycle Burnup B-14

Figure B-3 Moderator Temperature Coefficient versus Power Level Assuming Equilibrium Xenon B-15

Figure B-4 Moderator Temperature Coefficient versus Power Level Assuming No Xenon B-16

Figure C-1 Westinghouse Style A Swing Check Valve Models 03000CS88 and 04000CS88 ..... C-3

Figure C-2 Westinghouse Style B Swing Check Valve Models 03001CS88 and 04001CS88 ..... C-4

Figure C-3 Velan Swing Check Valve C-6

Figure C-4 Velan Swing Check Valve C-8

LIST OF ACRONYMS

AFW	Auxiliary Feedwater
AMSAC	ATWS Mitigation System Actuation Circuitry
AOT	Allowed Outage Time
ATWS	Anticipated Transient without Scram
BOL	Beginning of Life
CDF	Core Damage Frequency
CCDF	Conditional Core Damage Frequency
CPT	Critical Power Trajectory
CR	Control Rods
CRDM	Control Rod Drive Mechanism
CRI	Control Rod Insertion
CRMP	Configuration Risk Management Program
CVCS	Chemical and Volume Control System
DC	Doppler Coefficient
DNB	Departure from Nucleate Boiling
DSS	Diverse Scram System
EOL	End of Life
ESFAS	Engineered Safety Features Actuation System
FLB	Feedwater Line Break
HEP	Human Error Probability
HFP	Hot Full Power
HZP	Hot Zero Power
ICCDP	Incremental Conditional Core Damage Probability
ICLERP	Incremental Conditional Large Early Release Probability
IEV or IE	Initiating Event
IFBA	Integral Fuel Burnable Adsorber
IPE	Individual Plant Examination
LER	Large Early Release
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOL	Loss of Load
LONF	Loss of Normal Feedwater
LOSP	Loss of Offsite Power
LTS	Long-Term Shutdown
MD	Motor-Driven
MFW	Main Feedwater
MG	Motor-Generator
MOL	Middle of Life
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OA	Operator Action
ODSCC	Outer Diameter Stress Corrosion Cracking
PCV	Pressure Control Valve

LIST OF ACRONYMS (cont.)

PMTC	Positive Moderator Temperature Coefficient
PORV	Power-Operated Relief Valve
PR	Pressure Relief
PRA	Probabilistic Risk Analysis
PRT	Pressurizer Relief Tank
RCL	Reactor Coolant Loop
RCS	Reactor Coolant System
RI	Risk-Informed
RPS	Reactor Protection System
RSG	Replacement Steam Generators
RT	Reactor Trip
RTB	Reactor Trip Breakers
RTD	Resistance Temperature Detector
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
TD	Turbine-Driven
UET	Unfavorable Exposure Time
VCT	Volume Control Tank
W	Westinghouse
WABA	Wet Annular Burnable Absorber
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

The purpose of this program is to:

- Develop an approach and model for a risk-informed (RI) anticipated transient without scram (ATWS) analysis that can be implemented by all Westinghouse Owners Group plants to evaluate plant design changes, licensing issues, and plant operability concerns;
- Address the Nuclear Regulatory Commission's concerns with the risk-based approach presented in WCAP-11992 (Reference 1);
- Demonstrate the approach in a pilot plant application; and
- Clarify the regulatory requirements with respect to ATWS.

In the pilot plant application, the RI ATWS model was applied to the Braidwood Nuclear Generating Station. An objective of the pilot plant application is to delete the Braidwood Units 1 & 2 and Byron Units 1 & 2 Technical Specification 5.6.5b.5 that limits the ATWS unfavorable exposure time (UET) to 5% or less for each fuel cycle.

A UET limit effectively places additional constraints on the design moderator temperature coefficient. This restriction requires larger burnable absorber loadings, which can lead to higher fuel enrichments, larger fuel regions, and higher fuel cycle costs. To ensure that the UET restriction is met, Byron and Braidwood cores are designed with additional burnable absorbers and higher leakage loading patterns. Elimination of the 5% UET restriction will result in reduced reactor vessel fluence, less spent fuel, and lower fuel cycle costs.

This program uses a RI approach to demonstrate that the impact on risk of eliminating the 5% UET restriction is small. This is accomplished through the risk evaluation of a low reactivity core, a high reactivity core, and a bounding reactivity core, and demonstrating that the impact on risk for the high and bounding reactivity cores, with respect to the low reactivity core, is acceptable. The low reactivity core was designed to just meet the 5% UET restriction and has a maximum hot zero power (HZP) MTC of +3.5 pcm/°F. The high reactivity core represents a realistic core design that uses the PMTC Technical Specification to reduce burnable absorber inventories. It has a maximum HZP MTC of +5 pcm/°F. The bounding reactivity core has an even smaller burnable absorber inventory such that the HZP MTC is approximately equal to the +7 pcm/°F Technical Specification limit.

The approach used in this program is consistent with the NRC's approach for using probabilistic risk assessment in RI decisions on plant-specific changes to the current licensing basis. This approach is discussed in Regulatory Guide 1.174 (Reference 2) and Regulatory Guide 1.177 (Reference 3). The approach addresses, as documented in this report, the impact of the core design change on defense-in-depth and safety margins, as well as an evaluation of the impact on risk. The risk evaluation considers the impact on core damage frequency (CDF) and large early release frequency (LERF), in addition to assessing incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) for ATWS mitigation equipment out of service.

A detailed RI ATWS model was developed and quantified on a generic basis. This model was also applied to the Braidwood Station to assess the impact on risk of eliminating the Technical Specification that limits the ATWS UET to 5% or less for each fuel cycle. The key results from the program are summarized in the following:

- The CDF increase from the low reactivity core to the high and bounding reactivity cores in the generic analysis meets the ΔCDF acceptance guideline (<1.0E-06/yr) defined in Regulatory Guide 1.174. The CDF contribution from ATWS events to plant total CDF is small for all core designs.

- The LERF increase from the low reactivity core to the bounding reactivity core in the generic analysis slightly exceeds the acceptance guideline (<1.0E-07/yr) defined in Regulatory Guide 1.174. This is based on the conservative approach that applies the peak configuration specific RCS pressures across the whole cycle. For the sensitivity case that assumes the peak RCS pressures are applicable to 50% of the cycle, that is, the fraction of cycle time for each plant configuration that yields RCS pressures that exceed 3584 psi is 0.5, the impact on LERF meets the acceptance guideline. An RCS pressure of 3584 psi is the pressure where SG tubes will fail resulting in a large release. SG tubes were identified as the first component of the RCS pressure boundary that will fail as the RCS pressure increases during an ATWS event.

- ICCDP and ICLERP generic analyses show that PORV availability is not important to plant risk. Based on the RG 1.177 guideline, one PORV may be blocked for more than 3000 hours per year.

- All applicable acceptance criteria for the FSAR Chapter 15 design basis events will continue to be met with the implementation of this risk-informed approach. Therefore, all applicable safety margins will continue to be maintained.

- Tier 2 restrictions can be developed and implemented via a Configuration Management Program that address the defense-in-depth issue during unfavorable exposure times. This is not required to compensate for large impacts on plant risk, but rather to address the NRC's concern relative to possible degradation of defense-in-depth.

- The impact on CDF of removing the 5% UET core design restriction on the Braidwood Station is very small (ΔCDF = 2.3E-08/yr) and meets the guideline in RG 1.174 that defines a small impact on risk.

- For the Braidwood Station, a PORV can be blocked for a significant length of time (>3000 hours/year) based on the ICCDP calculation and the guidelines provided in RG 1.177.

- Tier 2 restrictions have been developed that can be implemented into the Braidwood Station Configuration Management Program to enhance maintaining defense-in-depth during unfavorable exposure times in the cycle.

A reload implementation process is proposed to demonstrate that core designs, with regard to ATWS risk, are acceptable. This can either be done with a best-estimate deterministic calculation to demonstrate that the UET for the reference plant conditions is less than or equal to 5% or by a RI approach and implementation of a Configuration (Risk) Management Program.

Based on the analysis presented in this report, it is concluded that UET limits for higher reactivity core designs should be eliminated. This is based on the RI approach which demonstrates that the impact on risk is small, safety margins are not impacted, and defense-in-depth can be addressed via a Configuration Management Program.

1 INTRODUCTION

The purpose of this program is to:

- Develop an approach and model for a risk-informed (RI) anticipated transient without scram (ATWS) analysis that can be implemented by all Westinghouse Owners Group (WOG) plants to evaluate plant design changes, licensing issues, and plant operability concerns;
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- Demonstrate the approach in a pilot plant application; and
- Clarify the regulatory requirements with respect to ATWS.

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A UET limit effectively places additional constraints on the design moderator temperature coefficient (MTC). This restriction requires larger burnable absorber loading, which can lead to higher fuel enrichments, larger fuel regions, and higher fuel cycle costs. To ensure the UET restriction is met, Byron and Braidwood cores are designed with additional burnable absorbers and higher leakage loading patterns. Elimination of the 5% UET restriction will result in reduced reactor vessel fluence, less spent fuel, and lower fuel cycle costs.

This program uses a RI approach to demonstrate that the impact on risk of eliminating the 5% UET restriction is small. This is accomplished through the risk evaluation of a low reactivity core, a high reactivity core, and a bounding reactivity core, and demonstrating that the impact on risk for the high and bounding reactivity cores, with respect to the low reactivity core, is acceptable. The low reactivity core was designed to just meet the 5% UET restriction and has a hot zero power (HZP) MTC of +3.5 pcm/°F. The high reactivity core represents a realistic core design that uses the PMTC Technical Specification to reduce burnable absorber inventories. It has a HZP MTC of +5 pcm/°F. The bounding reactivity core has an even smaller burnable absorber inventory such that the HZP MTC is approximately equal to the +7 pcm/°F Technical Specification limit.

The approach used in this program is consistent with the NRC's approach for using PRA in RI decisions on plant-specific changes to the current licensing basis. This approach is discussed in Regulatory Guide 1.174 ("An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Reference 2) and Regulatory Guide 1.177 ("An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Reference 3). The approach addresses, as documented in this report, the impact of the core design change on defense-in-depth and safety margins, as well as an evaluation of the impact on risk. The risk evaluation considers the impact on core damage frequency (CDF) and large early release frequency (LERF), in addition to assessing incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) for ATWS mitigation equipment out of service.

This report provides the background information relevant to the ATWS issue (Section 2) including recent WOG/NRC meetings, discusses the need for this change (Section 3), and provides the generic deterministic and probabilistic risk analyses supporting the justification for the change (Sections 4 and 5). Section 6 discusses the impact of the change on defense-in-depth and safety margins, as required by the RI approach, and Section 7 presents the approach to control plant operating configurations (equipment availability) that will maintain the ability of plants to prevent and mitigate ATWS events when operating in an unfavorable configuration. Section 8 presents the WOG ATWS model that utilities will need to implement in their plant specific PRA models to be able to evaluate plant design changes, licensing issues, and plant operability concerns related to ATWS. Section 9 provides the lead plant evaluation for Braidwood Station, which is consistent with the WOG ATWS model presented in Section 8. Section 10 describes the process utilities will need to follow to evaluate the impact of core reloads on plant risk from the ATWS perspective. Finally, Section 11 provides the conclusions from the study.

2 BACKGROUND

The following provides relevant background information important to this report. Short summaries of the ATWS Rule and WCAP-11992 are provided, in addition to a discussion of the UET Technical Specification requirement for Byron and Braidwood Stations. Summaries of the WOG/NRC meetings related to this current effort are also provided.

2.1 ATWS RULE

The final ATWS Rule (10CFR50.62) became effective on July 26, 1984. This rule was issued to require design changes to reduce expected ATWS frequency and consequences. The basis for the ATWS rule is provided in SECY-83-293 (Reference 4). SECY-83-293 included a simplified, risk-based analysis to determine the impact of several options to reduce ATWS consequences as measured by CDF. The analysis objective was to reduce the ATWS contribution to CDF to below 1E-05/yr. A key assumption in this risk-based analysis was that an unfavorable MTC would exist for 10% of the cycle for non-turbine trip events and 1% of the cycle for turbine trip events. An unfavorable MTC results in a pressure transient that exceeds 3200 psig, the pressure corresponding to the ASME Boiler and Pressure Vessel Code Service Level C stress limit, and it is assumed to result in reactor coolant system (RCS) piping failure and core damage. SECY-83-293 also included a value/impact assessment of several options for each Nuclear Steam Supply System (NSSS) vendor to determine the most cost-effective approach.

It should be noted that even though the risk analysis assumed unfavorable MTC values of 10% for non-turbine trip events and 1% for turbine trip events, the Westinghouse generic ATWS analyses performed in response to NUREG-0460 (Reference 5), documented in Westinghouse letter NS-TMA-2182 (Reference 6), did not support these values. In the Westinghouse generic analysis, a full power MTC of -8 pcm/°F was used with a sensitivity analysis using an MTC of –7 pcm/°F. In 1979, these values represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents a 95% confidence level of a favorable MTC.

Based on the results of the SECY-83-293 value/impact assessment, it was recommended that Westinghouse NSSS plants install the ATWS mitigating system actuation circuitry (AMSAC). The requirement for Westinghouse NSSS plants as stated in 10CFR50.62(b) is:

> "Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system."

This requirement is met by the installation of the AMSAC system for Westinghouse NSSS plants. AMSAC consists of equipment to trip the turbine and initiate auxiliary feedwater diverse from the reactor protection system.

WCAP-11992 specifically notes the following important points in SECY-83-293 as applied to Westinghouse NSSS plants:

- The objective of the ATWS Rule was to reduce the risk from ATWS events to an acceptable level. This is accomplished for Westinghouse reactors by the installation of AMSAC as demonstrated by SECY-83-293 results. These results show that with the addition of AMSAC for Westinghouse plants, the core damage frequency due to ATWS events is reduced to the target goal of no more than 1E-05/yr. The core damage frequency predicted for Westinghouse PWRs with AMSAC in the SECY-83-293 assessment is lower than that for the other PWR vendors with the installation of both AMSAC and DSS (diverse scram system) (e.g., 2.2E-05yr per Reference 4).

- The only requirement of the ATWS Rule for Westinghouse reactors is the installation of AMSAC. The acceptability of specific plant conditions as related to the ATWS events is determined within the context of total ATWS core damage frequency, per SECY-83-293.

- Implementation of the prescriptive rule was, in part, based on avoiding the requirement of extensive individual case analyses by licensees and the Staff. In addition, it was the judgement of the Staff as stated during the Commission briefing on SECY-83-293 on August 3, 1983, that ATWS need not be a design basis accident.

From this discussion it is concluded that: 1) with the installation of AMSAC, Westinghouse plants are in compliance with the ATWS Rule, and 2) ATWS is not a design basis event. The conclusion that ATWS is not a design basis event allows supporting analyses to be based on best estimate considerations.

2.2 WCAP-11992, "JOINT WESTINGHOUSE OWNERS GROUP/WESTINGHOUSE PROGRAM: ATWS RULE ADMINISTRATION PROCESS"

The objective of WCAP-11992 was to provide an approach for utilities to address continued ATWS Rule and basis compliance for Westinghouse PWRs, and to provide a means to assess the effects of plant configuration, fuel management, and operational changes. WCAP-11992 established a process for ATWS Rule administration for use by licensees in assessing the impact of changes in important parameters on ATWS CDF. It presented a probabilistic model consistent with SECY-83-293. The model assumed that ATWS overpressure occurs if the pressure limit corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit criterion (3200 psig) is exceeded. Exceeding this pressure is equated to core damage. As in the SECY study, the WCAP-11992 study set an ATWS CDF target of 1E-05/yr.

A reference plant approach was used in the supporting analyses. The reference plant was a typical Westinghouse 4-loop plant with model 51-series steam generators. Ranges of important parameters were identified based on current and expected operation. The impact of changes to these parameters on ATWS CDF were determined. The objective was to enable utilities to assess the impact of changes in plant parameters on ATWS CDF to ensure they were within acceptable ranges. If the CDF values were not within the acceptable range, then utilities could assess alternate approaches for reducing the predicted ATWS CDF. One of the plant parameters of interest was positive MTC (PMTC). The intent was to implement this approach using information contained in the report and to not require plant specific ATWS risk evaluations.

The concept of unfavorable exposure time was introduced and used in WCAP-11992. The UET represents the duration of a given fuel cycle, for a specific plant configuration, for which the core reactivity feedback is insufficient to preclude exceeding a peak RCS pressure of 3200 psig following an ATWS event. UETs are determined for twelve plant configurations considering success or failure of control rod insertion (CRI), amount of auxiliary feedwater (AFW) flow, and number of blocked pressurizer power-operated relief valves (PORVs). In this case, successful CRI is equated to 72 steps insertion of the lead bank.

Commonwealth Edison (currently Exelon Nuclear) referenced WCAP-11992 as part of a request for a license amendment to implement PMTC for the Byron and Braidwood Stations. The NRC rejected the approach described in the WCAP since it had not been formally reviewed and approved. WCAP-11992 was formally submitted to the NRC for review in May 1995 by the WOG in support of efforts to obtain generic approval of the methodology that a plant could use to demonstrate acceptability of ATWS results for PMTC cores. The NRC issued a letter summarizing their review and rejection of the approach on July 1, 1997 (Reference 7). The NRC noted concerns or issues in the following areas:

- Limitations regarding analytic completeness and treatment of uncertainties associated with parameters important to ATWS risk.
- The analysis does not establish an explicit link between MTC and risk.
- The potential for ATWS-induced steam generator tube rupture (SGTR) has not been considered.
- The approach described using a plant-specific ATWS-induced CDF numerical criterion of 1E-05/yr is not consistent with NRC's current direction with risk-informed regulation.

The issue identified in the last bullet arose with the NRC's move towards RI regulation after issuance of WCAP-11992 in 1988.

Since the ATWS risk model of WCAP-11992 is useful for assessing ATWS risk for issues including PMTC, plant power upgrades, and steam generators, and because a number of utilities have included the basic approach from WCAP-11992 in their Individual Plant Examinations (IPEs) and PRAs, there is value to the WOG in obtaining a more favorable closure to this issue.

2.3 BYRON/BRAIDWOOD PMTC LICENSE AMENDMENT

As noted in Section 2.2, Commonwealth Edison referenced WCAP-11992 as part of a request for a license amendment to implement PMTC for Byron and Braidwood Stations, and the approach presented in the WCAP was rejected by the NRC on the basis that the methodology was not reviewed and approved. It was proposed to use only the deterministic approach presented in the WCAP to justify specific MTC values for each operating cycle. This approach focused the NRC review on UETs and critical power trajectories. To meet this, the NRC restricted their review of WCAP-11992 to Sections 4.3.8, 4.6.8, and B.7.1.

The NRC found the approach to be acceptable (Reference 8). In their Safety Evaluation, the NRC stated:

> "The analysis must show that the UET, given the cycle design (including MTC), will be less than 5 percent, or equivalently, that the ATWS pressure limit will be met for at least 95 percent of the cycle. If the limit is not met the core design would be changed until the 95 percent level is achieved.
>
> This 95 percent probability level for the UET is equivalent to the probability level in the reference analyses for the ATWS Rule basis. In those analyses, staff requirements were that all parameters should be best estimate values with the exception of the MTC initial condition. That was to be at a level not to be exceeded (i.e., not less negative) at full power conditions for at least 95 percent of the cycle. The ComEd approach provides a similar level of assurance for the effectiveness of the reactivity feedback."

As part of the NRC's review and acceptance of PMTC, an additional requirement was added to the Byron and Braidwood Technical Specifications in the Administrative Controls Section. This follows:

TS 5.6.5	Core Operating Limits Report (COLR)
TS 5.6.5b	The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. Specifically those described in the following documents:
TS 5.6.5b.5	ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses - Reactivity Control System."

This Technical Specification requires core designs for Byron and Braidwood Stations to meet a 5% UET for the reference conditions. The reference conditions are no control rod insertion, 100% AFW, and no blocked PORVs.

2.4 NRC/WOG MEETINGS

Several meetings were held between the NRC and the WOG during the course of the program. The purpose of these meetings was to keep the NRC informed on the program and to give the NRC a chance to identify issues they felt needed to be addressed. The following sections provide short summaries of these meetings.

2.4.1 NRC/WOG December 17, 1998 Meeting

The NRC and WOG met on December 17, 1998 to discuss the WOG RI ATWS model and the program to address the NRC's comments on the WCAP-11992 Risk-Based ATWS Model. The objectives of the meeting were to:

- Present and discuss the WOG program to develop a RI ATWS model.
- Present and discuss the preliminary results from the WOG program.
- Obtain NRC feedback on the viability of the program and additional considerations that need to be addressed, with particular attention to elimination of MTC and UET as the primary variables which determine ATWS event acceptability and which can result in core design limitations.

The WOG Risk-Informed ATWS model and preliminary results were presented and discussed. The WOG response to the NRC's comments on WCAP-11992 were also presented and discussed. The NRC discussed their concerns with the approach and model. These concerns included the impact on defense-in-depth, part power risk considerations, RCS component aging and component response to ATWS peak pressures, improved understanding of the UET/MTC link, and control rod insertion requirements. The NRC noted as a possible "show-stopper" the modeling and understanding of the impact of the RCS pressure transient (above 3200 psig) on containment and containment safety systems, and the impact on LERF. The NRC also noted that moving from relying on a "natural" defense barrier (core feedback) to relying on equipment is a policy question they will need to address. They noted that significant uncertainty and sensitivity studies to ensure adequate understanding of the important issues and parameters will need to be done to support this change.

Appendix A of this WCAP contains a description of each of the specific issues identified by the NRC. Appendix A also provides responses to each issue.

2.4.2 NRC/WOG August 23, 2000 Meeting

The NRC and WOG met on August 23, 2000 to discuss the WOG's responses to NRC comments on the RI ATWS model identified during the previous NRC/WOG meeting on December 17, 1998. The objectives of the meeting were to:

- Obtain NRC concurrence that the WOG's RI approach to ATWS for addressing licensing issues, such as PMTC, is acceptable;
- Discuss the WOG's responses to the comments the NRC raised at the December 1998 meeting with regard to the WOG's RI approach to ATWS and obtain NRC feedback; and
- Discuss the WOG's approach to the ATWS regulatory issue and obtain NRC feedback.

The NRC was represented by members of the Probabilistic Safety Assessment and Reactor Systems Branches, and Research, who were primarily technical reviewers. In setting up the meeting, the WOG requested significant NRC Staff management attendance, similar to the December 1998 meeting, in an attempt to meet the first objective.

The WOG presented and discussed their responses to the NRC's issues identified at the December 1998 meeting. These were provided as an attachment to the NRC meeting summary (Reference 9). In regard to the licensing issue addressing what type of regulatory requirements are required for ATWS, the NRC Staff representatives indicated they could not provide concurrence that the WOG's approach to maintaining defense-in-depth via procedural requirements is acceptable at this point. Several members of the Staff did indicate that even if Reg. Guide 1.174 is used and all the requirements are met, there could be overriding deterministic arguments that guide their final decision. Several Staff members indicated that trading off a reduction in a "natural" defense-in-depth barrier for one controlled by procedures may not be acceptable.

The NRC issued a meeting summary and attached to it was a list of issues and additional information needs identified by the NRC. These issues involve:

- ICCDP for unavailable ATWS mitigation systems (PORVs in particular)
- Bounding core risk analysis
- Part power risk analysis
- RCS pressures that could lead directly to containment degradation
- Defense-in-depth considerations
- Availability of the event tree and fault tree models to the NRC

Appendix B of this WCAP provides the specific issues and information needs as provided by the NRC. Appendix B also provides the WOG response to each request.

2.4.3 NRC/WOG January 24, 2001 Meeting

NRC, WOG, and Exelon representatives met on January 24, 2001 to discuss policy issues related to the RI ATWS approach and the Byron/Braidwood pilot plant application. The WOG's objectives of the meeting were to:

- Communicate the need for resolution of ATWS issues.
- Communicate the status and plans for the WOG RI ATWS program and pilot application.
- Discuss and resolve RI ATWS policy issues.

The NRC was represented by members of the Probabilistic Safety Assessment and Reactor Systems Branches, and Research. NRC management was also in attendance. A summary of the key issues and discussions follow.

<u>Policy Issues</u>: The NRC specifically stated that their review and acceptance or rejection of the WOG RI approach to ATWS will be based on the rules and regulations that are currently in place. Specifically discussed rules and regulations included Regulatory Guide 1.174, 10CFR50.36 (Technical Specifications), and 10 CFR 50.65 (Maintenance Rule).

Benefits of Change: The NRC indicated that the safety benefits are very important to their acceptance or rejection of higher reactivity cores. It was noted by the WOG that benefits other than financial exist, specifically mentioned were reduced reactor vessel fluences and reduced number of spent fuel assemblies.

Bounding Results: The NRC is concerned with understanding the bounding impact on risk for higher reactivity cores. The WOG generic analysis used a core design with a hot zero power MTC of +5 pcm/°F, but some plant Technical Specifications limits are as high as +7 pcm/°F. The NRC is also concerned with the plant specific assessments following the WOG methodology and what approach, if any, will be used to keep the NRC informed regarding changes in risk due to higher reactivity cores.

Defense-in-Depth: The need and approach to address defense-in-depth was discussed. It was not clear from these discussions whether or not the WOG approach, to claim that a small impact on defense-in-depth is acceptable since the risk impact is small supplemented with procedural controls, will be acceptable to the NRC.

5% UET Limit: The WOG noted that a UET limit of 5% (for the condition of no CRI, all PORVs available, and all AFW) is more restrictive to the core design than the Technical Specification limit on MTC. In fact, with a 5% UET limit, the Technical Specification MTC limit cannot be reached.

Safety Margins: The Staff asked about the impact on safety margins. The WOG responded that all the FSAR analyses will be done consistent with the Technical Specification limits on MTC. That is, the most limiting conditions will be used depending on the accident being considered regardless of the core design. Therefore, the safety margins will not be impacted.

2.5 NRC LETTER, "WESTINGHOUSE OWNERS GROUP RISK-INFORMED ANTICIPATED TRANSIENT WITHOUT SCRAM APPROACH"

The NRC issued the letter "Westinghouse Owners Group Risk-Informed Anticipated Transients without Scram Approach" to the WOG on April 2, 2001 (Reference 10). The purpose of the letter was to provide NRC feedback to the WOG on the RI ATWS program. The letter concluded that the WOG approach is not in conflict with the basis of the ATWS rulemaking and that, if submitted, the Staff's review will focus on quantified risk findings, defense-in-depth, and margins. The letter also indicated that the Staff is expecting an effective configuration risk management program to be part of the submittal. In addition, the letter recommends that the WOG consider and/or address the following issues (taken directly from Reference 10):

Issue #1 – Peak Pressure, Meet ASME Service Level C (3200 psig)

In a PWR, the ATWS transient results in a primary system pressure rise, the magnitude of which is dependent upon the MTC, the primary relief capacity, and how much energy the steam generators can remove. If the pressure cannot be reduced, reactor coolant will be lost through the relief valves and the core will eventually be uncovered. If an ATWS occurs when the MTC is either positive or insufficiently negative to limit reactor power, the ATWS pressure increase will exceed the ASME Service Level C pressure and all subsequent mitigative functions are likely to be ineffective. The proposed WOG approach should address this situation.

Issue #2 – Technical Specification MTC=0 at Beginning of Cycle, Hot Standby, Zero Power

The MTC is a natural process that reduces the core reactivity as the water temperature increases. For a PWR with a negative MTC, an increase in the primary coolant temperature provides negative reactivity feedback to limit the power increase. During the first part of the fuel cycle below 100 percent power, the MTC can possibly be positive for a very short period of time. The MTC is more negative (less positive) at 100 percent power than at lower power. The MTC also becomes more negative (less positive) later in the fuel cycle. When the MTC is insufficient to maintain the primary system pressure below 3200 psig during an ATWS, it is designated in the basis of the ATWS rule as "unfavorable MTC" and in the WOG topical reports the equivalent condition is referred to as an UET. A Westinghouse analysis in December 1979 indicated that the MTC will be more negative than –8 pcm/°F for 95 percent of the cycle time, and more negative than –7 pcm/°F for 99 percent of the cycle time that the core is greater than 80 percent nominal power. The –7 pcm/°F was determined to be the point at which the core conditions became unfavorable. Under the approach proposed by the WOG, the values of the MTC and the doppler coefficient (DC) will have to be carefully examined to ensure that an accident does not result in a situation where the contribution from the MTC and DC effects results in an unacceptable reduction in the margin associated with the total temperature coefficient or results in a net positive reactivity feedback condition.

Responses to these issues are provided in Appendix C.

3 NEED FOR THE CHANGE

For a plant with a 5% UET restriction, the effective limit on MTC is much lower than that permitted by typical MTC Technical Specifications. Typical MTC Technical Specifications allow MTC values of +7 pcm/°F at low powers (up to 70% power), with a limit of 0 pcm/°F at full power. For a core design that just meets +7 pcm/°F at HZP, the corresponding expected MTC at full power, equilibrium xenon conditions would be approximately –3 pcm/°F. However, the "favorable MTC limit" for the reference plant employed in this study is approximately –7.5 pcm/°F, assuming the reference ATWS scenario and an inlet temperature of 600°F. This means that the MTC must be more negative than –7.5 pcm/°F for 95% of the operating cycle to ensure that the primary system pressure does not exceed 3200 psig for more than 5% of the cycle. This corresponds to a 5% UET. Effectively, then, a core design that just meets the low power MTC Technical Specification limit would not meet the full power MTC limit consistent with a 5% unfavorable exposure time.

Part of the difficulty in meeting this favorable MTC limit is that the MTC decreases very slowly with cycle burnup during the early portion of the operating cycle. Figure 3-1 shows the behavior of the HFP MTC versus cycle burnup for the low, high, and bounding reactivity cores used in this study. Note that the MTC changes very little for cycle burnups from 0 MWD/MTU to 5000 MWD/MTU, which represents about 23% of the operating cycle. The reason for this behavior has to do with the large burnable absorber inventories required by high energy cores (18 – 24 month cycle designs). These core designs require large burnable absorber inventories to control excess reactivity and maintain boron concentrations and moderator temperature coefficients within limits. Early in the cycle the depletion rate of the burnable absorbers is high such that the core excess reactivity remains approximately constant or, in some cases, even slightly increases. The result is an approximately constant or slightly increasing critical boron concentration which, in turn, leads to an approximately constant or slightly increasing (more positive) HFP MTC for roughly the first quarter of the operating cycle.

This effectively flat variability of the MTC with burnup early in the cycle means that MTC values that are just nominally more positive than the favorable MTC limit can lead to large UETs. For example, Figure 3-1 shows that the low reactivity core was very close to the –7.5 pcm/°F favorable MTC limit for the reference ATWS scenario during the early portion of the cycle. If, however, the MTC had been just slightly more positive, the UET would have jumped from ~0% to ~20% since the MTC is roughly constant during the first three or four months of the cycle.

Effectively, then, core designs with 5% UET restrictions (or equivalent MTC restrictions) cannot utilize the full range of the positive MTC technical specification. These core designs must employ significantly more burnable absorbers to reduce the BOL MTC to below the value that mitigates the ATWS pressure transient.

Additional burnable absorbers increase fuel cycle costs in several ways. First, the cost of the burnable absorbers themselves adds to total fuel cost. Wet Annular Burnable Absorbers (WABAs) are often used in core designs for MTC control since they deplete more slowly than integral fuel burnable absorbers and, therefore, are somewhat more effective in controlling the critical boron concentration. In the low reactivity core design developed for this study, 832 WABAs were used to reduce the MTC such that the UET for the reference ATWS scenario was ~5%. For the high reactivity core design—a core design which more aggressively utilizes the allowable range of the MTC technical specification—a total of only

48 WABAs were used (additional integral fuel burnable absorbers (IFBA) were used in this design). The cost of WABAs used for MTC control will add to the total fuel cycle cost incurred by the utility.

The second way in which WABAs add to fuel cycle cost is through increased enrichments or larger feed batch sizes. WABAs are efficient burnable absorbers, but they do have a small residual reactivity penalty since they displace moderator and have some parasitic neutron absorption. Use of large WABA inventories, then, may lead to small increases in feed enrichments or small increases in feed batch sizes to overcome this residual reactivity penalty.

Finally, since WABAs are separate core components, utilities will incur some costs in their handling and ultimate disposal.

While the total cost to the utility for use of these burnable absorbers will depend on a number of factors such as fuel management, cycle length, loading pattern, etc., the additional cost is not trivial. One utility has estimated that the additional fuel cost incurred due to ATWS related constraints is on the order of $500,000 per fuel cycle.

ATWS constraints may have other implications as well. One of the results of increasing burnable absorber inventories in the reactor core is that the core power distribution shape changes. Specifically, when burnable absorbers are loaded into the core interior for MTC control, the core power distribution tends to shift slightly toward the core periphery. This kind of power distribution shift will increase core leakage, which is an MTC benefit, but it can also have the undesirable side effect of increasing the fast neutron flux at the reactor vessel. For at least the early portion of the cycle, then, the fluence accumulation on the reactor vessel could be larger than if a smaller burnable absorber inventory had been used.

In summary, then, core design constraints related to ATWS can have real fuel cycle cost penalties. Additionally, increases in pressure vessel fluence are possible. These are the areas that will benefit from implementation of the risk-informed ATWS model described in this report.

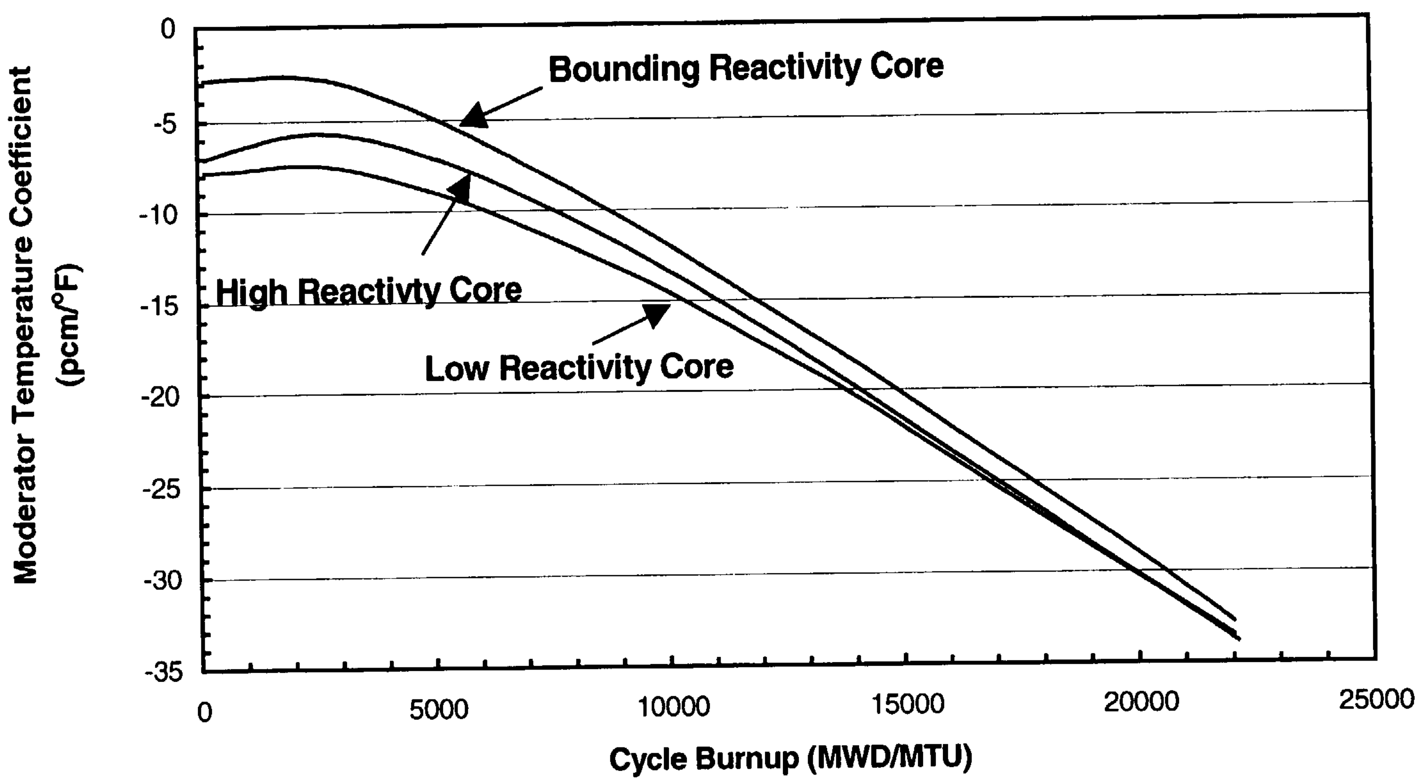

Figure 3-1 HFP Moderator Temperature Coefficient versus Cycle Burnup for the Low, High, and Bounding Reactivity Core Designs

4 DETERMINISTIC ANALYSIS

As documented in SECY-83-293 (Reference 4), the deterministic ATWS analyses that form the basis of the Final ATWS Rule, 10CFR50.62 (b), applicable to Westinghouse designed PWRs, are those documented in NS-TMA-2182 (Reference 6).

For the purpose of supporting the WOG Risk-Informed ATWS PRA program, additional deterministic analyses have been performed. These additional analyses consist of five specific analysis areas. These areas are: 1) the calculation of Critical Power Trajectories (CPT) for the generic Westinghouse ATWS analyses, 2) the calculation of Unfavorable Exposure Times for use in the Risk-Informed PRA model, 3) the calculation of peak RCS pressures at various conditions, 4) part-power ATWS analyses, and 5) the deterministic analyses for the Byron/Braidwood Risk-Informed ATWS PRA pilot program. In the areas that are directly related to the analysis of the ATWS transient conditions, the analytical methods and models used are consistent with those contained in the generic Westinghouse ATWS analyses presented in NS-TMA-2182. Each of these deterministic analysis areas is described in the subsections that follow.

4.1 CRITICAL POWER TRAJECTORY CALCULATIONS

The generic 1979 ATWS analyses that support the basis of the Final ATWS Rule applicable to Westinghouse PWRs are documented in NS-TMA-2182. These ATWS analyses were performed in accordance with NRC guidelines prescribed in NUREG-0460 (Reference 5) and included consideration of 2-Loop, 3-Loop, and 4-Loop Westinghouse PWR plant configurations with the various Westinghouse steam generator models applicable at that time. These ATWS analyses investigated the consequences of specific anticipated transients as prescribed by NUREG-0460 and assumed a full power moderator temperature coefficient (MTC) of -8 pcm/°F. Sensitivity analyses for variations in numerous conditions, including the use of a MTC of -7 pcm/°F, were also included as prescribed by NUREG-0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents a 95% confidence limit on favorable MTC for the fuel cycle.

From the generic ATWS analyses presented in NS-TMA-2182, the limiting condition of concern identified was the potential for RCS overpressurization following a complete loss of all main feedwater without reactor trip. The loss of all main feedwater was modeled to occur in both the Loss of Normal Feedwater (LONF) ATWS event and the Loss of Load (LOL) ATWS event. In the latter event, the complete loss of main feedwater is assumed to occur from the consequential loss of the condenser vacuum following the initiating turbine trip. In the analysis of the LONF ATWS event, a turbine trip is modeled to occur at 30 seconds. In both the LONF ATWS and LOL ATWS events, emergency auxiliary feedwater was assumed to be available at 60 seconds after event initiation. For both events, the peak RCS pressures reached are shown to be less than 3200 psig, the pressure established in Section 6.0 of NS-TMA-2182 as being that conservatively corresponding to the ASME Service Level C stress limit as prescribed by NUREG-0460. The actuation of the turbine trip (in the LONF ATWS) and the actuation of emergency auxiliary feedwater were assumed to occur based on what was later defined to be the AMSAC. As described in Section 2.1, the installation of AMSAC is the requirement of the Final ATWS rule, 10CFR50.62(b), as applicable to Westinghouse designed PWRs.

For use as input to the risk-informed PRA model, Unfavorable Exposure Time (UET), as described later in Section 4.2, must be determined. To determine UET, the reactivity feedback conditions of the core and plant conditions under consideration must be compared to the ATWS analysis conditions that lead to a peak RCS pressure at the pressure limit of 3200 psig. These variable conditions of significance to the resulting peak RCS pressure following the LONF and LOL ATWS events are total reactivity feedback (primarily MTC), primary-side pressure relief capacity, and auxiliary feedwater capacity. For a given primary-side pressure relief configuration and auxiliary feedwater capacity, reactivity feedback (MTC) can be adjusted in the ATWS analysis until the peak RCS pressure during the specific ATWS event equals 3200 psig. At these specific reactivity feedback conditions, the change in power with increasing temperature represents what is defined as the Critical Power Trajectory (or heatup/shutdown characteristic) for the specific plant configuration. The heatup/shutdown characteristics of a given core at various times in the cycle can then be compared to the Critical Power Trajectory (CPT) to establish UET for the given core at the specific plant configuration conditions.

For the purpose of this Risk-Informed ATWS PRA program, ATWS CPTs were generated for the two pressure limiting ATWS events (i.e., LONF and LOL ATWS) based on the 1979 generic ATWS analyses applicable for Westinghouse PWRs. Consistent with the 1979 generic ATWS analyses, the LOFTRAN computer program (Reference 20) was used for these analyses. The ATWS CPTs were generated based on the 4-Loop Westinghouse plant configuration with Model 51 steam generators to be consistent with the generic case presented in detail with sensitivity analyses in NS-TMA-2182. To bound operation at updated power conditions, conditions reflecting an NSSS power level of 3579 MWt were also considered. CPTs were generated for three different primary-side pressure relief configurations (0 PORVs, 1 PORV, 2 PORVs) and two auxiliary feedwater capacities (full AFW, half AFW). The resulting CPT values at elevated inlet temperatures (i.e., ≥ 600°F) are given in Tables 4-1 and 4-2 for the LONF ATWS and LOL ATWS events, respectively. These CPT values, which are given in terms of fraction of nominal power, are used in the determination of the UET values discussed in Section 4.2 that follows. It should be noted that the CPT and UET analyses employ best-estimate assumptions with respect to key input parameters.

Table 4-1 Loss of Normal Feedwater ATWS Critical Power Trajectory Data			
Fraction of 3579 MWt NSSS Power at constant 3200 psig RCS Pressure			
Loss of Normal Feedwater ATWS/Full AFW Capacity			
Tin (°F)	2 PORVs 3 of 3 PSVs	1 PORVs 3 of 3 PSVs	0 PORVs 3 of 3 PSVs
600	0.775	0.740	0.705
620	0.621	0.570	0.519
640	0.434	0.371	0.308
660	0.206	0.132	0.058
Loss of Normal Feedwater ATWS/Half AFW Capacity			
600	0.771	0.736	0.700
620	0.624	0.564	0.513
640	0.425	0.363	0.300
660	0.194	0.122	0.049

Table 4-2 Loss of Load ATWS Critical Power Trajectory Data			
Fraction of 3579 MWt NSSS Power at constant 3200 psig RCS Pressure			
Loss of Load ATWS/Full AFW Capacity			
Tin (°F)	2 PORVs 3 of 3 PSVs	1 PORVs 3 of 3 PSVs	0 PORVs 3 of 3 PSVs
600	0.734	0.697	0.661
620	0.561	0.508	0.456
640	0.360	0.294	0.229
660	0.120	0.042	–
Loss of Load ATWS/Half AFW Capacity			
600	0.720	0.684	0.649
620	0.541	0.490	0.439
640	0.335	0.271	0.207
660	0.091	0.014	–

4.2 UNFAVORABLE EXPOSURE TIME CALCULATIONS

This section documents the results of ATWS UET calculations for use in the WOG Risk-Informed ATWS PRA Program. Also included in this section are calculations of MTC values as a function of cycle burnup for HZP and HFP conditions and control rod worth data at other selected conditions. The latter information is for use in other deterministic ATWS analyses described in Section 4.3.

UET is defined as the time during the cycle when the reactivity feedback is not sufficient to prevent the RCS pressure from exceeding 3200 psig (the ASME Service Level C stress limit).

To calculate UET for a given plant condition and core model, the ANC computer code (Reference 21) is used first to determine the critical power as a function of inlet temperature at various cycle burnups. The "critical power" is the power that results in reactor criticality for a given set of conditions (inlet temperature, pressure, etc.). The ANC results are then compared to the Critical Power Trajectory data presented in Section 4.1 corresponding to the ATWS transient conditions that result in a peak RCS pressure of 3200 psig. The time that the ANC calculated critical power is greater than the ATWS Critical Power Trajectory power represents the time of unfavorable reactivity conditions. This time is termed the Unfavorable Exposure Time.

For the purpose of this program, three different core models were developed and analyzed with respect to ATWS UET. The three models correspond to three different levels of core excess reactivity.

1. A Low Reactivity Core Model was developed to have an approximate 5% UET for the ATWS base case (all PORVs available, full auxiliary feed capability, no credit for control rod insertion). This core has the largest burnable absorber inventory and a maximum HZP MTC of +3.5 pcm/°F. Of the three models developed, this model has the best ATWS performance.

2. A Bounding Reactivity Core Model was developed such that its most positive HZP moderator temperature coefficient was approximately +7 pcm/°F, consistent with the maximum part-power positive MTC Technical Specification limit licensed for Westinghouse plants. In this core model, all WABAs were removed and modifications were made to the IFBA inventory and fuel burnups to effectively tune the model to the desired MTC. This model, which has the least favorable ATWS response, was expressly developed to answer NRC questions with respect to ATWS performance for cores with minimum moderator temperature feedback.

3. A third model, called the High Reactivity Core Model, has a core excess reactivity that is between that of the Low and Bounding Reactivity Core models. This model represents an aggressive yet realistic use of the PMTC Technical Specification. Its most positive HZP MTC is +5 pcm/°F.

For each of these designs, a set of 24 UETs was calculated corresponding to different xenon assumptions, control rod insertion assumptions, and plant configuration assumptions (PORV and auxiliary feedwater capacity) and considering a coolant inlet temperature range of 600° to 660°F. Since the risk model includes an event that the rod control system will respond to the coolant temperature increase by inserting the lead control rod bank or that the operators can take an action to insert the lead control rod bank, UETs were calculated which assume one minute of insertion of the lead bank. These UETs are used in the risk model when control rod insertion is credited. For each of the models, HZP and HFP MTC values as a function of cycle burnup were calculated. Finally, for the Low and Bounding Reactivity Models, control rod worth data were generated for later use in other ATWS analyses (see Section 4.3). The UET, MTC, and rod worth data are summarized in the tables that follow. The UET data are used in the ATWS PRA model as described in Section 5.0.

Table 4-3 UETs for Low Reactivity Model with Equilibrium Xenon, No Control Rod Insertion							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	507.30	3.4	28.2	24.8	4.9
2	2	50	507.30	0.0	76.9	76.9	15.2
3	1	100	507.30	0.0	130.1	130.1	25.6
4	1	50	507.30	0.0	157.3	157.3	31.0
5	0	100	507.30	0.0	210.9	210.9	41.6
6	0	50	507.30	0.0	240.6	240.6	47.4

Table 4-4 UETs for Low Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps)							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	507.30	0.0	0.0	0.0	0.0
2	2	50	507.30	0.0	0.0	0.0	0.0
3	1	100	507.30	0.0	48.5	48.5	9.6
4	1	50	507.30	0.0	72.2	72.2	14.2
5	0	100	507.30	0.0	129.4	129.4	25.5
6	0	50	507.30	0.0	152.2	152.2	30.0

Table 4-5 UETs for Low Reactivity Model with No Xenon, No Control Rod Insertion							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	507.30	0.0	160.3	160.3	31.6
2	2	50	507.30	0.0	182.6	182.6	36.0
3	1	100	507.30	0.0	221.8	221.8	43.7
4	1	50	507.30	0.0	245.2	245.2	48.3
5	0	100	507.30	0.0	310.5	310.5	61.2
6	0	50	507.30	0.0	334.2	334.2	65.9

Table 4-6	UETs for Low Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps)						
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	507.30	0.0	91.5	91.5	18.0
2	2	50	507.30	0.0	117.1	117.1	23.1
3	1	100	507.30	0.0	153.7	153.7	30.3
4	1	50	507.30	0.0	173.0	173.0	34.1
5	0	100	507.30	0.0	215.3	215.3	42.4
6	0	50	507.30	0.0	231.4	231.4	45.6

Table 4-7	UETs for High Reactivity Model with Equilibrium Xenon, No Control Rod Insertion						
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	502.37	0.0	110.1	110.1	21.9
2	2	50	502.37	0.0	133.9	133.9	26.6
3	1	100	502.37	0.0	170.3	170.3	33.9
4	1	50	502.37	0.0	192.2	192.2	38.3
5	0	100	502.37	0.0	238.1	238.1	47.4
6	0	50	502.37	0.0	259.3	259.3	51.6

Table 4-8	UETs for High Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps)						
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	502.37	0.0	0.0	0.0	0.0
2	2	50	502.37	13.5	64.9	51.4	10.2
3	1	100	502.37	0.0	115.4	115.4	23.0
4	1	50	502.37	0.0	136.1	136.1	27.1
5	0	100	502.37	0.0	164.9	164.9	32.8
6	0	50	502.37	0.0	186.9	186.9	37.2

Table 4-9 UETs for High Reactivity Model with No Xenon, No Control Rod Insertion							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	502.37	0.0	194.4	194.4	38.7
2	2	50	502.37	0.0	212.8	212.8	42.4
3	1	100	502.37	0.0	246.2	246.2	49.0
4	1	50	502.37	0.0	270.4	270.4	53.8
5	0	100	502.37	0.0	319.6	319.6	63.6
6	0	50	502.37	0.0	343.0	343.0	68.3

Table 4-10 UETs for High Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps)							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	502.37	0.0	140.8	140.8	28.0
2	2	50	502.37	0.0	158.1	158.1	31.5
3	1	100	502.37	0.0	186.9	186.9	37.2
4	1	50	502.37	0.0	202.2	202.2	40.3
5	0	100	502.37	0.0	232.1	232.1	46.2
6	0	50	502.37	0.0	246.2	246.2	49.0

Table 4-11 UETs for Bounding Reactivity Model with Equilibrium Xenon, No Control Rod Insertion							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	513.75	0.0	150.8	150.8	29.4
2	2	50	513.75	0.0	168.2	168.2	32.7
3	1	100	513.75	0.0	198.4	198.4	38.6
4	1	50	513.75	0.0	216.5	216.5	42.1
5	0	100	513.75	0.0	255.6	255.6	49.7
6	0	50	513.75	0.0	277.7	277.7	54.1

Table 4-12 UETs for Bounding Reactivity Model with Equilibrium Xenon, 1 Minute of Control Rod Insertion (72 Steps)							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	513.75	0.0	106.8	106.8	20.8
2	2	50	513.75	0.0	123.3	123.3	24.0
3	1	100	513.75	0.0	148.5	148.5	28.9
4	1	50	513.75	0.0	161.9	161.9	31.5
5	0	100	513.75	0.0	194.4	194.4	37.8
6	0	50	513.75	0.0	208.1	208.1	40.5

Table 4-13 UETs for Bounding Reactivity Model with No Xenon, No Control Rod Insertion							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	513.75	0.0	215.9	215.9	42.0
2	2	50	513.75	0.0	231.3	231.3	45.0
3	1	100	513.75	0.0	260.6	260.6	50.7
4	1	50	513.75	0.0	280.8	280.8	54.7
5	0	100	513.75	0.0	331.9	331.9	64.6
6	0	50	513.75	0.0	365.8	365.8	71.2

Table 4-14 UETs for Bounding Reactivity Model with No Xenon, 1 Minute of Control Rod Insertion (72 Steps)							
Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	513.75	0.0	168.7	168.7	32.8
2	2	50	513.75	0.0	182.7	182.7	35.6
3	1	100	513.75	0.0	206.7	206.7	40.2
4	1	50	513.75	0.0	220.4	220.4	42.9
5	0	100	513.75	0.0	249.6	249.6	48.6
6	0	50	513.75	0.0	264.1	264.1	51.4

Table 4-15 HFP and HZP Moderator Temperature Coefficients for the Low Reactivity Core Model

Burnup (MWD/MTU)	Days	HFP MTC with HFP Eq. Xenon (pcm/°F)	HZP MTC with No Xenon (pcm/°F)
0	0.0	-4.39*	2.77
150	3.5	-7.79	3.00
1000	23.0	-7.68	3.27
2000	46.0	-7.42	3.50
3000	69.0	-7.59	3.47
4000	92.0	-8.20	3.16
5000	115.0	-8.95	2.71
6000	138.0	-9.86	2.18
8000	184.0	-12.08	0.81
10000	230.0	-14.53	-0.90
12000	276.0	-17.42	-2.87
14000	322.1	-20.32	-4.87
16000	368.1	-23.63	-7.11
18000	414.1	-26.94	-9.47
20000	460.1	-30.23	-11.98
21700	499.2	-33.11	-14.15
22100	508.4	-33.87	-14.64

* 0 MWD/MTU case has no xenon.

Table 4-16 HFP and HZP Moderator Temperature Coefficients for the High Reactivity Core Model			
Burnup (MWD/MTU)	Days	HFP MTC with HFP Eq. Xenon (pcm/°F)	HZP MTC with No Xenon (pcm/°F)
0	0.0	-3.58*	3.34
150	3.4	-7.05	3.58
1000	22.8	-6.39	4.30
2000	45.7	-5.77	4.85
3000	68.5	-5.75	5.00
4000	91.3	-6.26	4.73
5000	114.1	-7.07	4.27
6000	137.0	-8.06	3.64
8000	182.6	-10.54	1.97
10000	228.3	-13.36	-0.07
12000	273.9	-16.46	-2.36
14000	319.6	-19.82	-4.62
16000	365.3	-23.27	-7.13
18000	410.9	-26.65	-9.55
20000	456.6	-30.16	-12.12
22006	502.4	-33.50	-14.66

* 0 MWD/MTU case has no xenon.

Table 4-17 HFP and HZP Moderator Temperature Coefficients for the Bounding Reactivity Core Model

Burnup (MWD/MTU)	Days	HFP MTC with HFP Eq. Xenon (pcm/°F)	HZP MTC with No Xenon (pcm/°F)
0	0.0	0.53*	6.32
150	3.4	-2.84	6.56
1000	22.9	-2.72	6.91
2000	45.7	-2.63	7.10
3000	68.6	-2.99	6.92
4000	91.5	-3.85	6.44
5000	114.3	-4.90	5.76
6000	137.2	-6.12	4.96
8000	182.9	-8.89	3.07
10000	228.6	-11.92	0.94
12000	274.4	-15.19	-1.44
14000	320.1	-18.50	-3.85
16000	365.8	-21.95	-6.25
18000	411.5	-25.43	-8.82
20000	457.3	-28.93	-11.29
22000	503.0	-32.77	-13.96
22470	513.7	-33.46	-14.59

* 0 MWD/MTU case has no xenon.

Table 4-18 Differential and Integral Rod Worths for the Bounding Reactivity Core at 2000 MWD/MTU

Control Bank D Position (steps withdrawn)	Steps Inserted	Time After Initiation of Rod Insertion (sec)	Integral Rod Worth (pcm)	Differential Rod Worth* (pcm/step)
231	0	0.0	0.00	–
225	6	5.0	0.00	–
220	11	9.2	2.5	0.50
210	21	17.5	14.7	1.22
200	31	25.8	31.1	1.64
190	41	34.2	52.0	2.09
180	51	42.5	78.1	2.61
170	61	50.8	104.3	2.62
159	72	60.0	133.0	2.61

* The differential rod worth is the average value over the step interval. For example, the first value in the column is the average differential rod worth between 220 and 225 steps withdrawn. The top of the active fuel height is at ~225 steps withdrawn. The rod worth above the active fuel is assumed to be 0.0.

Table 4-19 Differential and Integral Rod Worths for the Low Reactivity Core at 2000 MWD/MTU				
Control Bank D Position (steps withdrawn)	**Steps Inserted**	**Time After Initiation of Rod Insertion (sec)**	**Integral Rod Worth (pcm)**	**Differential Rod Worth* (pcm/step)**
231	0	0.0	0.00	–
225	6	5.0	0.00	–
220	11	9.2	1.8	0.36
210	21	17.5	17.9	1.61
200	31	25.8	38.7	2.08
190	41	34.2	63.3	2.46
180	51	42.5	89.4	2.61
170	61	50.8	118.3	2.89
159	72	60.0	150.3	2.91

* The differential rod worth is the average value over the step interval. For example, the first value in the column is the average differential rod worth between 220 and 225 steps withdrawn. The top of the active fuel height is at ~225 steps withdrawn. The rod worth above the active fuel is assumed to be 0.0.

4.3 PEAK RCS PRESSURE CALCULATIONS

To support the WOG Risk-Informed ATWS PRA Program, maximum RCS pressures following the pressure limiting Loss of Load ATWS event were calculated for various conditions. These maximum RCS pressure values are for use in establishing upper bound RCS pressure levels to be considered in addressing LERF. These ATWS RCS pressure calculations were performed using the LOFTRAN code consistent with the ATWS analysis methodology used to support the generic ATWS analyses reported in NS-TMA-2182.

Peak RCS pressures following the pressure limiting Loss of Load ATWS event were determined for two different reactivity cores: 1) a bounding reactivity core model designed to a HZP MTC of approximately +7 pcm/°F (HFP MTC equivalent of -2.63 pcm/°F), and 2) a low reactivity core model with a HZP MTC of +3.5 pcm/°F (HFP MTC equivalent of -7.42 pcm/°F).

For these two core conditions, peak RCS pressures were determined for the generic ATWS model of a 4-Loop Westinghouse PWR with Model 51 steam generators at an uprated power level of 3579 MWt. Cases were considered with and without control rod insertion for both full and half auxiliary feedwater capacity with varying pressure relief capacities to reflect operation with 2, 1, or 0 PORVs. The cases assuming rod insertion credit 72 steps of rod insertion of the lead control bank (Bank D). The rod worths used in the modeling of rod control in the bounding reactivity and low reactivity core model cases are based on the information provided in Tables 4-18 and 4-19, respectively.

The resulting peak RCS pressures are summarized in Tables 4-20 and 4-21 for the bounding reactivity core model and the low reactivity core model, respectively.

Table 4-20 Loss of Load ATWS, Bounding Reactivity Core Model HZP MTC of ~+7 pcm/°F (HFP MTC = -2.63 pcm/°F)

Case	No. of PORVs	AFW Capacity (%)	Rod Control	Peak RCS Pressure (psia)
1	2	100	No	3546
2	1	100	No	3822
3	0	100	No	4093
4	2	50	No	3630
5	1	50	No	3955
6	0	50	No	4110
7	2	100	Yes	3333
8	1	100	Yes	3563
9	0	100	Yes	3914
10	2	50	Yes	3412
11	1	50	Yes	3670
12	0	50	Yes	4055

Table 4-21 Loss of Load ATWS, Low Reactivity Core Model HZP MTC of +3.5 pcm/°F (HFP MTC = -7.42 pcm/°F)				
Case	**No. of PORVs**	**AFW Capacity (%)**	**Rod Control**	**Peak RCS Pressure (psia)**
13	2	100	No	3090
14	1	100	No	3285
15	0	100	No	3563
16	2	50	No	3164
17	1	50	No	3374
18	0	50	No	3664
19	2	100	Yes	2924
20	1	100	Yes	3078
21	0	100	Yes	3308
22	2	50	Yes	2987
23	1	50	Yes	3162
24	0	50	Yes	3411

4.4 PART-POWER ATWS ANALYSES

To support the WOG Risk-Informed ATWS PRA Program and, in particular, the potential for operation at part-power conditions below the C-20 AMSAC actuation setpoint (i.e., ≤40% power), additional deterministic ATWS analyses at part-power conditions were performed. These analyses consisted of the determination of ATWS Critical Power Trajectory (CPT) data for operation at a power level of 40% without crediting AMSAC and the subsequent calculation of the corresponding UETs.

For the part-power conditions at an initial power level of 40%, ATWS CPTs were generated for the two pressure limiting ATWS events (i.e., LONF and LOL ATWS). For these conditions, the analyses again were based on the generic ATWS 4-Loop Westinghouse plant configuration with Model 51 steam generators to be consistent with the 1979 generic ATWS analyses. These ATWS analyses were performed using the LOFTRAN computer program and assumed conditions reflecting operation at 40% of a NSSS full power level of 3579 MWt. CPTs were generated for three different primary-side pressure relief configurations (0 PORVs, 1 PORV, 2 PORVs). No auxiliary feedwater was assumed in these analyses since no credit was taken for an AMSAC actuation. The resulting CPT values at elevated inlet temperatures (i.e., ≥600°F) are given in Tables 4-22 and 4-23 for the LONF ATWS and LOL ATWS events, respectively. These CPT values, which are given in terms of fraction of nominal power, are used in the determination of the UET values as follows. Note that in the data presented in Table 4-23, the power level for the CPT at 600°F with 2 PORVs available exceeds the initial power condition of 40% power. The reason this occurs is associated with the use of a constant RCS pressure assumption in the calculation of the CPT. During the ATWS transient, the RCS pressure is well below 3200 psig at RCS inlet temperatures less than approximately 630°F. A constant pressure of 3200 psig is conservatively assumed in the calculation of the CPT values and is consistent with the pressure assumption used in the corresponding ANC power search calculations performed to determine UET.

As described in Section 4.2, to determine UET, the ANC computer code is used to first determine the critical power as a function of inlet temperature at various cycle burnup conditions. The ANC results are then compared to the Critical Power Trajectory data corresponding to the ATWS transient conditions that result in a peak RCS pressure of 3200 psig. For the part-power UET calculations, the CPT data used is for the limiting LOL ATWS event initiated at 40% power as provided in Table 4-23. The time that the ANC calculated critical power is greater than the ATWS CPT represents the time of unfavorable reactivity conditions. This time is the UET.

For the part-power UET calculations, the ANC critical power search was initiated from a condition corresponding to 40% power. Two different xenon conditions were considered: HFP equilibrium xenon and no xenon. The HFP equilibrium xenon assumption reflects conditions associated with prolonged operation at full power and the postulated ATWS event occurring at 40% power while descending in power. The no xenon assumption reflects conditions associated with a postulated ATWS event occurring at 40% power during a power ascension following a prolonged shutdown or initial startup condition. For each condition, three different plant configurations were considered: 2 PORVs available, 1 PORV available, and no PORVs available (see Table 4-23).

The calculated UET values are presented in the attached tables for a low reactivity core model, a high reactivity core model, and the bounding reactivity core model as described in Section 4.2. For each model, the two xenon conditions and three plant configurations resulted in six UET calculations. Review of the tables indicates that the part-power UETs increase significantly as the xenon concentration decreases. Also, as in the full power cases considered in Section 4.2, the UETs increase with core excess reactivity, i.e., the UETs are the largest for the bounding reactivity core model and the smallest for the low reactivity core model. These results are consistent with expectations since lower xenon concentration and smaller burnable absorber inventory lead to higher critical boron concentrations, and, therefore, weaker negative moderator feedback.

The resulting UETs for the various part-power conditions are summarized in Tables 4-24 through 4-29. This part-power UET data is used in the ATWS PRA model as described in Section 5.

Table 4-22 Loss of Normal Feedwater ATWS Critical Power Trajectory Data			
Fraction of 3579 MWt NSSS Power at Constant 3200 psig RCS Pressure Loss of Normal Feedwater ATWS w/o AMSAC from 40% Initial Power			
Tin (°F)	**2 PORVs 3 of 3 PSVs**	**1 PORVs 3 of 3 PSVs**	**0 PORVs 3 of 3 PSVs**
600	0.365	0.340	0.315
620	0.315	0.278	0.238
640	0.221	0.173	0.122
660	0.068	0.013	–

Table 4-23 Loss of Load ATWS Critical Power Trajectory Data			
Fraction of 3579 MWt NSSS Power at Constant 3200 psig RCS Pressure Loss of Load ATWS w/o AMSAC from 40% Initial Power			
Tin (°F)	**2 PORVs 3 of 3 PSVs**	**1 PORVs 3 of 3 PSVs**	**0 PORVs 3 of 3 PSVs**
600	0.418	0.392	0.364
620	0.395	0.357	0.314
640	0.322	0.273	0.219
660	0.184	0.128	0.066

Table 4-24 UETs for Low Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition						
Case	**PORVs Available**	**Cycle Length (days)**	**UET Start (days)**	**UET End (days)**	**Total UET (days)**	**UET (%)**
1	2	507.30	0.0	0.0	0.0	0.0
2	1	507.30	0.0	0.0	0.0	0.0
3	0	507.30	0.0	0.0	0.0	0.0

Table 4-25 UETs for Low Reactivity Model with No Xenon, 40% Power Initial Condition

Case	PORVs Available	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	507.30	0.0	0.0	0.0	0.0
2	1	507.30	29.9	60.6	30.7	6.1
3	0	507.30	0.0	114.5	114.5	22.6

Table 4-26 UETs for High Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition

Case	PORVs Available	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	502.37	0.0	0.0	0.0	0.0
2	1	502.37	0.0	0.0	0.0	0.0
3	0	502.37	0.0	0.0	0.0	0.0

Table 4-27 UETs for High Reactivity Model with No Xenon, 40% Power Initial Condition

Case	PORVs Available	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	502.37	23.5	99.0	75.5	15.0
2	1	502.37	0.0	132.6	132.6	26.4
3	0	502.37	0.0	162.4	162.4	32.3

Table 4-28 UETs for Bounding Reactivity Model with HFP Equilibrium Xenon, 40% Power Initial Condition

Case	PORVs Available	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	513.75	0.0	0.0	0.0	0.0
2	1	513.75	0.0	84.0	84.0	16.4
3	0	513.75	0.0	113.9	113.9	22.2

Table 4-29 UETs for Bounding Reactivity Model with No Xenon, 40% Power Initial Condition

Case	PORVs Available	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	513.75	0.0	144.7	144.7	28.2
2	1	513.75	0.0	166.7	166.7	32.5
3	0	513.75	0.0	189.6	189.6	36.9

4.5 BYRON/BRAIDWOOD ANALYSIS

To support the Byron/Braidwood pilot application of the WOG Risk-Informed ATWS PRA Program, deterministic ATWS analyses were performed to reflect conditions representative of Exelon Nuclear's Byron and Braidwood Stations. These analyses consist of both ATWS Critical Power Trajectory (CPT) and UET calculations.

The Byron and Braidwood units are currently licensed to a NSSS power corresponding to 3600.6 MWt. The Byron 1 and Braidwood 1 units operate with BWI replacement steam generators (RSGs) whereas the Byron 2 and Braidwood 2 units operate with the original Westinghouse designed Model D5 steam generators. To reflect operation at the increased power level of 3600.6 MWt and to account for the differences in steam generators, plant-specific CPT data were generated for the Byron/Braidwood units for this application. To support the CPT calculations, the generic 4-Loop Westinghouse ATWS LOFTRAN model was modified to reflect the Byron/Braidwood plant licensed operating conditions and to reflect the differences in the steam generator design from those associated with the Model 51 steam generator used in the generic Westinghouse ATWS model.

As was done for the prior CPT calculations described in Section 4.1, ATWS CPTs were generated for the two pressure limiting ATWS events (i.e., LONF and LOL ATWS). Consistent with the 1979 generic ATWS analysis methodology, the LOFTRAN computer program was used for these analyses. CPTs were generated for three different primary-side pressure relief configurations (0 PORVs, 1 PORV, 2 PORVs) and two auxiliary feedwater capacities (full AFW, half AFW). The resulting CPT values at elevated inlet temperatures (i.e., ≥600°F) for the Byron 1 and Braidwood 1 units with BWI RSGs are given in Tables 4-30 and 4-31 for the LONF ATWS and LOL ATWS events, respectively. The CPT values for Byron 2 and Braidwood 2 with the Westinghouse Model D5 steam generators are given in Tables 4-32 and 4-33 for the LONF ATWS and LOL ATWS events, respectively. These CPT values, which are given in terms of fraction of nominal power, are used in the determination of the UET values for Byron and Braidwood as follows.

As described in Section 4.2, to determine UET, the ANC computer code is used to first determine the critical power as a function of inlet temperature at various cycle burnup conditions. The ANC results are then compared to the Critical Power Trajectory data corresponding to the ATWS transient conditions that result in a peak RCS pressure of 3200 psig. For the UET calculations performed to support the Byron/Braidwood pilot application of the WOG Risk-Informed ATWS PRA program, the CPT data used is that for the limiting LOL ATWS event with the BWI RSGs (i.e., Table 4-31). The ATWS CPT data for operation of the Byron 1/Braidwood 1 units with BWI RSGs bounds that for operation of Byron 2/ Braidwood 2 units with the Westinghouse Model D5 steam generators. The time that the ANC calculated critical power is greater than the ATWS CPT represents the time of unfavorable reactivity conditions. This is the UET.

For this pilot application, two sets of UETs are provided: the first set employs a core model that is characteristic of current Byron/Braidwood fuel management, while the second set employs a core model that is characteristic of future Byron/Braidwood fuel management, assuming the current 5% UET limit is lifted. For each core model, UETs assuming no control rod insertion and 72 steps of D-bank insertion are provided.

Tables 4-34 through 4-37 provide UET calculations for the two core design models. For each core model, two sets of UET calculations were performed: one set assuming no control rod insertion and a second set assuming 72 steps of D-bank insertion (approximately one minute of insertion). Each set of UET calculations comprises six ATWS scenarios covering various PORV and auxiliary feedwater assumptions. In all, a total of 24 UET calculations were performed. For each scenario, both the start and end of the unfavorable portion of the cycle are given. The total UET time in days and as a percentage of the cycle are also provided. This UET data is used in the ATWS PRA model for the Byron/Braidwood pilot application as described in Section 9.

Table 4-30 Loss of Normal Feedwater ATWS Critical Power Trajectory Data for Byron 1/ Braidwood 1 with BWI RSGs

Fraction of 3600.6 MWt NSSS Power at Constant 3200 psig RCS Pressure Loss of Normal Feedwater ATWS/Full AFW Capacity			
Tin (°F)	2 PORVs 3 of 3 PSVs	1 PORVs 3 of 3 PSVs	0 PORVs 3 of 3 PSVs
600	0.741	0.707	0.672
620	0.560	0.512	0.461
640	0.356	0.295	0.230
650	0.242	0.175	0.103
660	0.115	0.042	–
Loss of Normal Feedwater ATWS/Half AFW Capacity			
600	0.729	0.696	0.660
620	0.543	0.496	0.445
640	0.335	0.275	0.209
650	0.219	0.153	0.080
660	0.089	0.018	–

Table 4-31 Loss of Load ATWS Critical Power Trajectory Data for Byron 1/Braidwood 1 with BWI RSGs

Fraction of 3600.6 MWt NSSS Power at Constant 3200 psig RCS Pressure Loss of Load ATWS/Full AFW Capacity			
Tin (°F)	**2 PORVs 3 of 3 PSVs**	**1 PORVs 3 of 3 PSVs**	**0 PORVs 3 of 3 PSVs**
600	0.720	0.685	0.648
620	0.530	0.481	0.426
640	0.319	0.255	0.186
650	0.201	0.131	0.054
660	0.070	–	–
Loss of Load ATWS/Half AFW Capacity			
600	0.710	0.675	0.638
620	0.516	0.465	0.412
640	0.301	0.236	0.167
650	0.182	0.109	0.033
660	0.049	–	–

Table 4-32 Loss of Normal Feedwater ATWS Critical Power Trajectory Data for Byron 2/Braidwood 2 with W D5 SGs

Fraction of 3600.6 MWt NSSS Power at Constant 3200 psig RCS Pressure Loss of Normal Feedwater ATWS/Full AFW Capacity			
Tin (°F)	**2 PORVs 3 of 3 PSVs**	**1 PORVs 3 of 3 PSVs**	**0 PORVs 3 of 3 PSVs**
600	0.885	0.851	0.813
620	0.772	0.721	0.665
640	0.606	0.548	0.483
650	0.509	0.447	0.378
660	0.395	0.331	0.260
Loss of Normal Feedwater ATWS/Half AFW Capacity			
600	0.881	0.846	0.809
620	0.765	0.714	0.659
640	0.598	0.539	0.476
650	0.501	0.438	0.370
660	0.386	0.322	0.252

Table 4-33 Loss of Load ATWS Critical Power Trajectory Data for Byron 2/Braidwood 2 with W D5 SGs			
Fraction of 3600.6 MWt NSSS Power at constant 3200 psig RCS Pressure Loss of Load ATWS/Full AFW Capacity			
Tin (°F)	**2 PORVs 3 of 3 PSVs**	**1 PORVs 3 of 3 PSVs**	**0 PORVs 3 of 3 PSVs**
600	0.879	0.842	0.800
620	0.764	0.708	0.645
640	0.596	0.532	0.460
650	0.499	0.430	0.353
660	0.384	0.314	0.233
Loss of Load ATWS/Half AFW Capacity			
600	0.866	0.828	0.786
620	0.744	0.687	0.624
640	0.574	0.509	0.435
650	0.475	0.405	0.327
660	0.360	0.288	0.205

Table 4-34 UETs for Current Byron/Braidwood Core Designs, No Control Rod Insertion

Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	505.30	0.0	0.0	0.0	0.0
2	2	50	505.30	0.0	0.0	0.0	0.0
3	1	100	505.30	19.6	141.2	121.6	24.1
4	1	50	505.30	0.0	220.4	220.4	43.6
5	0	100	505.30	0.0	345.1	345.1	68.3
6	0	50	505.30	0.0	381.4	381.4	75.5

Table 4-35 UETs for Current Byron/Braidwood Core Designs, 1 Minute of Control Rod Insertion (72 Steps)

Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	505.30	0.0	0.0	0.0	0.0
2	2	50	505.30	0.0	0.0	0.0	0.0
3	1	100	505.30	0.0	0.0	0.0	0.0
4	1	50	505.30	0.0	0.0	0.0	0.0
5	0	100	505.30	28.9	149.1	120.2	23.8
6	0	50	505.30	3.5	191.6	188.1	37.2

Table 4-36 UETs for Future Byron/Braidwood Core Designs, No Control Rod Insertion

Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	500.92	0.0	141.2	141.2	28.2
2	2	50	500.92	0.0	166.8	166.8	33.3
3	1	100	500.92	0.0	231.3	231.3	46.2
4	1	50	500.92	0.0	256.1	256.1	51.1
5	0	100	500.92	0.0	332.5	332.5	66.4
6	0	50	500.92	0.0	362.1	362.1	72.3

Table 4-37 UETs for Future Byron/Braidwood Core Designs, 1 Minute of Control Rod Insertion (72 Steps)

Case	PORVs Available	Auxiliary Feed (%)	Cycle Length (days)	UET Start (days)	UET End (days)	Total UET (days)	UET (%)
1	2	100	500.92	0.0	0.0	0.0	0.0
2	2	50	500.92	0.0	80.8	80.8	16.1
3	1	100	500.92	0.0	142.9	142.9	28.5
4	1	50	500.92	0.0	162.7	162.7	32.5
5	0	100	500.92	0.0	208.4	208.4	41.6
6	0	50	500.92	0.0	225.2	225.2	45.0

5 PROBABILISTIC RISK ANALYSIS

Sections 5.1 and 5.2 present the generic ATWS event tree models, analysis, and results. This information is applicable to all anticipated transients in which main feedwater is lost. This is not applicable to LOSP events, inadvertent safety injection events, and inadvertent and manual reactor trip events. LOSP ATWS events are addressed in Section 5.3 of this report. Transient events initiated by a reactor trip by definition are not ATWS events. Inadvertent SI events are low frequency events in which an unnecessary SI occurred. These events are different from the limiting ATWS events because initially there is a power, temperature, and pressure reduction due to the excess boron. The inadvertent SI differentiates this event from the typical ATWS transient. In addition, these events are infrequent events and including them with the more frequent transient events would have a minor impact on the initiating event frequency and essentially no impact on the results of this assessment.

This analysis and the results are provided for a low reactivity core, a high reactivity core, and a bounding reactivity core. These cores are described as:

- The low reactivity core has a 5% UET for the reference condition of no CRI, 100% AFW, and all PORVs available. This core has the largest burnable absorber inventory and a maximum HZP MTC of +3.5 pcm/°F.
- The high reactivity core has a excess core reactivity between the low and bounding models. This core represents an aggressive, but realistic use of the PMTC Technical Specification with a most positive HZP MTC of +5 pcm/°F.
- The bounding reactivity core was developed such that its most positive HZP MTC is +7 pcm/°F which is consistent with the MTC Technical Specification for some plants. This core model was specifically developed to answer NRC questions related to ATWS performance of cores with minimum moderator temperature feedback.

CDF and LERF assessments are provided in the following sections as required by the RI approach described in Regulatory Guide 1.174. ATWS analyses are provided for the power operation regimes of start-up (power increase), shutdown (power decrease), and steady state at-power operation. These analyses consider mitigating system availability for these power conditions, in addition to the related equilibrium xenon concentrations. This analysis is applicable to all transient events with the failure of the reactor to trip except for the LOSP ATWS event. A separate analysis is provided for LOSP ATWS events.

Section 5.1 examines the impact on CDF of high reactivity and bounding reactivity cores relative to a low reactivity core for several different ATWS operating states. Section 5.2 extends this analysis to the impact on LERF by examining the expected RCS pressures and response of the RCS components to these pressures. Section 5.3 evaluates the CDF impact from LOSP ATWS events. Section 5.4 provides a summary of the results and conclusions.

5.1 ATWS CORE DAMAGE FREQUENCY ANALYSIS FOR THE LOW, HIGH, AND BOUNDING REACTIVITY CORES

In developing the ATWS risk model, it is necessary to consider several key plant operating factors. These are the power level and the plant power activity. Power level is important since below 40% power the AMSAC system is not in operation, and if an ATWS event occurred, AMSAC cannot be credited for starting AFW and tripping the turbine. Above 40% power AMSAC can be credited. The plant power activity refers to whether the plant is in a startup condition, shutdown condition, or operating at 100% (or near 100%) power. This is an important consideration with regard to equilibrium xenon concentration and reliability of systems, as discussed in later sections. Xenon concentration is an important consideration with regard to UETs.

Based on these key plant operating factors, five plant ATWS operating states have been defined. Table 5-1 defines these five states. Many current plant PRA models only consider two states; operation above and below 40% power with equilibrium xenon. These five states can be reduced to four by combining States 3 and 4 since they both have AMSAC available and equilibrium xenon. These four ATWS states are:

- ATWS State 1: Power Level <40%, Plant Startup
- ATWS State 2: Power Level ≥40%, Plant Startup
- ATWS State 3/4: Power Level ≥40%, Plant At-Power Operation and Plant Shutdown
- ATWS State 5: Power Level <40%, Plant Shutdown

5.1.1 ATWS State 3/4: Plant At-Power Operation and Plant Shutdown, Power Level ≥40%

This state represents plant operation when the power level is greater than or equal to 40% with equilibrium xenon concentration. AMSAC is operable in this state. UETs used in this evaluation are for the 100% power level with equilibrium xenon which can be applied conservatively to power levels down to 40%.

5.1.1.1 ATWS State 3/4 Event Tree

An ATWS event tree was developed based on the event tree in WCAP-11992. The overall approach uses the unfavorable exposure time concept. This concept determines the time during the cycle that the reactor cannot mitigate the ATWS overpressure transient, that is, the time the RCS pressure will exceed the pressure limit corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit criterion (3200 psig). This time is referred to as the unfavorable exposure time or UET. The UET is only important if the reactor fails to trip, that is, the rods fail to fall into the core. This failure can be due to failure of automatic RPS signals and manual actions, or mechanical failure of the rods or control rod drive mechanisms (CRDMs). The UETs for a given core are dependent on the availability of auxiliary feedwater to the steam generators for heat removal, partial insertion of the control rods (if rod insertion for reactor trip fails), and availability of RCS pressure relief.

Figure 5-1 shows the event tree. The first top event, IEV, is the frequency of a plant event that requires a reactor trip. The next four top events, RT (reactor trip, development of the trip signal), OAMG (operator action to trip the reactor via the motor-generator sets), CRI (operator action or rod control system to drive the control rods into the core), and CR (control rod insertion), are all related to equipment and operator action failures that lead to an ATWS event. The engineered safety features actuation system (ESFAS) signals and AMSAC are modeled as alternate methods to start AFW and trip the turbine given that an ATWS event has occurred. AFW100 and AFW50 model the probability of achieving 100% and 50% AFW flow. This, along with the availability of pressurizer safety valves and PORVs, are important in mitigating the overpressure event. PR (pressure relief) accounts for the unavailability or failure of safety valves and PORVs. The UETs are factored into the primary pressure relief top event. The UETs are dependent on the available AFW flow (100% or 50% flow), pressure relief available (number of PORVs available or not blocked), and success of partial control rod insertion. LTS (long-term shutdown) models the ability to shut down the reactor by boration after mitigation of the pressure transient.

Several important clarifications on the event tree follow:

- Control rod insertion (CR) is addressed following success of the reactor trip signal (RT) or failure of reactor trip signal and success of the operator to trip the reactor from the motor-generator (MG) sets (OAMG).
- The ESFAS is credited with starting AFW and tripping the turbine only for failures of reactor trip that cannot be associated with common cause failures between development of the reactor trip signal and ESFAS signals. The ESFAS signal is only credited if reactor trip fails due to failure of the control rods to fall into the core given a signal to trip was available.
- AMSAC is assumed to be a diverse means (diverse from the RPS) of actuating AFW and providing turbine trip.
- It is assumed that if an ATWS event has occurred, core damage will occur if AFW is not initiated or the turbine is not tripped. This is consistent with the ATWS approach in WCAP-11992 and the AMSAC criteria in the ATWS Rule.
- LTS is not addressed if CRI is successful. With successful CRI, it is assumed that the control rods will continue to be inserted and the reactor shut down.

The following applies to the path endstates, as defined in the "Class" column:

- CD-Sig identifies core damage endstates due to failure to generate signals to start AFW and trip the turbine. This is assumed to be a high RCS pressure (>3200 psi) core damage sequence.
- CD-AFW identifies core damage endstates due to failure to supply at least 50% AFW flow. This is assumed to be a high RCS pressure (>3200 psi) core damage sequence.
- CD-LTS identifies core damage endstates due to failure to provide long-term shutdown following successful mitigation of the pressure transient. This is a low RCS pressure core damage sequence.

- CD-PRA, CD-PRB, CD-PRC, and CD-PRD identify core damage endstates related to failure of adequate pressure relief. The specific endstate is based on the success or failure of achieving the equivalent of 72 steps lead bank reactivity insertion and the amount of AFW flow (100% or <100% and ≥50% or <50%). This is assumed to be a high RCS pressure (>3200 psi) core melt sequence.

The following describes the ATWS event tree and top events in more detail.

5.1.1.2 IEV: Initiating Event Frequency

This is the frequency of transient events that can lead to ATWS events. This includes all transient events with equilibrium xenon and initial power levels greater than 40% except, as previously noted, for LOSP, inadvertent safety injections, and inadvertent and manual reactor trips. The first year of operation is also eliminated since this is usually not typical of plant operation in the following years.

Consistent with the ATWS states previously discussed, reactor trips are divided into five different groups.

- Startup, power level <40%
- Startup, 40% ≤ power level < 95% power
- At-power, power level ≥95%
- Shutdown, 40% ≤ power level < 95% power
- Shutdown, power level <40%

Since plants operate at 100% power, or close to it, it was assumed that any trips in the 95% to 100% power range are at-power trips. It was also assumed that any trips in the 0% to 95% range occurred either during startup or shutdown since plants typically operate at or near 100% power. It was also assumed that startup trips occur prior to establishing equilibrium xenon and that shutdown trips occur after equilibrium xenon has been established.

Initiating event information from WCAP-15210 (Reference 11) was used to develop the trip frequencies for each of these states. WCAP-15210 is based on the events in INEEL/EXT-98-00401 ("Rates of Initiating Events at U.S. Commercial Nuclear Power Plants – 1987 through 1995," Reference 12) and updated with information from Licensee Event Reports for 1996 and 1997. Only data for Westinghouse NSSS plants was used. The number of trips have been decreasing in recent years and to get a representative initiating event frequency for current plant operation, data previous to 1993 was excluded, as were LOSP, inadvertent safety injections, inadvertent and manual reactor trips, and trips in the first year of operation. From this information, the number of trips initiated from below 40% power, from 40% to 95% power, and above 95% power are:

- Number of trips with power level <40% = 38
- Number of trips with power level ≥40% to <95% = 24
- Number of trips with power level ≥95% = 178

The number of startup trips and shutdown trips cannot be determined from the information contained in Reference 11 or 12. But a previous study on the reactor protection system (WCAP-14333, Reference 13) collected such information. In Section 8.4 of Reference 13, the probability of a reactor trip on startup is given as 0.088 and on shutdown as 0.068. From this information the number of trips at less than 40% power and from 40% to 95% power can be divided into startup and shutdown trips as follows:

- Number of startup trips from below 40% power = 38 x 0.088/(0.088 + 0.068) = 21.4
- Number of shutdown trips from below 40% power = 38 x 0.068/(0.088 + 0.068) = 16.6
- Number of startup trips from 40% to 95% power = 24 x 0.088/(0.088 + 0.068) = 13.5
- Number of shutdown trips from 40% to 95% power = 24 x 0.068/(0.088 + 0.068) = 10.5

Table 5-2 provides a summary of the number of events and trip frequency for the five ATWS states.

The events of interest in this part of the study are those initiated from a power level greater than 40% and with full power equilibrium xenon. These are trips with the power level greater than or equal to 95%, and greater than or equal to 40% during shutdown (ATWS states 3 and 4).

Trip frequency = 0.90 + 0.05 = 0.95 events/year (from Table 5-2). For this study a trip frequency of 1.0/yr is used.

5.1.1.3 RT: Reactor Trip Signal from the RPS

The reactor protection system (RPS) fault tree model, for the solid state protection system with the 7300 analog series signal processing, provided in NUREG/CR-5500, Vol. 2 (Reference 14) was used in the analysis. This fault tree models failure of a reactor trip signal and credits signals developed from two sets of analog (instrument) channels. For all transient events, reactor trip signals will be generated from at least two sets of analog channels, so this is appropriate. In addition, an operator action is credited to trip the reactor from the control room reactor trip switch. This operator action backs up failures in the RPS related to the analog channels and components in the solid state protection system, but not involving failures of the reactor trip breakers (RTB).

The component failure data used in the RPS fault tree is taken directly from Reference 14. The human error probability (HEP) for the operator action to trip the reactor is 1.0E-02. This is based on a review of HEPs used for this operator action in PRA models for W NSSS plants and represents a reasonably conservative value.

5.1.1.4 OAMG: Operator Action to Trip the Reactor via the MG Sets

The operator can take an action to trip the reactor by interrupting power to the CRDMs via the MG sets. Since this trips the reactor by interrupting power to the CRDMs, the control rods still need to drop into the reactor. If this action is successful, then the CR top event is addressed. If this action fails, then the operator can take an action to drive the control rods into the core. If the rod control system is in automatic, the rods will begin to move into the core automatically. This last action is addressed in top event CRI.

The failure probability used for OAMG depends on the reason RT failed. If the RT signal failed due to SSPS or channel signal processing (analog channels), then the OA included in the RT top event has also failed and there is a higher probability that this OA will also fail. If the RT signal failed due to RTB failure, then the OA in RT was most likely successful and OAMG can be assumed to be independent of, or not conditional on, other operator actions already taken.

The following human error probabilities are used:

- 0.5 is used when RT fails due to reasons related to the OA failure to trip the reactor in RT in conjunction with logic cabinet or analog channel processing failures – this is a conservative conditional failure probability (conditional on a previous OA already failing).

- 1E-02 is used when RT fails due to reasons not related to failure of the operator action to trip the reactor in RT, that is, when failures are related to RTB failures. This is based on a review of HEPs used for this operator action in PRA models for W NSSS plants and represents a reasonably conservative value.

5.1.1.5 CRI: Action to Drive the Control Rods into the Core

The rod control system may be in automatic or manual control. This is a plant specific decision. Assuming manual control is the most conservative approach since the automatic system will start inserting the rods before the operator can take action to do this.

If in manual, the operator can take the action to manually drive the control rods into the core using the rod control system. If the rod control system is in the automatic mode, the rods will start to insert automatically and the operator will continue to insert the rods, if necessary. This action needs to be taken within a very short time following event initiation (minutes) to limit the pressure transient. Success of this action provides 72 steps (negative reactivity) from the lead bank which is equivalent to one minute of insertion. Regardless of whether this action succeeds or fails, the ATWS event can still be mitigated depending on the availability of auxiliary feedwater and RCS pressure relief. The UETs are impacted by success of this action.

A value of 0.5 will be used for this event which can represent a high human error probability if the rod control system is in manual or an assumption that the rod control system is in automatic 50% of the time. A sensitivity study will be done assuming the system is in automatic 90% of the time. It should be noted that credit for manual rod insertion is possible, but depending on the plant, this may follow two other failed OAs. If so, then credit for this OA is very limited. CRI, whether it is for automatic rod insertion or manual rod insertion, is assumed to include the probability of the rods failing to insert, therefore, CR is not addressed following CRI success. If it was addressed separately, the probability of the rods failing to insert is extremely small compared to the CRI failure probability, so it would not impact the results.

The values to be used for CRI are:

- 0.5 probability the rod control system is in automatic (0.5 probability it is not in automatic, and therefore, fails)

- 0.9 probability the rod control system is in automatic will be used in a sensitivity study (0.1 probability it is not in automatic, and therefore, fails)

5.1.1.6 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

This top event models insufficient control rods fall into the core to shut down the reactor. If the actions, automatic or manual, to initiate reactor trip are successful, the control rods still need to fall into the core to shut down the reactor. With regard to the rod insertion, three outcomes are possible:

- Sufficient number of rods insert to bring the reactor subcritical.
- Sufficient number of rods insert to mitigate or partially mitigate the pressure transient, but not to bring the reactor subcritical. This is equivalent to the rods stepping in automatically by the rod control system or by the operator manually inserting the rods. Boration is still required to bring the reactor subcritical.
- Sufficient number of rods fail to insert so the pressure transient is not mitigated.

NUREG/CR-5500, Vol. 2 (Appendix E, Section E-4.2) calculates a probability of 1.2E-06/d for 10 or more rods failing to fully insert. The NUREG report assumes that failure of 10 control rods or more to insert results in a loss of shutdown capability and it does not matter which ten rods fail to insert. The NUREG notes that this is conservative. The number of rods that are required to insert to achieve a subcritical core is dependent on the core design and the location of the failed/successful control rods. In addition, the number of control rods required to insert to mitigate the pressure transient, but not provide shutdown, is also dependent on core design and control rod failure/success location.

The number of control rods required to insert to mitigate the pressure transient is less than the number of control rods required to bring the reactor subcritical. If sufficient information was known about the core design, control rod failure mechanisms, etc., it would be theoretically possible to calculate the probability of failing to insert multiple rods for different combinations of required rod locations to: 1) bring the core subcritical, and 2) mitigate the pressure transient while remaining critical. The NUREG assumption that failure of 10 or more rods to insert fails to shut down the core may be acceptable with respect to subcriticality, but is not appropriate for assuming the pressure transient will still occur, that is, the pressure transient will most likely be mitigated or significantly reduced. This is a conservative assumption (10 or more control rods fail to insert) with regard to the pressure transient since only D-bank insertion credit of 1 minute (72 out of 230 steps) has a significant impact on the UETs and this is significantly less than the number of control rods required to insert per the assumptions of the NUREG report.

It will be assumed in this model that failing to insert a sufficient number of control rods (failing CR) to provide an equivalent effect of failing to insert D-bank for one minute is not credible, that is, a sufficient number of control rods will always insert to equal the effect of 72 steps from D-bank. The pressure transient will still need to be mitigated, but the UET will be reduced to those values that assume D-bank insertion success. At this point the reactor will be critical, but at a lower power level and long-term shutdown (boration) will be required.

Therefore, the following approach will be used for CR in this analysis:

- A sufficient number of rods will always insert so that pressure transient will be mitigated or significantly reduced.
- Probability of failing to insert sufficient rods to bring the reactor subcritical is 1.2E-06/d.
- If CR fails, it is assumed that sufficient rods have been inserted to be the equivalent to 72 steps of D-bank insertion used in the UET calculations.

As noted above, CR is not addressed following success of CRI. The probability of rods failing to insert is assumed to be included in the probability of CRI failing (CR is very small compared to CRI).

5.1.1.7 ESFAS: Turbine Trip and AFW Pump Start by the ESFAS

A primary assumption regarding ATWS is that a common cause event occurs that disables the RPS and ESFAS completely inhibiting an ESFAS signal from being generated. But for certain equipment failures that lead to failure of reactor trip, such as control rods failing to drop into the core, the ESFAS signal will still be available for turbine trip and AFW pump start. The ESFAS signals are not available, assuming a common cause event inhibits all RPS signals, if reactor trip fails due to RTB, logic cabinet, or analog channel failures.

ESFAS signals to start AFW and trip the turbine will be credited only following failure to trip due to failure of the CR top event (rods fail to fall) following successful RT. The following value will be used:

- Failure Probability = 0.01 (RPS succeeds but reactor trip fails due to failure of rods to drop)

This is a conservative value that is significantly higher than the unavailability of ESF actuation signals as determined in other studies. A WOG program that analyzed the impact of allowed outage time changes on ESFAS reliability (Reference 13) showed that the unavailability of these signals varies from 3E-03 to 7E-04 depending on the specific signal being considered.

5.1.1.8 AMSAC: ATWS Mitigation System Actuation Circuitry

AMSAC is a diverse method (diverse from the RPS signals) to trip the turbine and start AFW. No detailed fault tree analysis of AMSAC has been done, but WCAP-11992 uses a conservative value of 1.0E-02/demand as a failure probability. This value has also been used in other studies.

- Failure probability = 0.01

5.1.1.9 AF100: AFW System Provides 100% Flow

As previously discussed, the UETs are dependent on available pressure relief and AFW flow. AFW is divided into 100% and 50% levels. AF100 represents 100% AFW flow from all AFW pumps to all four steam generators. For a AFW system design with 1 turbine-driven (TD) AFW pump and 2 motor-driven

(MD) AFW pumps, in which one MD pump provides ½ the flow as the TD pump, 100% flow is flow from the TD pump and both MD pumps.

The failure probability for this top event is based on a typical 4-loop plant with two motor-driven AFW pumps and one turbine-driven AFW pump and all support available.

- Failure probability = 8.82E-02 (from Vogtle IPE, Reference 15)
  Round off to 9.0E-02

5.1.1.10 AF50: AFW System Provides 50% Flow

AF50 represents less than 100% flow, but greater than or equal to 50% AFW flow to all four steam generators. The 50% flow requires flow from either both MD AFW pumps or the TD AFW pump. A conditional value is used since this event is addressed following failure of AF100. The value required is the probability of 50% flow failure given 100% flow has failed.

- Failure probability = 3.13E-03/8.82E-02 = 3.55E-02
  Round off to 4.0E-02

- where:
  3.13E-03 is failure of 50% or greater flow (Vogtle IPE, Reference 15)
  8.82E-02 is failure 100% flow

5.1.1.11 PR: Availability of Primary Pressure Relief

This event models the availability of primary pressure relief to mitigate the overpressure event. PR is dependent on the AFW flow (100% or 50%) and rod insertion (success or failure), and accounts for the UET, availability of PORVs, and failure probability of the safety valves. It also accounts for the frequency of initiators that can lead to ATWS events with regard to the time when the events occur during the cycle. UETs occur earlier in the cycle and transient events are more frequent earlier in the cycle also.

Fault trees are constructed for PR (see Appendix D). Altogether, four are required, one for each AFW/rod insertion combination, as follows:

- control rod insertion success, 100% AFW
- control rod insertion success, 50% AFW
- control rod insertion failure, 100% AFW
- control rod insertion failure, 50% AFW

Successful pressure relief requires opening all three safety valves and the required PORVs when not in an unfavorable exposure time. Each PR fault tree consists of four subtrees with each subtree modeling pressure relief requirements for a UET interval. The four UET intervals correspond to:

- pressure relief failure with 2 PORVs and 3 safety valves available
- pressure relief success requiring 2 PORVs and 3 safety valves

- pressure relief success requiring 1 PORV and 3 safety valves
- pressure relief success requiring 0 PORVs and 3 safety valves

For example, for the low reactivity core, with equilibrium xenon, with no control rod insertion, 100% AFW, and 1 PORV blocked, the UET is 130 days which is 26% of an 18 month fuel cycle (see Table 4-3). Therefore, if a transient event occurs while 1 PORV is blocked, if the reactor fails to trip, if the AFW system provides 100% flow, and if CRI fails, then the pressure transient cannot be mitigated for the initial 26% of the cycle. During a favorable portion of the cycle, the pressure transient can be mitigated by the available safety valves and PORVs. But if any safety valve or PORV fails, pressure relief will also fail.

At times plants operate with PORVs blocked, and blocked PORVs cannot be credited to mitigate an ATWS event since there is insufficient time to open the block valve. The following probabilities of blocked PORVs were assumed in this analysis. These values were chosen as a reasonably conservative set, but it is acknowledged they may not envelope all plants.

- probability that both PORVs are blocked = 0.05
- probability that PORV A is blocked = 0.10
- probability that PORV B is blocked = 0.10
- probability that neither PORV is blocked = 1 – (2 x 0.10 + 0.05) = 0.75
- probabilities of blocked PORVs are assumed to be randomly distributed throughout the fuel cycle

The following fault trees model primary pressure relief for the four noted AFW/RI conditions.

- PRA: with CRI and 100% AFW
- PRB: with CRI and 50% AFW
- PRC: without CRI and 100% AFW
- PRD: without CRI and 50% AFW

The probability of failure of the safety valves and PORVs are as follows:

- Safety valves (fail to open on demand): 1.0E-03/d (Reference 16)

- PORVs (fail to open on demand): 7.0E-03/d (Reference 16)

- Common cause failure of two PORVs = 7.0E-03/d x 0.1 = 7.0E-04/d
  Where:
  7.0E-03/d is the random failure per demand of one PORV
  0.1 is the Beta factor for common cause failure

The UETs provided on Tables 4-3, 4-4, 4-7, 4-8, 4-11, and 4-12 can be used directly assuming the probability of an event occurring throughout the fuel cycle is uniform. If not, then the UETs need to be modified, or weighted, to account for the higher frequency of trips during particular times in the cycle. Typically, transient events occur more frequently early in the fuel cycle. Since this is also the unfavorable portion of the cycle, the UETs need to be weighted based on the transient distribution during the fuel cycle.

The transient frequency distribution throughout the cycle was developed based on the information provided in Reference 11. Consistent with the discussion on initiating event frequency in Section 5.1.1.2 only transient events with initial power levels greater than 40%, except for LOSP events, inadvertent SI events, and manual or inadvertent reactor trips, were included. This includes the latest 5 years of data in the database (1993 to 1997).

The trip data from this database were sorted with respect to the time in life when the trip occurred. Table 5-3 provides a summary of this information with the number of trips provided in 30-day increments. The first 30-day increment is also shown divided into 5 days sub-increments. It is concluded from this that after the first 30 days the number of trips drops in about half. The raw data shows a wide variation in the number of trips occurring at different 30-day intervals. There is no reason to expect that the number of trips in any one particular 30-day interval should be significantly greater or less than a previous or following 30-day interval after the initial period of the cycle. The variation is expected to be random following the initial period. Towards the end of the cycle the frequency appears to tail off. This could be due to some plants not operating for the full 18-month cycle. For this analysis, it will be assumed that the trip rate remains constant through the end of the fuel cycle. The initial time period for the higher trip rate appears to be approximately 30 days. For this analysis, the distribution of trips for weighting the UETs is provided in Table 5-3 in the 4th column.

A sample calculation that demonstrates calculation of the weighted UETs follows:

From Table 4-11 the UET, for the bounding reactivity core, corresponding to no CRI, 100% AFW, and no PORVs blocked is from day 0 to day 151 of the operating cycle (or 0.29 fraction of cycle time).

> UET time: 30 days x 0.129 trip fraction + (151 - 30) days x 0.051 trip fraction = 10.04
> Non UET time: (514 - 151) days x 0.051 trip fraction = 18.51
> Weighted UET fraction = 10.04/(10.04 + 18.51) = 0.35
> Weighted non UET fraction = 18.51/(10.04 + 18.51) = 0.65
> where: 514 days is the number of days in an 18 month cycle

Tables 5-4, 5-5, and 5-6 summarize the weighted UETs for the low, high, and bounding reactivity cores.

These weighted UETs are used to derive the intervals (basic events PRXI1, PRXI2, PRXI3, and PRXI4 in PR fault trees PRA, PRB, PRC, and PRD; where the X represents A, B, C, or D). The following provides the calculations to determine these values for the bounding reactivity core. The weighted UETs are used to calculate the intervals. These values are summarized in Table 5-7 for the low, high, and bounding cores.

PRAI1: 0.27 - 0 = 0.27 (represents the weighted fraction of the cycle that pressure relief will fail with 2 PORVs and 3 safety valves available for the condition of CRI and 100% AFW)
PRAI2: 0.35 - 0.27 = 0.08 (represents the weighted fraction of the cycle that pressure relief can succeed with 2 PORVs and 3 safety valves available for the condition of CRI and 100% AFW)
PRAI3: 0.43 - 0.35 = 0.08 (represents the weighted fraction of the cycle that pressure relief can succeed with 1 PORV and 3 safety valves available for the condition of CRI and 100% AFW)
PRAI4: 1.0 - 0.43 = 0.57 (represents the weighted fraction of the cycle that pressure relief can succeed with 0 PORVs and 3 safety valves available for the condition of CRI and 100% AFW)

PRBI1: 0.30 - 0 = 0.30 (represents the weighted fraction of the cycle that pressure relief will fail with 2 PORVs and 3 safety valves available for the condition of CRI and 50% AFW)
PRBI2: 0.37 - 0.30 = 0.07 (represents the weighted fraction of the cycle that pressure relief can succeed with 2 PORVs and 3 safety valves available for the condition of CRI and 50% AFW)
PRBI3: 0.45 - 0.37 = 0.08 (represents the weighted fraction of the cycle that pressure relief can succeed with 1 PORV and 3 safety valves available for the condition of CRI and 50% AFW)
PRBI4: 1.0 - 0.45 = 0.55 (represents the weighted fraction of the cycle that pressure relief can succeed with 0 PORVs and 3 safety valves available for the condition of CRI and 50% AFW)

PRCI1: 0.35 - 0 = 0.35 (represents the weighted fraction of the cycle that pressure relief will fail with 2 PORVs and 3 safety valves available for the condition of no CRI and 100% AFW)
PRCI2: 0.44 - 0.35 = 0.09 (represents the weighted fraction of the cycle that pressure relief can succeed with 2 PORVs and 3 safety valves available for the condition of no CRI and 100% AFW)
PRCI3: 0.54 - 0.44 = 0.10 (represents the weighted fraction of the cycle that pressure relief can succeed with 1 PORV and 3 safety valves available for the condition of no CRI and 100% AFW)
PRCI4: 1.0 - 0.54 = 0.46 (represents the weighted fraction of the cycle that pressure relief can succeed with 0 PORVs and 3 safety valves available for the condition of no CRI and 100% AFW)

PRDI1: 0.38 - 0 = 0.38 (represents the weighted fraction of the cycle that pressure relief will fail with 2 PORVs and 3 safety valves available for the condition of no CRI and 50% AFW)
PRDI2: 0.47 - 0.38 = 0.09 (represents the weighted fraction of the cycle that pressure relief can succeed with 2 PORVs and 3 safety valves available for the condition of no CRI and 50% AFW)
PRDI3: 0.58 - 0.47 = 0.11 (represents the weighted fraction of the cycle that pressure relief can succeed with 1 PORV and 3 safety valves available for the condition of no CRI and 50% AFW)
PRDI4: 1.0 - 0.58 = 0.42 (represents the weighted fraction of the cycle that pressure relief can succeed with 0 PORVs and 3 safety valves available for the condition of no CRI and 50% AFW)

5.1.1.12 LTS: Long-Term Shutdown

This event requires the plant operators to establish long-term shutdown, which includes starting emergency boration. This is required on success paths that do not have full control rod insertion. For example, if CRI or CR succeed, then rod insertion has occurred and this is not addressed. Note that CRI requires the lead bank to insert 72 steps, with regard to mitigation of the RCS pressure spike, which is not full control rod insertion. It is assumed that with CRI the operators or automatic rod control system will continue to insert the rods until the core is shut down.

The failure probability for this event is dependent on an operator action for initiation of emergency boration. The following value is used based on a review of values used in several IPEs:

- Failure probability = 0.01

It is assumed that this action is independent of the previous OAs since it does not need to be completed in the same short time period as the OAs to trip the reactor, trip the MG sets, or manually drive in the control rods. The value of 0.01 is assumed to account for the HEP and equipment failure probabilities.

5.1.1.13 ATWS State 3/4: Core Damage Frequency Quantification

The ATWS model for the ATWS States 3/4 was quantified using a fault tree linking approach with the CAFTA computer code system. The event tree structure is provided in Figure 5-1. Fault trees were linked in for the top events RT and PR. The remaining top events are scalars. Sections 5.1.1.2 to 5.1.1.12 discuss these fault trees and scalars. The linked fault tree was quantified with a cutoff frequency of 1.0E-15.

The CDF quantification was completed for the low reactivity core (Case 3/4-1), high reactivity core (Case 3/4-2), and bounding reactivity core (Case 3/4-3). The change in cores is reflected in the model through the UET values and requires changing the values used for the PR intervals in the pressure relief fault trees. These values are provided in Table 5-7. All other basic event values remained the same between the three cases. The results, in terms of CDF, are provided in Table 5-8. Also shown are the increases in CDF for Cases 3/4-2 and 3/4-3 with respect to Case 3/4-1. Case 3/4-1 meets the 5% UET condition for no RI, 100%, AFW, and all PORVs available.

Table 5-1 Plant Operating States for the ATWS Event				
State	**Power Level**	**Plant Power Activity**	**AMSAC Available**	**Equilibrium Xenon Concentration**
1	<40%	Startup	No	No
2	≥40%	Startup	Yes	No
3	~100%	At-Power	Yes	Yes
4	≥40%	Shutdown	Yes	Yes
5	<40%	Shutdown	No	Yes

Table 5-2 Number of Transient Events and Frequency by ATWS State			
ATWS State	**Number of Events**	**Operating Years**	**Frequency (per yr)**
1 – Startup, Power <40%	21.4	197.4	0.11
2 – Startup, Power ≥40% to <95%	13.5	197.4	0.07
3 – At-power, Power ≥95%	178	197.4	0.90
4 – Shutdown, Power ≥40% to <95%	10.5	197.4	0.05
5 – Shutdown, Power <40%	16.6	197.4	0.08

Table 5-3 Transient Events Occurring in 30-Day Intervals Relative to Cycle Start

30-Day Interval	Number of Trips	Fraction of Trips	ATWS Analysis Distribution
1: 0-30 days 1-5 days 6-10 days 11-15 days 16-20 days 21-25 days 26-30 days	25 3 6 6 4 2 4	0.129	0.129
2: 31-60 days	15	0.077	0.051
3: 61-90 days	16	0.082	0.051
4: 91-120 days	15	0.077	0.051
5: 121-150 days	7	0.036	0.051
6: 151-180 days	12	0.062	0.051
7: 181-210 days	6	0.031	0.051
8: 211-240 days	11	0.057	0.051
9: 241-270 days	11	0.057	0.051
10: 271-300 days	9	0.046	0.051
11: 301-330 days	13	0.067	0.051
12: 331-360 days	14	0.072	0.051
13: 361-390 days	10	0.052	0.051
14: 391-420 days	11	0.057	0.051
15: 421-450 days	7	0.036	0.051
16: 451-480 days	8	0.041	0.051
17: 481-510 days	3	0.015	0.051
18: 511-540 days	1	0.005	0.051
Total	194	0.999	0.996

Table 5-4 Weighted UET Values for a Low Reactivity Core, Equilibrium Xenon, Power Level ≥40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0.00	0.17	0.32
RI, 50% AFW	0.00	0.21	0.36
No RI, 100% AFW	0.11	0.32	0.46
No RI, 50% AFW	0.22	0.37	0.52

Table 5-5 Weighted UET Values for a High Reactivity Core, Equilibrium Xenon, Power Level ≥40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0.00	0.29	0.38
RI, 50% AFW	0.14	0.33	0.43
No RI, 100% AFW	0.28	0.39	0.52
No RI, 50% AFW	0.33	0.43	0.56

Table 5-6 Weighted UET Values for a Bounding Reactivity Core, Equilibrium Xenon, Power Level ≥40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0.27	0.35	0.43
RI, 50% AFW	0.30	0.37	0.45
No RI, 100% AFW	0.35	0.44	0.54
No RI, 50% AFW	0.38	0.47	0.58

Table 5-7 Summary of Pressure Relief Intervals, Equilibrium Xenon			
PR Interval Basic Event	**Low Reactivity Core**	**High Reactivity Core**	**Bounding Reactivity Core**
PRAI1	0.00	0.00	0.27
PRAI2	0.17	0.29	0.08
PRAI3	0.15	0.09	0.08
PRAI4	0.68	0.62	0.57
PRBI1	0.00	0.14	0.30
PRBI2	0.21	0.19	0.07
PRBI3	0.15	0.10	0.08
PRBI4	0.64	0.57	0.55
PRCI1	0.11	0.28	0.35
PRCI2	0.21	0.11	0.09
PRCI3	0.14	0.13	0.10
PRCI4	0.54	0.48	0.46
PRDI1	0.22	0.33	0.38
PRDI2	0.15	0.10	0.09
PRDI3	0.15	0.13	0.11
PRDI4	0.48	0.44	0.42

Table 5-8 Yearly Core Damage Frequency Summary: ATWS State 3/4				
Plant At-Power Operation and Shutdown, Power Level ≥40% Standard Blocked PORV Probabilities Equilibrium Xenon				
Case	**Core**	**Rod Insertion (RI) Failure Probability**	**CDF (per year)**	**ΔCDF (per year)[1]**
3/4-1	Low Reactivity	0.5	1.09E-07	–
3/4-2	High Reactivity	0.5	1.70E-07	6.1E-08
3/4-3	Bounding Reactivity	0.5	4.69E-07	3.6E-07

Note:

1. Increase in CDF over Case 3/4-1 value.

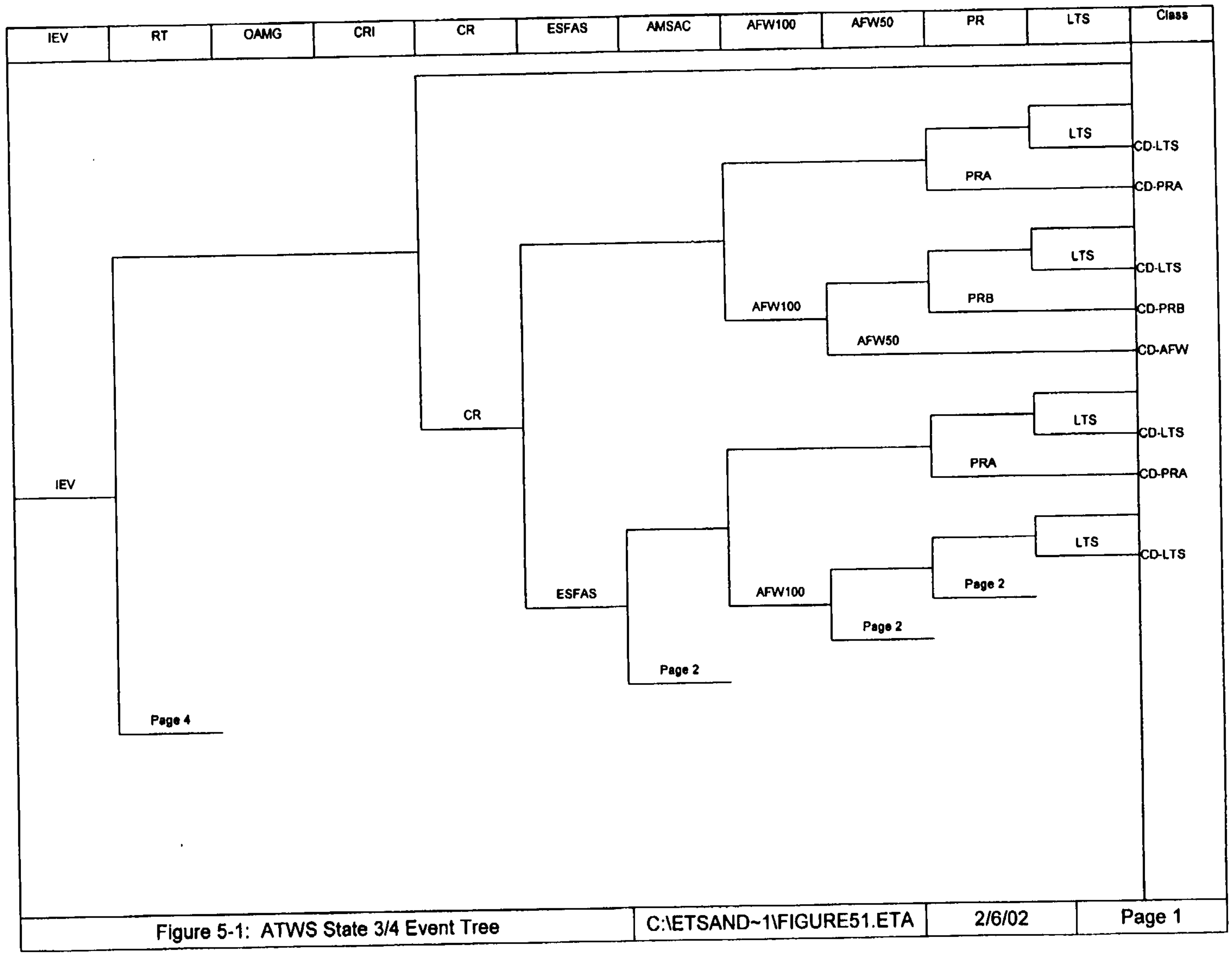

Figure 5-1 ATWS State 3/4 Event Tree

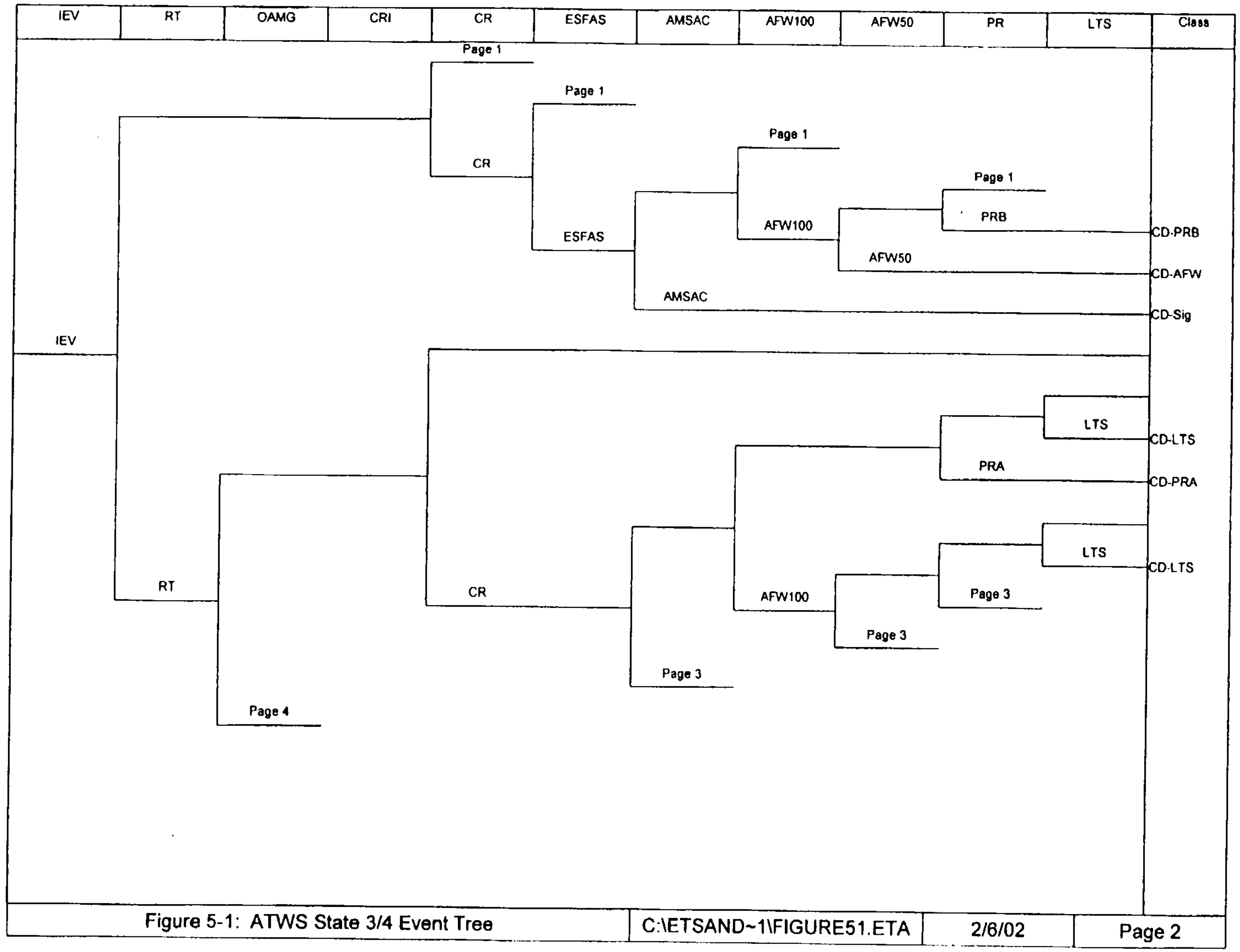

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

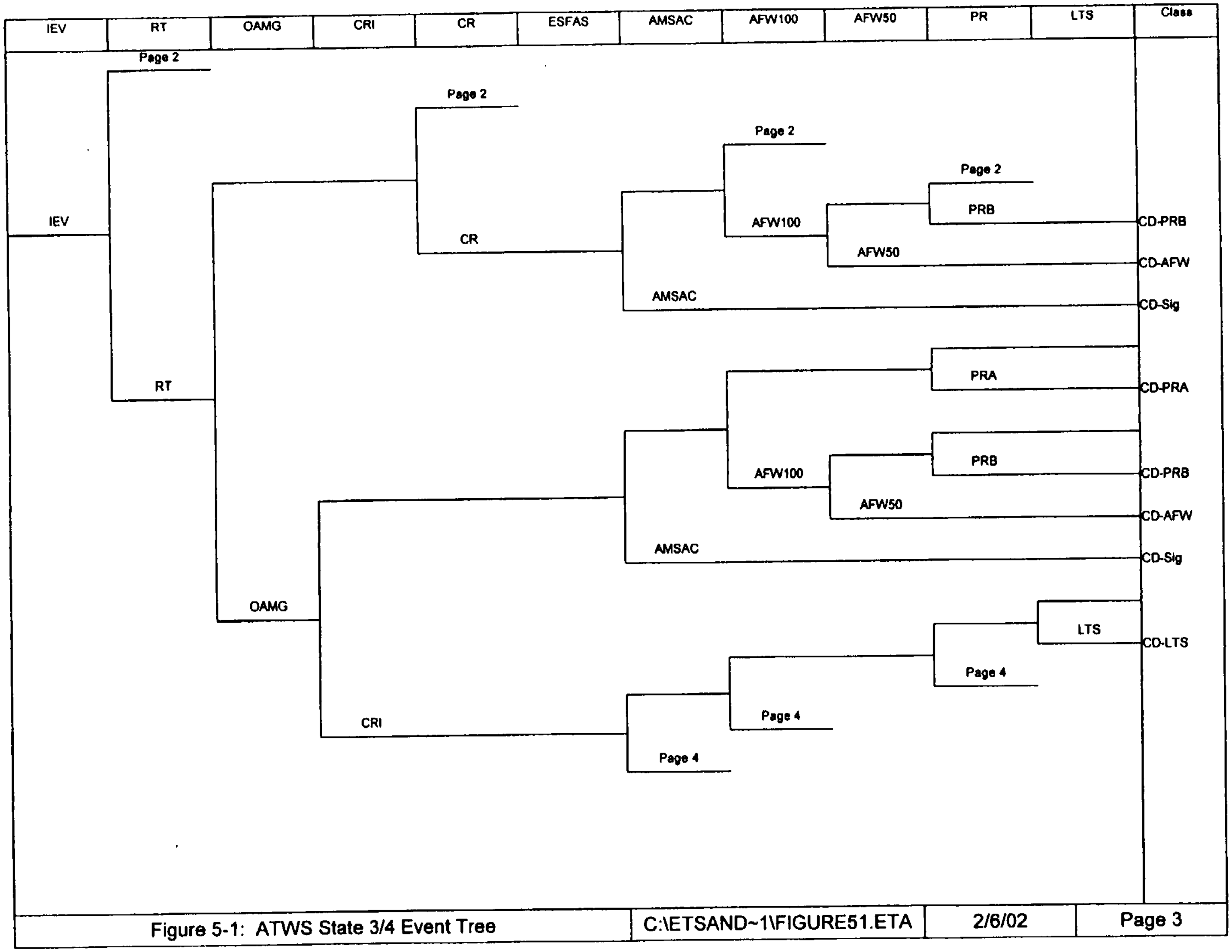

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

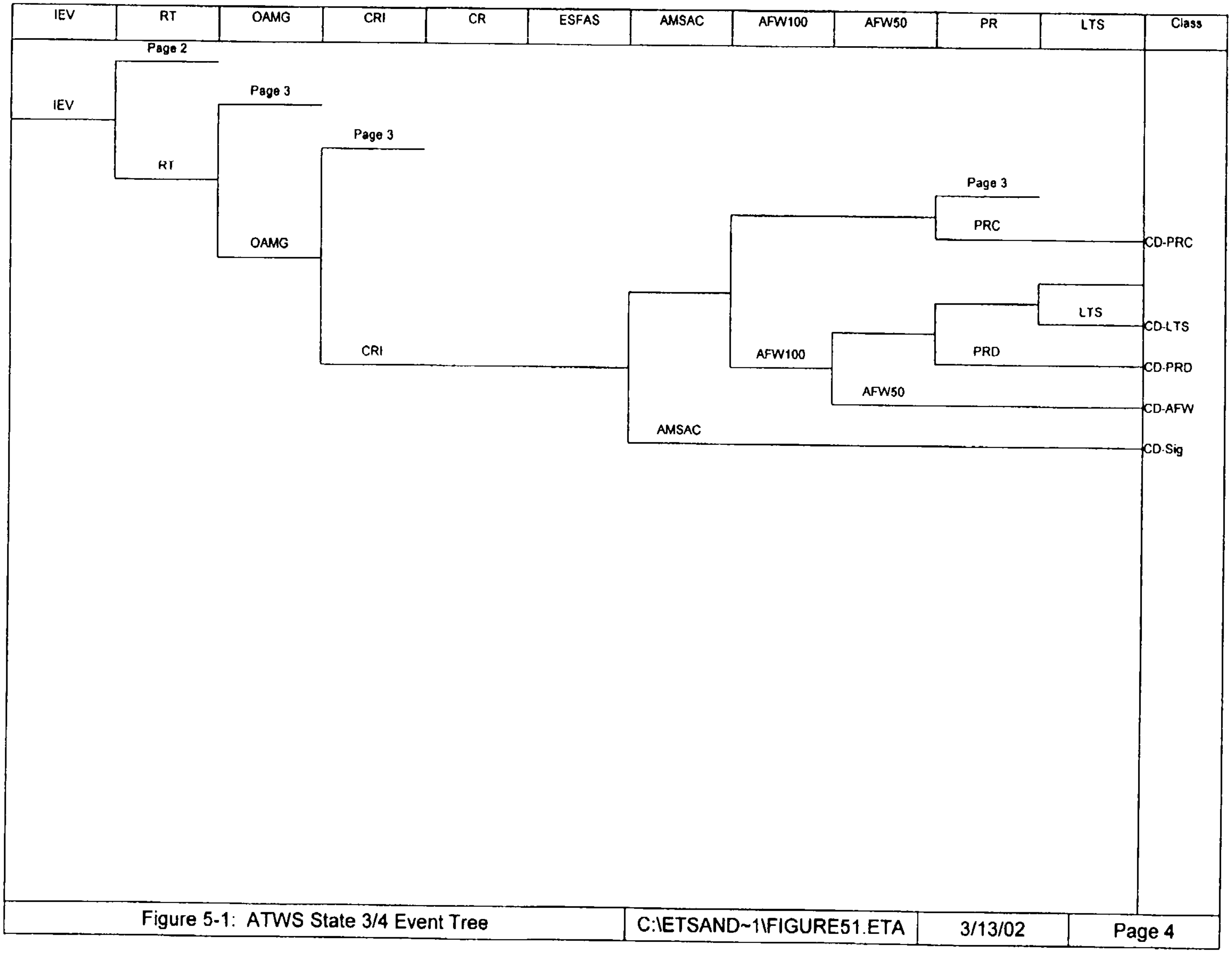

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

5.1.2 ATWS State 2: Plant Startup, Power Level ≥40%

This state represents plant operation when the power level is greater than or equal to 40% during startup conditions. During this phase of plant operation full power equilibrium xenon has not yet been established. It is conservatively assumed for this analysis that all plant startups follow plant shutdowns of sufficient length to deplete xenon. AMSAC is operable in this state. UETs used in this evaluation are for the 100% power level with no xenon, which can be applied conservatively to power levels down to 40%.

5.1.2.1 ATWS State 2 Event Tree

The event tree used to evaluate ATWS State 2 is the same as used for at-power ATWS evaluations. This is shown in Figure 5-1. The fault tree models for the event tree top events also remain the same. There are some changes to the fault tree basic event data inputs that are discussed in the following paragraphs, but the primary differences are the UETs during and immediately following the restart as related to the time it takes to attain equilibrium xenon. The UETs are provided for the low, high, and bounding reactivity cores with no xenon in Tables 4-5, 4-6, 4-9, 4-10, 4-13, and 4-14. Note that all three cores are unfavorable for all conditions at the beginning of the cycle.

As noted above, xenon buildup is important during the initial reactor operation following all shutdowns of sufficient length that allow the xenon concentration to deplete to a low enough level so that it does not provide the negative reactivity. A shutdown of approximately 3 days is sufficient in length to achieve xenon depletion. For relatively short shutdowns, the xenon concentration remains sufficiently high to eliminate this issue as an ATWS concern. Therefore, xenon concentration is an issue with regard to ATWS events for a reactor startup after any outage that is long enough to allow significant xenon depletion. A defined time for a "short outage" is not necessary for the following analysis since it will be conservatively assumed that all startups follow shutdowns of sufficient length to achieve xenon depletion.

The probability of an ATWS event is dependent on the reliability of the reactor trip system; development of trip signals and insertion of the control rods. A significant number of control rods failing to insert due to either: 1) failure to develop a trip signal either automatically or manually, or 2) failure of the control rods to drop due to mechanical problems results in an ATWS event. Studies done on the reliability of the reactor trip system assume that the plant is operating at power, and that specified test and maintenance activities demonstrate the operability of the reactor trip system on a periodic basis. The reliability of the reactor trip system for plant startups closely following a reactor trip is significantly higher. That is, a successful reactor trip demonstrates that the reactor trip system is fully operable and its reliability in the following startup is greater than during typical plant at-power operation when its operability is demonstrated only periodically. A plant startup also exercises the shutdown and control rods; both need to be withdrawn from the core via the CRDMs. This operation demonstrates their operability. In addition, during plant startup, test and maintenance activities that render parts of the reactor trip system unavailable will not be in progress.

The ATWS risk associated with plant startup needs to consider three types of startups under the noted conditions:

- Startup following refueling: zero xenon concentration level; control rods and CRDMs exercised for startup; no test or maintenance activities on the RPS in progress; no recent activities that demonstrated RPS operability other than typical periodic tests.

- Startup following a controlled plant shutdown: zero xenon concentration level (it will be assumed that the shutdown time was long enough for complete xenon depletion); control rods and CRDMs exercised for startup; no test or maintenance activities on the RPS in progress; no recent activities that demonstrated RPS operability other than typical periodic tests.

- Startup following a reactor trip: zero xenon concentration level (it will be assumed that the shutdown time was long enough for complete xenon depletion); control rods and CRDMs exercised for startup; no test or maintenance activities on the RPS in progress; the reactor trip that caused the shutdown demonstrated RPS operability.

The most conservative startup to evaluate with regard to risk is one following a refueling outage or one following a controlled shutdown and with the outage time of sufficient length to allow complete xenon depletion.

Other factors that need to be considered are the time to return to power and the xenon buildup during this time period. The return to power time for a new core is longer than for a core previously in operation due to restraints imposed by required startup tests, calibrations, and data collection. During this time period, xenon is building up. Theoretically, it is possible to determine unfavorable exposure times for various levels of xenon concentrations that could be used to construct a probabilistic model to determine ATWS risk during startup, but the level of effort would be high, the model complex, and the additional benefit marginal. This model would have lower xenon concentration levels at lower power levels, and as the elapse time from startup increased, the xenon concentration would also increase as would the power level.

To simplify this analysis, the approach used is conservative and encompasses all startup scenarios. The following assumptions apply:

- The startup will be assumed to be a rapid startup that will be considered a step change to full power, therefore, no credit will be taken for xenon buildup.

- The time the reactor is down following a reactor trip or shutdown is assumed to be long enough for the xenon to be depleted.

- Equilibrium xenon will be achieved within 50 hours.

- The startup will be assumed to follow a shutdown that did not require generation of a reactor trip signal, therefore, the probability of failure of the reactor trip signal is assumed to be the same as during power operation since there is no comprehensive testing of the RPS prior to startup.

- No test or maintenance activities are in progress that cause any part of the RPS to be unavailable.

- The startup, with the movement of the control and shutdown rods, demonstrates the operability of the control rods.

The following discusses the event tree top events in more detail.

5.1.2.2 IEV: Initiating Event Frequency

The value used for IEV is taken from Table 5-2 for start-up with power level ≥40% and <95%.

- IEV = 0.07/yr

This only accounts for the trips that occur while the plant is starting up and power level is ≥40%. In addition, there is a period of time while the plant is at power and before equilibrium xenon is achieved when a trip could also occur. This time period was previously noted to be 50 hours. Therefore, assuming that the startup is a step change, 50 hours of at-power operation also needs to be accounted for in the IE frequency. It is necessary to determine the number of at-power trips during a 50 hour time period. The first 30 day period in the cycle will be used since this is the time of the highest trip frequency.

Table 5-3 shows that 25 trips have occurred in the first 30 days of the cycle following startup. From this, the number of trips during a 50 hour time period in the first 30 days of the cycle is:

- Trips during a 50 hour period = 25 x 50 hrs/720 hrs = 1.74 trips

This number of trips is added to the startup trips (with the power level ≥40% and <95%).

- Total trip = 13.5 + 1.74 = 15.2 trips (note that the 13.5 trips is from Table 5-2)

From this the IE frequency is calculated to be 15.2 trips/197.4 years = 0.077/year.

5.1.2.3 RT: Reactor Trip Signal from the RPS

The unavailability of the reactor trip signals is discussed in Section 5.1.1.3. The same model is used in this analysis with the exception of several basic event values that eliminate test and maintenance activities as unavailability contributors. As previously discussed, these types of activities will not be scheduled during a startup. Per the Technical Specifications, the RPS is required to be available prior to entering Modes of applicability. This is done by setting the test and maintenance basic events for the analog channels and logic cabinet/RTB trains to 0.

5.1.2.4 OAMG: Operator Action to Trip the Reactor from the MG Sets

The same values are used as discussed in Section 5.1.1.4. These are:

- 0.5 is used when RT fails due to reasons related to the OA to trip the reactor in RT in conjunction with logic cabinet or analog channel processing failures – this is a conservative conditional failure probability (conditional on a previous OA already failing).

- 1.0E-02 is used when RT fails due to reasons not related to failure of the OA to trip the reactor in RT, that is, when failures are related to RTB failures.

5.1.2.5 CRI: Action to Drive the Control Rods into the Core

The same value is used as discussed in Section 5.1.1.5. This is:

- 0.5 probability that the rod control system is in automatic

5.1.2.6 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

The value used in Section 5.1.1.6 assumes normal reactor operation. This means that the reactor has been at power for some relatively long period of time and the control rods have not been fully exercised since the last startup. In the situation being considered in this ATWS state, trips prior to establishing equilibrium xenon, the reactor trip is required within 50 hours of startup when the rods were withdrawn from the core. Assuming that the probability of a component failing is directly related to the time from the last test (or time it was last exercised) leads to a component failure probability significantly lower than its probability of failure sometime between tests. Since the time from the last test is small in this situation, a value of 0 could be justified, but to be conservative, the value provided in Section 5.1.1.6 for failing to shut down the reactor due to insufficient rod insertion will be reduced by a factor of 10.

- 1.2E-07/d probability of the control rods failing to insert on demand

5.1.2.7 Other Top Events: ESFAS, AMSAC, AFW100, AFW50, LTS

These top events remain the same as discussed in Sections 5.1.1.7, 5.1.1.8, 5.1.1.9, 5.1.1.10, and 5.1.1.12. These are:

- ESFAS failure probability = 0.01
- AMSAC failure probability = 0.01
- AFW100 failure probability = 9.0E-02
- AFW50 failure probability = 4.0E-02
- LTS failure probability = 0.01

5.1.2.8 PR: Availability of Primary Pressure Relief

As discussed in Section 5.1.1.11, this event models the availability of primary pressure relief to mitigate the overpressure event. PR is dependent on the AFW flow (100% or 50%) and rod insertion (success or failure), and accounts for the UET, availability of PORVs, and failure probability of the safety valves. It also accounts for the frequency of initiators that can lead to ATWS events with regard to the time when the events occur during the cycle. UETs occur early in the cycle and transient events related to plant startups are more frequent early in the cycle also. In this case, the trip distribution is related to the plant startups.

The same fault trees are used for PR as discussed in Section 5.1.1.11 with the same basic event unavailabilities, or failure probabilities, except for the UET related values. Since all other values are the same as those discussed in Section 5.1.1.11, only the UET related inputs are further discussed in the following.

The UETs provided on Tables 4-5, 4-6, 4-9, 4-10, 4-13, and 4-14 need to be modified or weighted to account for the higher frequency of trips during particular times in the cycle. Typically, transient events occur more frequently early in the fuel cycle. Since this can also be the unfavorable portion of the cycle, the UETs need to be weighted based on the transient distribution during the fuel cycle.

The UET weighting needs to consider when the trip occurs during the cycle without equilibrium xenon. These trips occur during or immediately following plant startups. Therefore, the weighting needs to be done based on the distribution of plant startups throughout the cycle. Startups follow plant trips, controlled plant shutdowns, and at the beginning of the cycle following fuel loading. The following approach is used to determine the trip distribution.

- Startups following plant trips: Table 5-3 provides the distribution of trips over the cycle. Since there is approximately 1 trip per year or 1.5 per cycle (18 months), then a trip frequency distribution, or startup frequency distribution assuming a startup follows each trip, is determined by multiplying the trip distribution by the trip frequency. This is provided on Table 5-9.

- Startups following controlled plant shutdowns: No information is available on the frequency of controlled plant shutdowns throughout the cycle. For this study it will be assumed that a plant typically has one controlled shutdown per cycle and that this can occur with equal probability across the cycle.

- Startups at the beginning of the cycle: One startup occurs following every refueling outage.

Table 5-9 provides a summary of the startup information. The final column provides the startup distribution which is used for weighting the UETs. The weighting calculations are done as shown in Section 5.1.1.11.

Tables 5-10, 5-11, and 5-12 summarize the weighted UETs. These weighted UETs are used to derive the intervals (basic events PRXI1, PRXI2, PRXI3, and PRXI4; where the X represents A, B, C, or D). The calculations to determine these values are the same as shown in Section 5.1.1.11. The interval values used in the PR fault trees are summarized in Table 5-13 for the low, high, and bounding cores.

5.1.2.9 ATWS State 2: Core Damage Frequency Quantification

The ATWS model for the ATWS State 2 was quantified using the approach discussed in Section 5.1.1.13. The event tree structure is provided in Figure 5-1.

The CDF quantification was completed for the low reactivity core (Case 2-1), high reactivity core (Case 2-2), and bounding reactivity core (Case 2-3). The change in cores is reflected in the model through the UET values and requires changing the values used for the PR intervals in the pressure relief fault trees. These values are provided in Table 5-13. All other basic event values remained the same between the three cases. The results, in terms of CDF, are provided in Table 5-14. Also shown is the increase in CDF for Cases 2-2 and 2-3 with respect to Case 2-1. Case 2-1 meets the 5% UET condition for no RI, 100% AFW, and all PORVs available.

Table 5-9 Distribution of Plant Startups Across the Cycle

30-Day Interval	Trip Distribution[1]	Trip Frequency	Startup Frequency Following Trips[2]	Startup Frequency Following Controlled Shutdowns[3]	Startup Frequency Following Refueling[4]	Startup Frequency[5]	Startup Distribution[6]
1: 0-30 days	0.129	1.5	0.194	0.056	1	1.250	0.356
2: 31-60 days	0.051	1.5	0.077	0.056	0	0.133	0.038
3: 61-90 days	0.051	1.5	0.077	0.056	0	0.133	0.038
4: 91-120 days	0.051	1.5	0.077	0.056	0	0.133	0.038
5: 212-150 days	0.051	1.5	0.077	0.056	0	0.133	0.038
6: 151-180 days	0.051	1.5	0.077	0.056	0	0.133	0.038
7: 181-210 days	0.051	1.5	0.077	0.056	0	0.133	0.038
8: 211-240 days	0.051	1.5	0.077	0.056	0	0.133	0.038
9: 241-270 days	0.051	1.5	0.077	0.056	0	0.133	0.038
10: 271-300 days	0.051	1.5	0.077	0.056	0	0.133	0.038
11: 301-330 days	0.051	1.5	0.077	0.056	0	0.133	0.038
12: 331-360 days	0.051	1.5	0.077	0.056	0	0.133	0.038
13: 361-390 days	0.051	1.5	0.077	0.056	0	0.133	0.038
14: 391-420 days	0.051	1.5	0.077	0.056	0	0.133	0.038
15: 421-450 days	0.051	1.5	0.077	0.056	0	0.133	0.038
16: 451-480 days	0.051	1.5	0.077	0.056	0	0.133	0.038

Table 5-9 (cont.) Distribution of Plant Startups Across the Cycle

30-Day Interval	Trip Distribution[1]	Trip Frequency	Startup Frequency Following Trips[2]	Startup Frequency Following Controlled Shutdowns[3]	Startup Frequency Following Refueling[4]	Startup Frequency[5]	Startup Distribution[6]
17: 481-510 days	0.051	1.5	0.077	0.056	0	0.133	0.038
18: 511-540 days	0.051	1.5	0.077	0.056	0	0.133	0.038
Total	0.996		1.503	1.008	1	3.511	1.002

Notes:

1. From Table 5-3
2. Startup frequency following trips = Trip distribution x Trip frequency
3. One controlled shutdown per year is assumed with equal probability across the cycle.
4. One startup immediately following refueling
5. Startup frequency = Startup frequency following trips + Startup frequency following controlled shutdown + Startup frequency following refueling
6. Startup distribution = Specific startup frequency values/3.511

Table 5-10 Weighted UET Values for a Low Reactivity Core, No Equilibrium Xenon, Power Level ≥40%			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, 100% AFW	0.45	0.53	0.61
RI, 50% AFW	0.49	0.56	0.64
No RI, 100% AFW	0.54	0.62	0.74
No RI, 50% AFW	0.57	0.65	0.77

Table 5-11 Weighted UET Values for a High Reactivity Core, No Equilibrium Xenon, Power Level ≥40%			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, 100% AFW	0.52	0.58	0.64
RI, 50% AFW	0.54	0.60	0.66
No RI, 100% AFW	0.59	0.66	0.76
No RI, 50% AFW	0.62	0.69	0.79

Table 5-12 Weighted UET Values for a Bounding Reactivity Core, No Equilibrium Xenon, Power Level ≥40%			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, 100% AFW	0.55	0.60	0.65
RI, 50% AFW	0.57	0.62	0.67
No RI, 100% AFW	0.61	0.67	0.76
No RI, 50% AFW	0.63	0.70	0.81

Table 5-13 Summary of Pressure Relief Intervals, No Equilibrium Xenon			
PR Interval Basic Event	**Low Reactivity Core**	**High Reactivity Core**	**Bounding Reactivity Core**
PRAI1	0.45	0.52	0.55
PRAI2	0.08	0.06	0.05
PRAI3	0.08	0.06	0.05
PRAI4	0.39	0.36	0.35
PRBI1	0.49	0.54	0.57
PRBI2	0.07	0.06	0.05
PRBI3	0.08	0.06	0.05
PRBI4	0.36	0.34	0.33
PRCI1	0.54	0.59	0.61
PRCI2	0.08	0.07	0.06
PRCI3	0.12	0.10	0.09
PRCI4	0.26	0.24	0.24
PRDI1	0.57	0.62	0.63
PRDI2	0.08	0.07	0.07
PRDI3	0.12	0.10	0.11
PRDI4	0.23	0.21	0.19

Table 5-14 Yearly Core Damage Frequency Summary: ATWS State 2				
Plant Startup Operation, Power Level ≥40% **Standard Blocked PORV Probabilities** **No Equilibrium Xenon**				
Case	**Core**	**Rod Insertion (RI) Failure Probability**	**CDF (per year)**	**ΔCDF (per year)[1]**
2-1	Low Reactivity	0.5	1.17E-08	–
2-2	High Reactivity	0.5	1.31E-08	1.4E-09
2-3	Bounding Reactivity	0.5	1.36E-08	1.9E-09

Note:

1. Increase in CDF over Case 2-1 value.

5.1.3 ATWS State 1: Plant Startup, Power Level <40%

This state represents plant operation when the power level is less than 40% during startup conditions. During this phase of plant operation equilibrium xenon has not yet been established. It is conservatively assumed for this analysis that all plant startups follow plant shutdowns of sufficient length to deplete xenon. AMSAC is not operable in this state. UETs used in this evaluation are for the 40% power level without equilibrium xenon which can be applied conservatively to power levels down to 0%.

5.1.3.1 ATWS State 1 Event Tree

The event tree used to evaluate ATWS State 1 is based on the event tree used for at-power ATWS evaluations. The ATWS State 1 event tree is shown on Figure 5-2. The differences between the two event trees represents the plant response for events that occur above and below 40% power during startup. The key differences are related to the availability of a signal to trip the turbine and start AFW. AMSAC is not available below 40% power, therefore, signals are not available for these actuations. Similar to ATWS State 2 (startup, power level ≥40%), equilibrium xenon has not been established. Also similar to ATWS State 2, startups to be considered are those that follow refueling, plant trips, and required plant shutdowns.

UETs are provided for the low, high, and bounding reactivity cores with no xenon in Tables 4-25, 4-27, and 4-29 based on a power level of 40%. The UETs are only provided for the condition of no AMSAC with 2, 1, or 0 PORVs available. No AMSAC means that no credit is taken for AFW start or turbine trip. In addition, UETs are not provided with CRI. At low power levels, the position of the control rods with respect to the core is variable; they could be completely out or partially in. If completely out, the 72 steps insertion will not provide as much benefit as if they are starting from a position that is partially in. Due to the uncertainty of the control rod position and to simplify the analysis, no credit is taken for CRI.

The fault tree model for the event tree top event RT remains the same. The fault tree for the top event PR is also the same except only one condition is required that corresponds to no AMSAC (no AFW) and no CRI. AMSAC and CRI have been removed from the event tree. AFW is required for decay heat removal and must be started manually if the ESFAS signals are not available. ESFAS signals are available only when the ATWS is due RTB failures or failure of the control rods to insert with a trip signal available. There are some changes to the fault tree basic event data inputs that are discussed in the following paragraphs. The other top events are also discussed in the following paragraphs.

As discussed in Section 5.1.2.1, the following will be assumed:

- The startup will be assumed to be a rapid startup that will be considered a step change to full power.
- The time that the reactor is down following a reactor trip or shutdown is assumed to be long enough for the xenon to deplete.
- Equilibrium xenon will be achieved within 50 hours.

- The startup will be assumed to follow a shutdown that did not require generation of a reactor trip signal, therefore, the probability of failure of the reactor trip signal is assumed to be the same as for at-power operation since there is no comprehensive testing of the RPS prior to startup.

- No test or maintenance activities are in progress that cause any part of the RPS to be unavailable.

- The startup, with the movement of the control and shutdown rods, demonstrates the operability of the control rods.

The following discusses the event tree top events in more detail.

5.1.3.2 IEV: Initiating Event Frequency

The value used for IEV is taken from Table 5-2 for start-up with power level <40%.

- IEV = 0.11/yr

5.1.3.3 RT: Reactor Trip Signal from the RPS

The unavailability of the reactor trip signals is discussed in Section 5.1.1.3. The same model is used in this analysis with the exception of several basic event values that eliminate test and maintenance activities as unavailability contributors. As previously discussed, these types of activities will not be scheduled during a startup. Per the Technical Specifications, the RPS is required to be available prior to entering Modes of applicability. This is done by setting the test and maintenance basic events for the analog channels and logic cabinet/RTB trains to 0. This is consistent with Section 5.1.2.3.

5.1.3.4 OAMG: Operator Action to Trip the Reactor from the MG Sets

The same values are used as discussed in Section 5.1.1.4. These are:

- 0.5 is used when RT fails due to reasons related to the OA to trip the reactor in RT in conjunction with logic cabinet or analog channel processing failures. This is a conservative conditional failure probability (conditional on a previous OA already failing).

- 1.0E-02 is used when RT fails due to reasons not related to failure of the OA to trip the reactor in RT, that is, when failures are related to RTB failures.

5.1.3.5 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

As discussed in Section 5.1.2.6, the value used for CR is:

- CR = 1.2E-07/d.

5.1.3.6 PR: Pressure Relief

This event models the availability of PR to mitigate the overpressure event. PR for power levels less than 40% is dependent on PORV availability only. As previously discussed, no credit is taken for CRI (rod insertion) or AFW. Since AMSAC is not available, it is assumed that AFW will not be started by a signal. PR also accounts for the frequency of initiators that can lead to ATWS events with regard to the time when the events occur during the cycle. UETs occur early in the cycle and transient events due to startups, are more frequent early in the cycle also.

Only one fault tree is required for PR which corresponds to no AMSAC and no CRI. The fault tree structure is identical to that used for PR in event tree modeling for ATWS events with the power levels greater than 40%. The fault tree is provided in Appendix E. Only the basic event identifiers for the UET related values have changed. Since all other values are the same as those discussed in Section 5.1.1.11, only the UET related inputs are discussed further in the following.

The UETs provided on Tables 4-25, 4-27, and 4-29 need to be modified or weighted to account for the higher frequency of trips during particular times in the cycle. Typically, transient events occur more frequently early in the fuel cycle. Since the early portion of the cycle can be unfavorable, the UETs need to be weighted based on the transient distribution during the fuel cycle. The UET weighting discussed in 5.1.2.8 is applied in this ATWS state also. The weighting values are provided in Table 5-9. The final column provides the startup distribution that is used for weighting the UETs. The weighting calculations are done as shown in Section 5.1.1.11.

Tables 5-15, 5-16, and 5-17 summarize the weighted UETs. These weighted UETs are used to derive the intervals (basic events PRI1, PRI2, PRI3, and PRI4). The calculations to determine these values are the same as shown in Section 5.1.1.11. The interval values used in the PR fault trees are summarized in Table 5-18 for the low, high, and bounding cores.

5.1.3.7 ESFAS: Engineered Safety Features Actuation System

ESFAS will be credited as discussed in Section 5.1.1.7

- ESFAS failure probability = 0.01.

5.1.3.8 OAAFW: Operator Action to Start AFW

As previously noted, for ATWS events that occur with power levels less than 40%, AFW is not credited for mitigation of the primary pressure transient. But AFW is required for long-term decay heat removal which is required to be actuated by OA from the control room. The value used is conservative and based on the HEPs used in PRAs for W NSSS plants for actuating AFW for a transient event. Even though the event being analyzed is ATWS, the transient AFW value is appropriate since the AFW function for lower power ATWS events is for decay heat removal, not for RCS pressure mitigation.

- OAAFW human error probability = 1.0E-02

5.1.3.9 AFW: Auxiliary Feedwater System

The AFW system is required to remove decay heat and not required to mitigate the ATWS pressure transient. Several plant values for AFW unavailability for transient event mitigation are contained in the WOG PSA Model Methods and Results Comparison Database. These are all less than 5E-04. Based on this, a conservative value is used.

- AFW failure probability = 1.0E-03

5.1.3.10 LTS: Long Term Shutdown

Long-term shutdown is discussed in Section 5.1.1.12.

- LTS failure probability = 1.0E-02

5.1.3.11 ATWS State 1: Core Damage Frequency Quantification

The ATWS model for the ATWS State 1 was quantified using the approach discussed in Section 5.1.1.13. The event tree structure is provided in Figure 5-2.

The CDF quantification was completed for the low reactivity core (Case 1-1), high reactivity core (Case 1-2), and bounding reactivity core (Case 1-3). The change in cores is reflected in the model through the UET values and requires changing the values used for the PR intervals in the pressure relief fault trees. These values are provided in Table 5-18. All other basic event values remained the same between the three cases. The results, in terms of CDF, are provided in Table 5-19. Also shown is the increase in CDF for Cases 1-2 and 1-3 with respect to Case 1-1. Case 1-1 meets the 5% UET condition for no RI, 100% AFW, and all PORVs available.

Table 5-15 Weighted UET Values for a Low Reactivity Core, No Equilibrium Xenon, Power Level <40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.00	0.05	0.48

Table 5-16 Weighted UET Values for a High Reactivity Core, No Equilibrium Xenon, Power Level <40%			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.18	0.51	0.55

Table 5-17 Weighted UET Values for a Bounding Reactivity Core, No Equilibrium Xenon, Power Level <40%			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.52	0.55	0.58

Table 5-18 Summary of Pressure Relief Intervals, No Equilibrium Xenon, Power Level <40%			
PR Interval Basic Event	**Low Reactivity Core**	**High Reactivity Core**	**Bounding Reactivity Core**
PRI1	0.00	0.18	0.52
PRI2	0.05	0.33	0.03
PRI3	0.43	0.04	0.03
PRI4	0.52	0.45	0.42

Table 5-19 Yearly Core Damage Frequency Summary: ATWS State 1				
Plant Startup Operation, Power Level <40% **Standard Blocked PORV Probabilities** **No Equilibrium Xenon**				
Case	**Core**	**Rod Insertion (RI) Failure Probability**	**CDF (per year)**	**ΔCDF (per year)[1]**
1-1	Low Reactivity	0.5	1.32E-09	–
1-2	High Reactivity	0.5	7.00E-09	5.7E-09
1-3	Bounding Reactivity	0.5	1.34E-08	1.2E-08

Note:

1. Increase in CDF over Case 1-1 value.

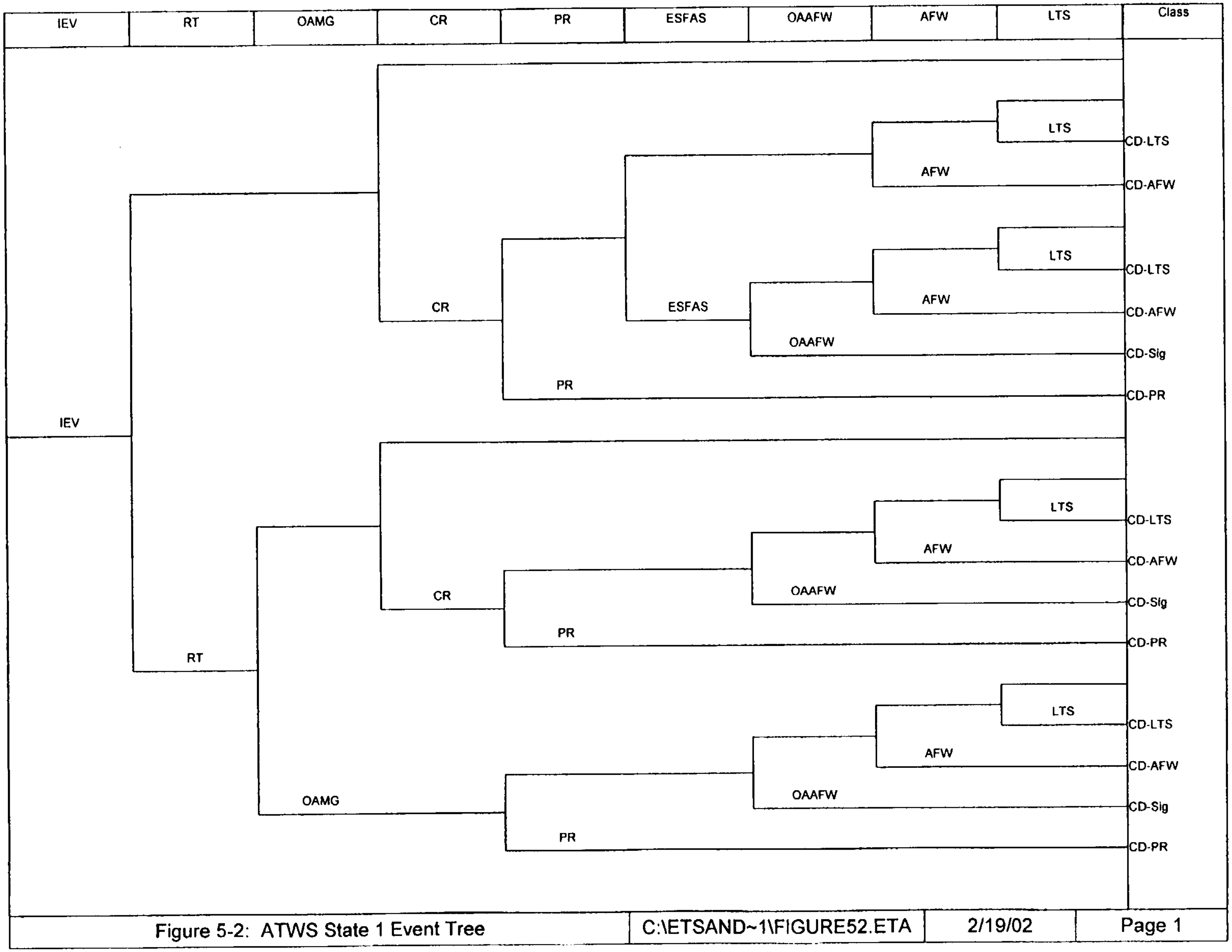

Figure 5-2 ATWS State 1 Event Tree

5.1.4 ATWS State 5: Plant Shutdown, Power Level <40%

This state represents plant operation when the power level is less than 40% during shutdown conditions. During this phase of plant operation equilibrium xenon has been established. It is assumed that the equilibrium xenon levels represent 100% power operation, that is, the plant was operating at the 100% power level prior to initiating the shutdown. This also assumes there is little xenon depletion during the power decrease to 40%. AMSAC is not operable in this state. UETs used in this evaluation are for the 40% power level with HFP equilibrium xenon which can be applied conservatively to power levels down to 0%.

5.1.4.1 ATWS State 5 Event Tree

The event tree used to evaluate ATWS State 5 is the same as that used for ATWS State 1. This is shown in Figure 5-2. The differences in the ATWS State 1 and ATWS State 5 evaluations are related to the availability of reactor trip signal, control rod insertion, and UETs, as discussed in the following sections. It is assumed in this analysis that the shutdown is occurring following plant operation at 100% power for a period of time long enough to establish equilibrium xenon consistent with 100% power level operation.

UETs are provided for the low, high, and bounding reactivity cores in Tables 4-24, 4-26, and 4-28 based on a power level of 40% and equilibrium xenon level consistent with hot full power operation prior to the shutdown. Note that the UETs for the low and high reactivity cores are all 0.0. The UETs are only provided for the condition of no AMSAC with 2, 1, or 0 PORVs available. No AMSAC means that no credit is taken for AFW start or turbine trip. In addition, UETs are not provided with CRI. As previously discussed, at lower powers the position of the control rods with respect to the core is variable; they could be completely out or partially in. If completely out, the 72 steps insertion will not provide as much benefit as if they are starting from a position that is partially in. Due to the uncertainty of the control rod position and to simplify the analysis, no credit is taken for CRI.

The fault tree model for the event tree top event RT remains the same. The fault tree for the top event PR is also the same, but again, only one condition is required that corresponds to no AMSAC (no AFW) and no CRI. AMSAC and CRI have been removed from the event tree. AFW is required for decay heat removal and must be started manually if the ESFAS signals are not available. ESFAS signals are available only when the ATWS is due RTB failures or failure of the control rods to insert with a trip signal available. There are some changes to the fault tree basic event data inputs that are discussed in the following paragraphs. The other top events are also discussed in the following paragraphs.

The following discusses the event tree top events in more detail.

5.1.4.2 IEV: Initiating Event Frequency

The value used for IEV is taken from Table 5-2 for shutdown with power level <40%.

- IEV = 0.08/yr

5.1.4.3 RT: Reactor Trip Signal from the RPS

The unavailability of the reactor trip signals is discussed in Section 5.1.1.3. The same model is used in this analysis. Note that test and maintenance activities on the RPS could be ongoing during a plant shutdown, therefore, component unavailabilities were included in the model for these activities. This is consistent with the at-power operation analysis presented in Section 5.1.1.

5.1.4.4 OAMG: Operator Action to Trip the Reactor from the MG Sets

The same values are used as discussed in Section 5.1.1.4. These are:

- 0.5 is used when RT fails due to reasons related to the OA to trip the reactor in RT in conjunction with logic cabinet or analog channel processing failures. This is a conservative conditional failure probability (conditional on a previous OA already failing).
- 1.0E-02 is used when RT fails due to reasons not related to failure of the OA to trip the reactor in RT, that is, when failures are related to RTB failures.

5.1.4.5 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

Since it is assumed that the plant has been in operation at 100% power for a period of time, no credit is taken for recent control rod movement other than surveillance test requirements. The value used for CR, consistent with Section 5.1.1.6, is:

- CR = 1.2E-06/d.

5.1.4.6 PR: Pressure Relief

This event models the availability of PR to mitigate the overpressure event. PR for power levels less than 40% is dependent on PORV availability only. As previously discussed, no credit is taken for CRI (rod insertion) or AFW. Since AMSAC is not available, it is assumed that AFW will not be started by a signal. PR also accounts for the frequency of initiators that can lead to ATWS events with regard to the time when the events occur during the cycle. UETs occur early in the cycle and transient events are more frequent earlier in the cycle also. For this analysis it is assumed that more plant shutdowns also occur earlier in the cycle. This is expected since earlier in the cycle it is more likely that plant or equipment problems will be identified that require a plant shutdown.

Only one fault tree is required for PR which corresponds to no AMSAC and no CRI. The fault tree structure is identical to that used for PR in event tree modeling for ATWS events with the power level greater than 40%. The fault tree is provided in Appendix E. Only the basic event identifiers for the UET related values have changed. Since all other values are the same as those discussed in Section 5.1.1.11, only the UET related inputs are further discussed in the following.

The UETs provided on Tables 4-24, 4-26, and 4-28, which correspond to the 40% power level with hot full power equilibrium xenon, need to be modified or weighted to account for the higher frequency of trips during particular times in the cycle. The weighting distribution that will be applied is that for the distribution of transient events throughout the cycle while at power. Typically, transient events occur more frequently early in the fuel cycle. Since the early portion of the cycle can be unfavorable, the UETs need to be weighted based on the distribution of expected shutdowns throughout the fuel cycle. As noted above, it is assumed that plant shutdowns will occur with a similar distribution throughout the cycle as plant transients. UET weighting discussed in 5.1.1.11 is applied in this ATWS state also. The weighting values are provided in Table 5-3. The final column provides the distribution which is used for weighting the UETs. The weighting calculations are done as shown in Section 5.1.1.11.

Tables 5-20, 5-21, and 5-22 summarize the weighted UETs. These weighted UETs are used to derive the intervals (basic events PRI1, PRI2, PRI3, and PRI4). The calculations to determine these values are the same as shown in Section 5.1.1.11. The interval values used in the PR fault trees are summarized in Table 5-23 for the low, high, and bounding cores.

5.1.4.7 ESFAS: Engineered Safety Features Actuation System

ESFAS will be credited as discussed in Section 5.1.1.7

- ESFAS failure probability = 0.01.

5.1.4.8 OAAFW: Operator Actuation to Start AFW

OAAFW will be credited as discussed in Section 5.1.3.8.

- OAAFW human error probability = 1.0E-02

5.1.4.9 AFW: Auxiliary Feedwater System

AFW will be credited as discussed in Section 5.1.3.9.

- AFW failure probability = 1.0E-03

5.1.4.10 LTS: Long Term Shutdown

Long-term shutdown is discussed in Section 5.1.1.12.

- LTS failure probability = 1.0E-02

5.1.4.11 ATWS State 5: Core Damage Frequency Quantification

The ATWS model for the ATWS State 5 was quantified using the approach discussed in Section 5.1.1.13. The event tree structure is provided in Figure 5-2.

The CDF quantification was completed for the low reactivity core (Case 5-1), high reactivity core (Case 5-2), and bounding reactivity core (Case 5-3). The change in cores is reflected in the model through the UET values and requires changing the values used for the PR intervals in the pressure relief fault trees. These values are provided in Table 5-23. All other basic event values remained the same between the three cases. The results, in terms of CDF, are provided in Table 5-24. Also shown is the increase in CDF for Cases 5-2 and 5-3 with respect to Case 5-1. Case 5-1 meets the 5% UET condition for no RI, 100% AFW, and all PORVs available.

Table 5-20 Weighted UET Values for a Low Reactivity Core, HFP Equilibrium Xenon, Power Level <40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.00	0.00	0.00

Table 5-21 Weighted UET Values for a High Reactivity Core, HFP Equilibrium Xenon, Power Level <40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.00	0.00	0.00

Table 5-22 Weighted UET Values for a Bounding Reactivity Core, HFP Equilibrium Xenon, Power Level <40%

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, No AMSAC	–	–	–
No RI, No AMSAC	0.00	0.23	0.29

Table 5-23 Summary of Pressure Relief Intervals, HFP Equilibrium Xenon, Power Level <40%

PR Interval Basic Event	Low Reactivity Core	High Reactivity Core	Bounding Reactivity Core
PRI1	0.00	0.00	0.00
PRI2	0.00	0.00	0.23
PRI3	0.00	0.00	0.06
PRI4	1.00	1.00	0.71

Table 5-24 Yearly Core Damage Frequency Summary: ATWS State 5

Plant Shutdown Operation, Power Level <40%
Standard Blocked PORV Probabilities
HFP Equilibrium Xenon

Case	Core	Rod Insertion (RI) Failure Probability	CDF (per year)	ΔCDF (per year)[1]
5-1	Low Reactivity	0.5	1.57E-09	–
5-2	High Reactivity	0.5	1.57E-09	0.0E-00
5-3	Bounding Reactivity	0.5	8.15E-09	6.6E-09

Note:

1. Increase in CDF over Case 5-1 value.

5.1.5 Summary of ATWS Core Damage Frequency Results

The results of the ATWS CDF analysis are summarized on Table 5-25. The CDF values are provided for each ATWS state and for the total for all ATWS states. Table 5-26 provides a summary of the important characteristics that define each ATWS state and the important model features for each ATWS state.

The following is concluded based on this analysis:

- The CDF increase from the low reactivity core to the high and bounding reactivity cores meets the ΔCDF acceptance guideline (<1.0E-06/yr) defined in Regulatory Guide 1.174.
- ATWS State 3/4, operation with the power level ≥40% and equilibrium xenon, is the largest contributor to CDF. This state contributes 88% or more to the total ATWS CDF, depending on the core reactivity.
- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, this state is the most important one to consider in plant specific PRA models. The other modes of operation are small contributors to plant risk and will not be important to the plant risk profile or to the risk-informed decision process involving changes to a plant.
- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, LERF assessments only need to consider this operating regime. The other ATWS states will be small contributors to LERF and ΔLERF.
- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, sensitivity studies provided in Section 5.1.6 are based on ATWS State 3/4.

Table 5-25 Summary of ATWS Core Damage Frequency by ATWS State

ATWS State				Core Reactivity					
				Low		High		Bounding	
ATWS State Identifier	Plant Activity	Power Level	Xenon Equilibrium	CDF	Percent[5]	CDF	Percent[6]	CDF	Percent[7]
1	Startup[1]	<40%	No	1.32E-09	1.1%	7.00E-09	3.6%	1.34E-08	2.7%
2	Startup[2]	≥40%	No	1.17E-08	9.4%	1.31E-08	6.8%	1.36E-08	2.7%
3/4	Power Operation & Shutdown[3]	≥40%	Yes	1.09E-07	87.9%	1.70E-07	88.5%	4.69E-07	93.1%
5	Shutdown[4]	<40%	Yes	1.57E-09	1.3%	1.57E-09	0.8%	8.15E-09	1.6%
Total ATWS CDF				1.24E-07	100%	1.92E-07	100%	5.04E-07	100%
CDF Increase Over Low Reactivity Core				NA	NA	6.8E-08	NA	3.8E-07	NA

Notes:

1. from Table 5-19
2. from Table 5-14
3. from Table 5-8
4. from Table 5-24
5. percent of total for low reactivity core
6. percent of total for high reactivity core
7. percent of total for bounding reactivity core

Several Key Elements:

- All CDF values are yearly values
- PORV blocked probabilities: 0.05 for two valves (A&B); 0.20 for one valve (A or B); 0.75 for no valves

Table 5-26 Summary of Important Modeling Differences Between ATWS States				
Parameter	**ATWS State 1**	**ATWS State 2**	**ATWS State 3/4**	**ATWS State 5**
Plant Activity	Startup	Startup	Power Operation and Shutdown	Shutdown
Power Level	<40%	≥40%	≥40%	<40%
Xenon Equilibrium	No	No	Yes (HFP)	Yes (HFP)
Control Rod Insertion Credit (CRI)	No	Yes (0.5)	Yes (0.5)	No
Control Rod Failure to Insert Value (CR)	1.2E-07/d	1.2E-07/d	1.2E-06/d	1.2E-06/d
AMSAC Available	No	Yes	Yes	No
RPS Test or Maintenance Activities Allowed	No	No	Yes	Yes
OA to trip reactor via MG Sets	Yes	Yes	Yes	Yes

5.1.6 ATWS Core Damage Frequency Analysis Sensitivity Studies

Based on the results and conclusions presented in Section 5.1.5, it is only necessary to consider ATWS State 3/4 in the sensitivity analysis. ATWS State 3/4 represents operation with the power level ≥40% and equilibrium xenon. The following sensitivity cases were evaluated:

Low Reactivity Core

- Case 1: Base Case (Case 3/4-1 in Section 5.1.1)
- Case 13: Worst Time in Cycle, Standard PORV Blocked Assumptions, CRI=0.5

High Reactivity Core

- Case 2: Base Case (Case 3/4-2 in Section 5.1.1)
- Case 4: Worst Time in Cycle, Standard PORV Blocked Assumptions, CRI=0.5
- Case 5: Worst Time in Cycle, One PORV Blocked, CRI=0.5
- Case 6: Worst Time in Cycle, No PORVs Blocked, CRI=0.5
- Case 7: End of Cycle, Standard PORV Blocked Assumptions, CRI=0.5
- Case 8: Yearly CDF, Standard PORV Blocked Assumptions, CRI=0.1
- Case 9: Worst Time in Cycle, Standard PORV Blocked Assumptions, CRI=0.1
- Case 10: Worst Time in Cycle, No PORVs Blocked, CRI=0.1

Bounding Reactivity Core

- Case 3: Base Case (Case 3/4-3 in Section 5.1.1)
- Case 11: Worst Time in Cycle, Standard PORV Blocked Assumptions, CRI=0.5
- Case 12: Worst Time in Cycle, Standard PORV Blocked Assumptions, CRI=0.1
- Case 14: Yearly CDF, Standard PORV Blocked Probabilities, CRI=0.1
- Case 15: Worst Time in Cycle, One PORV Blocked, CRI=0.5

Note the following on the sensitivity cases.

i. The worst time in the cycle is with regard to the UETs. The worst time is dependent on the core and is provided for the three cores below:

   - Low reactivity core – The worst time is during the interval from the 3rd to the 28th day. During this time period the plant will be able to mitigate the RCS pressure transient only with CRI, no PORVs blocked, and 100% or 50% AFW.

   - High reactivity core – The worst time is during the interval from the 14th to the 65th day. During this time period the plant will be able to mitigate the RCS pressure transient only with CRI, no PORVs blocked, and 100% AFW.

   - Bounding reactivity core – The worst time is the first 107 days of the cycle. During this time period the plant will not be able to mitigate the RCS pressure transient regardless of available mitigation equipment.

ii. The end of the cycle is the same for all the cores. The ATWS pressure transient can be mitigated in all twelve plant states. This means a minimum of 50% AFW and three safety valves are required.

iii. All the base evaluations presented in Sections 5.1.1 to 5.1.4 assumed a control rod insertion failure probability of 0.5. This is a conservative value if the rod control system is in automatic. The case with CRI set to 0.1 examines the importance of this value.

iv. All the base evaluations assumed the same probability for blocked PORVs (0.05 for two blocked, 0.20 for 1 blocked PORV). Several cases were quantified assuming one PORV is blocked and assuming no PORVs are blocked. These cases provide an indication of the importance of blocked PORVs to the CDF impact.

The results of these sensitivity cases are provide on Tables 5-27 to 5-31. The following discusses the results.

Table 5-27: This table provides an indication of the benefit of operating with a higher probability of having the rod control system in automatic over the full cycle. This shows that the CDF for the high reactivity core is expected to drop by 1.6E-08/yr (~9% of ATWS CDF) and for the bounding reactivity core by 1.9E-08/tr (~4%). Placing the rod control system in automatic increases the probability of successful partial reactivity insertion (72 steps by the lead bank). The impact of this is relatively low since this is not important later in core life because CRI is not necessary to mitigate the RCS pressure transient. It also has no impact early in life for the bounding core since all plant conditions, including those with CRI, have unfavorable exposure times. Although this is only a marginal benefit when averaged across the fuel cycle, it does provide a more significant benefit for the high reactivity core early in life. This is discussed further under the Table 5-30 discussion.

Table 5-28: This table provides the CDF values for the worst time in the cycle (at the beginning of the cycle, in this case), at the best time in cycle (end of the cycle), and the average CDF for the low reactivity core. The end of the cycle value is also applicable to the high and bounding reactivity cores since all the cores have favorable exposures in all configurations at the end of the cycle. The ATWS CDF, small to start, decreases significantly through the cycle.

Table 5-29: This table provides the same information as Table 5-28, except it is for the high reactivity core. Note that the ATWS CDF, which is small at the worst time in the cycle, decreases significantly through the cycle.

Table 5-30: This table examines the impact on CDF of several parameters for the high reactivity core during the worst time in the cycle. This is from the 14th to the 65th day during which the only condition that is favorable includes control rod insertion, 100% AFW and no blocked PORVs. By comparison with Cases 5 and 6, it is seen that a blocked PORV can have a significant impact on ATWS CDF. With no PORVs blocked the CDF is 2.19E-07/yr which increases by a factor of ~7 when a PORV is blocked. The results in this table also indicate that with no PORVs blocked, increasing the probability that the rod control system is in automatic decreases the ATWS CDF by 22% (Cases 6 and 10). These sensitivities indicate that by increasing the probability to achieve some control rod insertion and increasing PORV availability are beneficial during the worst time in the cycle. With these changes, the probability of operating in a favorable configuration is increased.

Table 5-31: This table examines the impact on CDF of several parameters for the bounding reactivity core. Cases 11 and 7 show the CDF at the worst time in the cycle and at the end of the cycle. Again, there is a significant difference in these values. Note that the CDF for Case 11 is the same as for Case 15. In both of these cases, at the worst time in the cycle, there is unfavorable exposure and the availability of a PORV provides no benefit.

Table 5-27 Sensitivity Studies: Core Damage Frequency Summary for All Cores, Standard Probabilities for Blocked PORVs

Case	Core	Rod Insertion (RI) Failure Probability	CDF (per yr)
1	Low Reactivity	0.5	1.09E-07
2	High Reactivity	0.5	1.70E-07
3	Bounding Reactivity	0.5	4.69E-07
8	High Reactivity	0.1	1.54E-07
14	Bounding Reactivity	0.1	4.50E-07

Table 5-28 Sensitivity Studies: Core Damage Frequency Summary for Low Reactivity Core, Standard Probabilities for Blocked PORVs

Case	Time in Cycle	Rod Insertion (RI) Failure Probability	CDF (per yr)
1	Yearly average	0.5	1.09E-07
7	End of cycle	0.5	2.30E-08
13	Worst time in cycle	0.5	4.54E-07

Table 5-29 Sensitivity Studies: Core Damage Frequency Summary for High Reactivity Core, Standard Probabilities for Blocked PORVs

Case	Time in Cycle	Rod Insertion (RI) Failure Probability	CDF (per yr)
2	Yearly average	0.5	1.70E-07
7	End of cycle	0.5	2.30E-08
4	Worst time in cycle	0.5	5.41E-07

Table 5-30 Sensitivity Studies: Core Damage Frequency Summary for High Reactivity Core, Worst Time in Cycle

Case	PORVs Available	Rod Insertion (RI) Failure Probability	CDF (per yr)
4	Standard Distribution	0.5	5.41E-07
5	1 (or 0)	0.5	1.51E-06
6	2	0.5	2.19E-07
9	Standard Distribution	0.1	4.92E-07
10	2	0.1	1.70E-07

Table 5-31 Sensitivity Studies: Core Damage Frequency Summary for Bounding Reactivity Core

Standard Probabilities for Blocked PORVs (except for Case 15 which has 1 PORV blocked)

Case	Time in Cycle	Rod Insertion (RI) Failure Probability	CDF (per yr)
3	Yearly average	0.5	4.69E-07
7**	End of cycle	0.5	2.30E-08
11	Worst time in cycle	0.5	1.51E-06
12	Worst time in cycle	0.1	1.46E-06
15	Worst time in cycle	0.5	1.51E-06*

* one PORV blocked
** value is from high reactivity case, but is also applicable to bounding core since the UETs are 0 at the end of the cycle

5.1.7 Incremental Conditional Core Damage Probability

Another risk measure of interest is the ICCDP. This is used to determine acceptable time periods equipment can be out of service, for example, how long can PORVs be blocked. The ICCDP calculation is generally used to assess changes to the completion times (allowed outage times, AOTs) specified in plant Technical Specifications. The ICCDP is defined in Reg. Guide 1.177 as:

$$ICCDP = (CCDF - CDF_{baseline}) \times AOT$$

where:

CCDF	=	conditional CDF with the subject equipment out of service
$CDF_{baseline}$	=	baseline CDF with nominal expected equipment unavailabilities
AOT	=	duration of single AOT under consideration

An acceptable AOT can be determined based on an acceptance guideline of ICCDP ≤ 5E-07 as provided in Regulatory Guide 1.177.

$$AOT(hr) = (5E\text{-}07 \times 8760\ hr/yr)/(CCDF - CDF_{baseline})/yr$$

Given this, the acceptable AOT, based on the worst time in the fuel cycle, to have a PORV blocked for a high and bounding reactivity core follow. Since the importance of the PORVs are dependent on the time in the cycle, the worst time in the cycle is used to develop a conservative AOT. Note that the CDF values for both cases are the same (1.51E-06/yr). The CDF corresponds to the conditions of one PORV blocked during the worst time in the cycle. Under these conditions, the pressure transient cannot be mitigated for either case, therefore, the CDF values are the same.

High reactivity core:

$$AOT = (5E\text{-}07 \times 8760)/(1.51E\text{-}06 - 1.70E\text{-}07) = 3269 \text{ hours} = 0.37 \text{ yr}$$

where:

1.51E-06/yr	=	CDF for high reactivity core, worst time in the cycle, with CRI = 0.5, one PORV blocked (Case 5)
1.70E-07/yr	=	CDF for high reactivity core, yearly average CDF, with CRI = 0.5, standard blocked PORV probabilities (Case 2)

Bounding reactivity core:

$$AOT = (5E\text{-}07 \times 8760)/(1.51E\text{-}06 - 4.69E\text{-}07) = 4207 \text{ hours} = 0.48 \text{ yr}$$

where:

1.51E-06/yr	=	CDF for bounding reactivity core, worst time in the cycle, with CRI = 0.5, one PORV blocked (Case 15)

4.69E-07/yr = CDF for bounding reactivity core, yearly average CDF, with CRI=0.5, standard blocked PORV probabilities (Case 3)

Both of these AOT values are based on using the yearly average CDF for the $CDF_{baseline}$ value. An argument could be made that the baseline CDF value should be the CDF for the core of interest (bounding or high) with CRI=0.5 and the standard blocked PORV probabilities during the worst time in the cycle, since the CDF with one PORV blocked is based on the worst time in the cycle. Since these baseline CDF values are larger that those used in the above calculations, the AOTs would be even greater.

The above ICCDP calculation indicates that PORV availability is not important to plant risk as measured by CDF. This is a direct result of the small contribution of ATWS to CDF and not because PORVs are not necessary for ATWS mitigation. PORVs are required during certain times in the cycle as evident from the UETs. From the sensitivity studies in Section 5.1.6, it was noted that increasing the availability of a PORV during the worst time in the cycle can have a significant impact on ATWS CDF. This appears to be inconsistent with the above conclusion. But the sensitivity study considers only ATWS CDF which is a small contributor to total plant CDF. The AOT calculation is based on a ICCDP guideline value (5E-07) which was developed based on total plant CDF. Therefore, increasing the availability of a PORV may have a significant impact on ATWS CDF, but only a small impact on total CDF.

5.2 ATWS LARGE EARLY RELEASE FREQUENCY ANALYSIS

This section discusses the analysis and provides the results of the analysis to determine the impact of the bounding reactivity core, relative to the low reactivity core, on LERF and the potential AOT for PORVs based on ICLERP.

At the December 17, 1998 meeting between the NRC and WOG, the NRC raised an issue regarding how the containment and safety systems inside containment will respond to the potentially large RCS pressure increase and ensuing high energy break that could occur during an ATWS event. The WOG approach to evaluate ATWS risk assumes core damage occurs if the pressure exceeds 3200 psig, and a study has been done to show that the RCS will remain intact up to this pressure. It is assumed that a loss-of-coolant accident (LOCA), that cannot be mitigated, will eventually occur and will relieve the RCS pressure in a relatively controlled manner. It is further assumed that containment systems and the containment will not be degraded. The specific NRC concern is directed at the level of confidence that the assumed LOCA will occur, as the RCS pressure exceeds 3200 psi, and relieves the pressure increase, as opposed to a catastrophic failure of the RCS that results in missile generation, degradation of containment safety systems, and possible containment failure resulting in a large early release (LER).

A three part approach was taken to address this issue. These are:

Part 1: A comprehensive examination of the RCS, and interfacing systems and components was undertaken to determine if these systems and components remain intact at the expected RCS pressures, or if missiles would be generated or RCS boundaries fail that would degrade or fail the containment. Details and results for this are provide in Appendix A (see the Response to Issues 2, 3, and 4). From this assessment of the RCS, it was determined that the SG tubes are the weak point and would be the path for a LER. From the response to Issues 2, 3, and 4 in Appendix A, the limiting RCS pressure that will result in SG tube failures is 3584 psi.

Part 2: RCS peak pressures corresponding to the possible core damage endstates related to the various combinations of CRI, AFW, and PORV availability were calculated. This is discussed and the results are provided in Section 4.3. Tables 4-20 and 4-21 provide the RCS pressures for the various configurations.

Part 3: The frequencies of reaching these RCS pressures were determined for the low and bounding reactivity cores based on a probabilistic LER model that addresses success and failure of CRI, level of AFW (100%; less than 100%, but greater than or equal to 50%; and less than 50%), pressure relief success (PORVs and safety valves). From this frequency information and the RCS pressure results, the frequency of reaching 3584 psi in the RCS and producing a LER was determined. This analysis is presented in the following section. Only LERF values for the low and bounding cores are provided. It is the intent of this analysis to show that even with changing the core design from a low reactivity to a bounding reactivity core the guideline for an acceptable impact on ΔLERF from Regulatory Guide 1.174 of 1E-07/yr is met.

5.2.1 ATWS Large Early Release Frequency for the Low and Bounding Reactivity Cores

Based on the results and conclusions presented in Section 5.1.5, it is only necessary to consider ATWS State 3/4 in the LERF assessment. ATWS State 3/4 represents operation with the power level ≥40% and equilibrium xenon. The following four cases were analyzed for LERF:

Low Reactivity Core

- Case LER1: Base Case, Standard PORV Blocked Assumptions, CRI=0.5 (this case corresponds to Case 3/4-1 in Section 5.1.1)

- Case LER2: One PORV blocked, CRI=0.5

Bounding Reactivity Core

- Case LER3: Base Case, Standard PORV Blocked Assumptions, CRI=0.5 (this case corresponds to Case 3/4-3 in Section 5.1.1)

- Case LER4: One PORV Blocked, CRI=0.5

To determine the frequency of a LER condition, the event tree shown in Figure 5-1 is used. All the top events are as described in Section 5.1 with the exception of PR. The PR top event when calculating CDF represents the failure of sufficient pressure relief to maintain the RCS pressure below 3200 psi. The pressure of interest now is 3584 psi, which leads to a LER. New pressure relief fault trees were developed that expand out the PORV and safety valve modeling. These are provided in Appendix F. UETs based on 3584 psi were not developed. This alternate approach was used since it is more versatile and can be applied to different RCS pressures limits. One key conservative assumption using this approach in the LERF analysis is that the RCS peak pressure is applied across the complete fuel cycle. This is similar to assuming that the LER unfavorable exposure time is 1.0 for plant conditions associated with the peak RCS pressure that exceed 3584 psi. That is, the RCS pressure is independent of the time in the plant operating cycle. This is a very conservative assumption since the RCS pressure will be dependent on the time in cycle and will only attain the peak pressure during the cycle's most adverse reactivity feedback conditions.

RCS pressures were calculated for a limited number of CRI/AFW/PR configurations. Those not specifically addressed are assumed to exceed the 3584 psi pressure limit. Tables 5-32 and 5-33 provide a summary of the pressures calculated for the various conditions for the low reactivity core and bounding reactivity core, respectively. Pressures that exceed 3584 psi are shown to be LER conditions. The RCS pressures are taken from Tables 4-20 and 4-21.

The LERF model was quantified for the cases previously listed. The results are provided in Table 5-34. This shows that the impact on LERF of a bounding reactivity core design is:

- ΔLERF = 1.28E-07/yr – 7.40E-09/yr = 1.21E-07/yr

This value is slightly larger that the ΔLERF guideline provided in Regulatory Guide 1.174. But it is based on a bounding reactivity core, not the high reactivity core that would provide a reduced impact on LERF, and it is based on the assumption that the peak RCS pressures will be attained at any time during the cycle. As previously noted, the second assumption is very conservative. A review of the weighted UETs for CDF for the bounding core (see Table 5-6), which are based on exceeding 3200 psi, indicates that these values are exceeded from 27% to 58% of the cycle, depending on the plant configuration. It is expected that if LERF UETs were calculated (for 3584 psi), they would be significantly less than 1.0 for the various plant configurations. Based on this, sensitivity cases were run that assumed the RCS pressure of 3584 psi would be exceeded 50% of the time for the plant configurations with pressures that exceed this limit. The results are provided in Table 5-34 as cases SenLER1 and SenLER3. These cases correspond to LER1 and LER3 except for the amount of the cycle the RCS pressure will reach the peak pressure. The impact on LERF is:

- ΔLERF = 6.78E-08/yr – 7.25E-09/yr = 6.05E-08/yr

In this case the ΔLERF guideline provided in Regulatory Guide 1.174 is met.

5.2.2 Incremental Conditional Large Early Release Probability

Another risk measure of interest is the ICLERP which is the equivalent of the ICCDP except the LERF is the basis. This can also be used to determine acceptable time periods equipment can be out of service, for example, how long can PORVs be blocked. The ICLERP calculation is generally used to assess changes to the completion times (allowed outage times, AOTs) specified in plant Technical Specifications. The ICLERP is defined in Reg. Guide 1.177 as:

$$ICLERP = (CLERF - LERF_{baseline}) \times AOT$$

where:

CLERF	=	conditional LERF with the subject equipment out of service
$LERF_{baseline}$	=	baseline LERF with nominal expected equipment unavailabilities
AOT	=	duration of single AOT under consideration

An acceptable AOT can be determined based on an acceptance guideline of ICLERP $\leq$ 5E-08 as provided in Regulatory Guide 1.177.

$$AOT(hr) = (5E\text{-}08 \times 8760\ hr/yr)/(CLERF - LERF_{baseline})/yr$$

Given this, the acceptable AOT to have a PORV blocked for the bounding reactivity core follows:

Bounding reactivity core:

$$AOT = (5E\text{-}08 \times 8760)/(1.97E\text{-}07 - 1.28E\text{-}07) = 6348 \text{ hours} = 0.72 \text{ yr}$$

where:

1.97E-07/yr	=	LERF for bounding reactivity core, with CRI = 0.5, one PORV blocked (Case LER4)
1.28E-07/yr	=	LERF for bounding reactivity core, with CRI = 0.5, standard blocked PORV probabilities (Case LER3)

The above ICLERP calculation indicates that PORV availability is not important to plant risk as measured by LERF. This is a direct result of the small contribution of ATWS to LERF and not because PORVs are not necessary for ATWS mitigation. PORVs are required during certain times in the cycle as evident from the UETs. This is consistent with the conclusions drawn from the results in Section 5.1.7 on ICCDP.

Table 5-32 Plant Conditions that Result in Large Early Releases[1], Low Reactivity Core					
CRI Success	AFW Flow (percent)	No. of PORVs	No. of Safety Valves	RCS Pressure (psi)	LER Contributor
Yes	100	2	3	2924	No
Yes	100	1	3	3078	No
Yes	100	0	3	3308	No
Yes	100	2	2	3308[2]	No
Yes	100	all other combinations		>3584	Yes
Yes	50	2	3	2987	No
Yes	50	1	3	3162	No
Yes	50	0	3	3411	No
Yes	50	2	2	3411[2]	No
Yes	50	all other combinations		>3584	Yes
Yes	<50	all combinations		>3584	Yes
No	100	2	3	3090	No
No	100	1	3	3285	No
No	100	0	3	3563	No
No	100	2	2	3563[2]	No
No	100	all other combinations		>3584	Yes
No	50	2	3	3164	No
No	50	1	3	3374	No
No	50	0	3	3664	Yes
No	50	2	2	3664[2]	Yes
No	50	all other combinations		>3584	Yes
No	<50	all combinations		>3584	Yes

Notes:

1. Defined as the RCS pressure exceeds 3584 psi.
2. The configuration of 0 PORVs and 3 safety valves is equivalent to 2 PORVs and 2 safety valves.

Table 5-33 Plant Conditions that Result in Large Early Releases[1], Bounding Reactivity Core

CRI Success	AFW Flow (percent)	No. of PORVs	No. of Safety Valves	RCS Pressure (psi)	LER Contributor
Yes	100	2	3	3333	No
Yes	100	1	3	3563	No
Yes	100	0	3	3914	Yes
Yes	100	2	2	3914[2]	Yes
Yes	100	all other combinations		>3584	Yes
Yes	50	2	3	3412	No
Yes	50	1	3	3670	Yes
Yes	50	0	3	4055	Yes
Yes	50	2	2	4055[2]	Yes
Yes	50	all other combinations		>3584	Yes
Yes	<50	all combinations		>3584	Yes
No	100	2	3	3545	No
No	100	1	3	3822	Yes
No	100	0	3	4093	Yes
No	100	2	2	4093[2]	Yes
No	100	all other combinations		>3584	Yes
No	50	2	3	3630	Yes
No	50	1	3	3955	Yes
No	50	0	3	4110	Yes
No	50	2	2	4110[2]	Yes
No	50	all other combinations		>3584	Yes
No	<50	all combinations		>3584	Yes

Notes:

1. Defined as the RCS pressure exceeds 3584 psi.
2. The configuration of 0 PORVs and 3 safety valves is equivalent to 2 PORVs and 2 safety valves.

Table 5-34 Summary of Large Release Frequencies				
ATWS State 3/4: Plant At-Power Operation and Shutdown, Power Level ≥40%				
Case	**Core**	**PORV Availability**	**CRI Value**	**LERF (per year)**
LER1	Low Reactivity	Standard Distribution	0.5	7.40E-09
LER2	Low Reactivity	One PORV Blocked	0.5	1.00E-08
LER3	Bounding Reactivity	Standard Distribution	0.5	1.28E-07
LER4	Bounding Reactivity	One PORV Blocked	0.5	1.97E-07
SenLER1	Low Reactivity	Standard Distribution	0.5	7.25E-09
SenLER3	Bounding Reactivity	Standard Distribution	0.5	6.78E-08

5.3 LOSS OF OFFSITE POWER ATWS CORE DAMAGE FREQUENCY ANALYSIS

The LOSP/ATWS event is not covered by the analysis presented in the previous sections since it is a different type of event than an ATWS with loss of main feedwater. During a LOSP event, the motor-generator sets, which provide power to the CRDMs, lose power and coast down which interrupts power to the CRDMs. The CRDMs, in turn, release the control rod assemblies which drop into the core. During a LOSP event it is not necessary to generate a reactor trip signal in the RPS to trip the plant. Therefore, the only way for an ATWS event with a LOSP event to occur is for the control rods to fail to insert due to control rod binding or mechanical problems associated with the CRDMs.

During this event the reactor coolant pumps lose power and coast down. The event is no longer an overpressure event, but a loss of flow/heatup event, with departure from nucleate boiling (DNB), not RCS overpressurization, as the issue. Therefore, the concept of UETs, as defining the time during the cycle when the pressure transient cannot be mitigated, is not applicable to LOSP/ATWS events.

Previous analyses (Reference 6) have demonstrated for low reactivity cores that there is sufficient DNB margin such that no core damage will occur. In the short term, the reactor power would be limited by a combination of negative reactivity additions (Doppler, MTC, and voiding). In the long term, the reactor would be shutdown by boration. In addition, decay heat removal via the AFW system will be required.

The MD AFW pumps and the charging pumps require AC power via the diesel generators, but the TD AFW pump does not. Therefore, operation of the diesel generators is required to power the MD AFW pumps and charging pumps. Other support systems that may be required, such as service water, are not directly addressed in the following analysis. These systems are more reliable than the DGs and would have only a minor impact, if any, on the results, and no impact on the conclusions.

It is assumed for this analysis that if DNB occurs, then core damage occurs. This is a conservative assumption since DNB does not equate to core damage, but is a simple way to define successful event mitigation that can be used to demonstrate that LOSP/ATWS events are not significant contributors to plant risk.

No analyses comparable to that in Reference 6 for low reactivity cores are available for higher reactivity cores. But preliminary studies of high reactivity cores show similar results. Based on this, a conservative analysis of LOSP/ATWS event has been done to demonstrate that the contribution to core damage is very small. This analysis assumes that AFW is required from all AFW pumps to all SGs.

Figure 5-3 shows the event tree for this event. The following defines the top events and failure probabilities for these events. This analysis uses "typical" failure probabilities to determine the expected contribution to CDF from LOSP/ATWS that is representative for all plants.

IE LOSP: Initiating Event Frequency from LOSP

The initiating event frequency for a LOSP event is based on the LOSP values for W NSSS domestic operating plants. The median of the LOSP initiating event frequencies is 0.044/yr. As a check on this, the EPRI Technical Report "Losses of Off-Site Power at U.S. Nuclear Power Plants – Through 1999" (Reference 17) reports a value of 0.034/yr based on events from 1988 to 1999.

- LOSP IE Frequency = 0.044/yr

CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

The value used for CR, as discussed in Section 5.1.1.6, is:

- CR Failure Probability = 1.2E-06

DG: Diesel Generator(s) Start and Run

On a loss of offsite power, the diesel generators are expected to start and run to provide AC power to safety related equipment. In this case, power to the MD AFW pumps and the charging system, plus supporting systems, is required. This analysis is based on a typical two train AC electrical power system (ESF buses) with one DG providing power to each ESF bus and a AFW system design with one TD pump and two MD pumps. Since it is assumed that all AFW pumps are required, both DGs are required to start and run.

The DG fail-to-start and fail-to-run values are based on the failure rates for W NSSS domestic operating plants.

DG fail-to-start = 6.9E-03/d
DG fail-to-run = 2.5E-03/hr

The initial high AFW flow rate will only need to continue until emergency boration becomes effective at shutting down the reactor. Eventually AFW to provide for decay heat removal will be sufficient. Since an analysis has not been performed to determine a mission time for the high AFW flow rate (1 TD pump and 2 MD pumps), it will be conservatively assumed to be the same as the LOSP mission time for a typical plant. This value varies depending on the detail of the utility's PRA model and depends on the probability of recovering offsite power. There is a high probability of recovering offsite power within a few hours. A conservative value typically used is 8 hours. Based on the above failure rates and mission time, the probability of failing the DG event is calculated.

- DG Failure Probability (1 of 2 DGs) = 5.4E-02.

AFW: Auxiliary Feedwater Flow Success

It is assumed that AFW flow to all four SGs from all the AFW pumps is required. As noted above, the AFW system configuration is assumed to be a design with one TD pump and 2 MD pumps. Therefore, success requires three of three pumps to four of four SGs. Section 5.1.1.9 provides an AFW failure probability for this configuration.

- AFW Failure Probability = 9.0E-02.

Boration: Emergency Boration

To finally shut down the reactor, emergency boration is required. This requires the plant operators to take an action. Section 5.1.1.12 provides a value for boration (long-term shutdown).

- Boration Failure Probability = 1.0E-02

Quantification of the Core Damage Sequences

The sequences leading to core damage follow. Substituting the appropriate values in the sequences provides the CDF for each sequence. Summing the CDF for the sequences provides the total CDF.

CD-DG = IE LOSP x CR x DG = 2.9E-09/yr

CD-AFW = IE LOSP x CR x (1-DG) x AFW = 4.5E-09/yr

CD-Bor = IE LOSP x CR x (1-DG) x (1-AFW) x BORATION = 4.5E-10/yr

CDF Total = 2.9E-09 + 4.5E-09 + 4.5E-10 = 7.9E-09/yr

This represents a very small contribution to total plant CDF and is also a minor contributor to the ATWS CDF for any of the three core types (low, high, and bounding reactivity) under consideration. If it is conservatively assumed that the LOSP/ATWS CDF contribution for the low reactivity core is 0 and the 7.9E-09/yr LOSP/ATWS CDF contribution is applicable to bounding core, this would represent an increase in CDF on 7.9E-09/yr related to the core change. This additional increase in CDF is very small compared to the increase in ATWS CDF for the different cores provided in Table 5-25 (3.8E-07/yr).

The above analysis provides a very conservative calculation since it assumes that all AFW is required which in turn requires both DGs to start and run. A basic premise for this assumption is that no control rods insert into the core. The failure probability for the CR top event assumes that 10 or more rods fail to insert, therefore, there is a high probability that some of the control rods have dropped into the core. This would provide a significant amount of negative reactivity insertion, which in turn would reduce the AFW and DG requirements. This would provide higher success probabilities for the AFW and DGs, and reduce the CDF contribution from these sequences. In addition, since the RCS pressure is not an issue, the response of the RCS and containment integrity are not issues, and consequently LERF is also not an issue.

The following is concluded from this analysis:

- LOSP/ATWS events are not significant contributors to plant CDF or plant ATWS CDF.

- LOSP/ATWS events to not produce high RCS pressures and do not impact RCS integrity.

- The increase in CDF from LOSP/ATWS events in moving from the low reactivity core to the bounding reactivity core is very small.
- Since the impacts on CDF and RCS integrity from LOSP/ATWS events are very small, this event will not be important to the plant risk profile or to the risk-informed decision process for assessing changes to a plant.

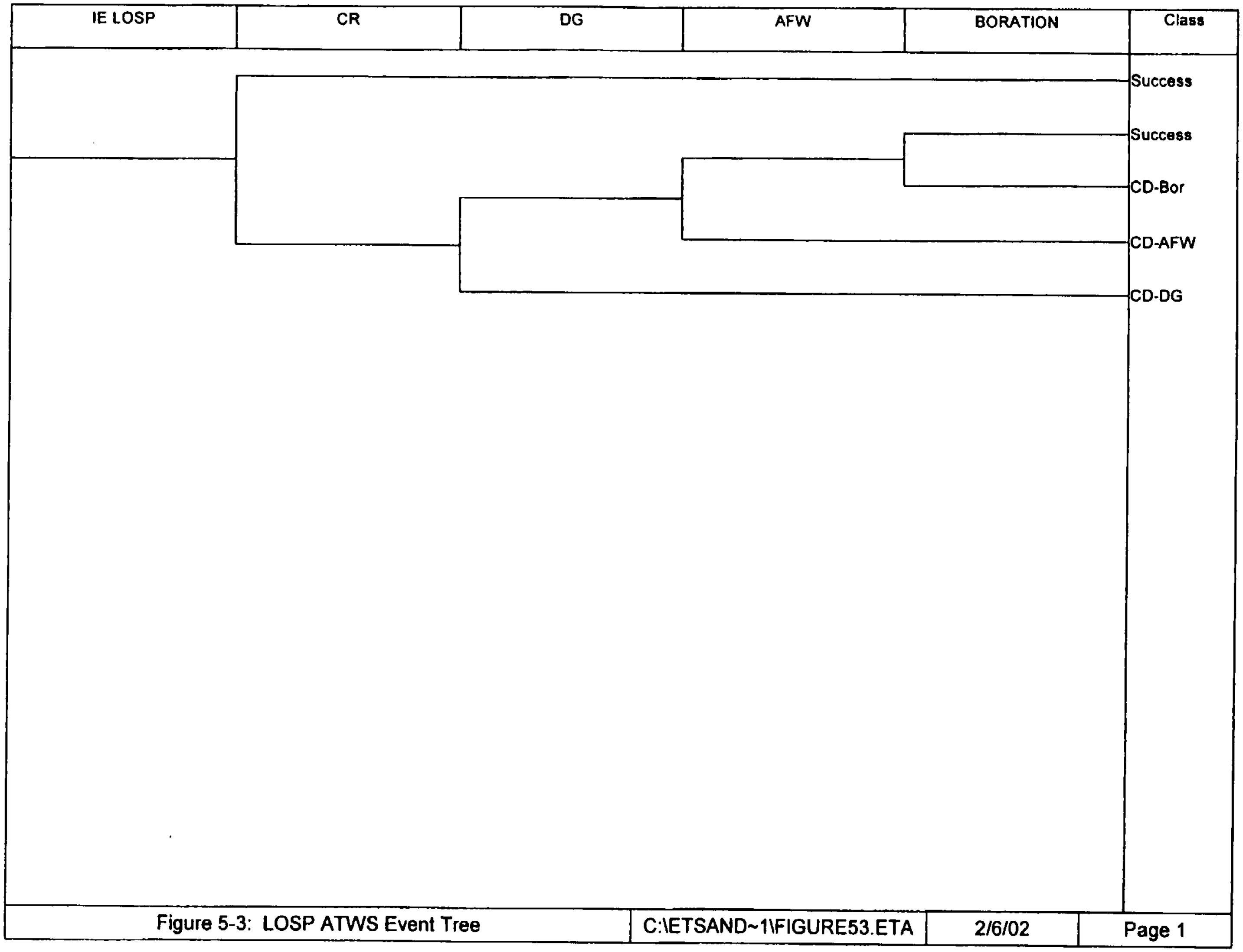

Figure 5-3 LOSP ATWS Event Tree

5.4 SUMMARY OF RESULTS FROM THE PROBABILISTIC RISK ANALYSIS

The following provides the key conclusions from the probabilistic part of the analysis. These are taken from the conclusions provided in Sections 5.1, 5.2, and 5.3.

- The CDF increases from the low reactivity core to the high and bounding reactivity cores meet the ΔCDF acceptance guideline (<1.0E-06/yr) defined in Regulatory Guide 1.174.
- The CDF contribution from ATWS events to plant total CDF is small for all core designs.
- ATWS State 3/4, operation with power level ≥40% and equilibrium xenon, is the largest contributor to CDF. This state contributes 88% or more to the total ATWS CDF depending on the core under consideration.
- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, this state is the most important one to consider in plant specific PRA models. The other modes of operation are small contributors to plant risk and will not be important to the plant risk profile or to the risk-informed decision process for assessing changes to a plant.
- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, LERF assessments only need to consider this operating regime. The other ATWS states will be small contributors to LERF and ΔLERF.
- Increasing the probability of partial control rod insertion, availability of AFW, and availability of pressure relief will reduce the ATWS contribution to plant risk even further.
- The LERF increase from the low reactivity core to the bounding reactivity core slightly exceeds the acceptance guideline (<1.0E-07/yr) defined in Regulatory Guide 1.177. This is based on the conservative approach that applies the peak configuration specific RCS pressures across the whole cycle.
- The LERF increase from the low reactivity core to the bounding reactivity core meets the acceptance guideline (<1.0E-07/yr) defined in Regulatory Guide 1.177 for the sensitivity case that assumes the peak RCS pressures are applicable to 50% of the cycle. That is, the UET is 0.5 for each plant configuration that yields RCS pressures that exceed 3584 psi. An RCS pressure of 3584 psi is noted as the pressure where SG tubes will fail, resulting in a large release.
- The LERF contribution from ATWS events to plant total LERF is small for all core designs.
- ICCDP and ICLERP analysis shows that PORV availability is not important to plant risk. Based on the RG 1.177 guideline, one PORV may be blocked for more than 3000 hours per year. This is not because PORVs are not required for ATWS mitigation, but as a result of the low importance of ATWS events to plant risk.
- LOSP/ATWS events are not significant contributors to plant CDF or plant ATWS CDF.

- LOSP/ATWS events do not produce high RCS pressures and do not impact RCS integrity.
- The increase in CDF from LOSP/ATWS events in moving from the low reactivity core to the bounding reactivity core is very small.
- Since the impacts on CDF and RCS integrity from LOSP/ATWS events are very small, this event will not be important to the plant risk profile or to risk-informed decision process for assessing changes to a plant.