Westinghouse Non-Proprietary Class 3

WCAP-15831-NP
Revision 1

September 2004

WOG Risk-Informed ATWS Assessment and Licensing Implementation Process

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15831-NP
Revision 1

WOG Risk-Informed ATWS Assessment and Licensing Implementation Process

G. R. André
G. G. Ament
R. D. Ankney
C. F. Doumont
P. J. Kotwicki
T. J. Matty
E. M. Monahan

September 2004

Approved:
G. G. Ament, Manager
Reliability and Risk Assessment

This work was performed under WOG Shop Order MUHP-1033 and MUHP-1034

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

LEGAL NOTICE

"This report was prepared by Westinghouse as an account of work sponsored by the Westinghouse Owners Group (WOG). Neither the WOG, any member of the WOG, Westinghouse, nor any person acting on behalf of any of them:

(A) Makes any warranty or representation whatsoever, express or implied, (I) with respect to the use of any information, apparatus, method, process, or similar item disclosed in this report, including merchantability and fitness for a particular purpose, (II) that such use does not infringe on or interfere with privately owned rights, including a party's intellectual property, or (III) that this report is suitable to any particular user's circumstance; or

(B) Assumes responsibility for any damages or other liability whatsoever (including any consequential damages, even if the WOG or any WOG representative has been advised of the possibility of such damages) resulting from any selection or use of this report or any information apparatus, method, process, or similar item disclosed in this report."

COPYRIGHT NOTICE

ACKNOWLEDGEMENTS

The authors acknowledge with appreciation those utility representatives and other personnel who assisted in providing information for this program, responding to questions, and providing analyses and document reviews.

Exelon Generation Company

Young In
Roy Linthicum
Dan Redden
Annie Wong

Westinghouse Electric Company

Jim Andrachek
Ken Balkey
Warren Bamford
Charles Brockhoff
Edward Carlin
William Carlson
Ike Ezekoye
Dick Haessler
Dave Katz
Robert Keating
Pat Kennevan
Robert Lutz
Dick Montoro
John Penkrot
April Rawluszki
Daniel Risher
Barry Sloane
Chuck Sterrett
Paul Stolinski
Carl Vitalbo
Derek Wenzel

Fauske and Associates, Inc.

Robert Henry

TABLE OF CONTENTS

TABLE OF CONTENTS (cont.)

TABLE OF CONTENTS (cont.)

LIST OF TABLES

LIST OF TABLES (cont.)

LIST OF TABLES (cont.)

LIST OF TABLES (cont.)

LIST OF TABLES (cont.)

LIST OF TABLES (cont.)

LIST OF FIGURES

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

IDENTIFICATION OF REVISIONS

This WCAP revision incorporates WOG responses to the following NRC clarification requests:

- Issues Identified in the NRC Letter "WCAP-15831-P, "WOG Risk-Informed ATWS Assessment and Licensing Implementation Process," July 2002 – Withdrawal of Previous Request for Additional Information (RAI) and Request for Further Clarifications Needed to Revise the Topical Report (TAC NO. MB5751)," dated March 25, 2004.
- Additional Issues Identified in a NRC/WOG Telecon on June 9, 2004 to Clarify NRC Issues Identified during a NRC/WOG Telecon on June 2, 2004.
- Additional Issues Identified in a NRC/WOG Telecon on June 25, 2004

This clarification information is added to the text of the WCAP and also included in Appendices G, H, and I. The following provides a summary of the revisions to the WCAP to respond to these clarification requests.

Revision	Location	Description
0	NA	Original Issue
1	Executive Summary	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 1	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 2.1	Text was added to state that the current applicability of the three bullets needs to consider today's risk-informed environment.
1	Section 2.2	Text was added to clarify that CRI is via the rod control system.
1	Section 2.3	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 2.4.2	Deleted the statement "Several members of the Staff did indicate ..."
1	Section 2.4.3	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 3	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 4.1	Text was added indicating where additional information on CPT calculations is located.
1	Section 4.2	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 4.2	Text was added to clarify that "... control rod insertion via the rod control system is credited."

IDENTIFICATION OF REVISIONS (cont.)

Revision	Location	Description
1	Section 4.2	Text was added indicating where additional information on UET calculations is located.
1	Section 4.3	Test was added to clarify that control rod insertion is 72 steps via the rod control system.
1	Section 4.3	Updated the last paragraph to include information on the time to reach 3200 psig.
1	Table 4-20	Revised the tables to include the time to reach 3214 psia.
1	Table 4-21	Revised the tables to include the time to reach 3214 psia.
1	Section 5	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 5.1.1.1	Text was added to clarify that CR (control rod insertion) is via the RPS.
1	Section 5.1.1.1	Text was added to the second bullet "... reactor trip fails due to failure of the control rods to fall into the core (mechanical failure)...".
1	Section 5.1.1.5	Text was clarified with regard to ATWS mitigation following success or failure of CRI. Text was added discussing the reliability of the rod control system.
1	Section 5.1.1.11	Text was clarified that control rod insertion requires 72 steps.
1	Section 5.1.1.11	Additional text was added to explain the difference between the total number of trips in Table 5-2 (240 trips) and in Table 5-3 (194 trips).
1	Table 5-26	Clarified that Control Rod Insertion Credit (CRI) in the parameter column is 72 steps.
1	Section 7	Eliminated Approach 2 "Application of the Plant CRMP" as a possible way to address configuration management.
1	Section 7	Added significant information on the key characteristics of the ATWS CMP including information on the applicability of the ATWS CMP, overall CMP structure and administrative control, compensatory actions, time allowed in an unfavorable configuration, core design considerations, and additional ATWS CMP requirements.
1	Section 8.2	Text was added to the second bullet "... reactor trip fails due to failure of the control rods to fall into the core (mechanical failure) given a reactor trip signal was generated."
1	Section 8.2.4	Text was clarified with regard to ATWS mitigation following success or failure of CRI. Text was added discussing the reliability of the rod control system.
1	Section 8.2.5	Text was added to the last paragraph further explaining why CR does not need to be addressed explicitly after success of CRI.
1	Section 8.3	Text changed in second paragraph related to calculations of ICCDP.
1	Section 9	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.

IDENTIFICATION OF REVISIONS (cont.)

Revision	Location	Description
1	Table 9-4	Text was added to further define the ATWS CMP.
1	Table 9-4	Text was added to parameter "OA to Drive Control Rods In (72 steps from lead bank)."
1	Section 10	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 10	Text was revised on the need for the LERF evaluation.
1	Section 10	Reference to Approach 2 was removed from the second option.
1	Section 11	Text was added to clarify that the configuration for the 5% UET restriction is the ATWS Rule reference configuration.
1	Section 11	Reference to Approach 2 was removed from the second option.
1	Appendices	Added Appendix G that contains responses to the NRC's request for further clarifications provided in a letter dated March 25, 2004.
1	Appendices	Added Appendix H that contains responses to additional issues identified in a NRC/WOG Telecon on June 9, 2004 to clarify NRC issues identified during a NRC/WOG Telecon on June 2, 2004.
1	Appendices	Added Appendix I that contains responses to additional issues identified in a NRC/WOG Telecon on June 25, 2004.

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, for the ATWS Rule reference configuration of no rod insertion, 100% auxiliary feedwater, and no blocked power operated relief valves, 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 for the ATWS Rule reference configuration 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 ATWS Rule reference configuration 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;
- Address the Nuclear Regulatory Commission's (NRC) 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 in the Braidwood Nuclear Generating Station probabilistic risk analysis (PRA) model. An objective of the pilot plant application is to delete Technical Specification 5.6.5b.5 for Braidwood Units 1 & 2 and Byron Units 1 & 2 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 (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 for the ATWS Rule reference configuration of no rod insertion, 100% auxiliary feedwater, and no blocked power operated relief valves, 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.

These important points are based on SECY-83-293 and the final ATWS Rule, and were applicable at the time those documents were issued. The current applicability of these points needs to consider today's risk-informed environment and changes in plant operation.

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) via the rod control system, 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 (ATWS Rule reference configuration). 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 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).

<u>Benefits of Change</u>: 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 ATWS Rule reference configuration 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 for the ATWS Rule reference configuration, 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 ATWS Rule reference configuration 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.

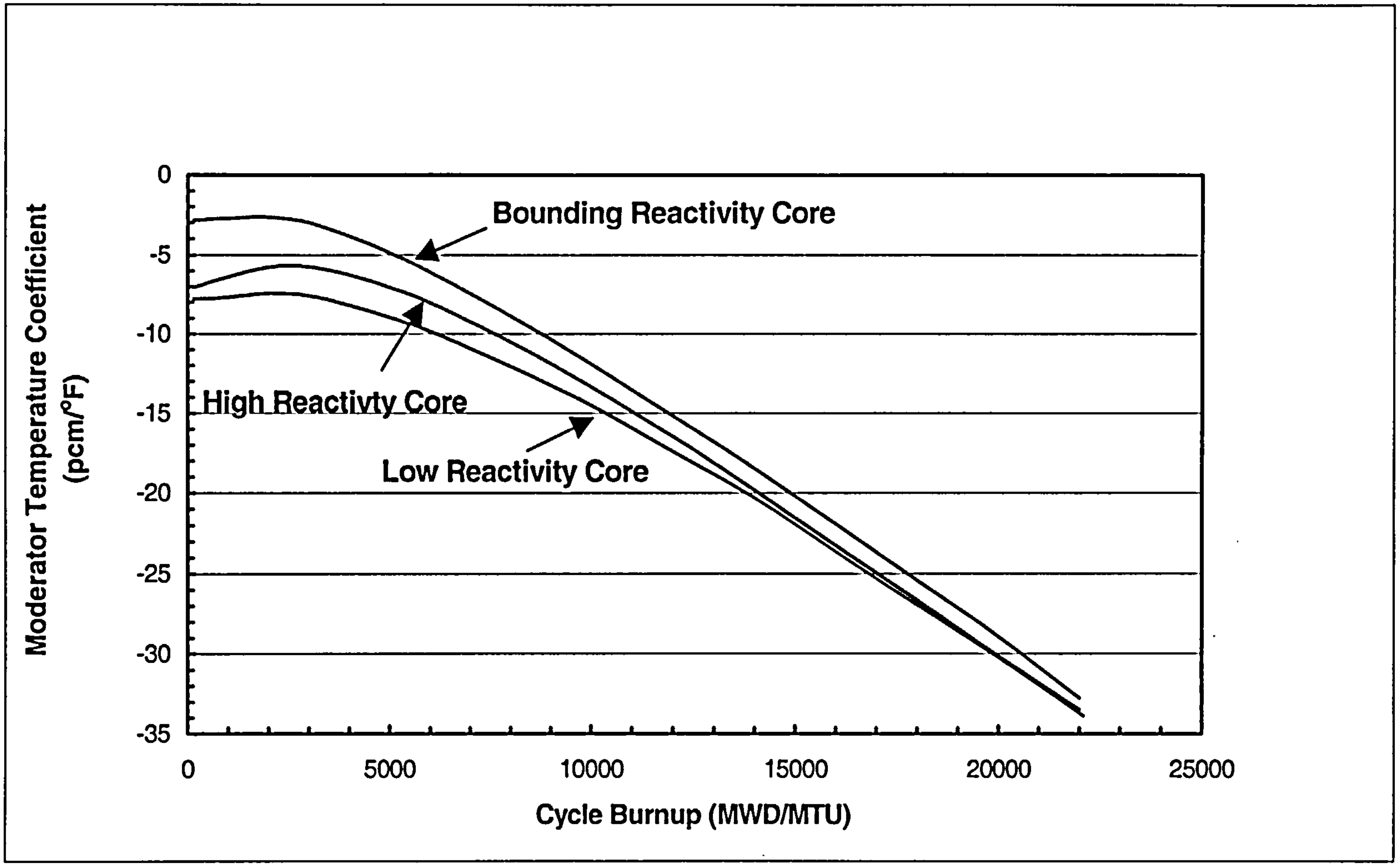

Figure 3-1 HFP Moderator Temperature Coefficient versus Cycle Burnup for the Low, High, and Bounding Reactivity Core Designs

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., $\geq 600°F$) 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.

Additional information related to the calculation of CPTs is provided in the response to Issue 6 in Appendix A and Technical Clarification 5 in Appendix G.

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; ATWS Rule reference configuration). 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 via the rod control system 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.

Additional information related to the calculation of UETs is provided in the response to Issue 6 in Appendix A and Technical Clarification 5 in Appendix G

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 (72 steps via the rod control system) 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 credited 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. Also shown in these tables are the times that the RCS pressures reached 3200 psig. It was found that the RCS pressure remains below 3200 psig for at least 90 seconds for all of the cases analyzed.

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)**	**Time RCS Pressure Reaches 3214 psia (sec)**
1	2	100	No	3546	99.5
2	1	100	No	3822	97.2
3	0	100	No	4093	94.5
4	2	50	No	3630	97.8
5	1	50	No	3955	95.7
6	0	50	No	4110	93.2
7	2	100	Yes	3333	109.2
8	1	100	Yes	3563	104.7
9	0	100	Yes	3914	101.2
10	2	50	Yes	3412	106.2
11	1	50	Yes	3670	102.7
12	0	50	Yes	4055	99.6

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)	Time RCS Pressure Reaches 3214 psia (sec)
13	2	100	No	3090	NA
14	1	100	No	3285	111.4
15	0	100	No	3563	104.9
16	2	50	No	3164	NA
17	1	50	No	3374	107.8
18	0	50	No	3664	102.8
19	2	100	Yes	2924	NA
20	1	100	Yes	3078	NA
21	0	100	Yes	3308	117.3
22	2	50	Yes	2987	NA
23	1	50	Yes	3162	NA
24	0	50	Yes	3411	113.4

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 ATWS Rule reference configuration 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 via the RPS), 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 (mechanical failure) 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. Some core designs will require CRI success during parts of the cycle to achieve successful pressure mitigation. In addition, it is possible to design a core that for part of the cycle the ATWS pressure transient cannot be mitigated regardless of the success or failure of CRI. But regardless of whether CRI succeeds or fails, auxiliary feedwater and pressure relief availability still need to be addressed. 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)

Given that the rod control system is in automatic, there is a high probability that it will function properly to step in the control rods. To determine a failure probability for the rod control system, a review of failure data on this system was performed based on failure events for 2002 and 2003 recorded in INPO's EPIX database. Eleven failure events were identified that were not immediately detectable, via alarms or unexpected control rod movements, that may cause failure of the rod control system in response to an ATWS event. Note that from the event descriptions in the database it was not always clear that the failure would have failed the rod control system in response to an ATWS event, but for simplicity, and to develop a conservative failure history, it was assumed that these 11 failures would lead to failure.

These failures that cannot be immediately detected will be detected in the short term by operators either noting the lack of control rod motion on a daily basis or noting a decreasing T_{avg}. The rod control system will compensate for changes in T_{avg} with control rod motion, therefore, every control rod step change required is equivalent to a surveillance test. Automatic control rod motions are expected on a daily basis, therefore, a failed rod control system could be detected within 1 day. Based on this information and assuming 50% of the W NSSS plants operate with the rod control system in automatic, a failure probability of 2.4E-03/demand is determined. The clarification to NRC question 1 in Appendix I provides additional information on this calculation and demonstrates that the reliability of the rod control system can be significantly lower than this value depending on several assumptions.

From this it is concluded that the rod control system is a highly reliably system and failures that do occur will be detected in a short period of time. Therefore, there is a low probability that the rod control system will fail to respond as require to an ATWS event.

It should be noted that the PRA model assumes an unavailability of 0.1 for the rod control system. Given that the rod control system is in automatic, this is a very conservative value.

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 72 steps of 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 (72 steps) success, 100% AFW
- control rod insertion (72 steps) success, 50% AFW
- control rod insertion (72 steps) failure, 100% AFW
- control rod insertion (72 steps) 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.

Note that the total number of trips in Table 5-2 is 240 and that the total in Table 5-3 is 194. The trips used to determine the distribution are those that occurred when the power level was greater than or equal to 40% which, from Table 5-2, is 202. This includes 8 trips that occurred after 18 months in the fuel cycle. Subtracting these 8 trips from the 202 trips leaves 194 trips which matches the total on Table 5-3. These eight trips were due to plants with significant downtime following startup after a refueling so the cycle time ran past 18 months. To maintain a set of data to determine the trip distribution over an 18 month fuel cycle, these were eliminated.

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.

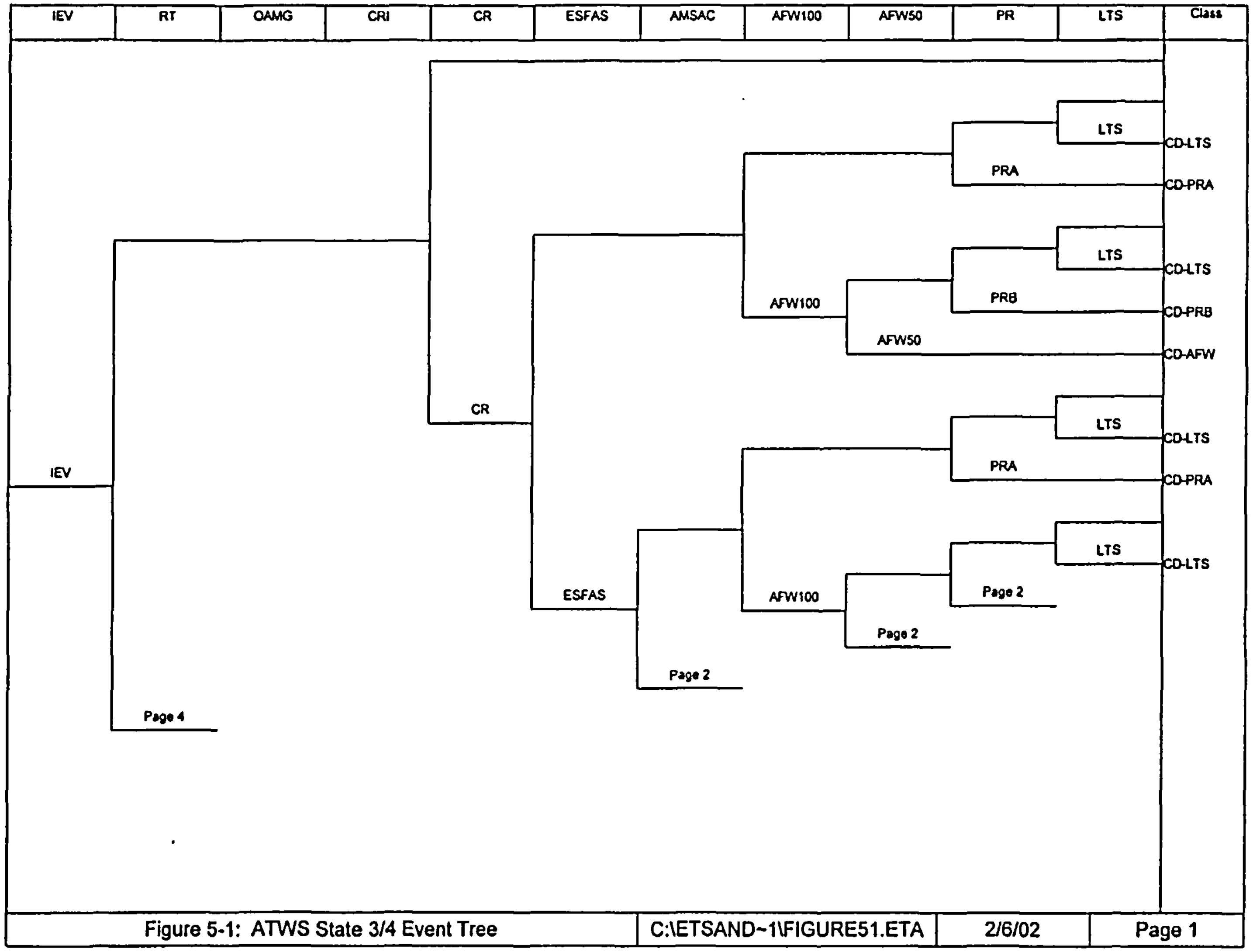

Figure 5-1 ATWS State 3/4 Event Tree

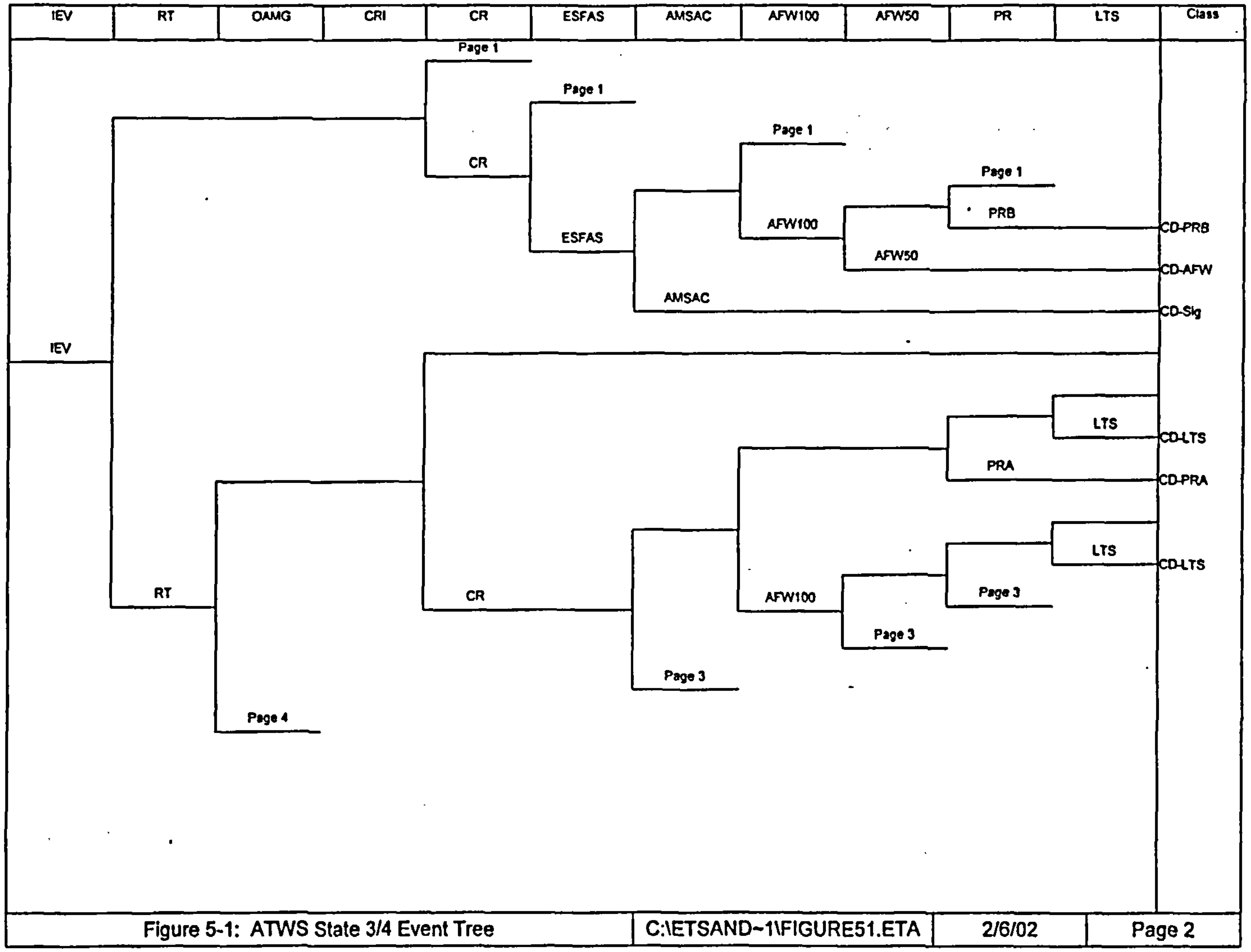

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

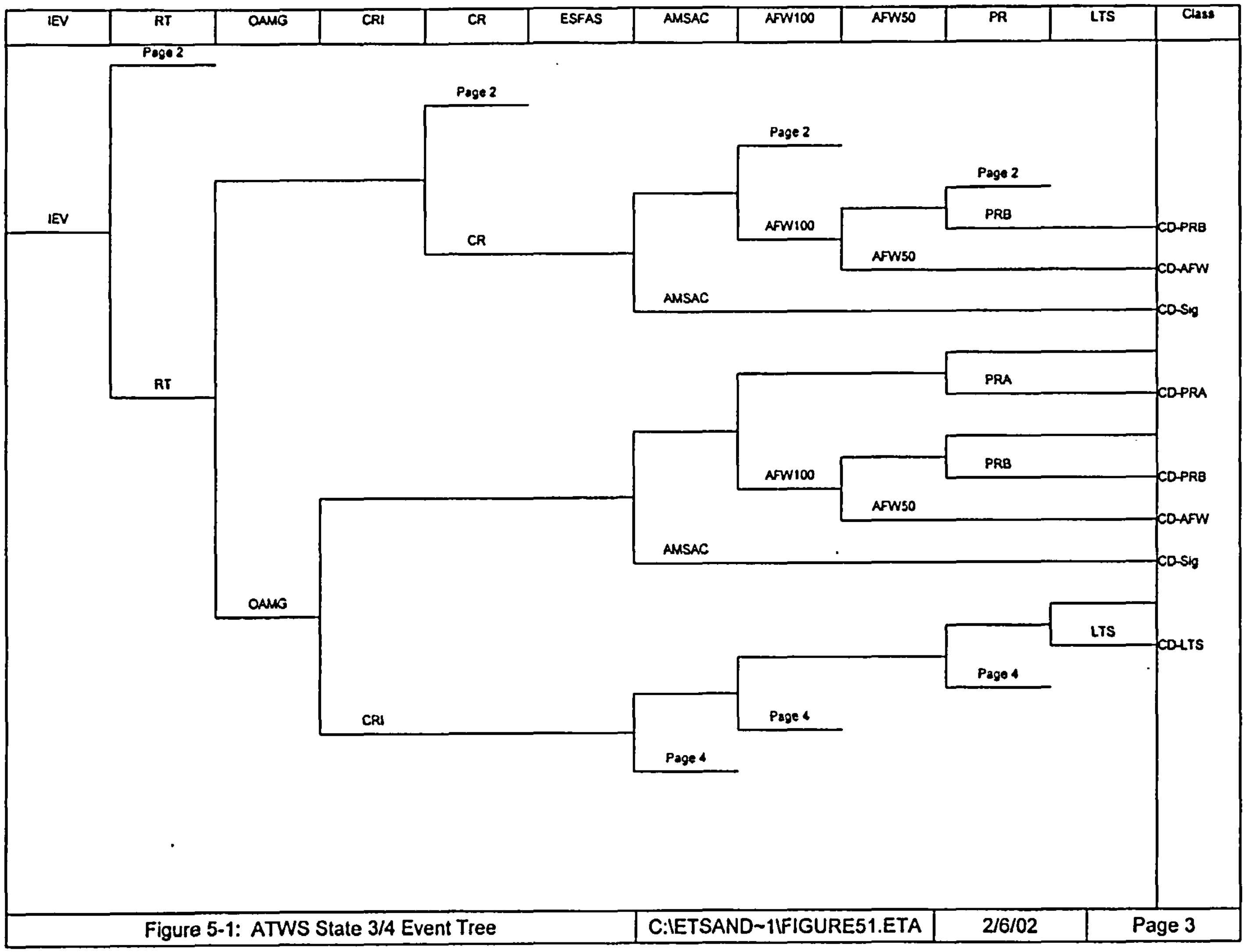

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

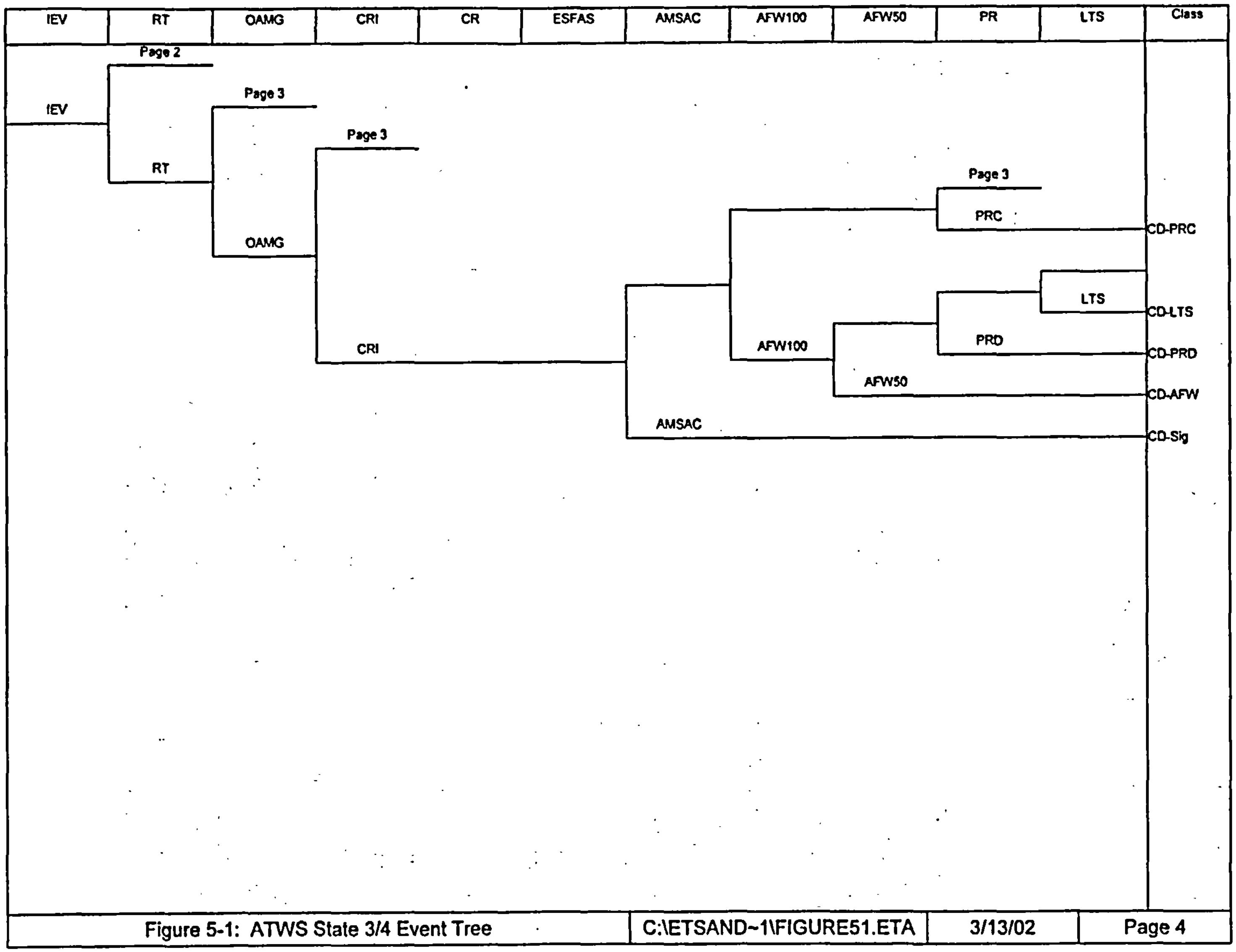

Figure 5-1 ATWS State 3/4 Event Tree (cont.)

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 Distribution of Plant Startups Across the Cycle (cont.)

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.

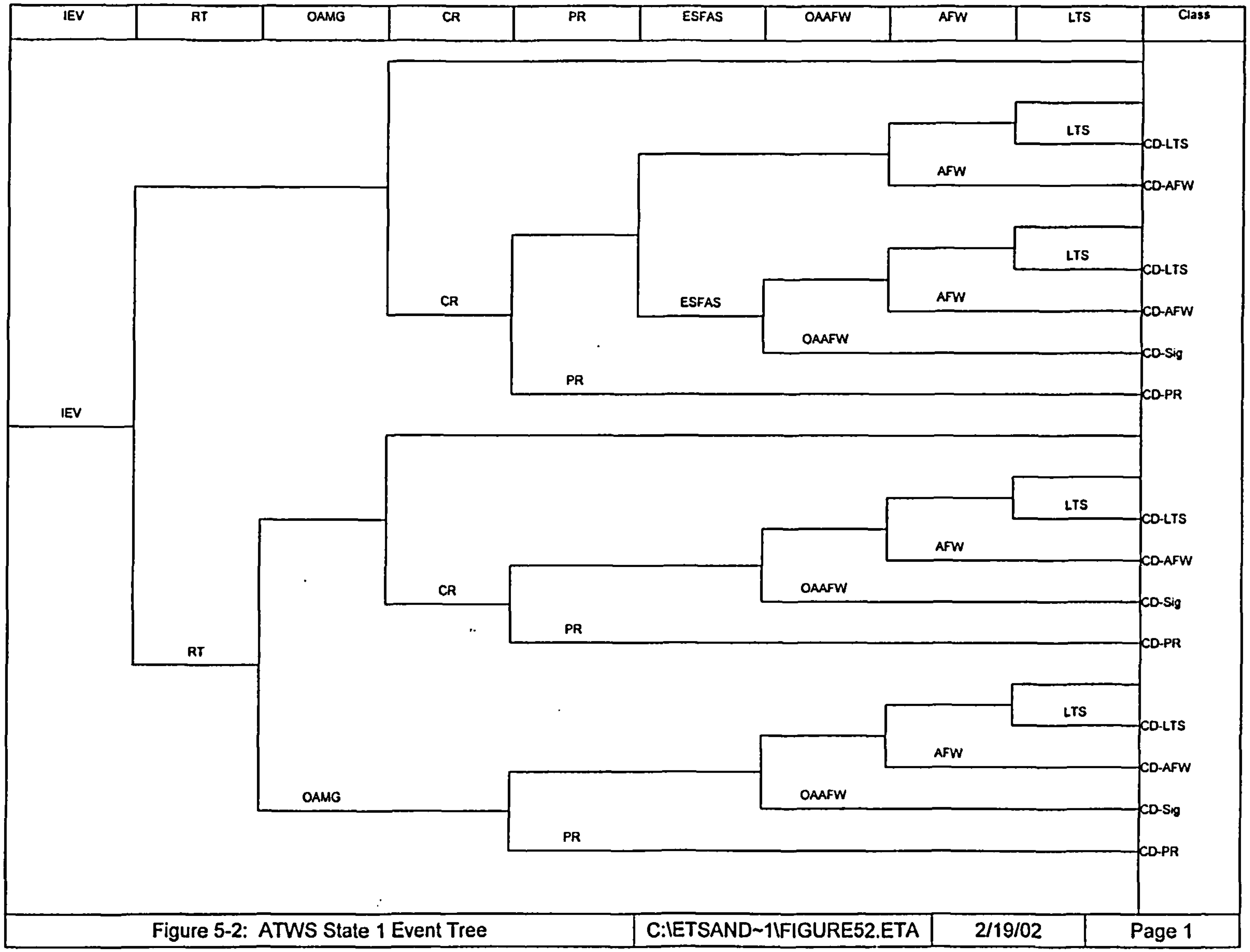

Figure 5-2 ATWS State 1 Event Tree

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 – 72 Steps (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:

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

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 $\leq$ 5E-07 as provided in Regulatory Guide 1.177.

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

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:

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

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:

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

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:

$$\text{ICLERP} = (\text{CLERF} - \text{LERF}_{\text{baseline}}) \times \text{AOT}$$

where:

CLERF	=	conditional LERF with the subject equipment out of service
$\text{LERF}_{\text{baseline}}$	=	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, 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, 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.

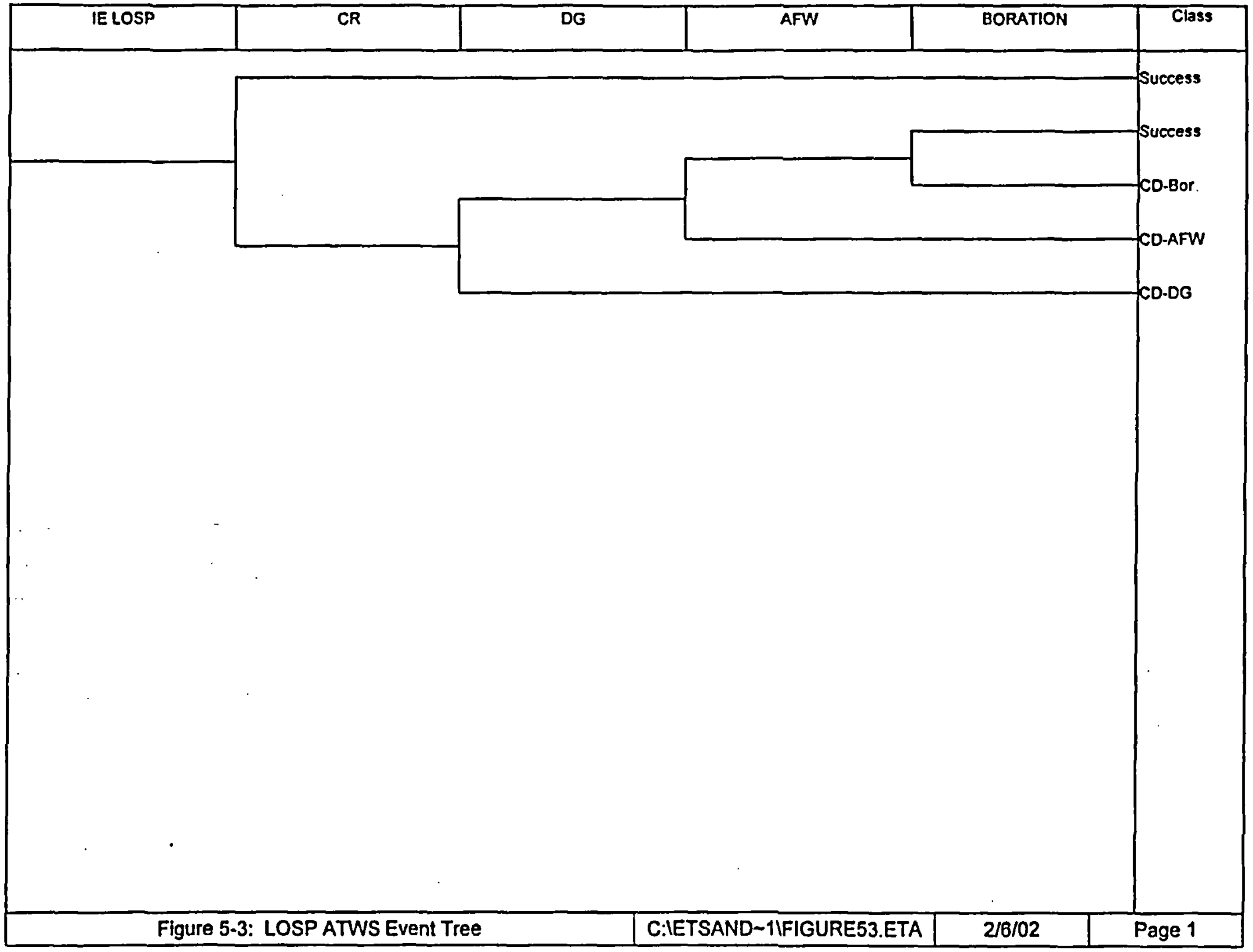

Figure 5-3 LOSP ATWS Event Tree

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.

6 IMPACT ON DEFENSE-IN-DEPTH AND SAFETY MARGINS

Section 5 discussed the impact of the changes on risk. According to the guidance in Regulatory Guide 1.174, the traditional engineering considerations also need to be addressed. These include defense-in-depth and safety margins. The fundamental safety principles on which the plant design is based cannot be compromised. Design basis accidents are used to develop the plant design. These are a combination of postulated challenges and failure events that are used in the plant design to demonstrate safe plant response. Defense-in-depth, the single failure criterion, and adequate safety margins may be impacted by the proposed change, and consideration needs to be given to these elements.

6.1 IMPACT ON DEFENSE-IN-DEPTH

Events that can occur in reactors can be mitigated by a number of safety systems that provide various levels of defense. Changes in the level of protection afforded by one level of defense, say due to equipment failure, can be compensated for by others. There are three basic levels of defense that ensure the reactor will be protected against RCS overpressurization and possible failure of the RCS pressure boundary with subsequent core damage from ATWS events. These include:

- Prevention: reactor trip with backup operator actions
- Control and Mitigation: the core physics defense barrier (reactor core and moderator feedbacks)
- Control and Mitigation: operation of existing systems to limit the potential pressure/temperature transient and provide reactor coolant inventory addition if necessary

Prevention: Reactor trip with backup operator actions

The first level of protection is provided by the RPS and backup operator actions. The RPS is an automatic system that will shut down the reactor if the RCS or core parameters exceed specified setpoints. The RPS consists of two redundant trains with each train consisting of logic cabinets and reactor trip breakers. The reactor trip breakers can be actuated automatically by two diverse mechanisms: the undervoltage trip and the shunt trip. Analog channels arranged in 2 of 3 or 2 of 4 combinational logic supply signals to each logic cabinet. The channels monitor plant operating parameters and provide signals to both logic cabinets that provide signals to open their respective reactor trip breakers. Reactor trip occurs when the trip combinational logic is met. Signals to trip the plant will be generated from at least two sets of channels for every anticipated transient event that can occur. If the automatic signal fails, then operators can take several actions, which follow, to trip the plant.

- Manually trip the reactor via the trip switch in the control room.
- Manually trip the reactor via interrupting power to the CRDMs from the MG sets (from the control room in many plants; locally at the MG sets near the control room in some plants).
- Manually drive in the control rods via the rod control system.

The first operator action listed provides a signal to open the reactor trip breakers, therefore, it is effective if the automatic trip failed due to failures in the logic cabinets or analog channels. If reactor trip failed due to reactor trip breaker failure or failure of a sufficient number of control rods to drop into the core, this action is ineffective. The second operator action listed interrupts the power to the CRDMs, therefore, it bypasses the RPS completely. This action is effective if the automatic trip failed due to failures in the logic cabinets, analog channels, or reactor trip breakers. If the reactor trip failed due to an insufficient number of control rods dropping into the core, then this operator action is also ineffective. (Note that, as discussed in Section 5.1.1.6, a very large number of control rods must fail to drop into the core in order to present an RCS integrity challenge via overpressure.) The third operator action listed requires the operator to drive the rods into the core by the rod control system. This action is taken if the rod control system is not in the automatic mode of operation. This action is effective if the automatic trip failed due to failures in the logic cabinets, analog channels, or reactor trip breakers. If the reactor trip failed due to an insufficient number of control rods dropping into the core, then this operator action may also be ineffective.

Table 6-1 provides a summary of the operator actions that are available to backup the various failures of the RPS.

One aspect of prevention is the industry trend, since the time that studies such as WCAP-11992 were performed in the late 1980s, to reduce annual plant trip challenges. As plants have matured and efforts to improve plant reliability have been implemented, the number of reactor trips has trended downward from roughly 4-8 per reactor-year to closer to 1 per reactor-year.

Control and Mitigation: Core physics defense barrier (reactor feedbacks)

An additional barrier in defense-in-depth is related to the design of the core with respect to the moderator reactivity feedback. The core is designed to provide negative moderator reactivity feedback to limit the reactor power and the RCS pressure transient if the RCS begins to heat up excessively. This is important for anticipated events, such as, loss of feedwater events that, without a rapid reactor trip, cause the reactor coolant system and core to increase in temperature. The negative reactivity reduces the reactor power and provides the operator time to borate the RCS to bring the reactor to shutdown conditions. Core designs with sufficiently negative reactivity feedback provide a "natural" barrier which limits events that could lead to core damage.

Control and Mitigation: Limit potential pressure transient

In addition to core reactivity feedbacks, in the defense-in-depth scheme, mitigation of the pressure transient by the RCS pressure relief system is also possible. This consists of pressurizer safety valves and PORVs. For a given core, the pressure transient that will need to be accommodated will depend on the time in cycle, the AFW flow rate, and the amount of negative reactivity insertion provided by the control rods. In many ATWS scenarios, partial control rod insertion will occur. In addition, as explained in the preceding paragraph, the operator can take action to manually drive the control rods into the core or the rod control system may be in the automatic mode, which would then automatically move the control rods into the core. Following successful mitigation of the pressure transient, the operator would have a substantial amount of time to borate the RCS to bring the reactor to shutdown conditions.

The AFW system will be started by either AMSAC or the ESFAS signals. AMSAC is a backup to the ESFAS. Signals from the ESFAS will be available to start AFW and trip the turbine under some, but not all, ATWS scenarios. Table 6-2 provides a summary of signals available to actuate the AFW and trip the turbine for the various failures of the RPS.

For ATWS events with peak pressures that do not exceed the safety valve setpoints, the event can be mitigated by emergency boration. Actuation of emergency boration requires an operation action.

Discussion

These barriers work together to provide a total level of plant protection and do not always offer three completely independent safety mechanisms. A partial degradation of one can be compensated for by another. For example, in many ATWS scenarios, partial insertion of the control rods is expected. This will reduce the severity of the pressure transient. For higher reactivity cores, the MTC may not be sufficient early in life to limit the pressure transient to below the pressurizer safety valve setpoints and pressure relief via these valves would be expected. Towards the end of life, pressure relief may not be required since negative reactivity feedback would be sufficient to limit the pressure transient.

If reactor trip fails, that is, a sufficient number of control rods do not drop into the core to shut it down, the pressure relief required to mitigate the potential pressure transient in the RCS will depend on a number of variables. These include core reactivity, time in core life, amount of negative reactivity provided by the controls rods that did drop, and AFW flow. It should also be noted that core design studies show that a large number of the control rod assemblies must fail to insert (i.e., a highly unlikely event) for a severe pressure transient to occur.

For higher reactivity cores, the MTC will be less negative (but always negative) at full power than for lower reactivity cores. The higher reactivity cores will result in higher pressure transients for similar conditions, time in life and AFW flow than low reactivity cores. But actions can be implemented during normal operation with higher reactivity core designs to counter this increased reactivity so that any higher pressure transients can be successfully mitigated.

Tables 4-3, 4-4, 4-7, and 4-8 show the UETs for the low reactivity and high reactivity core designs for 100% power and equilibrium xenon. As previously noted, a comparison of the UET values indicates the following:

- The higher reactivity core has longer UETs.
- Both cores can be operated with 0 UETs, but the lower reactivity core provides more flexibility to achieve this.
- To operate in a plant configuration with a low UET with the high reactivity core, it is important to maintain PORV availability, AFW availability, and control rod insertion from the lead bank (through either manual or automatic control rod insertion).

Tables 6-3 and 6-4 show the probabilities or split fractions for being in certain plant configurations dependent on the state of the rod control system, and PORV and AFW availability. Table 6-3 assumes that the rod control system is in manual, PORVs may be blocked, and AFW may be unavailable due to test or maintenance activities. Table 6-4 assumes that the rod control system is in automatic, a reduced probability that the PORVs are blocked, and the AFW system is available (although it may fail due to random or common cause component failures). A comparison of the information in these tables indicates it is possible to compensate for the degradation of one barrier with another. For example, plant configuration management scheme 2 (Table 6-4) ensures that the plant is operating in a configuration that can compensate for the degradation of the "natural" barrier. The probability of being in a 0 UET configuration is much higher in this scheme than in plant configuration management scheme 1 (Table 6-3).

In addition, and not illustrated in this example, it is also possible to restrict removal of RPS components from service for preventive type activities during unfavorable portions of the cycle. Extending test times to increase the availability of the RPS is also possible, but would require Technical Specification changes. These restrictions will increase the availability of the RPS during the portion of the cycle when the natural reactivity feedback mechanisms are less effective.

Based on the above discussion, it is seen that sufficient defense-in-depth barriers exist such that it is possible to compensate for limited degradation of one barrier with another and, therefore, maintain plant safety afforded by defense-in-depth requirements. This is an effective approach for managing the risk associated with ATWS events when implementing higher reactivity cores or other plant changes.

Elements of Defense-in-Depth

Regulatory Guide 1.174 defines the elements that comprise defense-in-depth that proposed changes need to meet. These elements and the impact of the proposed change on each follow:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

  The proposed change in core design has only a small calculated impact on CDF and LERF as discussed in Section 5. The proposed change impacts both CDF and LERF via higher RCS pressures if an ATWS event occurs. The LERF is impacted primarily from ATWS induced SG tube failures. The change in core design does not degrade core damage prevention and compensate with improved containment integrity nor does it degrade containment integrity and compensate with improved core damage prevention. The balance between prevention of core damage and prevention of containment failure is maintained. Consequence mitigation remains unaffected by the proposed change. Furthermore, no new accidents or transients are introduced with the requested change and the likelihood of an accident or transient is not impacted. The impacts on CDF and LERF are very small as demonstrated in Section 5.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design.

  The core design will change such that higher RCS pressures will occur if an ATWS event occurs. The magnitude of the RCS pressure will depend on the time in life when it occurs and the availability of pressure relief, AFW, and negative reactivity insertion. All safety systems, including the RPS, AFW system, RCS pressure relief capability, and rod control system will continue to function in the same manner with the same reliability, and there will be no additional reliance on additional systems or operator actions. The impact on risk is very small, but depending on the plant configuration, there could be an impact on defense-in-depth. This will be compensated for by plant configuration management programs that improve the preventive aspect or alternate mitigative capabilities as discussed in Section 7.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.

  No individual system redundancy, independence, or diversity will be impacted by the use of high reactivity cores.

- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.

  Defenses against common cause failures are maintained. The change requested does not impact or introduce any new common cause failure mechanisms. The probability of control rods failing to drop into the core will not be impacted by this change. This change does not impact ATWS preventive or mitigative systems, such as the RPS, AFW system, RCS pressure relief, or the rod control system.

- Independence of barriers is not degraded.

  The barriers protecting the public and the independence of these barriers are maintained. As previously indicated, there will be a small impact on the natural barrier, but it will remain independent of preventive barrier and the RCS pressure mitigation system (PORVs and safety valves). In addition, this change does not provide a mechanism that degrades the independence of the fuel cladding, RCS, and containment barriers.

- Defenses against human errors are maintained.

  No new operator actions related to the change are required to maintain plant safety. No additional operating, maintenance, or test procedures will be introduced or modified due to these changes. During the unfavorable exposure time, a configuration risk management program will be used to control other activities that could impact prevention or mitigation of ATWS events to compensate for an impact on defense-in-depth. This is discussed in Section 7.

6.2 IMPACT ON SAFETY MARGINS

With regard to safety margins, an acceptable guideline to follow, per Regulatory Guide 1.174, for demonstrating compliance with safety margins is as follows. With sufficient safety margins:

- Codes and standards or their alternatives approved for use by the NRC are met.

- Safety analysis acceptance criteria in the licensing basis (FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Consistent with these guidelines, implementation of the subject risk-informed approach to determine the impact of core design changes on plant safety will not eliminate the requirement to assess the impact of the change on the plant safety analysis licensing basis. 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. As such, the range of applicability of core design changes included in the risk-informed approach, including moderator temperature coefficient, are limited by the ability to meet applicable acceptance criteria of the FSAR Chapter 15 design basis events and by any existing plant specific Technical Specifications.

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

Failed RPS Element	Backup Operator Action		
	OA for Reactor Trip from the Control Room	OA to Interrupt Power to MG Sets from the Control Room	OA to Drive in the Control Rods
Analog Channels	Yes	Yes	Yes
Logic Cabinets	Yes	Yes	Yes
Reactor Trip Breakers	No	Yes	Yes
Control Rods	No	No	No

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

Failed RPS Element	Actuation Signal		Comments
	ESFAS	AMSAC	
Analog Channels	No	Yes	ESFAS signal is not available. Reactor trip and ESFAS signals are assumed to be failed due to common cause failure.
Logic Cabinets	No	Yes	ESFAS signal is not available. Reactor trip and ESFAS signals are assumed to be failed due to common cause failure.
Reactor Trip Breakers	Yes (AFW) No (turbine trip)	Yes	ESFAS is still available to start AFW, but the turbine trip signal will not be available since it is developed when a RTB closes. No common cause failure exists between ESFAS and reactor trip signals for reactor trip breaker failures.
Control Rods	Yes	Yes	ESFAS is still available to start AFW and trip the turbine. No common cause failure exists between ESFAS signals and the control rods failing to drop.

Table 6-3 Plant Configuration Probabilities, Plant Configuration Management Scheme 1			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
Rod Insertion 100% AFW	0.338	0.090	0.023
Rod Insertion 50% AFW	0.034	0.009	0.002
No Rod Insertion 100% AFW	0.338	0.090	0.023
No Rod Insertion 50% AFW	0.034	0.009	0.002

- Note: This assumes the following system/component failure probabilities and unavailabilities, and operator action failure probabilities.

- Rod control system in manual - 0.5 operator action failure to drive in control rods
- No PORVs blocked and none fail to open - 0.75
- One PORV blocked or fails to open - 0.20
- Two PORVs blocked or fail to open - 0.05
- 100% AFW = 0.90
- 50% AFW = 0.09
- <50% AFW = 0.01

Table 6-4 Plant Configuration Probabilities, Plant Configuration Management Scheme 2			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
Rod Insertion 100% AFW	0.848	0.045	0.009
Rod Insertion 50% AFW	0.036	0.002	>0.001
No Rod Insertion 100% AFW	0.045	0.002	>0.001
No Rod Insertion 50% AFW	0.002	>0.001	>0.001

- Note: This assumes the following system/component probabilities and unavailabilities.

- Rod control system in automatic - 0.95 reliability of rod control system
- No PORVs blocked and no PORVs fail to open - 0.94
- One PORV blocked or fails to open - 0.05
- Two PORVs blocked or fail to open - 0.01
- 100% AFW = 0.95
- 50% AFW = 0.04
- <50% AFW = 0.01

7 CONFIGURATION MANAGEMENT PROGRAM

The approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis, specifically Technical Specifications, requires the use of the three-tiered implementation approach. As noted in RG 1.177 (Section 3.1), "Application of the three-tiered approach is in keeping with the fundamental principle that the proposed change is consistent with the defense-in-depth philosophy. Application of the three-tiered approach provides assurance that defense-in-depth will not be significantly impacted by the proposed change." The three-tiered approach includes the following:

Tier 1, PRA Capability and Insights: Assess the impact of the change on CDF, ICCDP, LERF, and ICLERP. This is addressed in detail in Section 5.

Tier 2, Avoidance of Risk-Significant Plant Configurations: Provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when plant specific equipment is out of service consistent with the proposed Technical Specification change.

Tier 3, Risk-Informed Configuration Risk Management: Develop a program that ensures that the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity. This requirement is addressed by the Maintenance Rule.

Although the changes being proposed in this report are not related to Technical Specification requirements and do not impact the licensing basis of the plant, the NRC Staff has indicated on two occasions that they are concerned with how defense-in-depth will be maintained with higher reactivity cores. In the NRC's summary of the WOG/NRC meeting on December 17, 1998 (Reference 18) one major issue is identified as "The staff also noted that there remains a policy question as to what extent MTC would play a role in regulatory space. The staff is not clear as to how the defense-in-depth concept is maintained when MTC is unrestricted." The NRC further stated (Reference 10), "As we understand your proposal, the risk basis will include a configuration risk management program to assure high availability of components to mitigate the severity of ATWS events, such as automatic rod insertion, pressurizer power-operated relief valve (PORV) availability and auxiliary feedwater (AFW) availability. The effectiveness of this program will also be an important element of the staff's review focus. Additionally, in order to provide a sufficient risk informed basis, the staff notes that the WOG submittal should consider the risk impact of an effective configuration risk management program throughout the operating cycle, not solely during the "reference case" UET period (i.e., the UET period assuming all AFW and PORVs are available with rod insertion in manual mode)."

Based on this, the NRC is expecting the issue of potential degradation of defense-in-depth to be addressed by a configuration management program. The discussion on defense-in-depth in Section 6.1 states "... it is seen that sufficient defense-in-depth barriers exist such that it is possible to compensate for limited degradation of one barrier with another and, therefore, maintain plant safety afforded by defense-in-depth requirements. This is an effective approach for managing the risk associated with ATWS events when implementing higher reactivity cores or other plant changes." The following discusses the proposed approach to address this issue.

7.1 PROPOSED APPROACH TO CONFIGURATION MANAGEMENT

The objective of the ATWS configuration management program (CMP) is to operate the plant in a configuration that maintains defense-in-depth to ATWS events, that is, the configuration is favorable to ATWS pressure transient mitigation. But it is acceptable to operate in an unfavorable configuration, prior to implementing compensatory actions, for a limited length of (cumulative) time. The following provides background information for the ATWS CMP.

Tables 4-3, 4-4, 4-7, and 4-8 provide the UETs for the low and high reactivity cores for 100% power and equilibrium xenon. For the low reactivity core there are two configurations, near the start of the cycle, the plant can be operated in which result in a 0 UET. These are for conditions of successful partial rod insertion, both PORVs available, and at least 50% (of total available) AFW flow. For the high reactivity core there is one plant configuration, near the start of the cycle, in which the UET is 0. This is for successful partial rod insertion, both PORVs available, and all AFW available. These are the configurations for which defense-in-depth is not affected early in life. Under other conditions the degree of defense-in-depth, while not necessarily inadequate, may be lessened.

Currently plants can operate with PORVs blocked, with testing and maintenance activities in progress that result in the unavailability of parts of the AFW system (consistent with Tech Spec limitations on AOTs and Maintenance Rule requirements), and with the rod control system in either automatic or manual control. In addition, test and maintenance activities can also take place that result in parts of the reactor protection system being unavailable for short periods of time (again, consistent with the Technical Specifications and Maintenance Rule requirements). These activities can impact defense-in-depth.

By controlling the plant operating configuration, plants can maintain defense-in-depth capabilities. Plants can manipulate the plant configuration to ensure they are operating with favorable conditions with regard to UETs, and therefore ATWS events, by limiting the unavailability of systems important to ATWS event mitigation. Possible precautionary actions during UET periods can include the following:

- Operate with the rod control system in the automatic mode
- Limit blocking pressurizer PORVs
- Limit activities on the AFW system, AMSAC, and RPS that result in the unavailability of components within these systems.

These limitations would vary depending on the time in core life and become less restrictive further into the cycle. Certain routine maintenance activities and other non-regulatory activities on these systems could be moved to later in core life when the reactivity feedbacks are favorable.

Based on the PRA results presented and discussed in Section 5, it is seen that configuration restrictions are not required to compensate for large impacts on plant risk. Rather, configuration restrictions are being proposed to address the NRC's concern for possible degradation of defense-in-depth. As previously noted, the time in life when the plant mitigation systems cannot relieve sufficient RCS pressure is dependent on core design, time in core life, and the availability of rod insertion, pressure relief, and AFW. Table 7-1 presents the UET information from Tables 4-7 and 4-8 for the high reactivity core in the form of acceptable plant configurations for different times during the fuel cycle. In this case, defense-in-depth is the basis for acceptable configurations. This table simply shows the plant configuration required to maintain defense-in-depth, with regard to ATWS, at different times in life. It can be used to schedule acceptable times for removal of equipment from service. From this it is seen that later in cycle life offers more configurations that are acceptable from the defense-in-depth perspective. It should be noted that for the situation presented on Table 7-1, no credit for control rod insertion (72 steps) is given if the rod control system is in manual.

The following is an example of the use of this table. From day 14 through day 65, it is necessary to maintain the rod control system in automatic, restrict AFW maintenance activities, and maintain PORVs in the unblocked condition to maintain defense-in-depth. After day 65, AFW maintenance can be performed and defense-in-depth can still be maintained. Therefore, AFW maintenance activities would be scheduled after day 65, providing the other ATWS mitigation features are available.

When components are out of service that are important to ATWS mitigation, acceptable AOTs, or the equivalent of an AOT for systems not included in the Technical Specifications, can be calculated by use of ICCDP and ICLERP assessments. As previously shown in Sections 5.1.7 and 5.2.2, AOTs greater than 3000 hours can be justified for blocked PORVs. Although this is acceptable from a risk perspective, the NRC indicates this is not acceptable from a defense-in-depth perspective. To address the defense-in-depth issue, the following actions are proposed, where appropriate, when operating in an unfavorable time:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

The first action is directed at maintaining the defense-in-depth capabilities of the RPS and reducing the probability of the occurrences of an ATWS event. The next three actions will prevent further degradation of the configuration and will, in some cases, limit the RCS pressure to a level below which containment releases are a concern. Although these actions may not prevent core damage, they may prevent containment releases. The last action is directed at restoring defense-in-depth via pressure mitigation and then emergency boration.

As an example, consider a plant operating with the high reactivity core, with the rod control system in manual, the AFW system operable, and no PORVs blocked. At day 120 in the cycle, the plant is in a favorable operating configuration with regard to ATWS. If a PORV is now blocked, this becomes an unfavorable condition. Placing the rod control system in automatic, however, changes the plant back to a favorable condition. If the plant cannot be returned to a favorable condition, then voluntary activities that cause the RPS to be unavailable would be curtailed, reducing the probability of an ATWS event.

Several compensatory actions can also be taken when operating in an unfavorable configuration for an extended period of time. These include:

- Implement a back-up reactor trip

- Refine the calculation of UETs
- Reduce plant power to a point where the configuration is favorable

These are discussed in more detail in the following paragraphs.

None of these restrictions or compensatory actions involve changing operation to a plant mode where ATWS events are no longer applicable, such as moving to Mode 3. The risk analysis presented in Section 5 shows that the ATWS risk is small, even when operating in a condition with degraded defense-in-depth. Therefore, a risk argument will not support a plant shutdown. The risk from other potential events during a shutdown and subsequent startup, although small, is not necessarily less than the risk from an ATWS event with degraded defense-in-depth.

In summary, the approach to configuration management is to initially attempt to restore defense-in-depth. If this cannot be accomplished, then activities should be curtailed that cause the RPS and other ATWS mitigative features to be unavailable, and compensatory actions can be taken.

7.2 APPLICABILITY AND KEY CHARACTERISTICS OF THE ATWS CMP

The details of the ATWS CMP, with regard to how this program will be managed, controlled, implemented, and verified, will be specified on a plant specific basis. The following provides high level guidance for licensees to use to develop an ATWS CMP to ensure the plant is operating in a favorable ATWS configuration, consistent with the time in the cycle, and appropriate compensatory actions that can be taken if the cumulative unfavorable configuration time exceeds an acceptable limit.

7.2.1 Applicability of ATWS CMP

Plants will be divided into the following three groups based on consistency with the ATWS Rule and if the plant has a diverse scram system (DSS).

- Group 1: Plants with a DSS

- Group 2: Plants without a DSS, consistent with the ATWS Rule (installed AMSAC) and the basis for the ATWS Rule

- Group 3: Plants without a DSS, consistent with the ATWS Rule (installed AMSAC), but not the basis for the ATWS Rule

A plant consistent with the basis for the ATWS Rule will have either:

- A core design limit on UET of < 5% for the ATWS Rule reference configuration of no control rod insertion, all AFW available, and no PORVs blocked, or

- An MTC of < -8 pcm/°F for 95% of the cycle.

Plants in Groups 1 or 2 will not be required to implement the ATWS CMP. Plants in Group 3 will be required to implement the ATWS CMP.

7.2.2 Key Characteristic of the ATWS CMP

The ATWS CMP key characteristics are divided into three areas; overall CMP structure and administrative control, compensatory actions, and time allowed in an unfavorable condition. Each is discussed in the following paragraphs.

Overall CMP Structure and Administrative Control

By controlling the plant configuration, plants can maintain ATWS defense-in-depth capabilities. Plants can manipulate the plant configuration to ensure they are operating with favorable conditions with regard to UETs, and therefore ATWS events, by limiting the unavailability of systems important to ATWS event mitigation. Limitations on plant configuration vary depending on the time in the cycle and become less restrictive further into the cycle.

Configuration restrictions are proposed to address possible degradation of defense-in-depth. The time in life when the plant mitigation systems cannot relieve sufficient RCS pressure is dependent on core design, time in core life, and the availability of control rod insertion, pressure relief, and AFW. Table 7-1 presents UET information for a high reactivity core in the form of acceptable plant configurations for different times during the fuel cycle. This was developed from the UETs provided on Tables 4-7 and 4-8. In this case, defense-in-depth is the basis for acceptable configurations. This table defines the plant configurations required to maintain defense-in-depth, with regard to ATWS, at different times in the cycle. The information in this table can be used to schedule acceptable times for removal of equipment from service. It should be noted that for the situation presented on Table 7-1, no credit for control rod insertion (72 steps) is given if the rod control system is in manual.

Based on this, an ATWS CMP can be developed that is able to identify plant configurations that are acceptable or unacceptable with regard to maintaining defense-in-depth for ATWS events as plant configurations change with the time in the cycle. This will require licensees to have either cycle specific UETs or conservative UETs. Note that conservative UETs will overestimate the time of unfavorable exposure and may cause a plant to take unnecessary actions.

The ATWS CMP will have the following capabilities:

- Identify plant configurations (unfavorable configurations) that do not maintain defense-in-depth to an ATWS event.
- Track the time for individual occurrences when the plant is in an unfavorable plant configuration.
- Track the cumulative time per cycle when the plant is in an unfavorable plant configuration.
- Provide information on the length of time remaining in the UET for plant configurations.
- Provide compensatory actions to take if the unfavorable condition cannot be exited prior to expiration of the time allowed in the unfavorable configuration.

To maintain the proper level of control over the ATWS CMP and its use, it can be integrated into the Configuration Risk Management Program (CRMP) developed by utilities in response to the Maintenance Rule. The CRMP is typically contained within a plant's Technical Requirements Manual (TRM) or within plant procedures. The combined CRMP/ATWS CMP will be able to track the status of the ATWS mitigation systems and identify when the plant enters unfavorable configurations, and also track the cumulative time in these configurations.

Compensatory Actions

If a plant enters an unfavorable configuration, there are several compensatory actions that can be taken. Any one or a combination of these actions can be used to address the ATWS defense-in-depth concern. The proposed compensatory actions follow. Licensees can use those that provide the appropriate benefit.

1. Back-up Reactor Trip: Implement an alternate method to trip the reactor based on removing power to the CRDMs. This requires an operator action, that can be taken from the control room in a short time, to interrupt power to the motor-generator sets (of the CRDMs) or interrupt power from the motor-generator sets (of the CRDMs) to the CRDMs. This would provide a backup reactor trip signal that is diverse from the RPS. The only common components are the sensors and isolators that provide input to the RPS and control board indication. This operator action will need to be listed early in the plant's emergency operating procedures and the operators will need to be trained on the action. It is recommended that this action be placed in E-O of the plant's Emergency Operating Procedures ("Reactor Trip or Safety Injection").

   One of the issues with implementing this compensatory action is related to the coastdown time for the MG sets. That is, how long will it take the MG sets to coast down to a speed at which the voltage will degrade to a level that the CRDMs will release the control rods. If the RCS pressure increases to 3200 psi in approximately 90 seconds (see Section 4.3), the coastdown time needs to be relatively short to ensure the operators have sufficient time to diagnose the event and interrupt power to the MG sets. To meet this requirement, a coastdown time of approximately 30 seconds would be appropriate. There is no readily available information on the coastdown time. Running a test is possible, but this may damage the MG set. Initial analyses to determine a coastdown time concluded that the coast down will exceed the 30 second requirement. Therefore, relying on a short MG set coastdown time is not feasible.

   One possible alternative to relying on a short MG set coastdown time is to install equipment that will provide an undervoltage trip of the MG set breakers (the breakers on the output of the MG sets) based on an undervoltage signal from the buses that provide power to the MG sets. Some Westinghouse plants already have this capability. With this arrangement, interrupting power to the buses the MG sets are powered from will result in a quick trip of the reactor. Alternative methods to ensure this trip occurs within the required timeframe are also acceptable.

   As an alternative to a back-up reactor trip from the control room, a utility may consider locating a dedicated operator at the MG sets, to performed the MG set trip, if an unfavorable configuration exists beyond an acceptable time period. This would be beneficial for short durations, but may not be feasible for an extended time frame.

Once this compensatory action is implemented, further tracking of the time operating in an unfavorable configuration is not necessary.

2. UET Re-calculation: Re-calculation of UETs can be done based on plant specific information using analysis enhancements which may provide a better (shorter) estimate of the UETs. For example, if a plant is using a generic set of UETs for a representative plant that it is similar, but not identical, in design it may be possible to complete plant specific analysis that will provide shorter UETs. In addition, depending on the end-of-cycle burn-up assumptions for the previous cycle, using the actual end-of-cycle burn-up may also provide a benefit.

3. Power Reduction: The plant power level can be reduced to a level where the plant configuration becomes favorable. At the lower power level, RCS pressures following an ATWS event can be mitigated with reduced pressure relief capability. The plant can then operate at this reduced power level until the configuration becomes favorable as the time into the cycle increases.

   Reducing power will always reduce the UET, but will not eliminate it for all configurations. As stated above, in some cases reducing power is a viable mitigative strategy. However, it would require additional analyses to determine the power reduction needed to eliminate the UET for core and plant configuration of interest and possibly for additional configurations, depending on the length of time it would be necessary to remain at the reduce power level and the potential activities that could occur during this time period. These additional analyses would be similar to those completed for the full power cases. Iterative analyses would have to be completed to determine the power reduction required to eliminate the UET for the particular plant configuration. The analysis would include calculation of new CPTs at the reduced power level, followed by calculation of the UET for the plant and cycle specific core conditions following the same approach used to calculate UETs at full power.

Time Allowed in an Unfavorable Configuration

A 30-day cumulative time limit in an unfavorable condition is proposed. In some cases this length of time will provide sufficient time for the plant to exit the unfavorable configuration as the cycle progresses. This time can also be used to implement appropriate compensatory actions. The 30-day limit is based on the following:

- For a 500-day fuel cycle, a 5% UET would equate to 25 days. This is based on the 5% UET requirement placed on the Braidwood and Byron core designs.

- The industry, along with the NRC, is currently developing a risk-informed Technical Specification directed at flexible AOTs with a 30-day backstop. In this activity, a maximum time of 30 days would be allowed to return Technical Specification equipment to operable status if the risk analysis supports it.

- The risk associated with blocking a PORV is low. Following the approach in Regulatory Guide 1.177 for setting Technical Specification AOTs, a time of over 3000 hours can be justified via the risk analysis for blocking a PORV (see Section 5.1.7 and 5.2.2).

Note that this 30 day time period is cumulative.

7.2.3 Core Design Considerations

As previously noted, the approach to maintain defense-in-depth, or ATWS pressure transient mitigation capability, is to operate the plant in a configuration with a zero UET. Using this approach, the reload analysis will need to ensure that at least one plant configuration, with regard to CRI, AFW, and PORV availability, will have a zero UET. A zero UET should be demonstrated for the following conditions:

a. Hot full power moderator temperature coefficient
b. Equilibrium xenon
c. Nominal hot full power inlet temperature
d. 72 steps of control rod insertion of the lead bank
e. All PORVs operable
f. 100 percent (all) AFW flow available

The methodology licensees use to calculate UETs is discussed in Sections 4.1 and 4.2, and supplemented with information in response to Issue 6 in Appendix A and Technical Clarification 5 in Appendix G.

7.2.4 Additional ATWS CMP Requirements

The following lists a number of considerations for the use of the ATWS CMP during plant operation. Some of these are further discussed in response to Issue 1 in Appendix G.

- Effective full power days should be the basis for the ATWS CMP.

- Entries into unfavorable configurations to meet Technical Specification surveillance requirements and repair inoperable equipment are acceptable. Entries into unfavorable configurations to complete preventive or routine maintenance activities should be minimized. Some of the equipment important to mitigation of an ATWS pressure transient is also important to mitigation of design basis events. These design basis events typically are larger contributors to plant risk than the ATWS event, therefore, it is important to maintain the equipment operability for design basis event mitigation. The surveillance requirements demonstrate component operability, therefore, it is recommended that they continue to be completed at the specified interval. Similarly, inoperable components should be repaired to maintain design basis event mitigation capability.

- If component inoperability, due to surveillance requirements, maintenance activities, or repair activities, moves the plant into an unfavorable configuration, then simultaneous test and maintenance activities that compromise the availability of the reactor protection system (reactor trip signals, in particular) or that place the plant in a higher trip potential configuration should be rescheduled for when the plant returns to a favorable configuration.

- The time a plant may enter an unfavorable configuration to meet a surveillance requirement is relatively small, as demonstrated in the response to Issue 1.e in Appendix G. Therefore, it is proposed that this time does not need to be tracked against the total time allowed in unfavorable configurations, if it does place the plant in an unfavorable configuration.

- Startup testing is necessary to ensure equipment is operable to meet design basis event mitigation assumptions. Startup testing or the startup process should not be modified based on ATWS concerns. This is discussed further in the responses to Issue 1.f in Appendix G.

- Controls (procedures and actions) to ensure consistent implementation of the ATWS CMP following the high level guidance provided in Section 7.2.2 will be developed on a plant specific basis. These controls will be included in the appropriate plant document(s).

- There is no need to develop additional limiting conditions of operation, action statements, or surveillance requirements for inclusion in Technical Specifications to implement the ATWS CMP.

- The ATWS CMP needs to consider preventive maintenance and repair activities that cause systems important to ATWS mitigation to be unavailable. As discussed above, system unavailability due to surveillance tests does not need to be considered.

- Appropriate training should be provided to operators to ensure the ATWS CMP and associated compensatory actions are implemented correctly. This will include training in the following areas:

  a. Unfavorable exposure times as related to acceptable plant configurations and time in the cycle

  b. Equipment important to mitigating ATWS events

  c. Equipment important to preventing ATWS events

  d. Tracking UETs with effective full power days of operation

  e. Time allowed in unfavorable plant configurations and compensatory actions

  The detailed description of the training, tools, and procedures that will be provided to operators will be developed on a plant specific basis.

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

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS						
Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual		2	1	0
<14	X		Yes			X
14 to 65	X		No			X
>65	X		Yes			X
>110	X		Yes			X
		X	No			X
>115	X		Yes			X
	X		No		X	
		X	No			X
>134	X		Yes			X
	X		No		X	
		X	Yes			X
>136	X		Yes		X	
		X	Yes			X
>165	X		Yes		X	
	X		No	X		
		X	Yes			X
>170	X		Yes		X	
	X		No	X		
		X	Yes			X
		X	No		X	

Table 7-1 (cont.) Configuration Management Approach for the High Reactivity Core

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS						
Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual		2	1	0
>187	X		Yes	X		
		X	Yes			X
		X	No		X	
>192	X		Yes	X		
		X	Yes		X	
>238	X		Yes	X		
		X	Yes		X	
		X	No	X		
>259	X	X	Yes	X		

Note:

1. It is assumed that AFW availability will be controlled by Technical Specification for the AFW system, that is, only one pump is allowed to be out of service at a time. A shutdown is required for two pumps out of service.

8 WOG ATWS APPROACH AND MODEL

The following presents and discusses: 1) the recommended approach to address ATWS issues rising from higher reactivity cores, and 2) the recommended ATWS model for use in plant specific PRA models. The approach is consistent with RG 1.174 and addresses evaluating the impact on risk, in addition to the impact on defense-in-depth and safety margins.

The ATWS model discussed in the following is based on the model presented in Section 5 and is consistent with the model presented in WCAP-11992. If implemented as presented, it will provide a realistic assessment of ATWS risk, in terms of CDF, and can be used to assess the impact on CDF of PMTC, plant power upgrades, and SG issues. This model can also be used to assess the impact on ATWS risk related to the availability of pressurizer safety valves and PORVs, in addition to the reliability of the RPS and AMSAC.

ATWS events can be divided into five states as discussed in Section 5.1. These states are defined based on power level, which impacts the availability of AMSAC, and xenon concentration, which acts as a poison. Equilibrium xenon concentrations are achieved after approximately 50 hours of operation. During plant startups, that follow shutdowns of sufficient length to allow xenon depletion, the xenon cannot be credited in determining UETs. The five ATWS states are:

- State 1: Startup (no equilibrium xenon), Power level <40% (no AMSAC)
- State 2: Startup (no equilibrium xenon), Power level ≥40% (AMSAC)
- State 3: Power Operation (equilibrium xenon), Power level ~100% (AMSAC)
- State 4: Shutdown (equilibrium xenon), Power level ≥40% (AMSAC)
- State 5: Shutdown (equilibrium xenon), Power level <40% (no AMSAC)

All states have unique characteristics and are evaluated with distinct PRA models except for States 3 and 4. These states are very similar, with State 4 being bounded by State 3. The only difference is that State 4 trips would start from lower power levels which would result in lower RCS pressures.

As concluded and discussed in Section 5.1.5, most of these states do not contribute significantly to ATWS CDF. Power operation, including shutdown and power level ≥40% (State 3/4), is the largest contributor. In Table 5-25 the contribution of State 3/4 contributes at least 88% to CDF for the three core types. The other three states are relatively small contributors to ATWS CDF. Based on this, the WOG model only addresses State 3/4.

8.1 ACCIDENT PROGRESSION

The progression of the ATWS event for State 3/4 as described in the following is taken, in part, from WCAP-11992.

An ATWS event is composed of two different events; the first is an anticipated transient generating a reactor trip signal and the second is the failure to insert control assemblies into the core following the trip requirement. Two categories of ATWS events can be defined based MFW availability; events in which MFW continues to run and events in which MFW is unavailable. For ATWS events with MFW available, a less severe power mismatch between heat source and sink results and the RCS peak pressure is less severe. With the loss of MFW, a large mismatch between the heat source and sink occurs which in turn results in the RCS heatup. This heatup causes rising RCS temperature and pressure.

The water level in the SGs will drop as the remaining water in the secondary system, unreplenished by MFW flow, is boiled off. When the SG water level falls to the point where the SG tubes are exposed, the primary-to-secondary system heat transfer is reduced. The reactor coolant temperature and pressure continue to increase to the point where the PORVs and safety valves open. The peak pressure attained is dependent on the capability of the PORVs and safety valves to release the reactor coolant volumetric insurge into the pressurizer.

Depending on reactivity feedback conditions, these changes in the reactor coolant conditions cause the core power to be reduced. If the reactor control system is in the automatic mode, the control rods would begin to be inserted as the reactor coolant heatup begins, reducing power and mitigating the RCS overpressure.

There are several mechanisms by which a plant may be shut down following failure of the RPS to provide a trip signal. Plant procedures instruct operators to initiate a manual trip. This is done from the control board and requires the reactor trip breakers to open. If this fails, the operator can also trip the reactor by interrupting power from the motor-generator sets to the CRDMs. Due to the short period of time available for operator response, this can only be credited if the action can be done from the control room. The operator will also be instructed to manually insert the control rods. Then the operator is instructed to verify or manually trip the turbine, verify AFW started, and initiate emergency boration. Emergency boration will only be successful when the RCS pressure drops below the pressure limits of the charging pumps.

8.2 ATWS EVENT TREE MODEL

ATWS events can be initiated from a wide range of initiating events. The ATWS analysis for Westinghouse PWRs (Reference 6) established that the limiting events, with regard to RCS peak pressure, are the loss of load with subsequent loss of all MFW and complete loss of normal feedwater. These limiting events are both assumed to be initiated from normal operation at full power. If favorable reactivity conditions exist, the reactor core is expected to shut down prior to core damage following any anticipated transient without reactor trip provided the turbine trips and AFW flow is initiated in a timely manner. If unfavorable reactivity feedback conditions exist, there is the possibility that the allowable RCS component stress limits could be exceeded with possible loss of RCS integrity and core damage. The allowable component stress limits are based on the ASME Service Level C limit of 3200 psi.

As with previous ATWS assessments, core damage is conservatively assumed if any one of the following occur:

- Maximum RCS pressure exceeds the pressure limit corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion. This is defined as 3200 psi.
- RCS heat removal function is inadequate (either before or after the core is brought subcritical).
- The operator fails to initiate emergency boration in a timely manner.

As discussed in Section 5, 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 Service Level C limit of 3200 psi. 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 or manual actions, or mechanical failure of the rods or CRDMs. The UETs for a given core are dependent on the availability of AFW to the steam generators for heat removal, partial insertion of the control rods (if rod insertion for reactor trip fails), availability of RCS pressure relief, and negative reactivity feedback.

Figure 8-1 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 automatic trip signal), OAMG (operator action to trip the reactor from 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 ESFAS and AMSAC signal 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% AFW flow, and less than 100% but at least 50% AFW flow. This, along with the availability of pressurizer safety valves and PORVs, is important in mitigating the overpressure event. PR (pressure relief) accounts for the unavailability or failure of safety valves and PORVs. The UETs are also factored into this 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 (mechanical failure) given a reactor trip signal was generated.

- 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.

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

8.2.1 IEV: Initiating Event Frequency

This is the frequency of transient events that can lead to ATWS events. This includes all anticipated 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 equilibrium xenon requirement eliminates events that occur during plant startups. The first year of operation can be eliminated since this is usually not typical of plant operation in the following years.

The following guidelines can be used to determine an initiating event frequency:

- Since plants operate at 100% power, or close to it, trips in the 95% to 100% power range are at-power trips.
- Trips in the 0% to 95% range occur either during startup or shutdown since plants typically operate at or near 100% power.
- Startup trips occur prior to establishing equilibrium xenon and shutdown trips occur after equilibrium xenon has been established.
- The split between startup and shutdown trips can be determined from the probabilities of a trip during startup (0.088) and during shutdown (0.068). These values are discussed in Section 5.1.1.2 and are from WCAP-14333 (Reference 13).
- WCAP-15210 (Reference 11) is a source for trip events at Westinghouse plants.

The model presented in this section assumes MFW is lost for all anticipated transient events. If MFW continues to operate, then the event does not need to address the pressure relief response, including AFW and AMSAC, but only requires long-term shutdown. A split that accounts for MFW continuing to operate may be added to plant specific ATWS models if desired.

8.2.2 RT: Reactor Trip Signal from the RPS

This top event models the failure of the RPS to provide a reactor trip signal when required. Since the RPS provides the trip signal, the control rods still need to drop into the core. If this event is successful, then the CR event is addressed. If this fails, then alternate means to trip the reactor are addressed.

The RPS fault tree model should include the RTBs, either solid state logic cabinets or relay logic cabinets, and signal processing (analog or digital). The fault tree model for failure of a reactor trip signal should credit 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. In addition, an operator action should be 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 logic cabinets, but not failures involving the RTBs. Consideration also should be given to analog channel testing in the tripped or bypassed conditions.

The RPS fault tree models provided in NUREG/CR-5500, Vol. 2 (Reference 14) are acceptable. The RPS fault tree models provided in WCAP-15376 (Reference 19) or WCAP-14333 (Reference 13) can also be used. References 14 and 19 are also data sources for failures of components in the RPS.

The human error probability for the OA to trip the reactor from the control room is plant specific. It is the first action in a series of several OAs that can be taken to prevent or mitigate the ATWS event. Given that it is the first, there are no dependencies on previous actions that need to be considered.

8.2.3 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 core. To take credit for this action, it is necessary for it to be called out in the plant emergency operating procedures and it must be possible to complete the action from inside the control room. Due to the short timeframe available to respond to an ATWS event, actions outside the control room are not feasible. 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 or 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 other operator actions already taken.

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

This event models driving the control rods into the core via the rod control system. The rod control system may be under automatic or manual control. This is a plant specific decision. If the rod control system is in manual, the operator can take the action to manually drive the control rods into the core. 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 of insertion (negative reactivity) from the lead bank. Some core designs will require CRI success during parts of the cycle to achieve successful pressure mitigation. In addition, it is possible to design a core that for part of the cycle the ATWS pressure transient cannot be mitigated regardless of the success or failure of CRI. But regardless of whether CRI succeeds or fails, auxiliary feedwater and pressure relief availability still need to be addressed. The UETs are impacted by success or failure of this action.

The value used for this event will depend if the rod control system is in automatic or manual. In manual, the value will depend on success or failure of previous OAs.

- Rod control system in automatic: A conservative failure probability value of 0.1 can be used directly. The clarification provided to Item 1 in Appendix I provides a justification for a less conservative failure probability given that the rod control system is in automatic.

- Rod control system in manual (no credit taken for OAMG): For plants that do not credit OAMG, this OA will follow the OA to trip the reactor from the control room via the reactor trip switch. If the reactor trip signal failed (RT top event) 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 the OA to drive the rods in can be considered not dependent or conditional on the failure of the previous operator action.

- Rod control system in manual (credit taken for OAMG): For plants that credit OAMG, this OA will follow failure of at least one previous OA (failure to trip from the control room via the reactor trip switch) and possibly failure of two OAs (failure to trip from the control room via the reactor trip switch and OAMG). In the case of one failed OA, some credit can be taken for this OA, but it is conditional on the failure of a previous OA. In the case when two OAs have failed, very little to no credit should be taken for this OA.

Regardless, the value used will need to be plant specific.

8.2.5 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.

- 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 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 the core design and control rod failure/success locations.

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. The NUREG assumption of 10 or more rods failing to insert may be acceptable for shutting down the core, but significant negative reactivity is provided by those that do insert. That is, the pressure transient will be significantly mitigated. 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. D-bank insertion of 72 steps is significantly less than the number of control rods required to insert per the assumptions of the NUREG report.

The following guidelines are recommended:

- 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.
- Using this definition for failure of CR (10 or more rods failing to insert), it is assumed that the reactor will be critical, but at a lower power level, and long-term shutdown (boration) will be required.

Therefore, the following approach is recommended:

- A sufficient number of rods will always insert so that the pressure transient will be mitigated or severity 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.

Note that it is not necessary to explicitly address CR following success of CRI. It is understood that the control rods still need to move into the core, but 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).

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

A key 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 conditions when ESFAS signals are not available, assuming a common cause event inhibits all RPS signals, are if reactor trip fails due to RTB, logic cabinet, or analog channel failures.

ESFAS signals to start AFW and trip the turbine should only be credited following failure to trip due to failure of the CR top event (rods fail to insert) following successful RT. A detailed fault tree assessment of the ESFAS can be done to develop a failure probability or a conservative value of 0.01 can be used for failure of the signal. The 0.01 value is considered conservative since it 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 vary from 3E-03 to 7E-04 depending on the specific signal being considered.

8.2.7 AMSAC: ATWS Mitigation System Actuation Circuitry

AMSAC is a diverse method (diverse from the RPS signals) to trip the turbine and start AFW. No detail fault tree analysis of AMSAC has been done, but WCAP-11992 uses a conservative value of 0.01/demand as a failure probability. This value has also been used in other studies and is an appropriate value. A fault tree analysis would probably be required to justify a lower value.

8.2.8 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. The 50% level actually represents AFW flow that is less than 100% and greater than or equal to 50%. AF100 represents 100% AFW flow, which is flow from all the AFW pumps, to all steam generators. For an AFW system design with 1 TD AFW pump and 2 MD AFW pumps, in which one MD pump provides half the flow of the TD pump, 100% flow is flow from the TD pump and both MD pumps.

8.2.9 AF50: AFW System Provides 50% Flow

AF50 represents less than 100% AFW flow but at least 50% AFW flow to all steam generators. For a AFW system design with 1 TD AFW pump and 2 MD AFW pumps, in which one MD pump provides half the flow of the TD pump, 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 at least 50% flow failure given 100% flow has failed.

8.2.10 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 (PORVs blocked or fail to open), 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 are more frequent early in the cycle also.

Four sets of UETs are required that correspond to the various combinations of CRI and AFW. Four fault trees are required for PR, one for each set of UETs. Examples of the PR fault trees are provided in Appendix D. There is one fault tree for each AFW/rod insertion combination:

- Fault tree PRA: control rod insertion success, 100% AFW
- Fault tree PRB: control rod insertion success, 50% AFW
- Fault tree PRC: control rod insertion failure, 100% AFW
- Fault tree PRD: control rod insertion failure, 50% AFW

Successful pressure relief requires opening all three safety valves and the required PORVs when the reactivity feedbacks are favorable. 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

Plant specific UETs should be used when possible. A conservative set can also be used if available. Four UET sets will be needed with each set corresponding to a CRI/AFW combination. Within each set, UETs are required for 0, 1, or 2 PORVs available. For plants with 3 PORVs, UETs can be used for 0, 1, 2, or 3 PORVs available, providing the PR fault trees are modified to reflect the availability of 3 PORVs.

The UETs will need to be weighted according to the distribution of transient events over the cycle. This is required since the transient frequency is higher in the beginning of the cycle when unfavorable exposure times occur. The distribution in Table 5-3 can be used for this weighting. Weighting calculations are shown in Section 5.1.1.11. From the weighted UETs, the UET intervals that correspond to 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) are calculated. Sample calculations for this are also provided in Section 5.1.1.11.

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 to unblock the PORV. Plant specific values need to be developed for the probability that PORVs are blocked. Probabilities of blocked PORVs can be assumed to be randomly distributed throughout the fuel cycle unless other information is available that disputes this assumption. For plants with two PORVs, probabilities of blocked PORVs should be developed for each PORV and also for two PORVs. For plants with three PORVs, probabilities for blocked PORVs should be developed for each PORV, combinations of two PORVs, and for three PORVs. This is assuming that for plants with three PORVs, UETs will be used that correspond to the availability of one, two, and three PORVs.

Plant specific failure probabilities, if available, should be used for safety valve and PORV failure to open on demand and for common cause failure of multiple PORVs.

8.2.11 LTS: Long Term Shutdown

This event requires the plant operators to establish long-term shutdown which involves starting emergency boration. This is required on success paths that do not have full control rod insertion. 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. A plant specific value or fault tree should be used for boration. It can be 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.

8.2.12 Event Tree Sequence Endstates

The core damage endstates can be differentiated from each other according to RCS pressure, if required. Distinctions between high RCS pressure and low RCS pressure endstates are based on whether or not pressure relief was successful. Successful pressure relief maintains the RCS pressure below 3200 psi. The following defines the sequence endstates:

Low RCS pressure: Failure of LTS. LTS is only addressed if pressure relief is successful, so any sequence with LTS failure is a RCS low pressure condition.

High RCS pressure: All other core damage endstates are high pressure RCS conditions since they involve failure of pressure relief PRA, PRB, PRC, or PRD. Failure of AFW50 (less than 50% AFW flow) and failure of AMSAC are also equated to high RCS pressure endstates since insufficient AFW flow is available to mitigate the ATWS event with regard to pressure relief.

8.3 PRA MODEL QUANTIFICATIONS AND APPLICATION OF REGULATORY GUIDE 1.174

To demonstrate the acceptability of higher reactivity core designs, the requirements in Regulatory Guide 1.174 need to be met. This includes addressing the impact on risk, as well as the impact on defense-in-depth and safety margins, of higher reactivity cores. For the plant specific evaluation, only the impact on CDF is required to be evaluated for power operation. Power operation is considered to be the time that the plant is operating above 40% with full power equilibrium xenon. As shown previously, the other ATWS states do not contribute significantly to risk and do not need to be included in the evaluation. LERF also does not need to be addressed as long as the plant and core of interest are bounded by the bounding core evaluated in Section 5.2.

To evaluate the impact on CDF, the PRA model for the plant of interest should be quantified for the current core, that meets current requirements, and for the new higher reactivity core. The plant PRA should use plant specific models, parameters, and values, or values that are set to conservatively represent the plant. The change in CDF should be a small impact as defined in RG 1.174. Sensitivity quantifications can be completed on parameters that the results may be sensitive to, such as, the probability of blocked PORVs, credit for the rod control system being in automatic, and credit for operator actions. ICCDP values can also be determined for blocked PORVs. This is to provide an indication of the length of time PORVs can be blocked and meet the ICCDP guideline in RG 1.177 (5E-07).

The final step in the WOG approach is to address the impact on defense-in-depth and safety margins. The impact on safety margins is straight-forward and is addressed in Section 6.2. The impact on defense-in-depth should follow the approach in Section 6.1 and develop a configuration management program, which in this case is equivalent to defining Tier 2 restrictions (since these are predefined restrictions) similar to those proposed in Section 7.

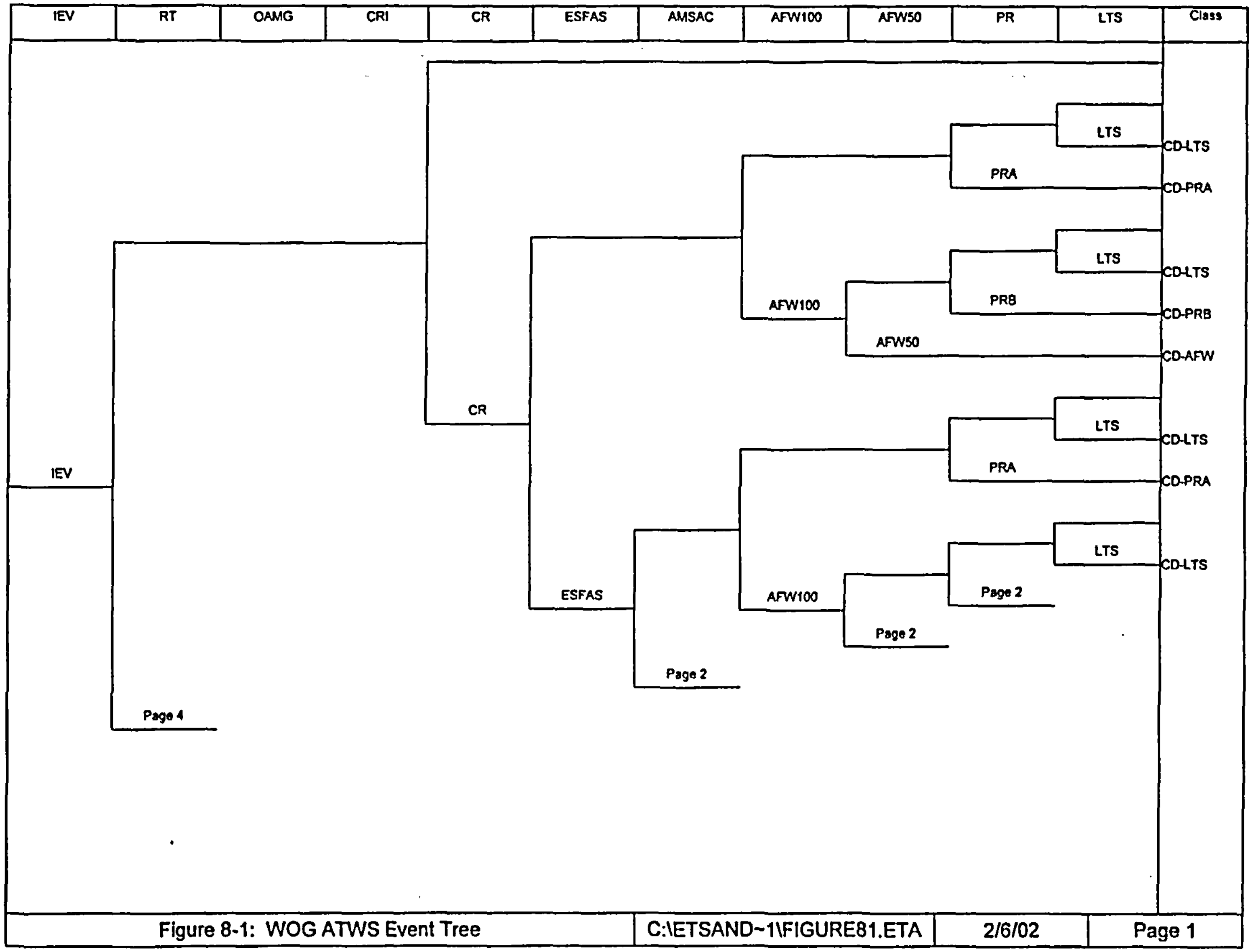

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40%

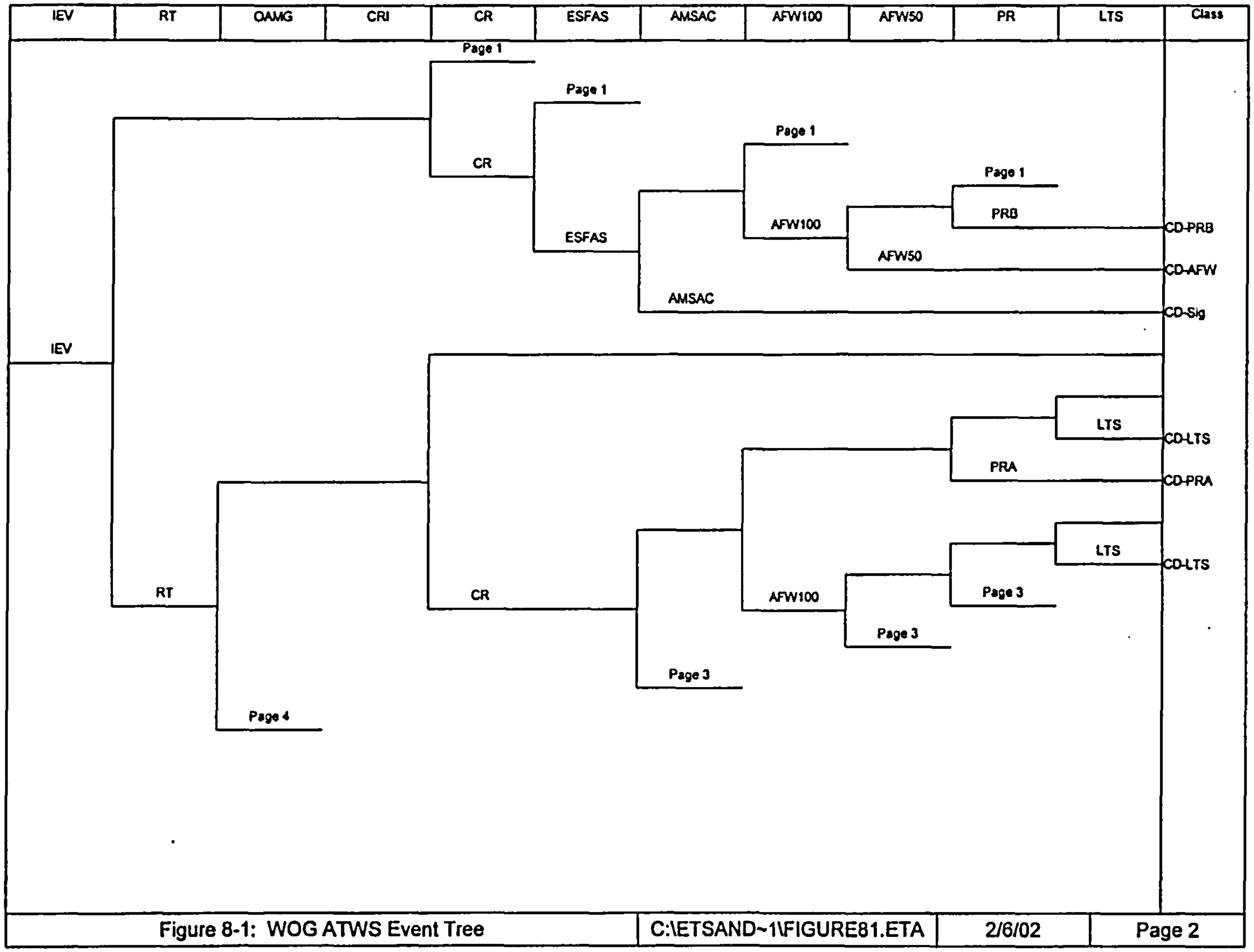

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

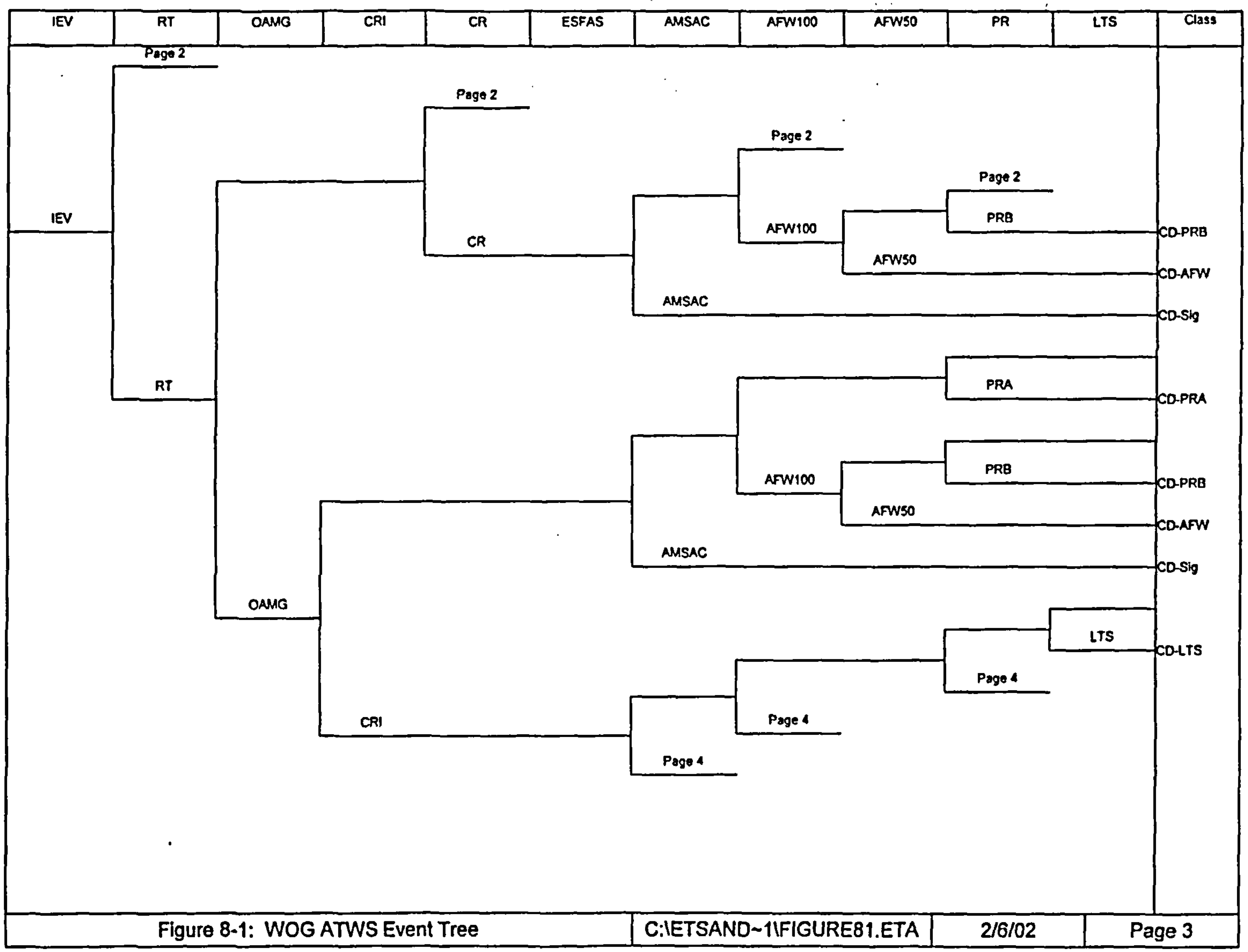

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

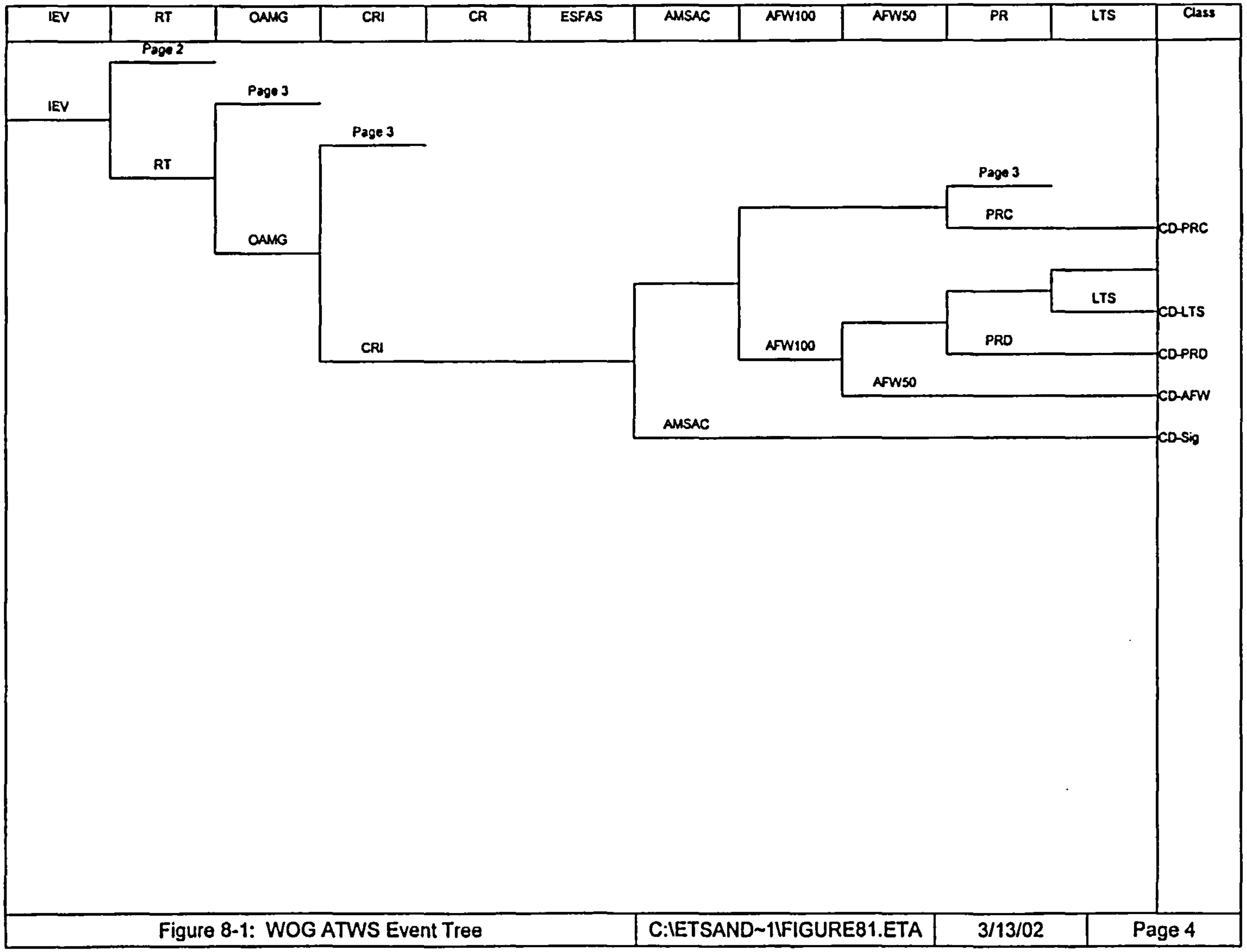

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

9 BRAIDWOOD LEAD PLANT EVALUATION

As discussed in Section 2.3, Byron and Braidwood referenced WCAP-11992 in their PMTC license amendment request. As part of the NRC's review and acceptance of this request, an additional requirement was added to the Byron and Braidwood Technical Specifications that requires core designs to meet a 5% UET for the ATWS Rule reference configuration of no rod insertion, 100% AFW, and no PORVs blocked, referred to as the reference conditions. One objective of the lead plant application is to remove this Technical Specification requirement using the risk-informed approach described in this WCAP.

In this evaluation, the Braidwood PRA model was modified to reflect the WOG model described in Section 8. UETs were developed for the current core design with the 5% UET restriction for the reference conditions and for a core design based on similar requirements, but without the 5% UET restriction. The PRA model was then quantified for both core designs to determine the impact on CDF. ICCDP values were calculated for blocked PORVs. A set of sensitivity evaluations was also completed. A configuration risk management program is provided to address defense-in-depth issues.

9.1 BRAIDWOOD ATWS PRA MODEL

The Braidwood PRA model uses the fault tree linking approach and the CAFTA code system for quantification. The model, as used in this evaluation, includes internal events. The ATWS model in the Braidwood PRA was reviewed for consistency with the WOG model and modified as appropriate. The following discusses each top event in the Braidwood model and how each conforms to the WOG model in Section 8. Table 9-4 provides a summary of the comparison of the WOG and Braidwood models and provides an assessment of the impact of modeling differences on the results. Figure 9-1 shows the Braidwood ATWS event tree.

ATWS Initiators

This top event represents anticipated transient events that have already proceeded to ATWS events. It includes the initiating event frequency for anticipated transients (IEV in the WOG model), failure of the reactor trip signals (RT in the WOG model), and failure to trip the reactor by interrupting power to the MG sets (OAMG in the WOG model).

The Braidwood PRA model uses IE frequencies based on industry and plant specific operating history. These include all events that occur above 40% power. The total IE frequency of anticipated transient events that can result in an ATWS event is approximately 1. This includes LOSP events and events without HFP equilibrium xenon above 40% power. LOSP events are not events that result in increased RCS pressures, but the LOSP IE frequency is small so including this contribution to the total IE frequency has essentially no impact on the results. Including the portion of events that occur above 40% power without equilibrium xenon will increase the IE frequency by a small amount and provide a slightly higher IE frequency.

The reactor trip signal model is based on the model provided in NUREG/CR-5500, Volume 2 (Reference 14) including the component failure probabilities. The Braidwood Station has a solid state protection system as modeled in the NUREG. The trip signal model includes an operator action to trip the reactor from the trip switch in the control room. The HEP for this action is 1.0E-02.

The operator action to trip the reactor by interrupting power to the CRDMs from the MG sets is included in the model. The value used is dependent on previous OA failures. If the previous failures include the OA in the control room from the trip switch, a value of 0.5 is used. If the failures do not include this previous OA, i.e., reactor trip breakers have failed, then a value of 1E-02 is used.

Main Feedwater (ATWS)

Main feedwater is addressed in the Braidwood PRA model, but not the WOG model. If MFW continues to run, then high RCS pressures are not a concern and only long-term shutdown is addressed. The probability that the MFW will not continue to run is 0.23.

Manual Rod Insertion (ATWS)

Manual rod insertion is the operator action to drive the rods into the core or it can represent the probability of the rod control system being in automatic. The value used, if an operator action is assumed, is dependent on the previous failures. If this action follows failure of the OA to trip the reactor via the trip switch in the control room and also failure of the OA to trip the reactor via the MG sets, then the failure probability value used is 1.0. If this follows only the failure of the OA to trip the reactor via MG sets, then a failure probability of 0.5 is used. Several cases use a value of 0.1, which represents unavailability of the automatic rod control system.

Control Rod Failure (mech. binding)

The value used is 1.21E-06/demand and is based on Reference 14.

AMSAC

AMSAC is included in the model to trip the turbine and start AFW. The value used is 1.0E-02.

Auxiliary Feedwater System

The availability of AFW is addressed by a three-way split in the event tree. The bottom path represents less that 50% flow, the middle path represents greater than or equal to 50% flow and less than 100% flow, and the top path represents 100% flow. The Braidwood AFW system consists of two pumps, one motor-driven and one diesel-driven. With this design, 100% flow requires both pumps to operate and 50% requires either pump to operate. Detailed fault trees of the AFW system are used to model this event.

Primary Pressure Relief

The primary pressure relief trees are identical to those used in the WOG model, but a number of inputs are different. These include UETs, probability of blocked PORVs, and failure rates for PORVs and safety values.

The UETs for the current core which meets the 5% restriction and for the higher reactivity core are provided in Tables 4-34 to 4-37. The weighted UETs were developed as discussed in the WOG model using the weighting factors from Table 5-3. The weighted UETs are provided in Tables 9-1 and 9-2. The pressure relief intervals are developed from the weighted UETs following the WOG approach. These are provided in Table 9-3.

The probability of PORVs being blocked are based on plant experience. The values used are:

- Both PORVs blocked = 0.0025
- PORV A blocked = 0.05
- PORV B blocked = 0.05
- No PORVs blocked = 0.8975

PORV failure is modeled by a detailed fault tree and the safety valve failure is modeled as a single basic event. The failure probability for a safety valve failure to open demand is 1.0E-03.

Shutdown of the Reactor

This is equivalent to the LTS (long-term shutdown) in the WOG model. A detailed fault tree for this event is used in the Braidwood model.

Miscellaneous

Note that the only top event not included in the Braidwood model is ESFAS. This models actuation of the AFW and turbine trip by the ESFAS. This is can only be credited if the ATWS event is due to failure of the control rods to insert due to mechanical binding, i.e., the reactor trip signal was present. This represents a small conservatism in the model in comparison to the WOG model.

A summary of the WOG and Braidwood ATWS models is provided on Table 9-4. It is concluded from this review that the Braidwood model follows the WOG model appropriately.

9.2 BRAIDWOOD ATWS CORE DAMAGE FREQUENCY QUANTIFICATIONS

A number of evaluations were performed. The first was for the current core design with the 5% UET restriction (Case B1) and the second was for a future core design without the 5% UET restriction (Case B2). Both assumed that the rod control system is in automatic and the standard probability for blocked PORVs. The only difference between these two cases is the UETs.

The following sensitivity cases were done for the future core design:

- Case B3: Worst Time in Cycle, Standard PORV Blocked Assumptions, Rod Control System in Automatic

- Case B4: End of Cycle, Standard PORV Blocked Assumptions, Rod Control System in Automatic

- Case B5: Yearly CDF, No PORVs Blocked, Rod Control System in Automatic
- Case B6: Yearly CDF, One PORVs Blocked, Rod Control System in Automatic
- Case B7: Yearly CDF, Two PORVs Blocked, Rod Control System in Automatic
- Case B8: Yearly CDF, Standard PORV Blocked Assumptions, Rod Control System in Manual

The results for these cases are provided on Tables 9-5 to 9-7. The following discusses the results:

Table 9-5: By comparison of Cases B1 and B2 it is seen that the impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) which meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr. A comparison of Case B8 to Case B2 shows the benefit of operating the plant with the rod control system in automatic. The CDF value decreases a relatively small amount. As discussed in Section 5.1.6, this impact is relatively small when examining the impact across the core life. Placing the rod control system in automatic increases the probability of successful partial rod insertion. Partial rod insertion is not important later in life since it is not necessary to mitigate the RCS pressure transient. Earlier in life the impact is more important since partial rod insertion has more of an impact on the pressure transient.

Table 9-6: 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 future core. The end of the cycle value is also applicable to the current core since both cores are favorable in all configurations at the end of the cycle. As seen in this table, the ATWS CDF, which is small to start, decreases significantly through the cycle.

Table 9-7: This table shows the impact of blocking PORVs on CDF. CDF values are provided for 0, 1, and 2 PORVs blocked for the complete cycle with the rod control system in automatic. The safety valves are still available for pressure relief in these cases. Also shown is the CDF for the standard blocked PORV probabilities. This demonstrates that even with both PORVs blocked, the ATWS CDF remains low.

9.3 INCREMENTAL CONDITIONAL CORE DAMAGE PROBABILITY

The ICCDP calculations are discussed in Section 5.1.7 for the generic analysis. As discussed in that section, the ICCDP is used to determine acceptable time periods equipment can be out of service, for example, how long can a PORV be blocked. The ICCDP calculation is generally used to assess changes to the completion times (AOTs) specified in plant Technical Specifications. As shown in Section 5.1.7, an acceptable AOT can be determined based on the acceptance guideline of ICCDP $\leq$ 5E-07 as provided in Regulatory Guide 1.177.

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

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

Given this, the acceptable AOT, based on the yearly average CDF, to have a PORV blocked for the future core follows.

$$AOT = (5E\text{-}07 \times 8760)/(1.67E\text{-}07 - 6.48E\text{-}08) = 42{,}857 \text{ hours} = 4.9 \text{ yr}$$

where:

1.67E-07/yr	=	CDF (yearly average) for future core with one PORV blocked (Case B6)
6.48E-08/yr	=	CDF (yearly average) for future core with standard blocked PORV probabilities (Case B2)

In the generic case presented in Section 5.1.7, the worst time in cycle was used to determine the most limiting time. In this case the yearly average CDF is used which is consistent with the guidelines in RG 1.177.

The calculated AOT value is high since the PORVs, with regard to being blocked, are not important to total plant CDF. With regard to ATWS risk, the CDF increases by a factor of approximately 3 when one PORV is blocked as opposed to no PORVs blocked (see Table 9-7). Even though there is a factor of 3 increase, the magnitude of the increase (1.1E-07/yr) is small since ATWS CDF is small.

9.4 CONFIGURATION MANAGEMENT PROGRAM

This section presents the Configuration Management Program that will be used to address the identified Tier 2 restrictions at Braidwood with higher reactivity cores in the future. The approach to these restrictions is discussed in detail in Section 7.

Tables 4-34 to 4-37 provide UETs for the current and future cores for Braidwood. For the current core there are a number of plant configurations, near the start of the cycle, the plant can be operated in which result in a 0 UET. For the high reactivity core there is one plant configuration, near the start of the cycle, for which the UET is 0. This is for successful partial rod insertion, both PORVs available, and all AFW available. These are the conditions, component and system unavailability, for which defense-in-depth is not affected early in life. For other conditions, the degree of defense-in-depth, while not necessarily inadequate, may be lessened.

As noted in Section 7, currently plants can operate with PORVs blocked, with testing and maintenance activities in progress that result in the unavailability of parts of the AFW system (consistent with Tech Spec limitations on AOTs and Maintenance Rule requirements), and with the rod control system in either automatic or manual control. In addition, test and maintenance activities can also take place that result in parts of the reactor protection system being unavailable for short periods of time (again, consistent with the Technical Specifications and Maintenance Rule requirements). These activities can impact defense-in-depth.

By controlling the plant operating configuration, defense-in-depth capabilities can be maintained. The plant configuration can be controlled to enhance the probability of operating with favorable conditions with regard to UETs, and therefore, ATWS events. The following were noted in Section 7 as possible precautionary actions to take during UET periods:

- Operate with the rod control system in the automatic mode
- Limit blocking pressurizer PORVs
- Limit activities on the AFW system, AMSAC, and RPS that results in the unavailability of components within these systems.

Based on the PRA results presented and discussed in Sections 9.2 and 9.3, it is seen that configuration restrictions are not required to compensate for large impacts on plant risk. Rather, configuration restrictions are being proposed to address the NRC's concern for possible degradation of defense-in-depth. Table 9-8 presents the UET information from Tables 4-36 and 4-37 for the future core in the form of acceptable plant configurations for different times during the fuel cycle. In this case, defense-in-depth is the basis for acceptable configurations. This table shows the plant configuration required to maintain defense-in-depth, with regard to ATWS, at different times in life.

When components are out of service that are important to ATWS mitigation, acceptable AOTs, or the equivalent of an AOT for systems not included in the Technical Specifications, can be calculated by use of ICCDP and ICLERP assessments. As previously shown in Section 9.3, very large AOTs can be justified for blocked PORVs. Although this is acceptable from a risk perspective, the NRC indicates this is not acceptable from a defense-in-depth perspective. To address the defense-in-depth issue, the following actions are proposed, where appropriate, when operating in an unfavorable exposure condition according to Table 9-8:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

The objectives of these actions are to restore defense-in-depth. If defense-in-depth cannot be restored, these actions will either prevent further degradation of the configuration or reduce the probability of an ATWS event.

Other aspects of the CMP will also be consistent with the CMP requirements provided in Section 7. Several key items being considered for the Braidwood CMP follow:

- 30 day cumulative time period in a UET.

- A backup reactor trip via operator action at the control board to interrupt power from the MG sets to the CRDMs as a compensatory action if the 30 day limit is exceeded.
- Modification of E-0 of the Emergency Operating Procedures to incorporate the operator action at the control board to interrupt power from the MG sets to the CRDMs.

Additional details of the ATWS CMP will be provided in the Braidwood License Amendment Request.

9.5 CONCLUSIONS FROM THE LEAD PLANT EVALUATION

The following provides the key conclusions from the Braidwood analysis.

- The Braidwood ATWS PRA model follows the WOG model appropriately and can be used to evaluate the impact of plant issues and design changes on ATWS contributions to CDF.
- The impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) and meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr.
- Operating with the rod control system in automatic and with the PORVs not blocked reduces the plant CDF. The impact of these actions on the total plant CDF is small, but is significant on the ATWS contribution to the total CDF.
- A PORV can be out of service, or blocked, for a significant length of time based on the ICCDP calculation and the guidelines provided RG 1.177.
- Tier 2 restrictions have been developed that can be implemented into the Braidwood Configuration Risk Management Program to enhance maintaining defense-in-depth during unfavorable exposure times in the cycle. This is not required to compensate for large impacts on plant risk, but rather to address the NRC's concern of possible degradation of defense-in-depth.

Table 9-1 Braidwood Weighted UET Values, Current Core Design with the 5% UET Restriction			
100% Power, Equilibrium Xenon			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0.00	0.00	0.22
RI, 50% AFW	0.00	0.00	0.42
No RI, 100% AFW	0.00	0.25	0.71
No RI, 50% AFW	0.00	0.48	0.78

Table 9-2 Braidwood Weighted UET Values, New Core Design without the 5% UET Restriction			
100% Power, Equilibrium Xenon			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0.00	0.35	0.47
RI, 50% AFW	0.23	0.38	0.50
No RI, 100% AFW	0.34	0.51	0.69
No RI, 50% AFW	0.39	0.55	0.75

Table 9-3 Braidwood: Summary of Pressure Relief Intervals		
100% Power, Equilibrium Xenon		
PR Interval Basic Event	Current Core Design With 5% UET Restriction	New Core Design w/o 5% UET Restriction
PRAI1	0.00	0.00
PRAI2	0.00	0.35
PRAI3	0.22	0.12
PRAI4	0.78	0.53
PRBI1	0.00	0.23
PRBI2	0.00	0.15
PRBI3	0.42	0.12
PRBI4	0.58	0.50
PRCI1	0.00	0.34
PRCI2	0.25	0.17
PRCI3	0.46	0.18
PRCI4	0.29	0.31
PRDI1	0.00	0.39
PRDI2	0.48	0.16
PRDI3	0.30	0.20
PRDI4	0.22	0.25

Table 9-4 Summary of Comparison of WOG and Braidwood ATWS PRA Models			
Parameter	**WOG Model**	**Braidwood Model**	**Comments**
IE Frequency	1.0/yr	~1.0/yr	No impact
Reactor Trip Model	NUREG/CR-5500	NUREG/CR-5500	No impact
OA to Trip via Trip Switch in the Control Room	1.0E-02	1.0E-02	No impact
OA to Trip via MG Sets	0.5 HEP after failure of one OA 0.01 HEP after failure of no OA	0.5 HEP after failure of one OA 0.01 HEP after failure of no OA	No impact
OA to Drive Control Rods In (72 steps from lead bank)	Automatic operation assumed 0.5 base value, 0.1 sensitivity	for automatic operation If in manual: 1.0 HEP after failure of two OAs 0.5 HEP after failure of one OA	Braidwood assumed automatic operation in the base model and later did sensitivities for manual operation of the rod control system
Main Feedwater Availability	Conservatively not addressed	Included in model	Including this results in fewer ATWS events and lower ATWS CDF
Control Rods Fail to Drop	1.2E-06/demand	1.21E-06/demand	No impact
ESFAS	Included in model	Conservatively not addressed	Including this results in slightly lower ATWS CDF
AMSAC	0.01	0.01	No impact
Auxiliary Feedwater	Scalars based on typical AFW system design	Detailed fault tree developed for Braidwood AFW system	Appropriate to use plant specific model
Primary Pressure Relief Model	Detailed fault trees	Detailed fault trees, same as WOG model	No impact

Table 9-4 Summary of Comparison of WOG and Braidwood ATWS PRA Models (cont.)

Parameter	WOG Model	Braidwood Model	Comments
UETs	Low reactivity (5%) core High reactivity core Bounding reactivity core	Current core (with 5% limitation) Future core (without 5% limitation)	WOG cores based on 4-loop plant with model 51 SGs Braidwood cores are plant specific and based on the replacement SGs
Blocked PORV Probabilities	0.10 for either PORV 0.05 for both PORVs Assumed conservative values	0.05 for either PORV 0.0025 for both PORVs Based on plant experience	The less the PORVs are blocked the lower probability of being in a UET. Provides lower CDF results
PORV Failure Probability	7.0E-03/demand	Detailed fault tree developed	Appropriate to use plant specific model
Safety Valve Failure Probability	1.0E-03/demand	1.0E-03/demand	No impact
Long-term Shutdown	1.0E-02	Detailed fault tree developed for Braidwood emergency boration	Appropriate to use plant specific model

Table 9-5 Core Damage Frequency Summary, Current and Future Cores			
Standard Probabilities for Blocked PORVs			
Case	**Core**	**Rod Control System**	**CDF (per yr)**
B1	Current Core	Automatic[1]	4.15E-08
B2	Future Core	Automatic[1]	6.48E-08
B8	Future Core	Manual[2]	7.11E-08

Notes:
1. Failure probability of rod control system = 0.1
2. OA HEP = 1.0 following failure to trip the reactor via the control room trip switch and via the MG sets; OA HEP = 0.5 following to trip the reactor via the MG sets only.

Table 9-6 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Time in Cycle			
Standard Probabilities for Blocked PORVs			
Case	**Time in Cycle**	**Rod Control System**	**CDF (per yr)**
B3	Worst time in cycle	Automatic[1]	1.18E-07
B2	Yearly average	Automatic[1]	6.48E-08
B4	End of cycle	Automatic[1]	4.10E-08

Note:
1. Failure probability of rod control system = 0.1

Table 9-7 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Blocked PORVs			
Rod Control System in Automatic			
Case	**Time in Cycle**	**Number of PORVs Blocked**	**CDF (per yr)**
B2	Yearly average	Standard Probabilities	6.48E-08
B5	Yearly average	0	5.30E-08
B6	Yearly average	1	1.67E-07
B7	Yearly average	2	2.09E-07

Table 9-8 Configuration Management Approach for the Future Braidwood Core						
Acceptable Operating Configurations Based on Defense-in-Depth for ATWS						
Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual[2]		2	1	0
≤81	X		No			X
>81	X		Yes			X
>141	X		Yes			X
		X	No			X
>143	X		Yes			X
	X		No		X	
		X	No			X
>163	X		Yes		X	
		X	No			X
>166	X		Yes		X	
		X	Yes			X
>208	X		Yes		X	
	X		No	X		
		X	Yes			X
>225	X		Yes	X		
		X	Yes			X

Table 9-8 (cont.) Configuration Management Approach for the Future Braidwood Core

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS						
Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual[2]		2	1	0
>231	X		Yes	X		
		X	No		X	
		X	Yes			X
>256	X		Yes	X		
		X	Yes		X	
>333	X		Yes	X		
		X	Yes		X	
		X	No	X		
>362	X		Yes	X		
		X	Yes	X		

Notes:

1. It is assumed that AFW availability will be controlled by Technical Specification for the AFW system, that is, only one pump is allowed to be out of service at a time.
2. If the rod control system is in manual, no credit is taken for partial rod insertion.

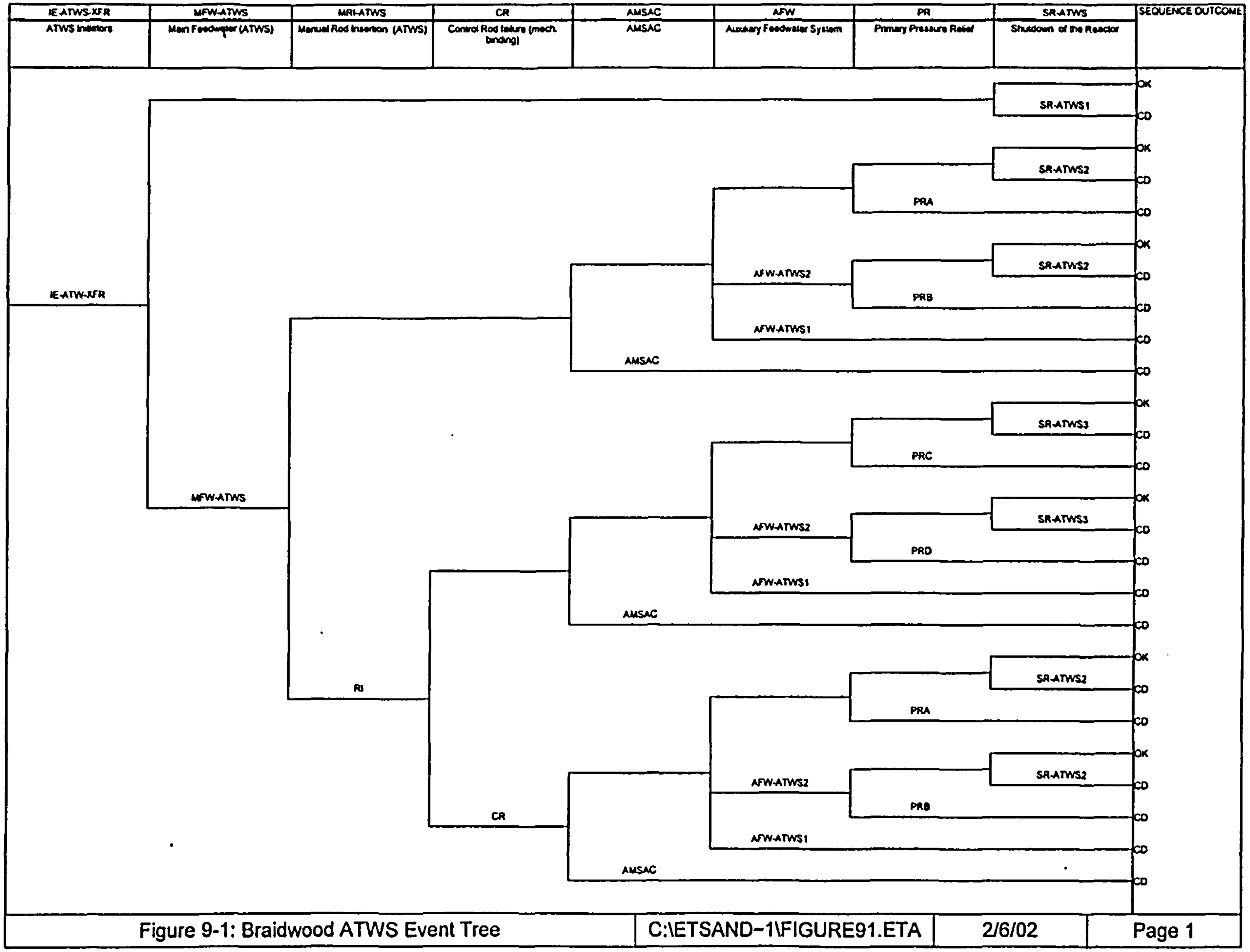

Figure 9-1 Braidwood ATWS Event Tree

10 RELOAD IMPLEMENTATION PROCESS

Plants, as currently licensed, only need to install AMSAC to meet the ATWS Rule (see Section 2.1). A key assumption in the NRC PRA for the ATWS Rule (Reference 4) 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. The Westinghouse generic analysis for ATWS used a full power MTC of –8 pcm/°F 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 an unfavorable MTC for 5% of the cycle. In more recent activities, the NRC imposed a Technical Specification on the Byron and Braidwood Stations, in response to a license amendment request for PMTC, that requires the UET to be no greater than 5% for the ATWS Rule reference configuration (no control rod insertion, 100% AFW, and no PORVs blocked). Although not explicitly stated as a licensing requirement in the ATWS Rule, this limit on UET is a restriction on core design for Byron and Braidwood.

Regulatory Guide 1.174 provides an approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis. In implementing RI decision-making, several key principles are expected to be met. These are (from RG 1.174, Section 2):

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

This report uses the RI approach to demonstrate the acceptability of eliminating the 5% UET restriction in the Byron and Braidwood Technical Specifications. Also, this report demonstrates that ATWS-specific MTC restrictions are unnecessary for all Westinghouse plants.

With the elimination of this requirement, the NRC is concerned about the design and use of reload cores with higher reactivity levels. This issue, higher reactivity reload cores, is applicable to Byron and Braidwood and other plants that have requested PMTC or have already implemented PMTC Technical Specifications. Currently the acceptability of reload cores is addressed via the 50.59 process and does not require prior approval of the NRC provided the provisions of 10 CFR 50.59 are met. The following is the proposed approach to address the NRC's concern.

The reload core implementation will continue under 10 CFR 50.59, but with the additional requirement of following the RI approach to evaluate changes to the plant's licensing basis. That is, the key principles specified above will be met. Applying this RI approach will only be done by utilities with plants that are not consistent with the bases for the ATWS Rule (Reference 6). This report demonstrates that these principles will be met on a generic basis for realistic or typical core designs that utilities would like to use in the future, as well as for a core design that is bounding for most Westinghouse plants. For reload cores, licensees will need to demonstrate that either the 5% UET restriction is met on a best estimate basis for the ATWS Rule reference configuration, or if not, that the generic RI analysis presented in this report is applicable. Licensees can demonstrate that the generic analysis is applicable by evaluating the impact of the new core design on CDF relative to a core design that meets a 5% UET for the ATWS Rule reference configuration. This should be done consistent with the WOG model presented in Section 8. The CDF impact should meet the acceptance criteria in RG 1.174. The CDF impact only needs to consider ATWS State 3/4 (power level $\geq$40% and HFP equilibrium xenon). There is no need to evaluate the other ATWS states since they are small contributors to ATWS risk. The LERF impact was evaluated and determined to be small for core designs considered in this analysis, therefore, it is not necessary to evaluate the LERF impact to demonstrate the applicability of this generic analysis. This plant specific assessment only needs to be done when initially transitioning to a high reactivity reload core (a core that does not meet the 5% UET for the reference configuration). Similar assessments for reload cores for following cycles are not required providing the analysis for the initial change to the high reactivity core remains applicable. In addition, configuration assessments based on either (ATWS) defense-in-depth or risk (as discussed in Section 7) will be required to assess the acceptability of plant operating configuration changes and identification of compensatory actions.

In summary, to demonstrate that a reload core is acceptable, given that the plant is not consistent with the bases for the ATWS Rule, licensees should either:

1. Demonstrate that the best estimate UET assuming no control rod insertion, 100% AFW, and no PORVs blocked is 5% or less.

   OR

2. Implement the WOG ATWS model to demonstrate that the impact on CDF meets the RG 1.174 acceptance guideline shown on Figure 3 of RG 1.174 AND implement a Configuration Management Program similar to the approach described in Section 7 of this report. Note that by meeting the CDF acceptance guideline using the WOG ATWS model, the licensee will demonstrate that the generic ATWS probabilistic risk analysis presented in this report remains applicable.

11 CONCLUSIONS

A WOG ATWS model was developed and presented that can be used to evaluate the impact of ATWS related issues on plant risk. The RI approach presented in RG 1.174, along with the WOG ATWS model, has been used in this program to demonstrate the acceptability of removing the stated or implied 5% restriction on UET for core designs. In particular, this model was used to evaluate the acceptability of high reactivity reload core designs. The following are the key conclusions from this study:

Key Conclusions from the Generic 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, and the CDF contribution from ATWS events to plant total CDF is small for all core designs.

- ATWS State 3/4, operation with the power level ≥40% and HFP equilibrium xenon, is the largest contributor to CDF. This state contributes 88% or more to the total ATWS CDF depending on the core reactivity. The other ATWS states (startup without equilibrium xenon for all power levels, and shutdown with xenon equilibrium for power level <40%) 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.

- 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 1.174. 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 that 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. An RCS pressure of 3584 psi is noted as 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 to fail as the RCS pressure increases during an ATWS event.

- ICCDP and ICLERP analysis 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. 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.

- The impacts on CDF and RCS integrity from LOSP/ATWS events are very small, therefore, this event is not important to the plant risk profile or to risk-informed decision process involving changes to a plant.

- Plant specific ATWS models and risk evaluations only need to consider CDF analyses for ATWS State 3/4 (power level ≥40% with HFP equilibrium xenon) since this state accounts for the largest contribution from ATWS events to plant risk. The plant specific model and evaluation can be used to assess the impact of plant changes on ATWS risk and also to demonstrate that the generic analysis and results are applicable to the individual plant.

- 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 addresses 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 of possible degradation of defense-in-depth.

Key Conclusions from the Braidwood Lead Plant Evaluation

- The Braidwood ATWS PRA model follows the WOG model appropriately and can be used to evaluate the impact of plant issues and design changes on ATWS contributions to CDF.

- The impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) and meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr.

- A PORV can be out of service, or blocked, for a significant length of time (greater than a year) based on the ICCDP calculation and the guidelines provided RG 1.177.

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

Reload Implementation Process

To demonstrate that a core reload is acceptable, with regard to ATWS considerations, given that the plant is not consistent with the bases for the ATWS Rule, licensees should either:

- Calculate the UET for the ATWS Rule reference configuration of no control rod insertion, 100% AFW, and no PORVs blocked to demonstrate it is 5% or less.

  OR

- Implement the WOG ATWS model and demonstrate that the impact on CDF meets RG 1.174 acceptance guideline shown on Figure 3 of RG 1.174 **AND** implement a Configuration Management Program similar to the approach described in Section 7 of this WCAP. Note that meeting the CDF acceptance guideline using the WOG ATWS model demonstrates that the generic analysis is applicable.

Based on the analysis presented in this WCAP, it is concluded that restrictions on UETs 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.

12 REFERENCES

1. WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988.

2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1988.

3. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1988.

4. SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram(ATWS) Events," USNRC, July 19, 1983.

5. NUREG/CR-0460, "Anticipated Transients Without Scram for Light Water Reactors," U.S. NRC, April 1978.

6. NS-TMA-2182, "ATWS Submittal," Westinghouse letter from T. M. Anderson to Dr. Stephen H. Hanauer (NRC), December 30, 1979.

7. "Review of WCAP-11992," U.S. NRC letter to D. V. VanderBurg, WOG Analysis Subcommittee Chairman, July 1, 1997.

8. "Issuance of Amendments (TAC NOS. M89092, M89093, M89072 and M89091)," NRC letter to D. L. Farrar (ComEd), July 27, 1995.

9. "Summary of Meeting Held on August 23, 2000 with the Westinghouse Owners Group (WOG) to Discuss Risk Informed Anticipated Transient Without Scram Program," NRC letter to Stuart A. Richards (NRC), October 26, 2000.

10. "Westinghouse Owners Group Risk-Informed Anticipated Transient without Scram Approach," NRC letter to Robert H. Bryan (WOG Chairman), April 2, 2001.

11. WCAP-15210, "Transient Initiating Event Operating History Database for U.S. Westinghouse NSSS Plants (1987-1997)," Rev. 1, November 1999.

12. INEEL/EXT-98-00401, Draft Report, "Rates of Initiating Events at U.S. Commercial Nuclear Power Plants – 1987 through 1995," April 1998.

13. WCAP-14333-P-A, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," Rev. 1, October 1998.

14. NUREG/CR-5500, Vol. 2, "Reliability Study: Westinghouse Reactor Protection System, 1984-1995," December 1998.

15. "Vogtle Electric Generating Plant, Unit 1 and 2, Individual Plant Examination Report in Response to Generic Letter 88-20," November 1992.

16. EPRI Report, "Advanced Light Water Reactor Utility Requirements Document," Volume 3, Chapter 1, Appendix A (PRA Key Assumptions and Groundrules).

17. EPRI Report, "Losses of Off-Site Power at U.S. Nuclear Power Plants – Through 1999," July 2000, Report Number 000000000001000158.

18. "Summary of December 17, 1998 Meeting with Westinghouse Owners Group Regarding Risk-Informed Anticipated Transients Without SCRAM Model," NRC letter to Thomas H. Essig (NRC/NRR), December 3, 1998.

19. WCAP-15376, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Interval and Reactor Trip Breaker Test and Completion Times," Rev. 0, October 2000.

20. "LOFTRAN Code Description," Burnett, T. W. T., et al., WCAP-7907-P-A (Westinghouse Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

21. "ANC: Westinghouse Advanced Nodal Computer Code," Davidson, S. L., ed., et al., WCAP-10965-P-A (Westinghouse Proprietary), WCAP-10966-P-A (Non-Proprietary), September 1986.

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

WOG Responses are Provided for Each Issue

Issue 1: Defense-in-Depth

The NRC has noted that maintaining existing defense-in-depth features of licensed nuclear power plants is important even when the impact of a desired plant change on CDF is small. With respect to ATWS in particular, a concern has been expressed that the use of core designs with more positive moderator temperature coefficients might be undesirable because it reduces the inherent core reactivity feedbacks (one of the defense-in-depth features of existing PWRs), which serve to shut down the core in the event of a plant transient. NRC views defense-in-depth in three layers for ATWS concerns: the first is core feedback, the second is the reactor trip system with backup operator actions, and the third is the set of plant features that serve to limit the pressure transient (or core heatup) that results from an ATWS event. The NRC requested information regarding how the loss of the "prevention" barrier is compensated for by the other barriers.

Response: The defense-in-depth issue is discussed in detail in Section 6.1. A process for compensating for the loss of the "prevention" barrier with other barriers is discussed in Section 7 (Configuration Management Program). Since the impact on risk has been shown to be small, this process is directed at maintaining defense-in-depth, and not to compensate for large impacts on plant risk.

Issues 2, 3, and 4: Large Early Release Frequency and Component Aging Considerations

The following response addresses three issues raised by the NRC. These are concerned with the structural integrity of the RCS pressure boundary during potential ATWS events. The basic issue concerns failure of the RCS and subsequent releases from containment either through containment failure, containment isolation failure, or containment bypass. Containment bypass could be via either the steam generator tubes or systems that interface with the RCS, such as the residual heat removal or letdown system. A statement of the issues follows.

Issue 2: Large Early Release Frequency

The NRC is concerned with 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 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 LOCA, that cannot be mitigated, will eventually occur that will relieve the RCS pressure in a relatively controlled manner; 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.

Issue 3: Steam Generator Tube Integrity

Current studies have indicated that the SG tubes will withstand an ATWS pressure peak that results in RCS failure. A 5% probability of SG tube failure is generally used if the RCS pressure increases to a point that the RCS fails (RCS pressure > 3200 psi). The NRC is concerned that with relaxation of SG tube structural requirements that ATWS induced SG tube ruptures could become an issue in the future. This was seen as an issue that the NRC and industry will need to keep in mind and re-visit as necessary, but no specific response is expected. (Note that even though no specific response was expected when the issue was stated by the NRC, SG tube integrity has been addressed in the response to Issues 2 and 4.

Issue 4: Component Aging Considerations

The NRC agrees that previous analyses done indicate that the RCS components will maintain their integrity up to 3200 psi, but these analyses assumed new or like-new component conditions. The concern is that with aged components this conclusion may not remain valid. This question arose with regard to valves that function to provide part of the RCS pressure boundary, and potentially interfacing system LOCAs and containment bypass issues.

Response: The following discusses the response of the RCS components to the potential high pressures during an ATWS event and the impact on large early release frequency. The RCS pressure during an ATWS event is dependent on the core design and time in core life, in addition to the availability of pressure mitigating systems and negative reactivity insertion. The systems and components that are important in mitigating the RCS pressure are the pressurizer PORVs and safety valves, the AFW system, and the rod control system.

As discussed in Section 5.2, a three part approach was taken to address LERF. Part 1 involves a comprehensive examination of the RCS, and interfacing systems and components, to determine if these components remain intact at the expected RCS pressures, or if missiles may be generated or interfacing systems fail such that containment integrity is degraded. Part 2 calculates the expected RCS pressures that correspond to the various combinations of control rod insertion, AFW, and PORV availability. These are used, in conjunction with the results from Part 1, to identify sequences that lead to containment degradation. Part 3 determines the frequencies of these sequences and calculates the LERF for the low, high, and bounding cores.

The RCS integrity assessment (Part 1) is provided in the following. The ATWS RCS pressure analysis and results (Part 2) are provided in Section 4. The ATWS LERF analysis and results (Part 3) are provided in Section 5.2.

RCS Integrity Assessment

A comprehensive examination of the RCS components, and systems and components that interface with the RCS was completed to identify any components that would fail at or below the RCS peak pressure for the bounding core (4110 psia). These components were divided in the following groups:

- Valves
- RCS Piping and Interfacing System Piping
- Pressurizer
- Steam generators
- Reactor vessel
- Reactor Coolant Pumps

A review of the design requirements of the components was completed as well as an assessment of the potential impact of aging on the component's structural integrity. It is important to note that the boundaries of this investigation are consistent with the traditional system boundaries of normally closed valves, isolation valves, check valves, and closed loop configurations. It was recognized that for closed loop configurations (i.e., steam generator tubes) a catastrophic failure of the wall would result in extended boundaries. The review considered external deadweight loads as the only additional source of stress beyond the pressure transient generated stress. Thermal expansion stresses will exist in many of these systems and may be reasonably large, but because of their nature, they tend to be self-limiting and redistribute with system deflections (unlike the pressure and deadweight stresses). The following sections discuss the findings for each group.

- Valves

a,b,c

a,b,c

a,b,c

- RCS Piping and Interfacing System Piping

The RCS and interfacing system piping in plants is designed in accordance with the requirements of ASME Section III or the equivalent B31.1 requirements. Under original design conditions, the design pressure of these systems is typically 2485 psi for design temperatures up to 680°F, and there is a nominal margin of safety for the pressure design of a factor of three. This piping is expected to retain structural integrity for the projected ATWS pressure of 4100 psi. Class 1 or piping with design pressure of 2500 psi would typically be schedule 160. Piping with design pressure of 1000 psi to 2000 psi would typically be schedule 80 or schedule 120 depending upon pipe size. The piping under discussion is typically at least schedule 80 or higher. Table A-1 provides a summary of stress intensity based on principal stress calculations for hoop, radial, and axial stress resulting from an applied pressure of 4100 psi to the straight stainless steel pipe typically attached to the RCS and interfacing systems. The only additional contributor to stress under the ATWS scenario is applied loads due to deadweight. The resulting stress for deadweight loads is typically less than 5000 psi for nuclear applications and, if added directly to the stress tabulated in Table A-1, would remain below recognized ASME Code limits for faulted one-time events. Clearly, the piping will not fail.

Table A-1 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi					
Nominal Pipe Size (inches)	Schedule				
	80	120	140	160	XXS
1/8	6512				
1/4	6825.6				
3/8	7783.3				
1/2	8208			6745.5	5044.5
3/4	9537.6			7122.6	5580.2
1	10180			7668.4	5859.1
1 1/4	11826			9321.6	6605.6
1 1/2	12822			9468.9	7070.8
2	14544			9641.8	7890.1
2 1/2	13954			10571	7607.4
3	15501			10967	8350.3
3 1/2	16631				
4	17593	13777		11560	9365.9
5	19434	14831		12084	10266
6	20057	15651		12470	10572
8		15908	14207	12847	13263
10		16828	14366	12891	
12		16844	15088	13091	
14		16902	14923	13386	
16		17310	14832	13485	
18		17267	15323	13571	
20		17568	15206	13633	
22		17823	15583	13875	
24		17458	15466	13734	

Table A-2 summarizes the resulting stress intensity in straight pipe for a principal stress based calculation for thickness reduced to 2/3 of nominal and applied deadweight loads equal to 5000 psi. This 33% allowance for potential wall thinning is impossible for stainless steel class 1 or class 2 piping and is included only to illustrate the margin available in piping. Again the calculated values remain within Code limits for stainless steel pipe, and so failure will not occur.

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

Nominal Pipe Size (inches)	Schedule				
	80	120	140	160	XXS
1/8	13987				
1/4	14483				
3/8	15973				
1/2	16627			14356	11513
3/4	18657			14948	12466
1	19631			15796	12932
1 1/4	22119			18328	14135
1 1/2	23619			18552	14867
2	26207			18815	16138
2 1/2	25320			20223	15702
3	27642			20823	16845
3 1/2	29338				
4	30779	25055		21718	18396
5	33536	26638		22508	19762
6	34468	27868		23089	20225
8		28254	25701	23656	24281
10		29632	25939	23722	
12		29657	27023	24023	
14		29744	26775	24467	
16		30355	26639	24616	
18		30291	27377	24744	
20		30742	27200	24839	
22		31123	27766	25202	
24		30577	27591	24990	

a,b,c

a,b,c

a,b,c

a,b,c

Table A-3 Letdown System Temperature Distribution

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

Table A-4 CRDM Stress Summary

a,b,c

a,b,c

a,b,c

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

a,b,c

a,b,c

Table A-6 RCP Stress Summary

a,b,c

References

1. CE Report: CENC-1273, "Analytical Report for Georgia Power Company Alvin W. Vogtle Unit No. 1 Reactor Vessel."

2. WCAP-12866, "Bottom-Mounted Instrumentation Flux Thimble Wear," January 1991.

3. WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors," September 1993.

4. Manufacturer's Literature on Swage Lock Fittings.

5. "Criteria for Design of Elevated Temperature Class 1 Components," Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, ASME, New York, 1976.

6. EM-4531, Rev 2.

7. EM-4860, Rev 1.

8. WCAP-13673, "Background and technical Basis: Handbook on Flaw Evaluation for the Sequoyah Units 1 and 2 Main Coolant System and Components," November 1993.

9. WCAP-14576, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," August 1999.

10. ASME Boiler and Pressure Vessel Code, Section XI, ASME, New York, 1995.

Issue 5: Part Power Considerations

The NRC is interested in the risk associated with part power operation. Particularly of concern is the risk when the reactor is initially started or re-started following a shutdown earlier in the cycle, when unfavorable exposure times exist. The concern is the risk related to the plant startup and the increased potential for a trip during this plant transient operation. The current ATWS models only include reactor trips at power levels greater than 40%. It was not clear if this risk is adequately addressed in ATWS models.

Response: The issue concerns plant risk due to ATWS events that occur during plant start-up following refueling and start-up following a reactor trip or required plant shutdown during the fuel cycle. The specific concern is the unfavorable exposure time during and immediately following the restart as related to the time it takes to build up equilibrium xenon concentration. The analysis used to determine the at-power UETs assumes that full power equilibrium xenon concentration exists. Without full power equilibrium xenon concentration, those UETs are not applicable.

The ATWS probabilistic risk analysis presented in Section 5 addresses ATWS risk due all phases of plant operation. This includes power operation before and after establishing equilibrium xenon concentration, and below 40% power without AMSAC and above 40% with AMSAC. Table 5-1 defines five different ATWS states analyzed. UETs were determined for each ATWS state and are provided in Section 4. Table 5-25 provides a summary of the ATWS CDF contributions for the five ATWS states. It was concluded from this study that the most important ATWS state, from the risk perspective, is with xenon equilibrium and the power level greater than or equal to 40%. The other states are relatively minor contributors to risk from ATWS events. Based on this it was concluded that plant PRA models that include ATWS events initiated from above 40% power with xenon equilibrium do adequately address ATWS risk and that the operating time prior to establishing xenon equilibrium does not need to be included in plant PRA models.

Issue 6: UET/MTC Link

The NRC is interested in the link between MTC (moderator temperature coefficient) and UET (unfavorable exposure time). They are concerned that all the inter-dependencies are not known and that some simplifications may lead to a secure feeling, but that a cliff may loom nearby. The NRC is interested in the range of the various coefficients that are used in the UET calculations. Sensitivity studies will need to be done to address this concern.

Response:

Unfavorable Exposure Time and Critical Power Trajectories

For a given plant and core design, the UET represents the period of time during the operating cycle when an ATWS event could lead to primary system pressures of greater than 3200 psi. The methodology used to determine the UET involves comparing two critical power trajectory (CPT) curves. The ATWS analysis is performed using LOFTRAN. The first CPT curve is calculated based on the reactivity feedback model used in the LOFTRAN analysis that results in a peak RCS pressure of 3200 psi. This CPT represents the change in power as a function of inlet temperature for this reactivity feedback model. To generate these curves, the transient analyst simulates an ATWS event and adjusts the moderator feedback (moderator density coefficient) in the point kinetics core model until the peak pressure limit is reached.

The second CPT curve is the set of inlet temperature and power level combinations that lead to criticality at the ATWS peak pressure in the actual core and using realistic feedback mechanisms. This second curve is generated by the core designer using a three-dimensional core model (ANC). This is the same core model that is used to assess key safety parameters for design basis events for the Reload Safety Evaluation. Realistic moderator, Doppler, and power feedbacks are employed. Using the core model, the core designer calculates a series of critical power levels as a function of inlet temperature and cycle burnup. The core designer then compares these critical power levels with the CPT curve from the system code. If, at a given cycle burnup step, the core critical power (CPT curve 2) is less than the peak pressure power (CPT curve 1), then that burnup is favorable with respect to meeting the 3200 psig limit. If, on the other hand, the critical power from the core model is greater than the peak pressure power from the system code, then that burnup is unfavorable. By calculating the fraction of the cycle that is unfavorable, the core designer determines the UET, usually in terms of number of effective full power days (EFPD) or percent of the cycle.

The limiting ATWS event for peak pressure is the Loss of Load event. Here, the increase in core inlet temperature drives the transient and the core response. As the core inlet temperature and system pressure increase, the natural core reactivity feedback mechanisms will respond and cause the power to drop. These feedback mechanisms effectively balance one another so that the core remains critical, albeit at a new statepoint condition. Briefly, a typical ATWS scenario is as follows: The core begins at steady state conditions operating at full power with nominal temperatures and pressures. When the ATWS event occurs, the inlet temperature rises causing a corresponding increase in system pressure. Since the full power moderator temperature coefficient is always negative, the core responds by dropping power. The positive reactivity increase caused by the drop in power effectively balances the negative reactivity effect of the increase in inlet temperature, resulting in a new critical condition. The two primary reactivity effects, then, are the moderator density feedback and the power coefficient feedback. The power feedback includes both moderator and Doppler components. These primary feedback mechanisms and their relationship to ATWS events are discussed below.

Moderator Density Feedback

Increases in coolant inlet temperatures will add negative reactivity to the core because of the negative moderator temperature coefficient. In response, the core power decreases, and equivalent positive reactivity is added to the core due to the combined effects of Doppler and moderator feedback (see power feedback discussion below).

ATWS events, however, involve not only an increase in core inlet temperature, but also an increase in system pressure. This complicates the moderator feedback since it becomes more than simply a temperature feedback at constant pressure; it involves a change in the moderator density associated with changes in inlet temperature, pressure, and power. For this reason, one cannot simply multiply the moderator temperature coefficient by the inlet temperature increase to determine the amount of negative reactivity added to the core during the event. This method will tend to overestimate the negative reactivity addition core since it doesn't account for the positive reactivity component associated with the pressure increase. Another reason that this simple approach will not work is that the moderator temperature coefficient is not a static value; it is a function of the dynamic reactor conditions, becoming more negative with decreasing moderator density and increasing moderator temperature.

Another factor that complicates the moderator feedback is axial flux redistribution. Whenever the inlet temperature or system pressure increases, the core axial power shape will change even if the reactor power is held constant. This change in axial power shape affects core reactivity since the axial burnup distribution of the core is not uniform. Generally, the net effect of an increase in both system pressure and core inlet temperature is a shift in the axial power distribution toward the bottom of the core, making the core less reactive due the higher fuel burnup there and higher axial power peaking. Reactivity changes due to redistribution are subtle reactivity effects that are usually implicitly included in the moderator, Doppler, and power coefficients.

As the above suggests, the moderator feedback during this kind of event has several components and complicating factors. Consequently, in any assessment of the core reactivity balance for an ATWS event, moderator feedback must be accounted for as part of an integrated reactivity effect between reactor states.

Doppler Feedback

Doppler feedback comes into play in association with the inlet temperature increase and power feedback (see power feedback discussion below). Generally, Doppler temperature feedback is a function of fuel type and power density. It is not a strong function of the core loading pattern or fuel burnup. Consequently, for a given plant, Doppler temperature feedback will not vary much from cycle to cycle or within a cycle.

The negative Doppler feedback that occurs due solely to the moderator temperature increase does not play a dominant role in an ATWS event, but it is important and must be accounted for in the overall reactivity balance. As the coolant temperature increases, the fuel temperature will also increase, adding negative reactivity to the core. Like the moderator feedback, Doppler feedback also has a redistribution component associated with changes in axial power shape and peaking factors. Higher power peaking and more highly skewed power shapes yield increased Doppler feedback.

More important than the Doppler feedback due to moderator temperature increase is the positive Doppler feedback in conjunction with the drop in core power. This is discussed in the following section.

Power Feedback

In the ATWS reactivity balance, a drop in reactor power effectively balances the negative reactivity effects associated with the inlet temperature increase. The overall power feedback is the sum of the moderator, Doppler, and redistribution reactivity components associated with this drop in reactor power, with the moderator component being the most dominant in the latter half of the cycle.

As reactor power drops, moderator density increases, fuel temperatures decrease, and power shifts toward the top of the core. Each of these effects adds positive reactivity to the core. The critical power level is that reactor power that just balances the net negative reactivity due to the inlet temperature increase. Because the moderator temperature coefficient generally becomes more negative with cycle burnup, power feedback becomes stronger with cycle burnup. Early in the cycle, the critical boron concentration is at its highest value. During this time, the moderator temperature feedback is at its weakest. As the core burns and the critical boron concentration decreases, the MTC becomes increasingly negative with cycle burnup. For a given inlet temperature increase, then, a larger drop in reactor power will occur at end-of-life than at beginning-of-life. For this reason, the unfavorable portion of the cycle is always nearest the beginning of the cycle.

ATWS Reactivity Balance

To characterize the interplay of these various reactivity components, reactivity balances were quantified for the low, high, and bounding reactivity cores designs for selected core inlet temperatures and cycle burnups. The reactivity balance is associated with five successive reactor states defined so as to separate the reactivity components:

1. Nominal HFP Steady State Condition (2250 psi, 556.6°F T_{in}, 3565 MWt)

2. Increased Pressure Condition at Nominal Inlet Temperature (3200 psi, 556.6°F T_{in}, 3565 MWt)

3. Increased Pressure with Higher Inlet Temperature, Moderator Feedback Held Constant (3200 psi, T_{in} of 600° to 660°F, 3565 MWt, moderator feedback same as State 2)

4. Increased Pressure and With Higher Inlet Temperature, Moderator Feedback Included (3200 psi, T_{in} of 600° to 660°F, 3565 MWt)

5. Critical Power Condition (3200 psi, T_{in} of 600° to 660°F, critical power level)

States 1 and 5 are critical states representing the initial and final reactor states. State 2 is a supercritical state resulting from the pressure increase. States 3 and 4 add in the negative Doppler and moderator density feedback, respectively. State 4 is always subcritical because of the negative reactivity associated with the inlet temperature increase (decreased moderator density). State 5 is the final critical power condition where the power is decreased to balance the negative reactivity resulting from States 1-4.

Tables A-7, A-8, and A-9 show these reactivity balances as well as moderator density coefficients, Doppler temperature coefficients, pressure coefficients, and power coefficients for the low, high, and bounding reactivity cores, respectively. In these tables, the pressure coefficient was calculated using the core k_{eff} values from States 1 and 2 above. The Doppler coefficient was calculated using States 2 and 3. The moderator density feedback was calculated using States 3 and 4. Finally, the power coefficient was calculated using States 4 and 5. Note that these coefficients represent average values between the reactor states. Furthermore, slightly different values would have been obtained if the order of the reactor states were changed. For example, if the inlet temperature were increased in State 2 and the pressure increased in State 3, the coefficient values would change somewhat. The above order was chosen primarily to avoid coolant voiding in the model, which would occur in the high inlet temperature cases if the pressure were not increased first.

Tables A-7, A-8, and A-9 also provide the HFP MTC at nominal conditions, the calculated critical powers, and the critical power limits for 3200 psig system pressure. These critical power limits correspond to the reference ATWS scenario, which assumes all PORVs available and full auxiliary feedwater.

Tables A-7, A-8, and A-9 illustrate the differences between cores with different excess reactivities with respect to ATWS performance. The low reactivity core achieves more negative MTC values early in the cycle through the use of a much larger loading of burnable absorbers. As a result, this core exhibits lower critical NSSS powers early in the cycle and a much smaller UET overall. With increasing cycle burnup, the critical powers of the high and bounding reactivity cores approach those of the low reactivity core. This occurs since, as burnup progresses and the burnable absorbers deplete, the cores have similar reactivity coefficients and reactivity balance values, i.e., their reactivity feedbacks become comparable.

Figure A-1 illustrates how the critical power varies with HFP MTC. Figure A-1 plots the calculated critical powers for the low, high, and bounding reactivity cores as a function of HFP MTC for both the 600° and 640°F inlet temperature cases. The plotted values come from Tables A-7, A-8, and A-9. Note that all three cores follow the same critical power versus MTC trendlines. This means that for a given inlet temperature and HFP MTC, these cores would be expected to have similar critical powers.

Note also that the MTC that yields a "favorable" critical power is slightly different for the two inlet temperatures. For this particular core and for inlet temperatures of 600°F, the MTC must be more negative than approximately -7.5 pcm/°F to achieve a favorable critical power. For the 640°F inlet temperature, the "favorable" MTC value is about -7 pcm/°F. Thus, the MTC requirement for a favorable critical power will vary somewhat depending on the inlet temperature assumption. For these scenarios, then, the UET is determined by the fraction of the cycle for which the HFP nominal MTC is less negative than these values.

The MTC requirement will also vary depending on the ATWS scenario being considered (number of PORVs available, auxiliary feedwater assumption, control rod insertion assumption) and the plant specific operating conditions (nominal power level, nominal inlet temperature, etc.) since these assumptions affect the peak pressure critical power limits calculated by LOFTRAN. If, for example, one were to assume a different ATWS scenario where only one PORV was available instead of two, the critical power limits corresponding to 3200 psig would be lower, and a more negative MTC value would be required to achieve a favorable critical power. Conversely, for a given core design and MTC versus burnup behavior, these lower critical power limits would lead to a higher UET for this particular ATWS scenario relative to the reference case.

In the risk-informed approach being proposed, all of the reactivity effects discussed above (Doppler, moderator, and power feedbacks) are implicitly included in the evaluation of each ATWS scenario through the reactivity balance that is inherent in the critical power and UET calculations. In this way, the particular feedback characteristics of a specific core design and the critical power limits appropriate for a particular plant are accounted for in the overall risk evaluation.

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

Initial Conditions and Final T_{in}	T_{in} = 600°F				T_{in} = 620°F				T_{in} = 640°F				T_{in} = 660°F			
Cycle Burnup (MWD/MTU)	150	3000	10000	21700	150	3000	10000	21700	150	3000	10000	21700	150	3000	10000	21700
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2624	2588	2331	2021	1971	1911	1533	1059	1255	1173	702	146	431	313	<0*	<0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	-3	-39	-296	-606	-37	-97	-475	-949	-33	-115	-586	-1142	2	-116	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	80	77	163	403	80	77	162	403	80	77	162	403	79	77	162	404
Doppler Reactivity	-51	-52	-55	-57	-74	-75	-79	-83	-95	-96	-101	-106	-114	-116	-122	-128
Moderator Reactivity	-444	-447	-813	-1785	-851	-871	-1476	-3044	-1481	-1533	-2427	-4692	-2579	-2688	-3919	-7025
Power Reactivity	416	422	705	1439	845	869	1393	2723	1496	1552	2365	4394	2613	2727	3706	5666
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-173*	-1083*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11
Pressure Coefficient (pcm/psi)	0.084	0.081	0.171	0.425	0.084	0.081	0.171	0.425	0.084	0.081	0.171	0.424	0.084	0.081	0.171	0.425
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.29	-1.35	-1.41	-1.26	-1.28	-1.35	-1.40	-1.25	-1.27	-1.34	-1.40	-1.25	-1.27	-1.34	-1.39
Moderator Density Coef. ($\Delta\rho/gm/cm^3$)	0.078	0.079	0.143	0.316	0.096	0.098	0.167	0.345	0.118	0.122	0.194	0.377	0.150	0.156	0.227	0.410
Power Coefficient (pcm/%)	-15.5	-15.2	-20.1	-32.9	-18.7	-18.6	-24.3	-38.5	-23.0	-23.0	-29.3	-45.6	-29.6	-29.8	-37.1	-56.7

* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

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

Initial Conditions and Final T_{in}	T_{in} = 600°F				T_{in} = 620°F				T_{in} = 640°F				T_{in} = 660°F			
Cycle Burnup (MWD/MTU)	150	3000	10000	22006	150	3000	10000	22006	150	3000	10000	22006	150	3000	10000	22006
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2656	2688	2374	2014	2014	2053	1600	1041	1312	1351	788	117	495	527	<0*	<0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	29	61	-253	-613	6	45	-408	-967	24	63	-500	-1171	66	98	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	71	55	147	408	71	55	148	408	71	55	147	408	71	55	147	407
Doppler Reactivity	-51	-52	-55	-57	-74	-75	-79	-82	-95	-96	-101	-105	-114	-116	-122	-127
Moderator Reactivity	-412	-361	-751	-1800	-800	-734	-1377	-3066	-1401	-1328	-2280	-4720	-2426	-2360	-3701	-7055
Power Reactivity	393	359	659	1450	803	754	1308	2740	1424	1370	2234	4418	2469	2421	3588	5641
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-87*	-1133*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5
Pressure Coefficient (pcm/psi)	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.29	-1.34	-1.40	-1.26	-1.28	-1.34	-1.39	-1.26	-1.28	-1.34	-1.39	-1.25	-1.27	-1.33	-1.38
Moderator Density Coef. ($\Delta\rho$/gm/cm^3)	0.073	0.063	0.132	0.319	0.090	0.083	0.155	0.348	0.112	0.106	0.182	0.379	0.141	0.137	0.215	0.412
Power Coefficient (pcm/%)	-15.2	-14.3	-19.5	-33.0	-18.3	-17.6	-23.6	-38.5	-22.4	-21.9	-28.5	-45.5	-28.5	-28.3	-35.9	-56.4

* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

Table A-9 Bounding Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario																
Initial Conditions and Final T_{in}	T_{in} = 600°F				T_{in} = 620°F				T_{in} = 640°F				T_{in} = 660°F			
Cycle Burnup (MWD/MTU)	150	3000	10000	22470	150	3000	10000	22470	150	3000	10000	22470	150	3000	10000	22470
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2923	2862	2420	2014	2399	2303	1661	1041	1775	1647	862	117	1012	855	<0*	≤0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	296	235	-207	-613	391	295	-347	-967	487	359	-426	-1171	583	426	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	22	23	131	407	22	23	131	407	22	23	131	407	22	23	131	406
Doppler Reactivity	-52	-52	-55	-57	-74	-75	-79	-82	-95	-96	-101	-106	-114	-116	-122	-127
Moderator Reactivity	-211	-231	-684	-1799	-478	-527	-1268	-3066	-938	-1032	-2123	-4720	-1794	-1951	-3479	-7061
Power Reactivity	240	260	608	1449	530	578	1217	2741	1011	1105	2093	4419	1886	2044	3449	5648
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-21*	-1133*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46
Pressure Coefficient (pcm/psi)	0.023	0.025	0.137	0.429	0.023	0.024	0.138	0.429	0.023	0.025	0.137	0.428	0.023	0.025	0.137	0.428
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.28	-1.35	-1.40	-1.26	-1.28	-1.35	-1.40	-1.25	-1.28	-1.34	-1.39	-1.25	-1.27	-1.34	-1.38
Moderator Density Coef. ($\Delta\rho$/gm/cm^3)	0.037	0.041	0.120	0.318	0.054	0.059	0.143	0.348	0.075	0.082	0.169	0.379	0.104	0.113	0.202	0.412
Power Coefficient (pcm/%)	-13.0	-12.9	-18.7	-33.0	-16.0	-16.2	-22.6	-38.5	-20.0	-20.4	-27.5	-45.5	-26.2	-26.8	-34.5	-56.5
* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.																

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

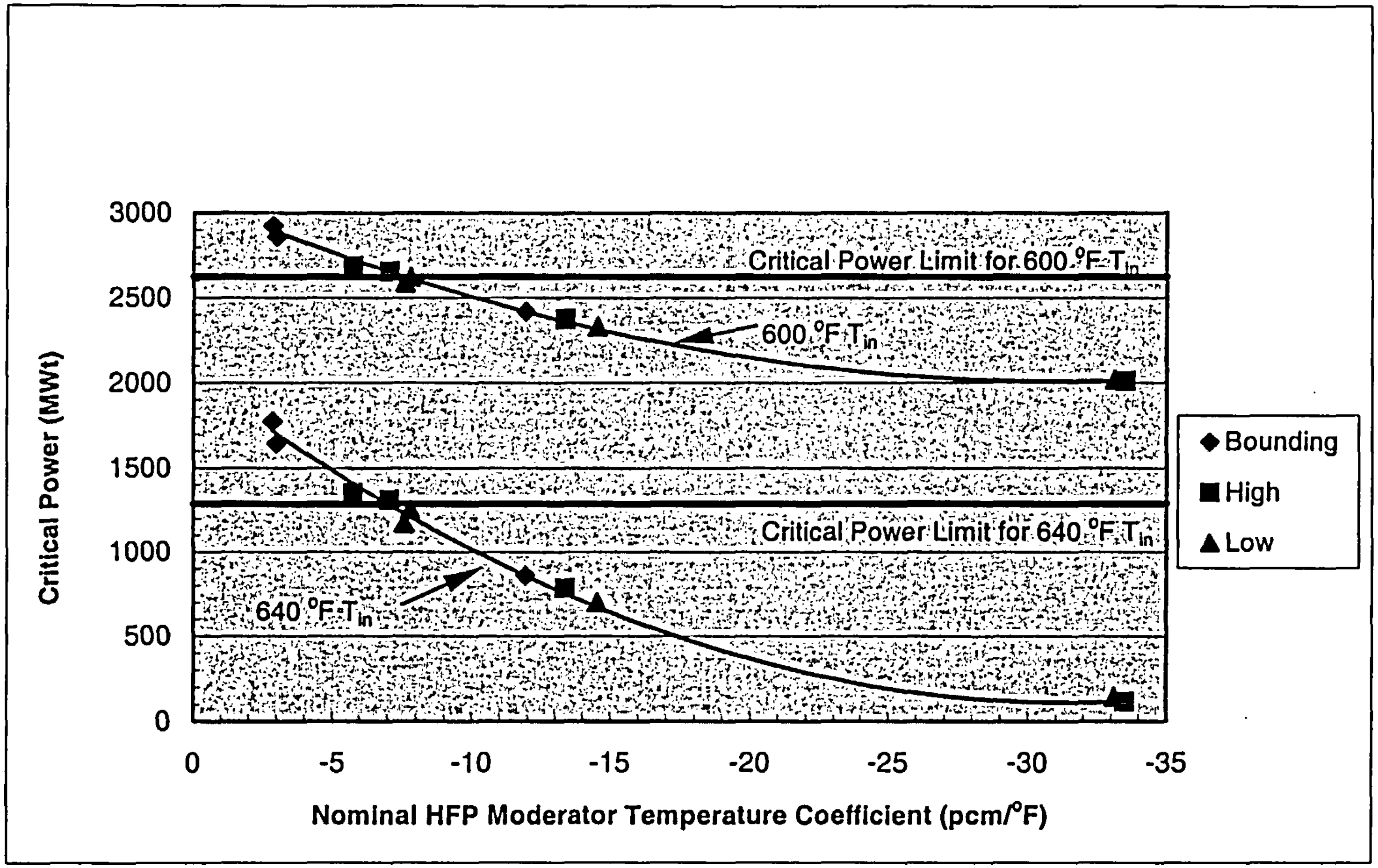

Figure A-1 Critical Powers for the Bounding, High, and Low Reactivity Core Designs for the Reference ATWS Scenario

Issue 7: Impact on Safety Margins

Requirements from other Chapter 15 events need to be maintained. It will be necessary to show that there is no impact on design basis event margins. The NRC noted that this issue is not directly a PRA issue.

Response: The current WOG program is developing a risk-informed approach consistent with Regulatory Guide 1.174 that can be used on a plant specific basis to demonstrate that the impact of core design changes, specifically those related to the moderator temperature coefficient, on plant safety is acceptable. Regulatory Guide 1.174 requires that the impact on plant risk, as measured by core damage frequency and large early release frequency, in addition to the impact of the change on defense-in-depth and plant safety margins be assessed. The impact on safety margins is discussed in Section 6.2. But in summary, 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.

Issue 8: Loss of Offsite Power with ATWS Events

Failure of the control rods to insert following a loss of offsite power (LOSP) event is not specifically addressed in the generic PRA ATWS model. The NRC would like to see this addressed on a generic and/or plant specific basis; whichever is necessary.

Response: Section 5.3 addresses the LOSP/ATWS event on a generic basis and contains a discussion of the analysis. The following is concluded in Section 5.3:

- 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 risk-informed decision process involving changes to a plant.

Therefore, the LOSP/ATWS event does not need to be included in plant specific PRA models.

Issue 9: Control Rod Insertion

The model currently assumes there is no link between burnup and control rod insertion requirements. This will need to be addressed to either: 1) show it is not important, 2) use a conservative value on the control rod insertion requirements, or 3) use different requirements for different times in the fuel cycle. Another comment on control rod insertion requirements is related to the event used to determine the number of rods required to insert. It was asked if this assumption covers all events.

Response:

During an ATWS event, the reactor coolant inlet temperature increases. The natural reactivity feedback mechanisms (moderator and Doppler) respond by reducing the core power level, effectively limiting the primary system pressure transient. Near beginning-of-life (BOL), the natural reactivity feedback mechanisms are weaker relative to middle-of-life (MOL) and end-of-life (EOL) due to a less negative moderator temperature coefficient. This results in higher peak pressures near BOL for a given inlet temperature increase.

Along with the natural reactivity feedback mechanisms, automatic or manual control rod insertion can mitigate the system pressure transient by introducing negative reactivity, further reducing the core power level. The effectiveness of the control rods in reducing the power level is assessed at all cycle burnups through the calculation of burnup dependent critical powers. Calculations are performed assuming a pressure of 3200 psi, a range of inlet temperatures, and D-Bank insertion of 72 steps. The resulting power levels are compared to the critical power trajectory (CPT) curves, generated by Transient Analysis, that yield a peak pressure of 3200 psi. If the calculated critical power for a given burnup is less than the CPT curve value, then that burnup is "favorable." If the calculated critical power is greater than the CPT curve value, then that burnup is "unfavorable." By quantifying the fraction of the cycle that is unfavorable, we obtain the unfavorable exposure time (UET) for the given scenario.

For the most probable plant configurations (e.g., full auxiliary feed, 0 PORVs blocked), cycle burnups beyond the first half of the cycle are generally not limiting since the natural feedback mechanisms are strong enough to limit the peak system pressure to less than 3200 psi. Thus, control rod insertion is primarily a benefit near BOL for these scenarios since the negative reactivity of the control rods augments the natural reactivity feedback, significantly reducing the UET for the cycle. For the reference ATWS scenario (loss of normal feedwater with 0 PORVs blocked and full auxiliary feed), 72 steps of control rod insertion reduces the UET to 0% for the low and high reactivity core models; i.e., the 3200 psig limit is never reached at any time during the cycle.

In the risk-informed approach being proposed, credit for rod insertion is taken based upon the probability of operator action to drive in the control rods or the probability of the automatic rod control system to function properly. When credit for rod insertion is taken in this fashion, only insertion of the lead control bank (Control Bank D) is credited and only one minute of rod insertion is assumed (~72 steps of insertion). The amount of control rod insertion assumed is not event specific; i.e., the same assumptions are made for all ATWS events in evaluating whether the peak pressure limit is met. (With regard to ATWS events caused by mechanical binding of the control rods, it is expected that a sufficient number of rods will insert to provide the equivalent of 72 steps insertion of the lead bank.) Furthermore, the probability of control rod insertion is not a function of the cycle burnup or the burnup of the fuel assemblies in control rod positions. There are currently no specific burnup restrictions or limits on fuel assemblies placed in control rod locations. Control rods are expected to insert properly into all fuel assemblies that meet the generic licensed fuel burnup limit.

Control rod insertion, even the modest amount of rod insertion assumed here, is very effective in reducing the core power level. Credit for rod insertion, within the framework of a risk-informed approach, is justified based upon the high probability that a sufficient number of control rods will insert or that manual rod insertion or automatic rod insertion through the rod control system will be successful.

Issue 10: Regulatory Issue

The NRC is concerned with how plant operation would be regulated with regard to ATWS. The NRC asked how would a plant that tripped early in its cycle and wanted to restart with one power operated relief valve blocked and a main feedwater pump unavailable, as permitted by Tech Specs, be treated from the regulatory perspective.

Response: Limitations on the unavailability of systems important to mitigation of an ATWS event during UETs may be implemented if required to maintain plant safety for higher reactivity core designs. As discussed in Section 7, and based on the PRA results in Section 5, configuration restrictions are not necessary to compensate for large impacts on plant risk related to higher reactivity core designs. It was shown in Section 5 that the impact on plant risk of higher reactivity cores is small. But a concern does exist relative to the impact of higher reactivity core designs on defense-in-depth. To address this, configuration restrictions via plant specific configuration management programs are being proposed.

Section 7 presents and discusses the proposed configuration management program. The objective of the program is to maintain defense-in-depth though out the cycle by managing the plant configuration and, if that cannot be accomplished, to take actions to reduce the probability of an ATWS event. This is done by the following when operating in an unfavorable exposure time:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

Details of the approach are provided in Section 7.

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

WOG Responses are Provided for Each Issue and Information Need

Issue 1: The WOG did not address the potential for no auxiliary feedwater (AFW) to be available (e.g., resulting from maintenance and/or equipment failures) as a parameter of the possible plant configurations (i.e., auto or manual rod insertion, power-operated relief valve (PORV) availability, and AFW availability). The inclusion and consideration of no AFW being available would aid the staff in future reviews.

Response: The availability of AFW is considered in the analysis and in the configuration management approach as discussed in the following.

Unfavorable Exposure Time Analysis: UETs are developed for success or failure of rod insertion (72 steps from the lead bank), number of blocked PORVs (0, 1, or 2), and amount of AFW flow to the steam generators. AFW flow level of 100% and 50% are considered. 100% AFW flow is flow from all AFW pumps and 50% AFW flow is half the 100% flow value. UETs from these different conditions, including AFW, are provided in Section 4.

Probabilistic Risk Analysis: The PRA model provided in Section 5 includes AFW flow as an event in the ATWS event tree (see Figure 5-1). Splits between 100%, 50% (<100% but $\geq$50%), and <50% are provided. These correspond to the UET AFW dependencies. The AFW unavailability values used for these split include component failure probabilities, common cause failure of components, and component unavailability due to test and maintenance.

Configuration Management Approach: The configuration management approach to enhance defense-in-depth is discussed in Sections 6.1 and 7. AFW availability is a key part of the configuration management approach for maintaining a plant configuration corresponding to a favorable exposure. This is shown on Figure 7-1 in the column "AFW Maintenance Acceptable" and also as discussed in Section 7 where it's noted that possible precautionary actions during UET periods can include, among others, limiting activities on the AFW system that result in its unavailability. Eliminating scheduled AFW maintenance and testing activities that result in partial AFW system degradation, as in the unavailability of a single pump, increases the probability that 100% AFW flow will be supplied if an ATWS event occurs.

Conclusion: The AFW plays an important role in the plant configuration management approach to maintain defense-in-depth and is included in the approach to configuration management described in Section 7.

Issue 2: The WOG did not provide any information on the incremental conditional core damage probability (ICCDP) when operating in the various plant configurations. For each core design, calculating the ICCDP for operating in each plant configuration during: 1) the estimated operating time within the configuration's unfavorable exposure time (UET) period (e.g., based on a plant configuration management scheme and considering random failures), 2) operating throughout the UET period (i.e., assuming the plant operates throughout the entire UET period in that configuration), and 3) operating throughout the entire cycle (i.e., assuming the plant remains in that configuration throughout the cycle) would aid the staff in future reviews.

Response: ICCDP calculations have not been completed for the three different core designs for the different plant configurations for the three conditions as requested above. First, it's not possible to estimate the amount of time plants will operate within each configuration. This is a plant specific decision and will be impacted by the requirements to maintain defense-in-depth via a configuration management program. Second, plants cannot operate in a number of these configurations for a significant length of time due to other limitations, primarily the Technical Specification requirements on AFW. The AFW Technical Specification would not allow the AFW system to be degraded for a significant length of time. Typically, 72 hours is the maximum for one AFW train inoperable.

Two plant configuration cases were evaluated to determine acceptable configuration specific operating times based on the ICCDP and the Regulatory Guide 1.177 acceptance guideline of 5E-07. 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 (in this case the acceptable configuration specific operating time)

An acceptable configuration specific operating time can be determined based on the acceptance guideline of ICCDP $\leq$ 5E-07.

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

Two cases are considered. The first is a bounding case that represents any plant configuration that cannot mitigate the pressure transient. This will be based on the high reactivity core. The second case is for blocked PORVs, which will be provided for both the high and bounding reactivity cores. This second case is identical to those presented in Section 5.1.7.

Case 1: Configurations in which the pressure transient cannot be mitigated (any configuration with a UET)

CCDF = 1.51E-06/hr (CDF given the pressure transient cannot be mitigated. This is taken from Table 5-31/Case 11 and is for the worst time in the cycle for the bounding core. For the bounding core, at the worst time in the cycle, the pressure transient cannot be mitigated even with all equipment available. This value also represents the CCDF for any core in a plant configuration in which the pressure transient cannot be mitigated.)

$CDF_{baseline}$ = 1.70E-07/yr (high reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 1.70E-07/yr) = 3269 hr = 0.37 yr

Case 2A: Configurations in which one PORV is blocked – high reactivity core

This turns out to be the same as Case 1 if we consider the time in the cycle a UET exists with a blocked PORV.

CCDF = 1.51E-06/yr (from Table 5-30/Case 5)

$CDF_{baseline}$ = 1.70E-07/yr (high reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 1.70E-07/yr) = 3269 hr = 0.37 yr

Case 2B: Configurations in which one PORV is blocked – bounding reactivity core

The CCDF used is the same as Case 1 since the pressure transient cannot be mitigated during the worst time in the cycle regardless of the number of PORVs available.

CCDF = 1.51E-06/yr (from Table 5-31/Case 15)

$CDF_{baseline}$ = 4.69-07/yr (bounding reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 4.69E-07/yr) = 4207 hr = 0.48 yr

Conclusion: Although the ICCDPs requested have not been provided, conservative acceptable configuration specific operating times were determined based on the ICCDP and Regulatory Guide 1.177 acceptance guideline of 5E-07. Based on this it is seen that ATWS pressure mitigating components can be unavailable for significant lengths of time.

Issue 3: The number of days within a UET condition for the various plant configurations for the bounding core design was not provided. A chart that identifies the number of days within the UET condition for the various plant parameters for the bounding core design would aid the staff in future reviews.

Response: The UETs for the bounding core are provided in Tables 4-11 to 4-15, 4-28, and 4-29.

Issue 4: Since there are different UETs calculated for plant configurations in which the only parameter that changed is the rod insertion mode (i.e., Auto or Manual), it appears that at least some partial rod insertion is credited in the analyses when in the Auto rod insertion mode. An explanation of how the WOG addresses this parameter in the UET calculations would aid the staff in future reviews.

Response: In the risk-informed approach being proposed, credit can be taken for insertion of the lead bank either through operator action (manual rod insertion) or through actuation of the rod control system when the reactor is in automatic rod control (auto rod insertion). When credit is taken for manual or auto rod insertion, only the lead control bank (D-Bank) is assumed to insert and the credit for insertion is limited to 72 steps (~1 minute of control rod insertion). In the UET calculation, there is no distinction between manual or auto rod insertion. For a given plant configuration, the UET is calculated with and without rod insertion assumed. When control rod insertion is assumed, D-Bank is inserted 72 steps from the all rods out position for the critical power calculation. This has the effect of reducing core reactivity and dropping reactor power to a lower value than would be achieved with no control rod insertion. The net result is a smaller UET.

Issue 5: Based on the meeting discussions, apparently all maximum ATWS pressure calculations were performed with equilibrium xenon levels at 100 percent power, except for the part-power calculations, which actually were 100 percent power with no xenon. The information provided gave UETs and probabilities for these conditions. A table of the peak pressures for the part-power conditions would aid the staff in future reviews.

Response: The probabilistic analysis discussed in Section 5 evaluated the ATWS CDF for all ATWS states which are:

- ATWS State 1: power level <40%, without equilibrium xenon
- ATWS State 5: power level <40%, with equilibrium xenon
- ATWS State 2: power level ≥40%, without equilibrium xenon
- ATWS State 3/4: power level ≥40%, with equilibrium xenon

From this assessment it was determined that ATWS State 3/4 is the largest contributor to core damage frequency; 88% for the low reactivity core, 89% for the high reactivity core, and 93% for the bounding reactivity core (see Table 5-25). Since this state is the dominant contributor to ATWS CDF, peak RCS pressures have been calculated only for this state. The remaining states are small contributors to CDF so they were not considered in the LERF assessment, therefore, peak RCS pressures have not been calculated. Peak RCS pressures are provided in Tables 4-20 and 4-21 only for ATWS events that initiate at 100% power with equilibrium xenon.

Table B-1 provides a summary of the CDF contributors for the low and bounding reactivity cores for each ATWS state. Also shown are the increases in CDF from each state. This again shows that ATWS State 3/4 is the dominant contributor to the increase in CDF, accounting for ~ 95% of the increase.

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

ATWS State	Low Reactivity Core CDF (per yr)	Bounding Reactivity Core CDF (per yr)	ΔCDF (per yr)	Percent of Total ΔCDF (per yr)
1	1.32E-09	1.34E-08	1.21E-08	3.2%
2	1.17E-08	1.36E-08	1.90E-09	0.5%
3/4	1.09E-07	4.69E-07	3.60E-07	94.7%
5	1.57E-09	8.15E-09	6.58E-09	1.7%
Total	1.24E-07	5.04E-07	3.80E-07	100.1%

As stated in Section 5.2.1, LERF values were only calculated for ATWS State 3/4. If it is very conservatively assumed that all the core damage sequences for the other ATWS states proceed to large early release sequences, the increase in LERF, from the low reactivity core to the bounding reactivity core, for the other ATWS states is:

$$\Delta\text{LERF (ATWS States 1, 2, \& 5)} = 1.21\text{E-}08 + 1.90\text{E-}09 + 6.58\text{E-}09 = 2.1\text{E-}08/\text{yr}$$

Adding this to the ΔLERF for ATWS State 3/4 from Section 5.2.1, for the sensitivity case that assumes the peak pressures are applicable to 50% of the cycle, the total LERF impact is then:

$$\Delta LERF = 6.05E\text{-}08/yr + 2.1E\text{-}08/yr = 8.2E\text{-}08/yr$$

Even with this conservative approach, the impact on LERF from the low to bounding reactivity core meets the LERF guideline in Regulatory Guide 1.177 (<1E-07/yr defines a small impact), and the conservative estimate for ATWS States 1, 2, and 5 account for approximately 25% of the total ΔLERF. This also shows that the LERF contributions from these states are not significant. Based on this, peak RCS pressures were not calculated for part power conditions, and it was concluded that ATWS risk from part power conditions plays only a small role in ATWS risk and can be neglected in the decision-making process.

Issue 6: It is not clear that the peak ATWS pressures at 100% power with no xenon bound the peak ATWS pressures at lower powers with no xenon, especially since the moderator temperature coefficient (MTC) can be more positive at lower powers (i.e., at 100 percent power MTC is limited by technical specifications to 0 pcm/°F, but at 70% percent power MTC can be as high as +5 pcm/°F). The identification of the part-power levels that produce the bounding peak ATWS pressures for the low, high, and bounding core cases and what bounding pressures are reached would aid the staff in future reviews.

Response: As discussed in the response to Issue 5, the ATWS states without equilibrium xenon are very small contributors to CDF. Also as discussed, if it is assumed that all core damage sequences in ATWS states without equilibrium xenon are assumed to proceed to large early release sequences, the impact on LERF, from the low to bounding reactivity core, meets the LERF guideline in Regulatory Guide 1.177.

Based on this conservative analysis, it was concluded that the RCS pressures for conditions without equilibrium xenon are not necessary and would not provide any benefit in the decision-making process.

Issue 7: For some cores, the peak ATWS pressures would occur at some point in time substantially after the beginning of the fuel cycle, when the different rates of burnable poison depletion, uranium depletion, fission product accumulation and plutonium breeding create the maximum net surplus reactivity. The full power UETs described in the information provide account for these factors. However, it is not clear if the peak pressures from an ATWS during a restart from a forced outage (>3 days) in the period of maximum core reactivity is bounded by the pressures calculated for other conditions. The bounding pressure for the above conditions would aid the staff in future reviews.

Response: As noted in the response to Issue 5, peak pressures have only be calculated and provided for 100% power operation with equilibrium xenon. By the probabilistic risk analysis provided in Section 5, it was shown that the condition of power level ≥40% with equilibrium xenon accounts for the majority of the ATWS CDF and it was concluded from this that LERF analysis and the supporting RCS pressure analysis only needs to be done for these conditions.

Also as discussed in the response to Issue 5, if it is assumed that all core damage sequences in ATWS states without equilibrium xenon are assumed to proceed to large early release sequences, the impact on LERF, from the low to bounding reactivity core, meets the LERF guideline in Regulatory Guide 1.177.

Based on this conservative analysis, it was concluded that the RCS pressures for conditions without equilibrium xenon are not necessary and would not provide any benefit in the decision-making process.

Issue 8: Identifying the initiating event conditions that result in the highest pressures and what pressures are reached would aid the staff in future reviews. Specifically, a table of the pressures reached for the different cores if no AFW, no PORVs, and no rod insertion are available for the initiating event that results in the highest pressure would aid the staff in future reviews.

Response: Tables 4-20 and 4-21 provide the RCS peak pressures for the twelve UET conditions for the low and bounding reactivity cores. These are based on the pressure limiting loss of load ATWS event for a 4-loop W PWR with model 51 steam generators at an uprated power level of 3579 MWt.

Issue 9: The information provided includes distinct values for moderator temperature coefficient (MTC) at 150, 4000, 9000, and 21,512 MWD/MTU. Though helpful, this does not provide insight into the MTC behavior at low power and in-between these four points. A plot of MTC and peak pressure as a function of time for the limiting power level and limiting initiating event while in the UET domain for each case would aid the staff in future reviews.

Response: Figure B-1 gives the HFP moderator temperature coefficients as a function of cycle burnup for the low, high, and bounding reactivity cores. Similarly, Figure B-2 gives the HZP moderator temperature coefficients as a function of burnup. These figures clearly show that the bounding core has the weakest moderator feedback early in the operating cycle. The low reactivity core, on the other hand, has the strongest moderator feedback. As the cycle burnup proceeds and the burnable absorbers deplete, all three cores tend to approach roughly the same moderator coefficient values. Note in Figure B-2 that the bounding reactivity core reaches the +7 pcm/°F limit at a burnup of about 2000 MWD/MTU.

Figure B-1 can be used to approximate the UET for a given ATWS scenario and inlet temperature assumption. For the reference ATWS scenario (all PORVs available, full auxiliary feed, no rod insertion) and an assumed ATWS inlet temperature of 600°F, the "favorable" MTC is approximately -7.5 pcm/°F (see response to the UET/MTC link issue, Appendix A, Issue 6). Thus, the portion of the cycle for which the MTC is less negative than -7.5 pcm/°F is the unfavorable portion of the cycle. Figure B-1 shows that the low reactivity core is always more negative than -7.5 pcm/°F; therefore, for this ATWS scenario and inlet temperature assumption, the UET would be ~0% of the cycle. For the bounding reactivity core, the MTC does not become more negative than -7.5 pcm/°F until a burnup of about 6800 MWD/MTU. Since the EOL burnup is ~22,000 MWD/MTU, this corresponds to a UET of approximately 30% of the cycle. For the high reactivity core, the favorable portion of the cycle occurs for burnups greater than about 4900 MWD/MTU, corresponding to a UET of about 22%. The "favorable" MTC value will depend upon the ATWS scenario being considered (number of PORVs available, etc.) and the inlet temperature assumption.

Figures B-3 and B-4 show the MTC behavior as a function of power level. In Figure B-3, equilibrium xenon was assumed at each power level, while in Figure B-4 no xenon was assumed. (At HZP, no xenon and equilibrium xenon are the same condition.) As these figures show, the MTC is a monotonically decreasing function of power level. Figure B-3 demonstrates that even for the bounding core, the expected MTC at full power with equilibrium xenon is negative (-2.7 pcm/°F was calculated). In Figure B-4, the MTC values are more positive than in Figure B-3 due to the higher critical boron concentrations that result from the no xenon assumption. While Figure B-4 gives MTC values at HFP with no xenon, this is not a realistic condition. Typical power ramp rates are slow enough such that significant xenon build-up prior to reaching full power is expected. If necessary, control rod withdrawal limits can be specified as a function of boron concentration to ensure that the MTC Technical Specification is met. This is consistent with current MTC Technical Specifications.

The plots of peak pressure as a function of time have not been developed. As discussed in Section 5.2, a conservative approach was used in the LERF assessment that assumed the peak pressures are applicable to the full cycle length for each rod insertion, AFW flow, and blocked PORV configuration. The peak pressures are provided in Section 4.3.

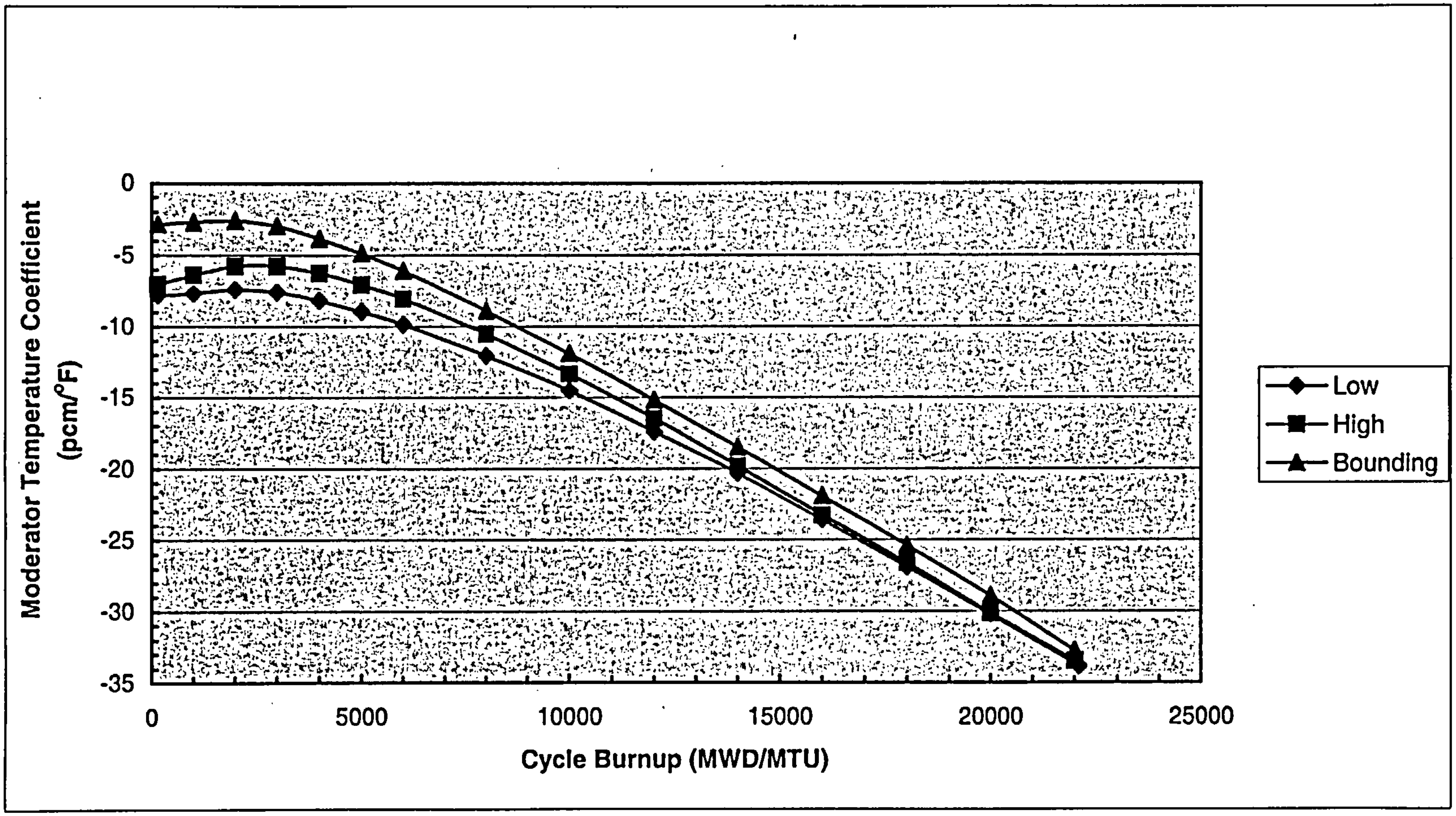

Figure B-1 HFP Moderator Temperature Coefficient versus Cycle Burnup

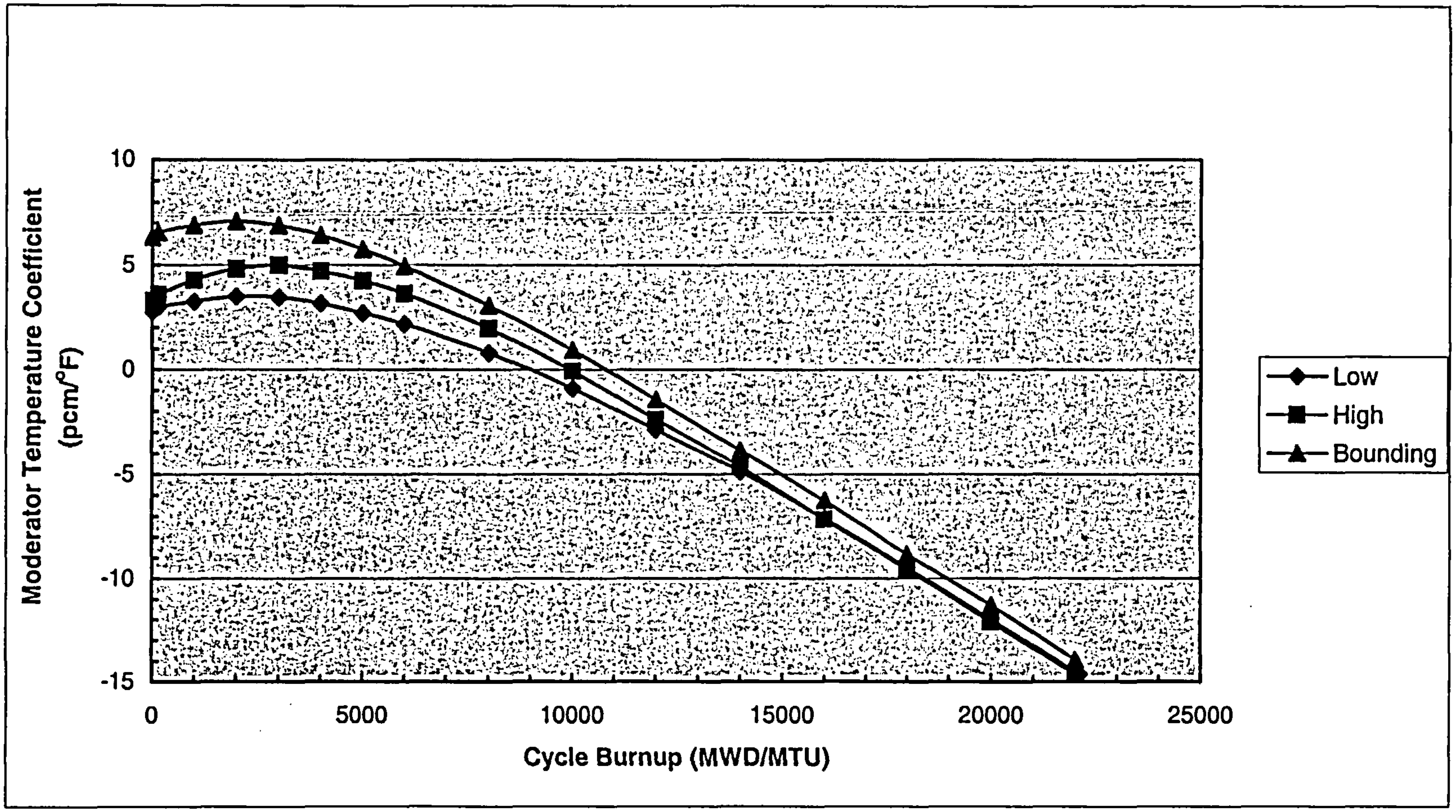

Figure B-2 HZP Moderator Temperature Coefficient versus Cycle Burnup

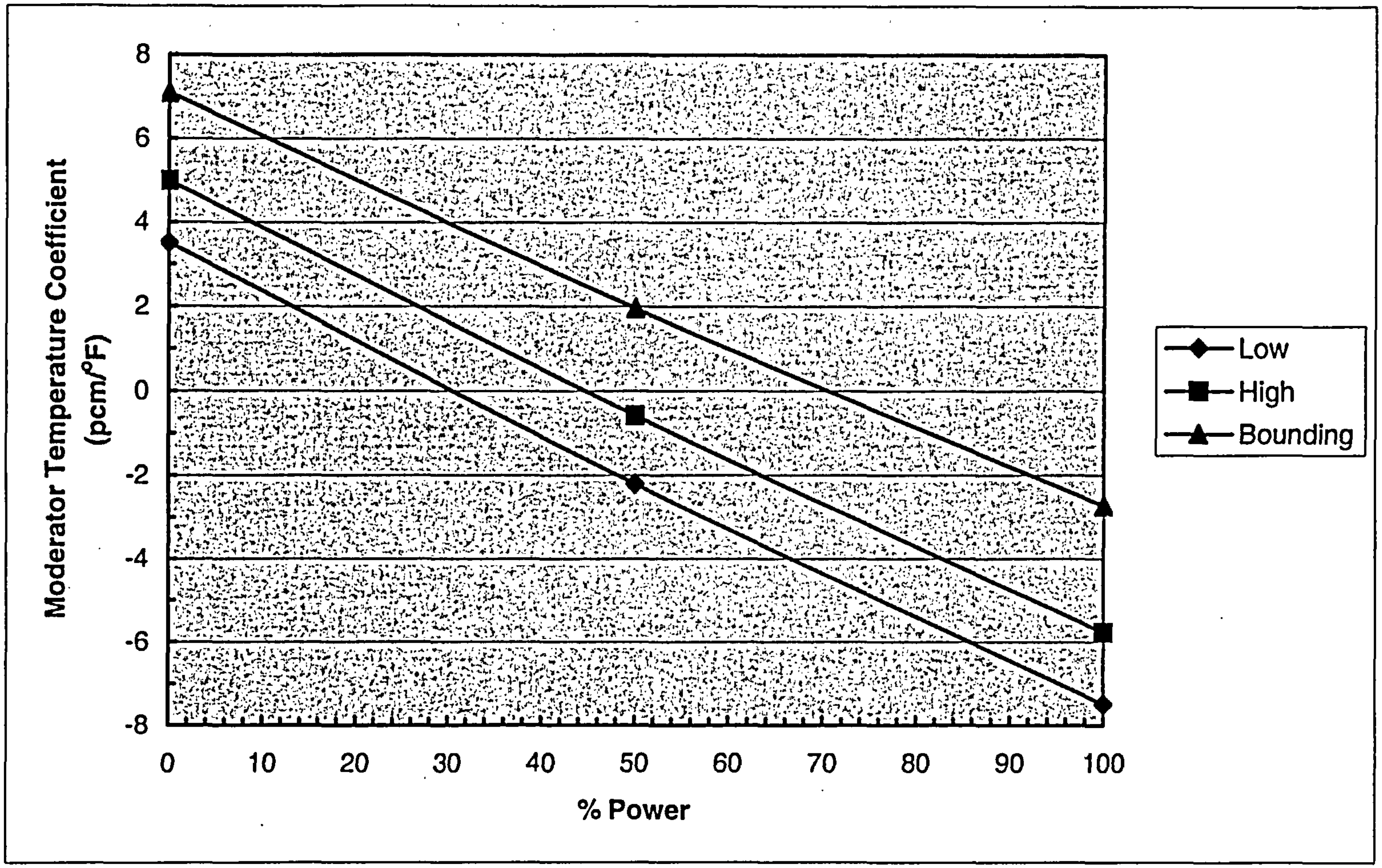

Figure B-3 Moderator Temperature Coefficient versus Power Level Assuming Equilibrium Xenon

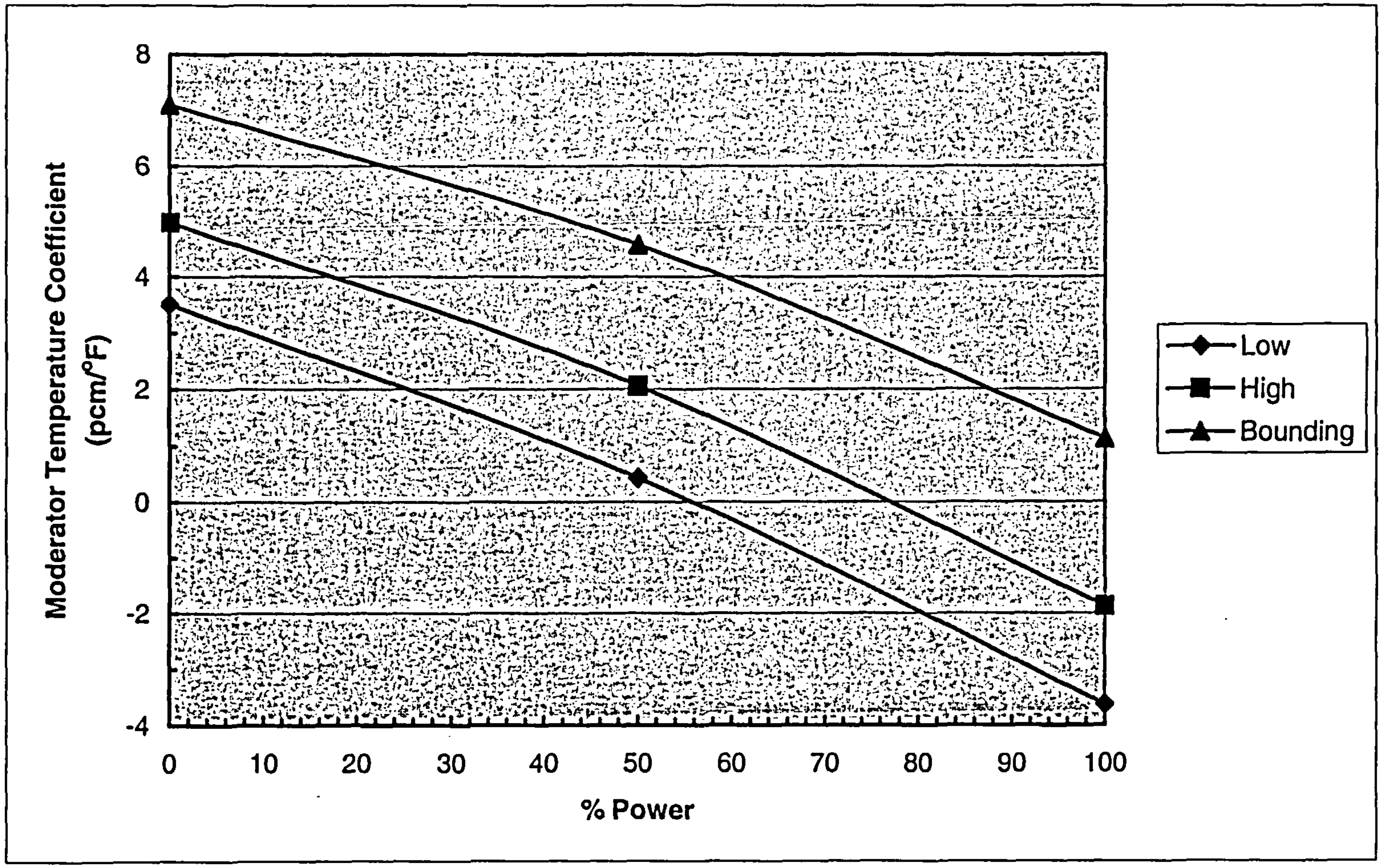

Figure B-4 Moderator Temperature Coefficient versus Power Level Assuming No Xenon

Issue 10: Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," addresses risk-informed approaches. As part of the engineering analysis, the RG indicates that consideration should be given to defense-in-depth and safety margins. One of the conditions for maintaining consistency with defense-in-depth philosophy is "Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided." The WOG-proposed approach includes compensating for increased UETs (i.e., periods of inadequate plant capability to withstand the peak pressures resulting from an ATWS) that results from using higher reactivity core designs by some form of plant configuration management. The current information provided by the WOG may not result in any limitation on unfavorable MTC, and concomitantly on UET. The WOG needs to justify how their approach satisfies the above condition for maintaining consistency with the defense-in-depth philosophy. The WOG also needs to address how the plant configuration management schemes should be controlled by the utility.

Response: Section 6.1 addresses the impact of the higher reactivity cores on defense-in-depth. Section 6.1 also addresses the elements that comprise defense-in-depth as defined in Regulatory Guide 1.174. Section 7 presents the configuration management approach that will be used to maintain defense-in-depth. It is important to note that the objective of the configuration management program is to operate the plant in a configuration that preserves defense-in-depth. As noted in Section 7, this requirement is not the result of a need to address a large impact on risk. The analysis in Section 5 demonstrates that the risk impact is small and the Section 5 analysis did not credit a configuration management program to show this.

With regard to the issue concerning over-reliance on programmatic activities to compensate for weaknesses in plant design, the following is provided in Section 6.1.

> *The core design will change such that higher RCS pressures will occur if an ATWS event occurs. The magnitude of the RCS pressure will depend on the time in life when it occurs and the availability of pressure relief and AFW, and available reactivity insertion. All safety systems, including the RPS, AFW system, RCS pressure relief capability, and rod control system will continue to function in the same manner with the same reliability, and there will be no additional reliance on additional systems or operator actions. The impact on risk is very small, but depending on the plant configuration, there could be an impact on defense-in-depth. This will be compensated for by plant configuration management programs that improve the preventive aspect or alternate mitigative capabilities as discussed in Section 7.*

Control of the configuration management program will depend on the licensee. It is expected that the program will be incorporated into the plant's program to address (a)(4) of the Maintenance Rule. This requirement states:

> *Before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.*

Plants with a UET > 5% for the reference case (no rod insertion, 100% AFW, and no PORVs blocked) will develop a set of acceptable operating configurations that address maintaining defense-in-depth for ATWS events. An example is provided in Figure 7-1. This is also discussed in Section 10 (Reload Implementation Process).

Issue 11: The updated event tree and fault tree models used in support of the analyses were not provided. To aid the staff in future reviews, the updated models, the identification of the dominant cutsets and/or sequences, and the bases and/or references for the failure rates or unavailability values used for the basic events in the models should be provided.

Response: The updated event trees and fault tree models are provided on the enclosed disk. The output files for the base quantifications for the low, high, and bounding reactivity cores are also provided on the disk. The basic event failure rates and unavailability values are provided in Section 5 of the WCAP.

Appendix C
Issues Identified in NRC letter "Westinghouse Owners Group Risk-Informed Anticipated Transient without Scram Approach"

WOG Responses are Provided for Each Issue

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.

Response: In the WOG approach, if the RCS pressure exceeds 3200 psig, core damage is assumed to occur. If the pressure remains below 3200 psig, the RCS remains intact and mitigative functions remain operable. The function of particular interest after the pressure transient has been mitigated is boration. Check valves connecting the boration system to the RCS, and providing part of the RCS pressure boundary, will be exposed to the 3200 psig RCS pressure. The following provides the assessment regarding the operability for these valves following the pressure transient.

An evaluation of the charging line check valves described below has been performed to determine the effect of an ATWS transient on their integrity and operability. The ATWS transient would result in a 3200 psig differential pressure across the disc at a maximum fluid temperature of 550°F with the valves closed. The evaluation included the effect of the transient on the pressure boundary components which consist of the main flange joint, bonnet or cover, and disc to determine if the valve would function following the transient.

Base on the evaluation of the four design configurations that makeup a majority of the installed valves, the valves should operate following the ATWS transient condition. Following the transient some seat leakage may be present which may require the valve be disassembled and inspected.

Following is a description of the four design configurations and with a summary of each evaluation.

1. Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves
2. Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves
3. Velan 3" Swing Check Valves built to drawing 78409 (B10-3114B-13MS)
4. Velan 3" Swing Check Valve built to drawing 78431

1. Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves

These swing check valves were manufactured to the requirements of the ASME Code Section III Class 1 requirements and supplied over a range of applicable Codes of construction. Therefore, the effect of the applicable Codes of construction was considered in the evaluation. Also, the design changes, i.e., main flange bolting hole depth, etc., to the valve model has been taken into account in the evaluation. In addition to the Code of construction and the design changes, the valve main flange bolting remains SA-453 Gr 660 material, the bonnet (cover) is SA-240 Type 316 or SA-182 F316 material and the disc is SA-182 F316 material.

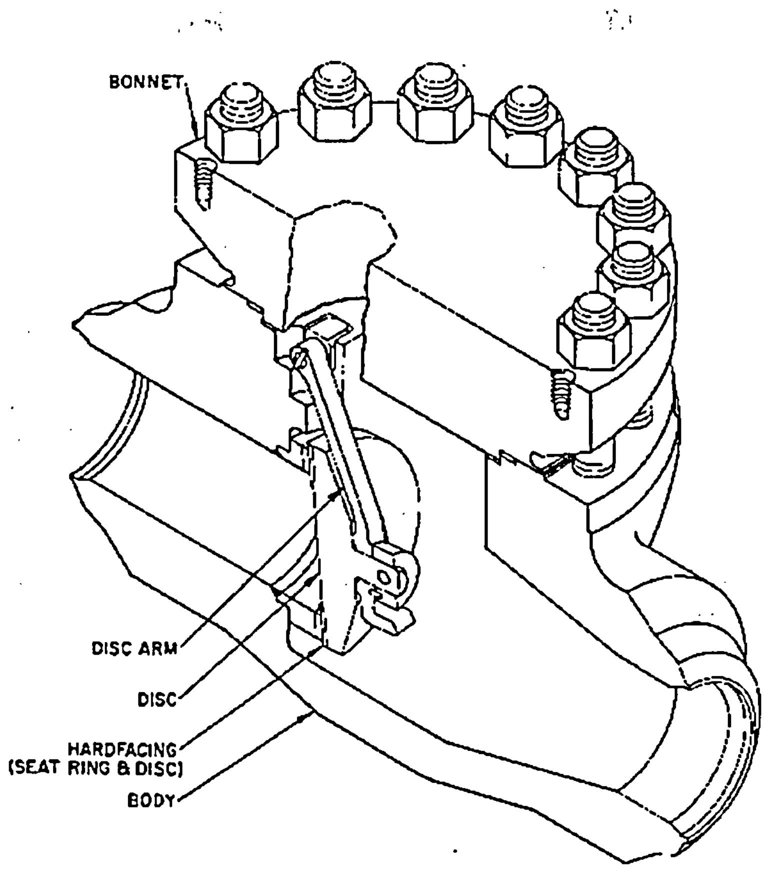

Figure C-1 Westinghouse Style A Swing Check Valve Models 03000CS88 and 04000CS88

Main Flange Joint

The main flange joint was evaluated using the methodology from the original design analysis of the valve that was derived from the ASME Code Section III. This methodology was applied to the analysis of the main flange joint at the ATWS transient conditions to determine if the main flange and main flange bolting stresses were acceptable. The evaluation concluded that the main flange and bolts stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Bonnet (Cover)

The cover was evaluated using classical methods used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the cover stresses are less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Disc

The disc was evaluated in the closed position using classical methods based on a flat plate theory used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Summary

The results of the evaluation show that the Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves should operate following the ATWS transient condition described.

2. Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves

These swing check valves were manufactured to the requirements of the ASME Code Section III Class 1 requirements and supplied over a range of applicable Codes of construction. Therefore, the effect of the applicable Codes of construction was considered in the evaluation. Also, the design changes, i.e., main flange bolting hole depth, etc., to the valve model has been taken into account in the evaluation. In addition to the Code of construction and the design changes, the valve main flange bolting remains SA-453 Gr 660 material, the bonnet (cover) is SA-240 316 or SA-182 F316 material and the disc is SA-564 Gr 630 H1150 material.

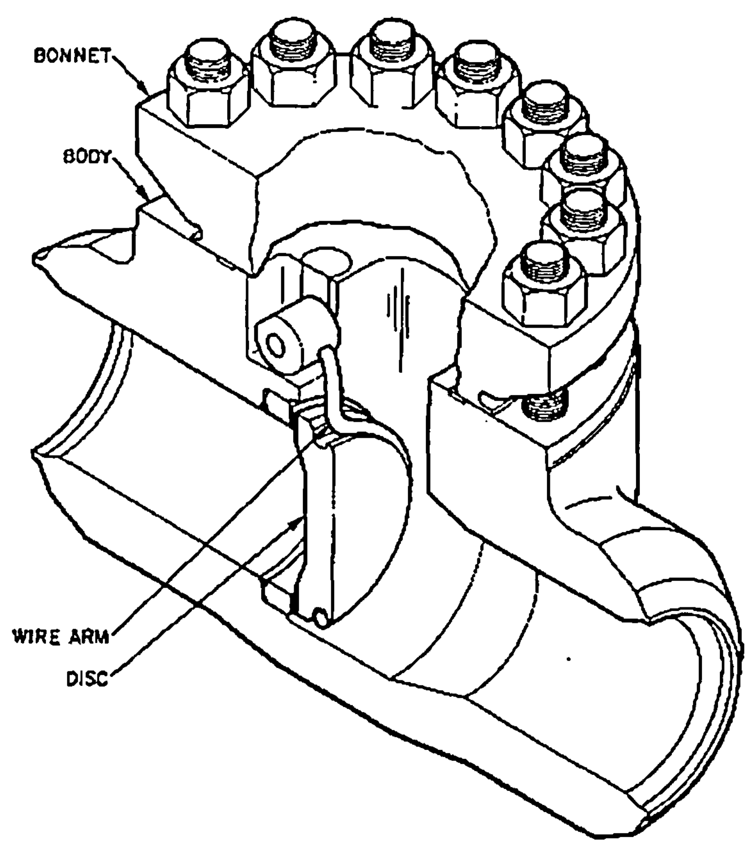

Figure C-2 Westinghouse Style B Swing Check Valve Models 03001CS88 and 04001CS88

Main Flange Joint

The main flange joint was evaluated using the methodology from the original design analysis of the valve that was derived from the ASME Code Section III. This methodology was applied to the analysis of the main flange joint at the ATWS transient conditions to determine if the main flange and main flange bolting stresses were acceptable. The evaluation concluded that the main flange and bolts stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Bonnet (Cover)

The cover was evaluated using classical methods used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the cover stresses are less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Disc

The disc was evaluated in the closed position using classical methods based on a flat plate theory used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Summary

The results of the evaluation show that the Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves should operate following the described ATWS transient condition.

Velan 3" Swing Check Valves (B10-3114B-13MS) built to drawings 78409 and 78431

A review of records show that the swing check valves have had some design and material changes to the pressure retaining parts since the time of initial construction. Major design changes have been made to the disc that could affect the results of this evaluation. These changes consist of design changes to the configuration of the disc and the materials used for its construction. Therefore, the evaluation considered the various disc design evolutions that could be installed in these valves.

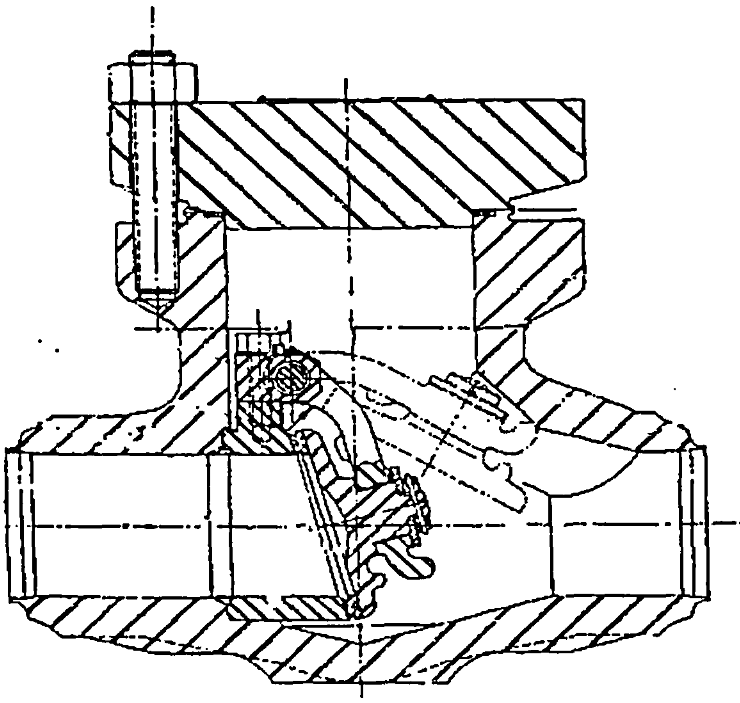

Figure C-3 Velan Swing Check Valve

3. Velan Swing Check Valves built to drawing 78409

These valves were built prior to the use of the ASME Code Section III. They were manufactured to the requirements of ANSI B16.5.

Following are the major components used in this valve design.

Component	Part Number(Drawing)	Material
Body	7904-2-13 (Dwg 79042)	A-182 F316
Cover	8164-5-13 (Dwg 8164-004)	A-182 F316
Disc- Forged*	8204-5-35 (Dwg 8204-15)	A-182 F316
Disc – Cast	8204-5-35 (Dwg 82045)	A-351 CF8 casting pattern 82044E
Main Flange Bolts	9144-2-54 (3/4"-10UNC x 4.25" lg)	A-193 Gr B7

* Forged discs were supplied as replacement parts.

Main Flange Joint

The main flange joint was evaluated using the methodology in the ASME Code Section III for flanged joints at the ATWS transient conditions, and the flange and bolts stresses were less than normal ASME Code allowable stresses.

Cover (Bonnet)

The cover was evaluated using the design rules in the ASME Code Section III for blind flanges at the ATWS transient conditions. The evaluation concluded that the cover stresses were less than normal ASME Code allowable stresses.

Disc

There are various design disc configurations that could be installed in the valves. They include both cast and forged discs.

- Cast Disc

The cast disc identified above was evaluated by both elastic and plastic analysis. The conclusion of the elastic analysis is that, while the stresses in the disc exceed normal allowable stresses at 3200 psig and 550°F, the stresses do not exceed faulted condition allowable stresses at that temperature. Since the stresses exceed normal allowable stresses localized yielding may occur that could result is seat leakage, but the valve should still function as required. The stresses at the plant design conditions do not exceed normal allowable stresses.

The plastic analysis provided further evidence that the cast disc identified above will survive the ATWS transient.

Older revisions of the drawing may be present in some of the valves, but sufficient detail of the actual configuration of the pattern is not available to determine the effect of the pattern changes have on the analysis.

- Forged Disc

The replacement part forged disc was evaluated in the closed position using classical methods based on a flat plate theory at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code allowable stresses.

Summary

The results of the evaluation show that the Velan swing check valves built to drawing 78409 (B10-3114B-13MS) should operate following the ATWS transient condition described.

4. Velan Swing Check Valves built to drawing 78431

These valves were built to the requirements of the ASME Code Section III.

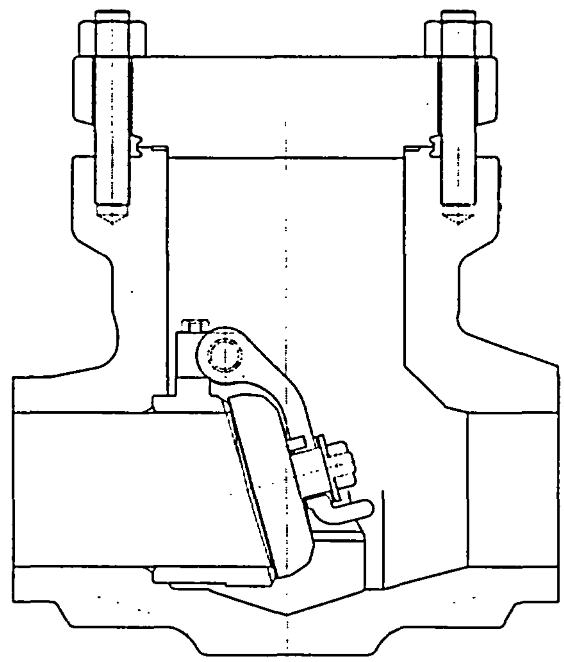

Figure C-4 Velan Swing Check Valve

Following are the major components used in this valve design.

Components	Part Number (Drawing)	Material
Body	8155-19-13 (Dwg 81551)	SA/A 182 F316
Cover	8164-7-13 (Dwg 81647)	SA/A-182 F316
Disc- Casting	8204-7-139 (Dwg 82047 Pattern 82044E)	SA/A-351 CF3M
Disc – Forged*	8204-5-35 (Dwg 8204-15)	A-182 F316

* Forged discs were supplied as replacement parts.

Main Flange Bolts	8244-3-163 (3/4"-10UNC x 4.25" Lg)	SA/A-453 Gr 660

Main Flange Joint

The main flange joint was evaluated using the methodology in the ASME Code Section III for flanged joints at the transient conditions, and the flange and bolts stresses were less than normal ASME Class 1 Code allowable stresses.

Cover (Bonnet)

The cover was evaluated using the design rules in the ASME Code Section III for blind flanges at the transient conditions. The evaluation concluded that the cover stresses were less than normal ASME Code allowable stresses.

Disc

There are various design disc configurations that could be installed in the valves. They include both cast and forged discs.

- Cast Disc

The cast disc identified above was evaluated by both elastic and plastic analysis. The conclusion of the elastic analysis is that, while the stresses in the disc exceed normal allowable stresses at 3200 psig and 550°F, the stresses do not exceed faulted condition allowable stresses at that temperature. Since the stresses exceed normal allowable stresses localized yielding may occur that could result is seat leakage, but the valve should still function as required.

The plastic analysis provided further evidence that the cast disc identified above will survive the ATWS transient.

Older revisions of the drawing may be present in some of the valves, but sufficient detail of the actual configuration of the pattern is not available to determine the effect of the pattern changes have on the analysis.

- Forged Disc

The forged disc was evaluated in the closed position using classical methods based on a flat plate theory at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code allowable stresses.

Summary

The results of the evaluation show that the Velan swing check valves built to drawing 78431 should operate following the described ATWS transient condition.

Issue 2: MTC/UET

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 of nominal power. The -7 pcm/°F was determined to be the point at which the core conditions become unfavorable. Under the approach proposed by the WOG, the values of 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.

Response: As part of the Reload Safety Evaluation process, the reactivity coefficients for each reload core are evaluated to ensure that they are within the bounding values assumed in the reference safety analyses. This process is described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology." The reload core evaluations include moderator coefficients, Doppler temperature coefficients, and Doppler-only power coefficients. Most negative and least negative limits are checked since, depending on the transient, strong or weak reactivity feedback may be limiting. For plants with positive moderator temperature coefficient technical specifications, the reference safety analyses assume the technical specification limit values for those transients where least negative moderator feedback is limiting (e.g., heatup transients). The MTC values are evaluated for each reload core design to ensure that the MTC technical specification limit bounds the expected MTC values.

Even for cores with positive moderator temperature coefficient technical specifications, the total power coefficient will be significantly negative over the power operating range. For example, total power coefficients were examined for the bounding reactivity core at 2000 MWD/MTU, the cycle burnup where the MTC reached the tech spec limit of +7 pcm/°F. At HFP, equilibrium xenon, nominal conditions, the total power coefficient (TPC) was -9.4 pcm/% power. At HZP, the TPC was -10.5 pcm/% power. Consequently, despite the positive moderator temperature coefficient, the overall reactivity feedback due to an increase in power was negative over the power operating range.

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

PRA: Control rod insertion success, 100% AFW
PRB: Control rod insertion success, 50% AFW
PRC: Control rod insertion failure, 100% AFW
PRD: Control rod insertion failure, 50% AFW

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".

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

PR: No control rod insertion, No AMSAC (no AFW)

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".

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

PRA: Control rod insertion success, 100% AFW
PRB: Control rod insertion success, 50% AFW
PRC: Control rod insertion failure, 100% AFW
PRD: Control rod insertion failure, 50% AFW

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".

Appendix G
Issues Identified in the NRC Letter "WCAP-15831-P, "WOG Risk-Informed ATWS Assessment and Licensing Implementation Process," July 2002 – Withdrawal of Previous Request for Additional Information (RAI) and Request for Further Clarifications Needed to Revise the Topical Report (TAC No. MB5751)," dated March 25, 2004

WOG Responses are Provided for Each Issue

Response to NRC's Request for Further Clarifications

TECHNICAL

Technical Clarification 1. To respond to the staff's concern about the potential degradation of defense-in-depth, Section 7 of WCAP-15831-P discusses an anticipated transient without scram (ATWS) configuration management approach (i.e., Approach 1) that can be implemented by utilities. The topical report (TR) provided limited details regarding the development and implementation of this approach. During meetings with the Westinghouse Owners Group (WOG), the staff agreed to review a detailed description of this approach for ensuring defense-in-depth capabilities. The staff has compiled the following topics that the WOG should address in its revision to WCAP-15831-P.

Technical Clarification 1.a. The ATWS configuration management program should have as its fundamental goal, minimizing unfavorable exposure time (UET) conditions at Westinghouse plants. The TR should describe how this program will be managed, controlled, implemented, and verified to ensure UETs are minimized.

Response: The objective of the ATWS Configuration Management Program (CMP) is to operate the plant in a configuration that maintains defense-in-depth to ATWS events, that is, the configuration is favorable to ATWS pressure transient mitigation. It is acceptable to operate in an unfavorable configuration, prior to implementing compensatory actions, for a cumulative time that will be specified as part of the ATWS CMP.

The details of the ATWS CMP, with regard to how this program will be managed, controlled, implemented, and verified, will be specified on a plant specific basis. The revision to the WCAP will provide high level guidance that will be followed by licensees to ensure the plant is operating in favorable ATWS configurations, consistent with the time in the cycle, and implementing appropriate compensatory actions if the cumulative unfavorable configuration time exceeds an acceptable limit. The WCAP will specify acceptable compensatory actions and the cumulative time a plant is allowed to operate in unfavorable conditions, in addition to implementation requirements to ensure the program is managed and controlled appropriately. The WCAP will also provide the detailed approach for calculating unfavorable exposure times.

In general, the ATWS CMP will have the following key characteristics.

ATWS CMP Key Characteristics

Plants will be divided into the following three groups based on consistency with the ATWS Rule and if the plant has a diverse scram system (DSS).

- Group 1: Plants with a DSS
- Group 2: Plants without a DSS, consistent with the ATWS Rule (installed AMSAC) and the basis for the ATWS Rule

- Group 3: Plants without a DSS, consistent with the ATWS Rule (installed AMSAC), but not the basis for the ATWS Rule

A plant consistent with the basis for the ATWS Rule will have either:

- a core design limit on UET of < 5% for the ATWS Rule reference configuration/condition of no control rod insertion, all auxiliary feedwater (AFW) available, and no PORVs blocked, or

- an MTC of < -8 pcm/°F for 95% of the cycle.

Plants in Groups 1 or 2 will not be required to implement the ATWS CMP.

The ATWS CMP key characteristics are divided into three areas; overall structure and administrative control, compensatory actions, and time allowed in an unfavorable condition. Each is discussed in the following paragraphs.

<u>Overall CMP Structure and Administrative Control</u>

By controlling the plant configuration, plants can maintain ATWS defense-in-depth capabilities. Plants can manipulate the plant configuration to ensure they are operating with favorable conditions with regard to UETs, and therefore ATWS events, by limiting the unavailability of systems important to ATWS event mitigation. Limitations on plant configuration vary depending on the time in the cycle and become less restrictive further into the cycle.

Configuration restrictions are proposed to address possible degradation of defense-in-depth. The time in life when the plant mitigation systems cannot relieve sufficient RCS pressure is dependent on core design, time in core life, and the availability of control rod insertion, pressure relief, and AFW. Table 7-1 of the WCAP presents UET information for a high reactivity core in the form of acceptable plant configurations for different times during the fuel cycle. This was developed from the UETs provided on Tables 4-7 and 4-8 of the WCAP. In this case, defense-in-depth is the basis for acceptable configurations. This table defines the plant configurations required to maintain defense-in-depth, with regard to ATWS, at different times in the cycle. The information in this table can be used to schedule acceptable times for removal of equipment from service. It should be noted that for the situation presented on Table 7-1, no credit for control rod insertion is given if the rod control system is in manual.

Based on this, an ATWS CMP can be developed that is able to identify plant configurations that are acceptable or unacceptable with regard to maintaining defense-in-depth for ATWS events as plant configurations change with the time in the cycle. This will require licensees to have either cycle specific UETs or conservative UETs. Note that conservative UETs will overestimate the time of unfavorable exposure and may cause a plant to take unnecessary actions.

The ATWS CMP will have the following capabilities:

- Identify plant configurations (unfavorable configurations) that do not maintain defense-in-depth to an ATWS event.

- Track the time for individual occurrences when the plant is in an unfavorable plant configuration.
- Track the cumulative time per cycle when the plant is in an unfavorable plant configuration.
- Provide information on the length of time remaining in the UET for plant configurations.
- Provide compensatory actions to take if the unfavorable condition cannot be exited prior to expiration of the time allowed in the unfavorable configuration.

To maintain the proper level of control over the ATWS CMP and its use, it can be integrated into the Configuration Risk Management Program (CRMP) developed by utilities in response to the Maintenance Rule. The CRMP is typically contained within a plant's Technical Requirements Manual (TRM) or within plant procedures. The combined CRMP/ATWS CMP will be able to track the status of the ATWS mitigation systems and identify when the plant enters unfavorable configurations, and also track the cumulative time in these configurations.

Compensatory Actions

If a plant enters an unfavorable configuration, there are several compensatory actions that can be taken. Any one or a combination of these actions can be used to address the ATWS defense-in-depth concern. The proposed compensatory actions follow. Licensees can use those that provide the appropriate benefit.

1. Back-up Reactor Trip: Implement an alternate method to trip the reactor based on removing power to the control rod drive mechanisms (CRDMs). This requires an operator action, that can be taken from the control room in a short time, to interrupt power to the motor-generator sets (of the CRDMs) or interrupt power from the motor-generator sets (of the CRDMs) to the CRDMs. This would provide a backup reactor trip signal that is diverse from the reactor protection system (RPS). The only common components will be the sensors and isolators that provide input to the RPS and control board indication. This operator action will need to be listed early in the plant's emergency operating procedures and the operators will need to be trained on the action.

   As an alternative to a back-up reactor trip from the control room, a utility may consider locating a dedicated operator at the MG sets if an unfavorable configuration exists beyond an acceptable time period. This would be beneficial for short durations, but may not be feasible for an extended time frame.

   Once this compensatory action is implemented, further tracking of the time operating in an unfavorable configuration is not necessary.

2. UET Re-calculation: Re-calculation of UETs can be done based on plant specific information using analysis enhancements which may provide a better (shorter) estimate of the UETs. For example, if a plant is using a generic set of UETs for a representative plant that is similar in design, but not identical, it may be possible to complete plant specific analysis that will provide shorter UETs. In addition, depending on the end-of-cycle burn-up assumptions for the previous cycle, using the actual end-of-cycle burn-up may also provide a benefit.

3. Power Reduction: The plant power level can be reduced to a level where the plant configuration becomes favorable. With the lower power level, RCS pressures following an ATWS event can be mitigated with reduced pressure relief capability. The plant can then operate at this reduced power level until the configuration becomes favorable as the time into the cycle increases.

Time Allowed in an Unfavorable Configuration

A 30-day cumulative time limit in an unfavorable condition is proposed. In some cases this length of time will provide sufficient time for the plant to exit the unfavorable configuration as the cycle progresses. This time can also be used to implement appropriate compensatory actions. The 30-day limit is based on the following:

- For a 500-day fuel cycle, a 5% UET would equate to 25 days. This is based on the 5% UET requirement placed on the Braidwood and Byron core designs.

- The industry, along with the NRC, is currently developing a risk-informed Technical Specification directed at flexible allowed outage times (AOTs) with a 30 day backstop. In this activity, a maximum time of 30 days would be allowed to return Technical Specification equipment to operable status if the risk analysis supports it.

- The risk associated with blocking a PORV is low. Following the approach in Regulatory Guide 1.177 for setting Technical Specification AOTs, a time of over 3000 hours can be justified via the risk analysis for blocking a PORV (see Sections 5.1.7 and 5.2.2 of the WCAP).

Note that this 30 day time period is cumulative.

Technical Clarification 1.b. The TR should clearly describe the methodology licensees would use to develop a plant- and cycle-specific ATWS configuration management program. Specifically, the TR should describe how the UETs would be calculated based on the plant- and cycle-specific parameters, such as core design and current cycle operating history (e.g., downpowers, shutdowns, etc.), and how these calculations will be controlled and verified prior to, during, and following plant startup.

Response: The response to Technical Clarification 5.d provides the requested information.

Technical Clarification 1.c. The plant-specific reload analysis for each cycle should ensure that the UET is minimized based on established criteria for specific conditions. For example, a 0 percent UET could be set as a core reload acceptance criteria based on the following assumed conditions:

i. Hot full power moderator temperature coefficient;
ii. Equilibrium xenon;
iii. Nominal hot full power inlet temperature;
iv. One minute of automatic control rod insertion (CRI) (i.e., 72 steps of insertion of the lead bank);
v. All power operated relief valves (PORVs) operable; and
vi. 100 percent auxiliary feedwater (AFW) flow available.

Response: The approach to maintain defense-in-depth, or ATWS pressure transient mitigation capability, is to operate the plant in a configuration with a zero UET. Using this approach, the reload analysis will need to ensure that at least one plant configuration, with regard to CRI, AFW, and PORV availability, will have a zero UET. It is agreed that the six conditions listed above are the appropriate core reload acceptance criteria. As a clarification to Condition iv, 72 steps of control rod insertion of the lead bank is the key part of this condition. Specification of one minute of automatic rod insertion is not necessary.

Technical Clarification 1.d. The configuration management program should be based on the effective full power days of operation at the plant.

Response: The WOG agrees that effective full power days of operation should be the basis for the ATWS CMP.

Technical Clarification 1.e. The configuration management program should be designed to prevent the voluntary entry into an UET. The performance of routine surveillances and routine maintenance or testing could be reasonably scheduled and performed either prior to or after time intervals where it would cause entry into an UET. The TR should establish the criteria or conditions governing voluntarily entry into an UET. This includes criteria such as the controls and limitations on these entries, when these voluntary entries will be allowed or not allowed, what compensatory actions will be implemented prior to and during any planned voluntary entries, how long specific voluntary entries will be allowed, and what if any compensatory actions may be taken to allow an extension of the voluntary entry.

Response: Entries into unfavorable configurations to meet Technical Specification surveillance requirements and repair inoperable equipment are acceptable. Entries into unfavorable configurations to complete preventive or routine maintenance activities should be minimized. Some of the equipment important to mitigation of an ATWS pressure transient is also important to mitigation of design basis events. These design basis events typically are larger contributors to plant risk than the ATWS event, therefore, it is important to maintain the equipment operability for design basis event mitigation. The surveillance requirements demonstrate component operability, therefore, it is recommended that they continue to be completed at the specified interval. Similarly, inoperable components should be repaired to maintain design basis event mitigation capability.

If component inoperability, due to surveillance requirements, maintenance activities, or repair activities, moves the plant into an unfavorable configuration, then simultaneous test and maintenance activities that compromise the availability of the reactor protection system (reactor trip signals, in particular) or that place the plant in a higher trip potential configuration should be rescheduled when the plant returns to a favorable configuration.

The time a plant may enter an unfavorable configuration to meet a surveillance requirement is relatively small, as demonstrated in the following. The systems/components of interest are:

- Auxiliary feedwater
- Pressurizer PORVs
- Pressurizer PORV block valves
- Pressurizer safety valves
- Rod control system

- AMSAC
- Turbine trip (steam stop valves and steam control valves)

The surveillance requirements identified in the Technical Specifications (NUREG-1431) for these systems/components along with surveillances required by plant procedures, from a generic perspective, are summarized on Table G-1. Similar information is provided on Table G-2 for Braidwood including the times the surveillance requirements cause the system/component to be inoperable. It is concluded from this that the time these systems/components are unavailable to meet surveillance requirements is small in comparison to the (proposed) 30-day total time allowed in an unfavorable configuration. Therefore, it is proposed that this time does not need to be tracked against the total time allowed in unfavorable configurations, if it does place the plant in an unfavorable configuration.

Technical Clarification 1.f. Entry into an UET should be permitted for performance of actions necessary to startup the plant and ensure proper operability and testing of equipment (e.g., manual rod control during startup to enable rod operability testing, etc.) However, these actions should be scheduled and controlled to prevent excessive entry into UET conditions.

Response: Startup testing is necessary to ensure equipment is operable to meet design basis event mitigation assumptions. There are no plans to modify startup testing or the startup process based on ATWS concerns. The applicable UETs during a startup are those without equilibrium xenon and the configuration is unfavorable at the beginning of the cycle for all configurations for the low, high, and bounding reactivity cores (see Tables 4-5, 4-6, 4-9, 4-10, 4-13, and 4-14 of the WCAP). Therefore, at the beginning of the cycle and without equilibrium xenon, delaying removal of equipment from service based on ATWS considerations provides no benefit. But general good practices will be followed to maintain ATWS prevention and mitigation capabilities during startup.

Technical Clarification 1.g. With the understanding that unforeseen circumstances may arise, the configuration management program should limit the time UET conditions are permitted to persist for events beyond the control of the licensee (i.e., equipment failure, emergency actions, etc.). Procedures should be developed and actions should be identified such that the plant will exit an UET condition in a timely fashion. A list of proposed actions and procedures should be developed and described in the TR to ensure that time spent in an UET condition by a licensee is limited. For example, controls similar to technical specification limiting conditions of operation, action statements, and surveillance requirements (SRs) should be clearly identified to allow proper monitoring of UET conditions and minimize operations under UET conditions. Additionally, the staff requests that these compensatory actions be clearly described and supported by technical justifications which ensure that UETs are minimized.

Response: The response to Technical Clarification 1.a provides high level guidance on the time limit a plant will be allowed to operate in unfavorable configurations and also the compensatory actions to be taken if a plant exceeds this time limit. Controls (procedures and actions) following this high level guidance will be developed on a plant specific basis and included in the appropriate plant document(s). This level of detail will not be added to the WCAP. In addition, there are no plans to develop additional limiting conditions of operation, action statements, or surveillance requirements for inclusion in Technical Specifications.

Technical Clarification 1.h. The configuration management program needs to consider not just maintenance-related activities, which seems to be the only focus under the current approach, but also any time ATWS-related components/systems are out of service, unavailable, or not in their expected state/condition (e.g., testing, discovery of inoperable or failed conditions) such that they are unable to perform their functional response to an ATWS event.

Response: The WOG agrees with the NRC's comment. It is not the intent of the ATWS CMP to consider only maintenance-related activities, but all activities (testing, preventive maintenance, and repair) that cause the systems important to ATWS mitigation to be unavailable.

Technical Clarification 1.i. The TR should explicitly address how plants will respond to conditions in which the ATWS-related equipment is unavailable, as identified above. The staff does not accept the concept that there are no situations that may require changing operation to a plant mode where ATWS events are no longer applicable, such as moving to Mode 3. There should be administrative requirements to proactively respond to these conditions to minimize and/or eliminate the UET, which may include actions to lower power, shutdown, extend an outage, or terminate start-up, as appropriate.

Response: As discussed in the response to Technical Clarification 1.a, the WCAP will provide the high level guidance that licensees will use to develop detailed plant specific guidance. A plant will identify compensatory actions to take when the cumulative time allowed in unfavorable conditions is exceeded. These actions include one or more of the following:

- Implement a back-up reactor trip actuated by operator action
- Perform UET re-calculations
- Initiate a power reduction

The plant power level can be reduced to a level where the plant configuration becomes favorable. With the lower power level, RCS pressures following an ATWS event can be mitigated with reduced pressure relief capability. The plant can then operate at this reduced power level until the configuration becomes favorable as the time into the cycle increases. The response to Technical Clarification 7 provides additional information regarding UETs for lower power levels.

Plant startups will be done consistent with current plant procedures and Technical Specifications. If a plant starts up with equipment inoperable, such that the configuration is unfavorable, startup can continue consistent with plant procedures and Technical Specifications, and the time in the unfavorable configuration will be accumulated against the total time allowed in an unfavorable configuration.

Plant shutdown is not an acceptable alternative. From a risk perspective, the risk level associated with a plant shutdown and the following startup, although small, is not zero and comparable to the ATWS risk from continued at-power plant operation.

Technical Clarification 1.j. The SRs that will be implemented in support of the ATWS configuration management approach should be identified and justified as acceptable in periodically assuring that ATWS-related equipment is available and functional, consistent with the cycle-specific ATWS configuration management approach matrix.

Response: No additional surveillance requirements will be required to demonstrate operability of the equipment monitored with the ATWS CMP that is required to maintain ATWS defense-in-depth capability. The equipment of interest and the current surveillance requirements are:

- Auxiliary feedwater: Technical Specification (TS) Surveillance Requirements 3.7.5.1, 3.7.5.2, 3.7.5.3, 3.7.5.4, and 3.7.5.5
- Pressurizer PORVs: TS Surveillance Requirements 3.4.11.2, 3.4.11.3, and 3.4.11.4
- Pressurizer PORV block valves: TS Surveillance Requirements 3.4.11.1, and 3.4.11.4
- Pressurizer safety valves: TS Surveillance Requirements SR 3.4.10.1
- Rod control system: TS Surveillance Requirement 3.1.4.2 and Plant Procedures
- AMSAC: Plant Procedures
- Turbine trip (steam stop valves and steam control valves): Plant Procedures

Note: The Technical Specification (TS) requirements are based on the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Vol. 1, Rev. 2, April 2001.

Technical Clarification 1.k. A detailed description of the training, tools, and procedures supplied to operators to permit them to assess the significance of plant operating conditions should be provided including time-in-life, equipment availability, and current cycle operating history (e.g., down-powers, shutdowns). From this description it should be clear that operators will have the necessary training and understanding to ensure that entry into UETs is to be minimized and that the prompt return to a non-UET condition is essential.

Response: Appropriate training will be provided to operators to ensure the ATWS CMP and associated compensatory actions are implemented correctly. This will include training in the following areas:

- Unfavorable exposure times as related to acceptable plant configurations and time in the cycle
- Equipment important to mitigating ATWS events
- Equipment important to preventing ATWS events
- Tracking UETs with effective full power days of operation
- Time allowed in unfavorable plant configurations and compensatory actions

A detailed description of the training, tools, and procedures that will be provided to operators will be developed on a plant specific basis following high level requirements specified in the WCAP.

Table G-1	General Assessment: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient		
System	**Surveillance Requirement Number[1]**	**Surveillance Requirement and Frequency**	**Component Availability to Respond to an ATWS Event during Performance of Surveillance**
Pressurizer Safety Valves	Tech Spec SR 3.4.10.1	SR: Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%. Frequency: In accordance with the Inservice Testing Program	This surveillance is performed with the plant shutdown, therefore, the safety valves will be available at power.
Pressurizer PORVs	Tech Spec SR 3.4.11.1	T.S. Notes: 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. SR: Perform a complete cycle of each block valve. Frequency: 92 days	The block valve will be closed for only a short time period since it is only required to be cycled. The PORVs will not be available during this surveillance.
	Tech Spec SR 3.4.11.2	T.S. Note: Only required to be performed in MODES 1 and 2 SR: Perform a complete cycle of each PORV. Frequency: [18] months	The block valve will be closed for this surveillance. The PORVs will not be available during this surveillance.
	Tech Spec SR 3.4.11.3	SR: [Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. Frequency: [18] months]	The PORVs will not be available during this surveillance if done at power.
	Tech Spec SR 3.4.11.4	SR: [Verify PORVs and block valves are capable of being powered from emergency power sources. Frequency: [18] months]	PORVs and block valves will be available during this surveillance.

Table G-1 (cont.)	General Assessment: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient		
System	**Surveillance Requirement Number[1]**	**Surveillance Requirement and Frequency**	**Component Availability to Respond to an ATWS Event during Performance of Surveillance**
AFW	Tech Spec SR 3.7.5.1	T.S. Note: [AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.] SR: Verify each AFW manual, power operated, and automatic valve in each water flow path, [and in both steam supply flow paths to the steam turbine driven pump,] that is not locked, sealed, or otherwise secured in position, is in the correct position. Frequency: 31 days	AFW will be available during this surveillance.
	Tech Spec SR 3.7.5.2	T.S. Note: [Not required to be performed for the turbine driven AFW pump until [24 hours] after > [1000] psig in the steam generator.] SR: Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head. Frequency: In accordance with the Inservice Testing Program.	These surveillances are performed on recirculation flow. AFW will not be available during this surveillance unless AFW start signals re-align valves for flow to the SGs.
	Tech Spec SR 3.7.5.3	T.S. Note: [AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.] SR: Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. Frequency: [18] months	This surveillance actuates components to their required position for AFW flow. AFW will be available during this surveillance.

Table G-1 (cont.) General Assessment: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient

System	Surveillance Requirement Number[1]	Surveillance Requirement and Frequency	Component Availability to Respond to an ATWS Event during Performance of Surveillance
	Tech Spec SR 3.7.5.4	T.S. Notes: • [Not required to be performed for the turbine driven AFW pump until [24 hours] after > [1000] psig in the steam generator.] • [AFW trains(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.] SR: Verify each AFW pump starts automatically on an actual or simulated actuation signal. Frequency: [18] months	If this surveillance is performed with the plant shut down, AFW will be available at power. If done at power, valves may re-align on AFW start signal.
	Tech Spec SR 3.7.5.5	SR: [Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator. Frequency: Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days]	This surveillance is performed prior to the plant entering Mode 2, therefore, AFW will be available at power.
Steam Stop & Control Valves	Plant Procedures	SR: Cycle the valves closed. Frequency: Based on turbine overspeed protection requirements.	The valves are cycled to the closed position which is the required position for tripping the turbine, therefore, valves will be available at power.

Table G-1 (cont.)	General Assessment: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient		
System	**Surveillance Requirement Number[1]**	**Surveillance Requirement and Frequency**	**Component Availability to Respond to an ATWS Event during Performance of Surveillance**
Rod Control System	Tech Spec 3.1.4.2	SR: Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core > 10 steps in either direction. Frequency: 92 days	Rod control system placed in manual during this surveillance, therefore, automatic rod control will not be available at power.
	Plant Procedures	SR: Axial flux difference calibration Frequency: Utility dependent	Rod control system placed in manual during this surveillance, therefore, automatic rod control will not be available at power.
	Plant Procedures	SR: Turbine governor valve testing Frequency: Plant specific	Rod control system placed in manual during this surveillance, therefore, automatic rod control will not be available at power.
AMSAC	Plant Procedures	SR: AMSAC testing Frequency: Plant specific	AMSAC will not be available during this surveillance.

Note:
1. Based on Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Vol. 1, Rev. 2, April 2001.

Table G-2 Braidwood Specific: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient			
System	**Surveillance Requirement Number[1]**	**Surveillance Requirement and Frequency**	**Time Component is Unavailable to Perform Surveillance**
Pressurizer Safety Valves	Tech Spec SR 3.4.10.1	SR: Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%. Frequency: In accordance with the Inservice Testing Program	0 hours; This surveillance is performed with the plant shutdown, therefore, the safety valves will be available .
Pressurizer PORVs	Tech Spec SR 3.4.11.1	T.S. Notes: Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. SR: Perform a complete cycle of each block valve. Frequency: 92 days	0.1 hours; The block valve will be closed for only a short time period since it is only required to be cycled. The PORVs will not be available during this surveillance.
	Tech Spec SR 3.4.11.2	T.S. Note: Only required to be performed in MODES 1 and 2 SR: Perform a complete cycle of each PORV. Frequency: 18 months	0.1 hours; The block valve will be closed during this surveillance. The PORVs will not be available during this surveillance.
	Tech Spec SR 3.4.11.3	SR: Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. Frequency: 18 months	0 hours; Surveillance is performed with the plant shutdown, therefore, the PORVs will be available .
AFW	Tech Spec SR 3.7.5.1	SR: Verify each AFW manual, power operated, and automatic valve in each water flow path that is not locked, sealed, or otherwise secured in position, is in the correct position. Frequency: 31 days	0 hours; AFW will be available during this surveillance.
	Tech Spec SR 3.7.5.2	SR: Verify day tank contains > 420 gal of fuel oil. Frequency: 31 days	0 hours; AFW will be available during this surveillance.
	Tech Spec SR 3.7.5.3	SR: Operate the diesel-driven AFW pump for > 15 minutes. Frequency: 31 days	0 hours; AFW will be available during this surveillance.

Table G-2 (cont.)	Braidwood Specific: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient		
System	**Surveillance Requirement Number[1]**	**Surveillance Requirement and Frequency**	**Time Component is Unavailable to Perform Surveillance**
	Tech Spec SR 3.7.5.4	SR: Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head. Frequency: In accordance with the Inservice Testing Program.	0 hours; AFW will be available during this surveillance.
	Tech Spec SR 3.7.5.5	SR: Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. Frequency: 18 months	0 hours; This surveillance actuates components to their required position for AFW flow. AFW will be available during this surveillance.
	Tech Spec SR 3.7.5.6	SR: Verify each AFW pump starts automatically on an actual or simulated actuation signal. Frequency: 18 months	0 hours; Valves closed during this surveillance automatically open on actual AFW/AMSAC actuation signal. AFW will be available.
	Tech Spec SR 3.7.5.7	SR: Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator. Frequency: Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days]	0 hours; This surveillance is performed prior to the plant entering Mode 2, therefore, AFW will be available at power.
	Tech Spec SR 3.7.5.8	SR: Verify fuel oil properties are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program Frequency: In accordance with the Diesel Fuel Oil Testing Program	0 hours; AFW components remain available during this surveillance.
Steam Stop & Control Valves	Plant Procedures	SR: Cycle the valves closed. Frequency: Based on turbine overspeed protection requirements	0 hours; The valves are cycled to the closed position which is the required position to trip the turbine, therefore, valves are available at- power.

Table G-2 (cont.) Braidwood Specific: Summary of Surveillance Requirements for Systems Important to Mitigation of the ATWS Pressure Transient

System	Surveillance Requirement Number[1]	Surveillance Requirement and Frequency	Time Component is Unavailable to Perform Surveillance
Rod Control System	Tech Spec 3.1.4.2	SR: Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core > 10 steps in either direction. Frequency: 92 days	1 hour; Rod control system placed in manual during this surveillance, therefore, automatic rod control will not be available.
	Plant Procedures	SR: Axial flux difference calibration Frequency: 4 channels @ 92 days per channel	< 0.5 hour/channel; Rod control system placed in manual during this surveillance (only while the NI channel is being placed in or taken out of test). Therefore, automatic rod control will not be available.
	Plant Procedures	SR: Turbine governor valve testing Frequency: 92 days	1 hour; Rod control system placed in manual during this surveillance, therefore, automatic rod control will not be available.
AMSAC	Plant Procedures	SR: AMSAC testing Frequency: Semi-annual	4 hours; AMSAC will not be available during this surveillance.

Note:
1. Based on Braidwood Unit 1 and Unit 2 Technical Specifications.

<u>Technical Clarification 2</u>. Since an ATWS is a beyond design basis accident, crediting the rod control system for limiting the peak pressures experienced may be acceptable to the staff. However, the rod control system is a control system that it is not assumed to mitigate the consequences of Chapter 15 design basis transients and accidents. Therefore, the TR should contain sufficient information to provide the staff with a reasonable assurance that the rod control system will function as assumed in WCAP-15831-P. Additionally, with regard to the assumptions used in Item 1c to determine the plant- and cycle-specific UET, the 72-step automatic CRI credit is contrary to the assumption of no CRI credit used in the basis for the ATWS rule. In order for the staff to find that this credit is acceptable, the WOG should provide technical analyses demonstrating that the rod control system will be capable of performing the required mitigative safety function under core conditions representative of an ATWS, i.e., high temperatures and pressures.

Response: The WOG model credits 72 steps, in some sequences, from the lead control rod bank in response to an ATWS event. In response to the increase in RCS temperature, as measured by the resistance temperature detectors (RTDs), the rod control system will begin stepping the control rods into the core. The only components of the rod control system that will be exposed to the high temperature RCS environment are the RTDs. The RTDs provide the T_{avg} that is used in conjunction with the T_{ref} to control rod movement. The T-hot and T-cold RCS measurements are used to calculate the T_{avg} signal in the process equipment. The signal to the rod control system will saturate at about 630°F and then continue to provide a constant signal. The T_{avg}/T_{ref} temperature difference will continue to be great enough for the rod control system to continue to move the rods in at 72 steps per minute.

This type of response has been verified for the 7300 and Eagle-21 process protection systems.

Technical Clarification 3. In Section 8.2.4, it is stated that "[r]egardless of whether this action succeeds or fails, the ATWS event can be mitigated depending on the availability of AFW and RCS pressure relief." This statement and resulting logic modeling does not appear to address the conditions for some fuel designs (e.g., bounding reactivity) in which the UET exists even with all equipment available with the exception that the rod control system is in manual (cf. Table 4-36). By definition, for the fuel designs that create an UET even with all equipment available and the rod control system in manual, a success state cannot be achieved if top events reactor trip (RT), operator action to M-G set (OAMG), and CRI all fail (or if top event [control rod] CR is failed by itself). The text and ATWS event tree logic models should be revised to address these potential fuel design-specific conditions. Also, identify if there are any other situations in which the ATWS event tree logic is not consistent with any of the analyses presented in Chapter 4 and the resulting ATWS configuration management approach.

Response: It is agreed that if CRI fails (automatic or manual action to drive in the control rods), then successful ATWS event mitigation may not be possible, depending on the particular core design, time in the cycle, and AFW and pressure relief availability. In the extreme case for a core design has non zero UETs for all twelve plant configurations (for the various combinations of control rod insertion, AFW flow, and PORV availability), then the ATWS pressure transient cannot be mitigated for some part of the cycle regardless of the plant configuration. As the event tree is currently constructed, the top event PR (primary pressure relief) accounts for this possibility. Similarly for a core design that has non zero UETs for all plant configurations except the single condition with CRI success (72 steps of control insertion), 100% AFW flow, and no PORVs blocked. If CRI fails, then the ATWS pressure transient cannot be mitigated. But again, the top event PR accounts for this possibility.

The words in Section 8.2.4, and also Section 5.1.1.5, will be changed to state that with some core designs CRI success is necessary during parts of the cycle to achieve successful pressure mitigation. It will also be noted that it is possible to design a core that for part of the cycle the ATWS pressure transient cannot be mitigated regardless of the success or failure of CRI. No changes to the event tree logic are required since the top event PR accounts for the UETs and ATWS pressure mitigation equipment availability.

<u>Technical Clarification 4.</u> With regard to Sections 4.1 through 4.5 of WCAP-15831-P, the WOG should provide a listing of key assumptions and plant conditions used in performing the deterministic analyses. Specifically, a table should be provided that contains the same parameters as those listed in Tables 3.1 and 3.2 of NS-TMA-282, "ATWS Submittal," dated December 30, 1979. To provide a good comparison of the data, this table should consist of columns containing the parameter information for the following cases: (1) the bounding reactivity model described in WCAP-15831-P, and (2) the most limiting WOG 4-loop, 3-loop, or 2-loop plant intended to be covered by WCAP-15831-P. For Case 2, the TR should justify why the particular plant chosen is the most limiting. Additionally, for each parameter in Case 2 which is not bounded by the value used in Case 1, a justification should be provided for the difference that demonstrates that the results obtained in WCAP-15831-P are bounding for this plant.

Response: The WCAP analyses are not meant to be bounding analyses that eliminates the need for plant specific evaluations. The WCAP analyses examined a wide range of core designs to demonstrate that even with a core design with the maximum part-power positive MTC Technical Specification limit licensed for Westinghouse plants, that the ATWS risk is very small. In addition, this showed that the risk impact of moving to such a core, from a core that meets a 5% UET for the reference conditions (no rod insertion, 100% AFW, no PORVs blocked) is also very small.

The term "bounding core" was used to denote a core designed to the Technical Specification limits. This was not meant to infer that the deterministic and risk analyses with the bounding core enveloped all Westinghouse NSSS plants and no additional plant specific work is required. The reload implementation process, described in Section 10 of the WCAP, requires that licensees use plant specific or bounding UETs. The WOG will decide on the appropriate approach. If bounding UETs are used, then a number of UET sets will be developed and the various plants will be binned into the various UET groups. At that point, if bounding UETs are used, the WOG will develop the justification for the plant binning process.

At this point, the requested information is not provided since the results in the WCAP are not meant to be bounding in the sense that plant specific analysis is no longer required. Since plant specific analysis is still required, the values used in the plant specific work will be justified, not the values used in the "bounding" reactivity core analysis.

Technical Clarification 5. Regarding the calculation of critical power trajectory (CPT) and UETs, the staff requests the following information:

Technical Clarification 5.a. The CPTs were calculated for the two pressure-limiting ATWS events based on the 1979 generic Westinghouse ATWS analyses (i.e., Westinghouse letter NS-TMA-2182). The two events result in the complete loss of all main feedwater without a reactor trip and are identified as the loss-of-normal feedwater ATWS and the loss-of-load ATWS. A description of the evaluation performed should be provided to demonstrate that these events remain the limiting ATWS events for current core designs and operations.

Response: The Loss of Normal Feedwater and Loss of Load events are limiting because they result in the largest heat source/heat sink mismatch. Current core designs and operations, such as upratings, only make these events more limiting since the loss or degradation of the secondary heat sink increases the mismatch and results in a larger heat source with no corresponding heat sink.

The following provides a qualitative review of the other ATWS events and why they are not more limiting than the Loss of Normal Feedwater and Loss of Load events. The following events are discussed:

- Rod Withdrawal at Power
- Rod Withdrawal from Subcritical
- Partial Loss of Flow
- Accidental RCS Depressurization
- Excessive Load Increase
- Loss of AC Power
- Feedwater Malfunction
- Boron Dilution
- Startup of an Inactive Loop
- Dropped Rod

The Rod Withdrawal at Power (RWAP) event results in a pressurization of the RCS, but this pressure increase is limited by the rod worth and the total reactivity that can be inserted until the rods are fully withdrawn. The secondary heat sink is also intact and available for heat removal. Following withdrawal of the rods, nuclear feedback causes core power to decrease to the secondary side load. The plant will reach an equilibrium condition at an elevated Tavg. The results will therefore be less limiting than both the Loss of Load and Loss of Normal Feedwater events.

The Rod Withdrawal from Subcritical event is similar to the Rod Withdrawal at Power event. The limited amount of reactivity that can be inserted, plus the availability of the secondary heat sink, make this event less severe than the Loss of Normal Feedwater and Loss of Load events.

The Partial Loss of Flow event results in a primary side heatup and pressure increase. However, the loss of coolant flow would result in voids in the core and an associated reduction in reactivity and core power. Since the secondary side is intact, substantial heat removal can be maintained. The results of this event will therefore be less limiting than those of the Loss of Normal Feedwater and Loss of Load events.

The Accidental RCS Depressurization event results in a decrease in primary side pressure, and is therefore less limiting than the Loss of Normal Feedwater and Loss of Load events.

The Excessive Load increase event results in an increase in core power. However, due to the primary side cooldown resulting from the increase in steam flow, the RCS pressure decreases during the event. Therefore, this event is also less limiting than the Loss of Normal Feedwater and Loss of Load events.

The Loss of AC Power event yields results that will be less limiting than those from the Loss of Normal Feedwater event. During a Loss of AC Power event, the RCP coastdown will dominate the beginning of the transient causing an increase in core fluid temperature, which subsequently causes the core power to decrease. The core power will be lower in the Loss of AC Power event than in the Loss of Normal Feedwater event when steam generator inventory decreases to the point at which steam generator heat transfer is degraded. Therefore, the results of the Loss of AC Power event will be less severe than the Loss of Normal Feedwater.

The Feedwater Malfunction event results in an excessive increase in heat removal and an increase in core power due to a cooldown in the primary side. Although the core power increases, the cooldown yields a decrease in the RCS pressure. This event is therefore less limiting than either the Loss of Normal Feedwater or the Loss of Load events.

The Boron Dilution event is similar to the RWAP event. The dilution causes an increase in reactivity, which increases core power and results in a pressurization of the RCS. The increase in core power and pressure is much slower in the boron dilution event than the RWAP event. As in the RWAP event, the power increase is eventually offset by reactivity feedback. Since the secondary heat sink is intact, the plant can reach a new equilibrium condition. Based on this, this event is also less limiting than either the Loss of Normal Feedwater or Loss of Load events.

The Startup of an Inactive Loop event occurs when the cooler water in an inactive loop is inadvertently mixed with the remaining RCS inventory. This results in a cooldown of the RCS, and an increase in core reactivity and power. Since this is a cooldown event, there is only a small increase in RCS pressure due to the power increase. Any pressure increase that does occur is much less limiting than those in the Loss of Normal Feedwater and Loss of Load events.

In the Dropped Rod event, the response of the automatic rod control system to a dropped rod or bank of rods results in a power overshoot, and potential increase in RCS pressure. The plant reaches a new equilibrium condition after the overshoot. A reactor trip is not normally credited in the generic Westinghouse methodology for this event as described in WCAP-11394-P-A. The RCS pressure does not approach 3200 psig for this event, and is less limiting than either the Loss of Normal Feedwater or Loss of Load event.

Based on the above discussions, the Loss of Normal Feedwater and Loss of Load events are the limiting ATWS events.

<u>Technical Clarification 5.b.</u> The CPT calculations are described in Section 4.1 of the TR and are based on a nuclear steam supply system (NSSS) power level of 3579 MWt. Since some plants are either currently licensed to or pursuing uprates for power levels greater than 3579 MWt, please discuss how

power levels in excess of 3579 MWt can be accounted for in the CPT and UET calculations described in WCAP-15831-P to extend the validity of the TR as power levels increase. The staff also requests the WOG to provide a list of resulting limitations and conditions for the use of the TR (e.g., power levels, peaking factors, steam generator type).

Response: The CPT and UET calculations presented in the topical report represent an example of the process that will be used on a plant specific basis. These example calculations are not intended to bound all plant configurations. The CPTs and UETs will be justified when this process is implemented for a given plant. The particular plant's power level, peaking factors, and steam generator type will be appropriately accounted for at that time.

Technical Clarification 5.c. The CPTs are calculated using the LOFTRAN computer code and the UETs are calculated using the ANC computer code. Please demonstrate that all restrictions and limitations are satisfied for the present application of these codes.

Response: The LOFTRAN code remains valid for this application. The calculation of the CPTs does not exercise the code in any radically different way from its current and past applications, including its use in the NS-TMA-2182 ATWS study. The Safety Evaluation Report (SER) on LOFTRAN, as found in WCAP-7907-P-A, specifically calls out the use of the code for ATWS.

The qualification of the PHOENIX-P/ANC nuclear design system is documented in WCAP-11596-P-A (T. Q. Nguyen, et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988). PHOENIX-P is the cross section lattice code, and ANC is the 3D nodal core simulator. WCAP-11596 qualified PHOENIX-P/ANC for the multi-dimensional nuclear analysis of PWR cores. This WCAP, including the SER, indicates no specific limitations or restrictions on the use of ANC. With respect to the qualification of ANC, the SER stated the following: "...these comparisons cover the areas expected to be studied for a nodal code system to be used for PWR analyses, and they provide a wide range of relevant parameters and variations to test the capabilities of the system. The selected reactors were suitably representative and reasonably diverse, and provided sufficient approach to extremes of expected operating parameters."

The ANC calculations performed to calculate ATWS UETs involve calculations of critical powers at the peak pressure limit over a range of inlet temperatures (600°F to 660°F) and burnups. These critical power search calculations are similar to other design basis calculations, e.g., steamline break calculations, which are routinely performed for each reload design. The critical powers that are calculated for the UET analysis are all within the range of 0% to 100% power. The range of core average moderator densities encountered is approximately 0.59 g/cm^3 to 0.70 g/cm^3. Because of the high pressure employed in these calculations, coolant voiding does not occur. Consequently, the moderator feedback effects are within the code's capabilities and are adequately modeled.

Technical Clarification 5.d. The CPT results are presented in Tables 4-1 and 4-2, but the TR does not provide a clear description of how these values are generated. For at least one representative point in the tables, please provide a detailed explanation of the methodology used to generate the CPT value. This should include the LOFTRAN generated plots for the key system parameter values used in the evaluation of CPT and a sample calculation demonstrating the method used to calculate the UETs provided in Section 4.2 of the TR. The explanation should include a detailed description of the ANC computer code model (e.g., noding, full core) and provide output plots for the key parameters generated by ANC. The WOG should also demonstrate how the ANC output is compared to the CPTs to determine UETs.

Response: The generation of the CPTs for the Loss of Load (LOL) event, at T_{in} values from 600°F to 660°F, full AFW capacity, with 2 PORVs/3 of 3 PSVs in operation is described below. These are the CPT values of 0.734, 0.561, 0.360, and 0.120 from Table 4-2 of the WCAP.

Calculation of the CPTs is a two-step process. The first step is to run the LOFTRAN code to determine the reactivity feedback conditions that result in a peak RCS pressure of approximately 3200 psig. This is completed for each PORV/PSV and AFW flow assumption set. For this case, the assumptions are a LOL event with full AFW capacity, and 2 PORVs/3 of 3 PSVs in operation. The transient is analyzed from full power conditions. Reactivity coefficients, primarily MTC, are adjusted iteratively until the peak RCS pressure in the transient is approximately 3200 psig. The final reactivity coefficients are the only output used in the next step of the CPT calculation. For this case, the final MTC is approximately -6 pcm/°F.

In the second part of the analysis, the reactivity feedback conditions calculated in step #1 are used as a fixed boundary condition in the CPT calculations. In this calculation, it is assumed that the reactor is just critical at the defined values of core inlet temperature and core pressure (from LOFTRAN calculation in step #1). An iterative search on reactor power is completed for each subsequent pair of inlet temperature and pressure. The pairs for this case are 600°F/3200 psig, 620°F/3200 psig, 640°F/3200 psig, and 660°F/3200 psig. The reactivity calculation is performed using the same methods that are used in the LOFTRAN code. For each new temperature and pressure pair, an iterative search on reactor power is completed to find the critical condition (zero excess reactivity). It was found that core power fractions of 0.734, 0.561, 0.360, and 0.120 of nominal are calculated for the temperature/pressure pairs listed above, using the reactivity feedback from the step #1 LOFTRAN run.

The discussion below describes the ANC calculations used to determine the UET for the High Reactivity Core Model for ATWS Case 1 assuming equilibrium xenon and no control rod insertion. ATWS Case 1 assumes full auxiliary feedwater capability and all PORVs available. From Table 4-7 of WCAP-15831-P, the UET for this case was 110.1 EFPD.

ANC core models employ 2x2 nodes radially and, typically, 24 to 28 nodes axially to model an individual fuel assembly. The number of axial nodes may vary from plant to plant and cycle to cycle depending upon the details of the core design, e.g., the lengths of burnable absorbers, axial blankets, etc. Quarter-core symmetric models are typically used unless an asymmetry in the core loading pattern requires full core modeling. In the ATWS UET calculations, uniform inlet temperatures are assumed, so that use of quarter-core models is typical.

The ANC calculations to determine the UET are reasonably straightforward. For a given cycle burnup step, a precondition case is performed at nominal conditions (HFP, nominal temperature and pressure, equilibrium xenon). This case establishes the nominal critical boron concentration at this cycle burnup. The boron concentration and xenon distribution are held constant for the remaining calculations at this burnup step. Next a series of critical power calculations are performed at the peak pressure limit with various inlet temperatures (typically 600, 620, 640, and 660°F). Thus, for each burnup step, four critical power values are obtained.

ANC does not provide output plots of key parameters for ATWS calculations. The key output parameter of interest is the converged critical power for each burnup step and inlet temperature. Table G-3 below provides the results of these critical power calculations for this case. In Table G-3, the ANC critical powers are given in terms of absolute megawatts and include the pump heat. For comparison, Table G-3 also indicates the CPT values in terms of fraction of NSSS power and absolute megawatts. The 14 MWt values in the last column, over the burnup range 10000-22006 MWD/MTU, are not true critical powers. These values correspond to the pump heat. For these cycle burnups, the core power level calculated was 0 MWt, indicating that the core was subcritical at the inlet temperature of 660°F.

For each inlet temperature and burnup step, a parameter termed the "unfavorable power" is then calculated. This is simply the difference between the ANC critical power and the CPT power. These values are given in Table G-4. Positive values indicate an unfavorable ATWS response since the ANC critical power is larger than the power level required to reach the peak pressure limit. Conversely, negative values indicate a favorable ATWS response since the ANC critical power is less than the power level required to reach the peak pressure limit. The values from Table G-4 are plotted in Figure G-1 (only values corresponding to burnups of up to 8000 MWD/MTU are plotted).

The UET is determined by the range of cycle burnups for which the unfavorable power is positive for any inlet temperature. As Figure G-1 shows, the curve for 600°F crosses the zero line at the largest burnup. This burnup is 4822 MWD/MTU, which corresponds to 110.1 EFPD. Since the UET starts at the beginning of the cycle (Figure 1 shows that BOL is unfavorable for all inlet temperatures), the total UET for this case is, therefore, 110.1 EFPD.

The other UET values in Table 4-7 of WCAP-15831-P were calculated in a similar fashion using CPT values appropriate for each case. Table 4-8 of the WCAP includes the effect of 72 steps of control rod insertion from the lead bank. The ANC calculations for this table are performed in the same fashion as described above, but the ANC critical power calculations include the negative reactivity effect of 72 steps of insertion of the lead control bank. This negative reactivity addition results in lower critical powers.

Implementation of this methodology for actual core designs will involve calculating the UETs for the ATWS cases in Tables 4-7 and 4-8 for the reload core (or confirming that previously employed UETs remain valid). These calculations will be performed using the same quality assurance procedures as are used for design basis analyses performed as part of the Reload Safety Evaluation. The ANC models used for the UET calculations will be the same models that are used for other reload core analyses. With these models, the UETs will be evaluated using best estimate global core reactivity and reactivity feedback predictions.

Variations in the operating cycle burnup can affect the next cycle's core reactivity, which in turn, could affect calculated UETs. These variations are addressed in the Reload Safety Evaluation process for the next cycle by generating separate ANC models based on low and high estimates of the current operating cycle's shutdown burnup (i.e., the operating cycle burnup window). The same process will be used for the ATWS UET calculations. If the operating cycle shutdown burnup is known, however, then UET calculations for the next cycle will be based on the actual operating cycle's burnup. This would be the case, for example, if the UET calculations are performed after the current cycle shuts down for refueling but before the next cycle operation begins. The UETs will be transmitted to the utility as part of the normal transfer of reload core operational and physics data. The utility will then prepare the configuration management program using the UET information. The UET data identifies which plant configurations are favorable and which are unfavorable for any cycle burnup. Since the UETs will be based upon cycle burnup or, equivalently, effective full power days (EFPD), the configuration management program will not be affected by shutdown periods or periods of part power operation.

Technical Clarification 5.e. Table 4-2 provides the loss of load ATWS CPTs. For no PORVs available and an inlet temperature (T_{in}) of 660°F, a dash is shown in the table (i.e., no value is given). Discuss the meaning of this dashed line and how a UET is calculated for this condition. What is the UET associated with these ATWS conditions?

Response: The dashed line indicates that the power level corresponding to an RCS pressure of 3200 psig and a T_{in} of 660°F, at the fixed reactivity conditions associated with the given case, is below zero. In other words, the reactivity feedback at that high temperature and pressure would shut the core down.

For cases in which the CPT is a "negative" power, the UET was conservatively calculated using an inlet temperature corresponding to a core power of zero (i.e., the point at which the CPT line intersects a power of zero).

Technical Clarification 5.f. Tables 4-1 and 4-2 of the TR provide CPT results for core inlet temperatures (T_{in}) ranging from 600°F to 660°F. Discuss the basis for the range of inlet temperatures (T_{in}) used. Does this temperature range bound all ATWS scenarios?

Response: A range of inlet temperatures between a typical full power, nominal T_{in} (~555°F) and a T_{in} of 660°F is assumed. The maximum value of 660°F has been found to yield critical powers close to zero. The reactivity feedback at an inlet temperature of 660°F and pressure of 3200 psig is sufficient to reduce the power to zero or near zero. The minimum value is chosen to be modestly elevated above the initial Tin (~555°F). Since the T_{in} value drives the pressure increase, the minimum T_{in}'s must be high enough to result in peak pressures of 3200 psig.

Table G-3 ATWS Case 1 Critical Power Trajectory and ANC Critical Powers for the High Reactivity Core Model with Equilibrium Xenon and No Control Rod Insertion					
Inlet Temperature (°F)		600	620	640	660
CPT Fraction of 3579 MWt		0.734	0.561	0.360	0.120
CPT Power (MW_t)		2627	2008	1288	429
Cycle Burnup (MWD/MTU)	EFPD	ANC Critical Power (MWt) for T_{in} of 600°F	ANC Critical Power (MWt) for T_{in} of 20°F	ANC Critical Power (MWt) for T_{in} of 640°F	ANC Critical Power (MWt) for T_{in} of 660°F
150	3.4	2659	2018	1315	499
1000	22.8	2691	2064	1376	574
2000	45.7	2702	2078	1383	574
3000	68.5	2695	2060	1358	534
4000	91.3	2659	2014	1301	470
5000	114.1	2620	1957	1230	388
6000	137.0	2570	1886	1148	296
8000	182.6	2467	1739	966	96
10000	228.3	2370	1597	784	14
12000	273.9	2285	1465	624	14
14000	319.6	2210	1351	488	14
16000	365.3	2153	1258	381	14
18000	410.9	2096	1173	281	14
20000	456.6	2046	1098	189	14
22006	502.4	2010	1041	117	14

Table G-4 ATWS Case 1 Unfavorable Power (ANC Power – CPT Power) for the High Reactivity Core Model with Equilibrium Xenon and No Control Rod Insertion

Cycle Burnup (MWD/MTU)	EFPD	Unfavorable Power (MWt) for T_{in} of 600°F	Unfavorable Power (MWt) for T_{in} of 620°F	Unfavorable Power (MWt) for T_{in} of 640°F	Unfavorable Power (MWt) for T_{in} of 660°F
150	3.4	32	10	27	69
1000	22.8	64	56	87	144
2000	45.7	75	70	95	144
3000	68.5	68	52	70	105
4000	91.3	32	6	13	41
5000	114.1	-7	-51	-59	-41
6000	137.0	-57	-122	-141	-134
8000	182.6	-160	-268	-323	-333
10000	228.3	-257	-411	-504	-415
12000	273.9	-342	-543	-665	-415
14000	319.6	-417	-657	-800	-415
16000	365.3	-474	-750	-907	-415
18000	410.9	-531	-835	-1007	-415
20000	456.6	-581	-910	-1100	-415
22006	502.4	-617	-967	-1171	-415

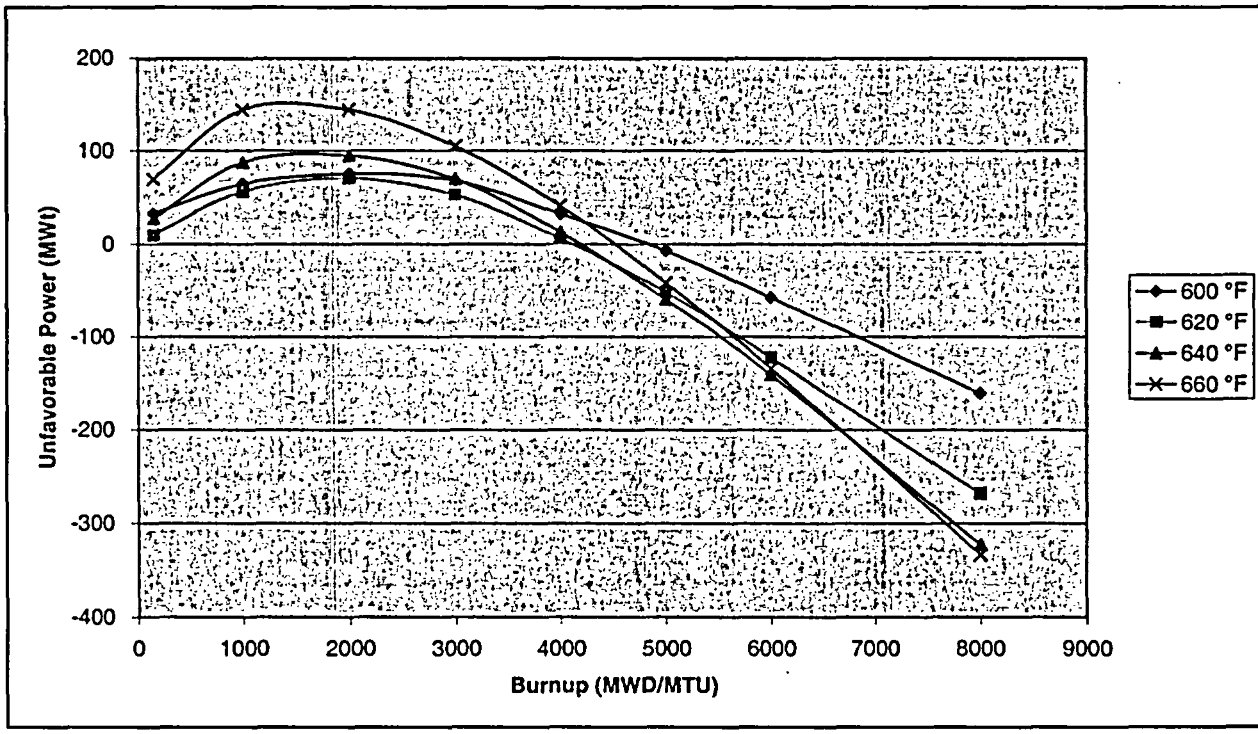

Figure G-1 Unfavorable Power (MWt) for ATWS Case 1 of the High Reactivity Core Model as a Function of Cycle Burnup and Inlet Temperature

<u>Technical Clarification 6.</u> For Approach 1 in Section 7, the TR states that the identified actions are proposed "... to restore defense-in-depth." While in an UET condition, four of the identified five actions restrict further activities that would not change the existing condition of being vulnerable to an ATWS event. Though these actions might prevent the plant condition from worsening, they do not consider the fact that the plant is already in an unacceptable configuration if an ATWS event occurs. Thus, these actions do not restore defense-in-depth, except for possibly the situations in which placing the rod control system in automatic could (if CRI is credited) eliminate the unfavorable configuration. The text should be revised to ensure unfavorable configurations are eliminated when they occur in accordance with the established controls supporting Item 1 above.

Response: Defense-in-depth against ATWS events is provided by prevention of the event and mitigation of the event if prevention fails. Prevention is provided by inserting the control rods into the core. This can be done automatically by the reactor protection system (RPS) or by an operator action to trip the plant. The reactor protection system provides the reactor trip signals (RTS). The typical RTS circuit consists or analog channels (field transmitters or process sensors and process control and protection system), combinational logic cabinets (solid state or relay protection system), and reactor trip switchgear (reactor trip breakers). The analog channels provide signals to each of the logic cabinets and are typically arranged in two of three or two of four logic to meet reliability requirements. The logic cabinets and reactor trip switchgear are arranged in two trains with either one capable of tripping the reactor. Therefore, there is a degree of defense-in-depth built into the RPS. In addition, an operator action to trip the reactor can be taken from the control room. This provides a backup to failures of analog channels and logic cabinets. The reactor trip breakers are required to open with this operator action. Therefore, this operator action provides an additional degree of defense-in-depth from the prevention perspective.

If an ATWS event does occur (prevention fails), then mitigation is provided by pressurizer safety valves and PORVs, along with AFW, for pressure relief and shutdown is provided by emergency boration.

The actions listed in Section 7 of the WCAP are:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

The first action is directed at maintaining the defense-in-depth capabilities of the RPS and reducing the probability of the occurrences of an ATWS event. The next three actions will prevent further degradation of the configuration and will, in some cases, limit the RCS pressure to a level below which containment releases are a concern. Although these actions may not prevent core damage, they may prevent containment releases. The last action is directed at restoring defense-in-depth via pressure mitigation and then emergency boration. This will be clarified in the revision to this WCAP.

Technical Clarification 7. In Section 4.3.3 of WCAP-11992, it is stated that "... an initial power less than 70 percent will not result in RCS pressures greater than that corresponding to the ASME Level C service criterion ..." The staff interprets this statement as meaning that an UET is not possible at less than 70 percent power. Is this a correct interpretation and is this situation still valid for all current and expected plant cases and fuel designs (e.g., the bounding reactivity case)? If this situation is not valid for these conditions, please explain what has changed since the development (and recent efforts to get approval) of WCAP-11992 that make this statement not correct for WCAP-15831-P. If this situation is still valid, then the staff suggests that the mitigative strategies address how this power limitation could be used as a proactive response to potentially prolonged UETs, consistent with Item 1 above.

Response: WCAP-10858 Revision 1, the AMSAC Generic Design Package, states the following with respect to power "Short-term protection against high reactor coolant system pressures is not required until 70% of nominal power. However, in order to minimize the amount of reactor coolant system voiding during an ATWS, AMSAC should operate at and above 40% of nominal power. Furthermore, the potential exists for spurious AMSAC actuations during start-up at the lower power levels. To assure the above requirements are met, AMSAC will be automatically blocked at turbine loads less than 40% by the C-20 permissive." Thus, although the peak pressure limit may not be violated below 70% power, the AMSAC arming setpoint was conservatively established as 40% power.

The conclusion that the ASME Level C service criterion is met at power levels below 70% was based on the core configuration utilized in the ATWS analyses presented in NS-TMA-2182. As shown in Section 4.4, use of a high reactivity or bounding reactivity core model can yield non-zero UETs for some plant configurations even at a reduced power level. Reducing power will always reduce the UET, but will not eliminate it in all cases. Therefore, in some cases, reducing power would be a viable mitigative strategy. However, it would require additional analyses to determine the power reduction needed to eliminate the UET for all of the various core and plant configurations. These additional analyses would be similar to those completed for the full power cases. Iterative analyses would have to be completed to determine the power reduction required to eliminate the UET for the particular plant configuration. The analysis would include calculation of new CPTs at the reduced power level, followed by calculation of the UET for the plant and cycle specific core conditions following the same approach discussed in response to Technical Clarification 5.

EDITORIAL

Editorial Clarification 1. Throughout the TR, reference is made to an UET that is conditioned by a specific plant configuration (i.e., 100 percent PORV capacity available, 100 percent AFW system availability, no control rod insertion capability, and 100 percent ATWS mitigating system actuation circuitry (AMSAC) availability). Though this conditional definition was used in WCAP-11992 and was allowed as part of the current method of calculating and controlling the UET for some licensees, the staff does not believe this configuration condition is a valid aspect of the basic UET definition and can lead to misunderstandings. A more basic definition of UET would be the time in which the reactor core reactivity feedback is not sufficient to prevent RCS pressure from exceeding 3200 psig following an ATWS event. With this definition, the UET is defined by the plant's pressure response, which can change as the plant conditions and configurations change. Thus, for example, with all equipment operable a plant might not be in an UET condition, but if a specific ATWS-significant component becomes unavailable, the plant could then immediately enter a UET condition. This definition is then very similar to the definition of unfavorable moderator temperature coefficient (MTC) that is used in the supporting technical bases of the ATWS rule (10 CFR 50.62). To avoid confusion, whenever referring to the specific plant configuration consisting of 100 percent PORV capacity, 100 percent AFW system availability, no control rod insertion capability, and 100 percent AMSAC availability, it should be identified as the "ATWS rule reference case UET" or similar phrase that distinguishes this conditional definition from the more basic UET definition. It should also be recognized that this "ATWS rule reference case UET" may be a small portion of the actual UET experienced at a plant. Please revise WCAP-15831-P accordingly.

Response: The WOG definition of UET is provided in the second paragraph on Page 4-4 of the WCAP. The definition is given as "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)." This is consistent with the definition proposed above by the NRC.

UETs are determined for the following plant configurations. All assume AMSAC and the pressurizer safety valves are available.

1. Control rod insertion (72 steps from lead bank), 100% AFW, 2 of 2 PORVs available
2. Control rod insertion (72 steps from lead bank), 50% AFW, 2 of 2 PORVs available
3. Control rod insertion (72 steps from lead bank), 100% AFW, 1 of 2 PORVs available
4. Control rod insertion (72 steps from lead bank), 50% AFW, 1 of 2 PORVs available
5. Control rod insertion (72 steps from lead bank), 100% AFW, 0 of 2 PORVs available
6. Control rod insertion (72 steps from lead bank), 50% AFW, 0 of 2 PORVs available
7. No control rod insertion, 100% AFW, 2 of 2 PORVs available
8. No control rod insertion, 50% AFW, 2 of 2 PORVs available
9. No control rod insertion, 100% AFW, 1 of 2 PORVs available
10. No control rod insertion, 50% AFW, 1 of 2 PORVs available
11. No control rod insertion, 100% AFW, 0 of 2 PORVs available
12. No control rod insertion, 50% AFW, 0 of 2 PORVs available

All twelve conditions are used in the ATWS risk model. Plant configuration 7 is used in the ATWS Rule. Where appropriate, this will be referred to as the "ATWS rule reference configuration."

<u>**Editorial Clarification 2**</u>. The TR includes the statement that SECY-83-293 demonstrates that the installation of AMSAC reduces the risk from ATWS events to an acceptable level. It should be noted that the SECY-83-293 supporting risk analysis and other related analyses performed in support of the ATWS rule were performed in the late 1970s and early 1980s based on plant operating conditions (i.e., plant equipment configurations and availability, fuel design, etc.) at that time. These analyses were performed well before the advent of the risk-informed decision-making processes within the NRC, such as described by Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," et. al. As such, the risk analyses developed in support of the ATWS rule were relatively simplistic and made some significant assumptions regarding plant operating conditions. Current plant operating conditions may be considerably different from those assumed in these analyses. Based on the above, it may be misunderstood to state that the SECY-83-293 analyses (performed almost two decades ago) demonstrates (present tense) an acceptable level of risk from ATWS events with the installation of AMSAC, when some of the most significant assumptions of those analyses may no longer be valid. Please revise WCAP-15831-P accordingly.

Response: Since this statement, in Section 2.1 of WCAP-15831-P, is taken directly from Section 2.1 of WCAP-11992 ("Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988) the text will not be changed, but the following clarification will added.

"These important points are based on SECY-83-293 and the final ATWS Rule, and were applicable at the time those documents were issued. The current applicability of these points needs to consider today's risk-informed environment and changes in plant operation."

As noted, this statement is taken from Section 2.1 of WCAP-11992. WCAP-11992 has been reviewed by the NRC.

<u>Editorial Clarification 3</u>. The TR includes the statement that the ATWS Rule only required the installation of AMSAC for Westinghouse reactors and that "[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." Though the staff agrees that the only requirement for Westinghouse reactors in the ATWS Rule was the installation of AMSAC, the staff has not been able to identify in SECY-83-293 where it states the acceptability of specific plant conditions is solely determined within the context of core damage frequency (CDF). Please clarify the intent of this statement in the TR and/or revise WCAP-15831-P accordingly.

Response: The ATWS Rule is based on a value/impact assessment of several options to reduce ATWS risk to an acceptable level. The SECY-83-293 study used an ATWS "risk" goal of "no more than about 1E-05 per year". Core damage was equated with public risk for the value/impact study. The plant/design options for ATWS event mitigation were developed to reduce core damage frequency (risk) to an acceptable level while considering the cost of implementation. Since the target core damage frequency was 1E-05/reactor year, the acceptability of plant conditions is based on core damage frequency.

No changes to WCAP-15831-P or WCAP-15831-NP will be made based on this comment.

Note that this statement is taken from Section 2.1 of WCAP-11992 ("Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988) which was reviewed by the NRC.

<u>Editorial Clarification 4</u>. There is an incorrect statement in Section 2.4.2 of the TR. The statement is: "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." In applying RG 1.174, an applicant must address probabilistic and engineering aspects of the licensing basis change. At the NRC/WOG August 23, 2000 meeting, the staff emphasized the need for the WOG to fully address the deterministic aspects in its TR and not rely solely on probabilistic arguments. Sections 2.2.1.1 and 2.2.1.2 of RG 1.174 indicate that engineering evaluations must be performed to ensure that adequate defense-in-depth and safety margins are maintained. Please revise WCAP-15831-P accordingly.

Response: With this clarification of the Staff's intent, the statement in question will be removed from the WCAP.

Editorial Clarification 5. Table 5-2 identifies 240 transient events that have occurred by "ATWS State" while Table 5-3 identifies only 194 transient events. Please explain the difference in the total number of transient events between these two tables and also please explain why there are a fractional number of events identified for the various ATWS states in Table 5-2. Please revise WCAP-15831-P accordingly.

Response: Table 5-2 of the WCAP provides the trip frequency for all power levels. The total number of trips in this table is 240. Table 5-3 of the WCAP provides the fraction of trips in 30-day intervals and is used to determine the trip distribution over the fuel cycle. The total in this table is 194. This value only considers trips when the power level is greater than or equal to 40%. From Table 5-2 the number of trips with a power level greater than or equal to 40% is 202. This includes 8 trips that occurred after 18 months in the fuel cycle. Subtracting these 8 trips from the 202 trips leaves 194 trips which matches the total on Table 5-3. These eight trips were due to plants with significant downtime following startup after a refueling so the cycle time ran past 18 months. To maintain a set of data to determine the trip distribution over an 18-month fuel cycle, these were eliminated. Additional text will be added to Section 5.1.1.11 of the WCAP to explain this difference.

Table 5-2 provides the number of events in the five ATWS states. Several of these are fractional numbers. Calculation of these values is demonstrated in Section 5.1.1.2 of the WCAP. To recap, the total number of events over all power levels is 240. Thirty-eight for these occurred with the power level < 40% and 24 occurred with the power level ≥ 40% and < 95%. The fractions arise when these numbers are divided between startup and shutdown trips. WCAP-14333 collected information on startup and shutdown trips, and provides the probability of a reactor trip on startup as 0.088 and on shutdown as 0.068. These two values were then used, as shown in Section 5.1.1.2, to divide the number of trips in the two power level ranges between startup and shutdown trips, which resulted in fractional number of events.

<u>Editorial Clarification 6.</u> In Section 8.2 of the TR, the clarifying bullet regarding when the engineered safety features actuation system (ESFAS) is credited seems to be internally inconsistent and confusing. The first sentence states that ESFAS is only credited if the reactor trip signal failure is not a common cause failure (CCF) that can also be associated with the ESFAS signal. However, the second sentence states that the ESFAS signal is only credited if the reactor trip fails due to failure of the control rods to fully insert into the core, which the staff assumes is referring to top event CR. Please clarify when ESFAS is and is not credited in the ATWS probabilistic analyses and revise WCAP-15831-P accordingly.

Response: It is assumed that if the reactor trip signal fails, then the ESF actuation signal also fails. If the ESF actuation signal was available, then the AMSAC signal would not be necessary. But due to the potential for common cause failures, it is assumed that the ESF actuation signal will not be available whenever the reactor trip signal fails. Therefore, the ESF actuation signal is only credited when the ATWS event occurs due to the control rods failing to drop into the core (mechanical failure) given a reactor trip signal has been generated. The top event modeling the control rods failing to drop into the core is CR.

Additional text will be added to Sections 8.2 and 5.1.1.1 of the WCAP to further explain this modeling.

Editorial Clarification 7. The relationship between top events CR and CRI needs to be clarified throughout WCAP-15831-P in accordance with the following specific comments:

Editorial Clarification 7.a. The phrase "control rod insertion" is not used consistently in the TR. In some cases it refers to top event "CRI" and in other cases it refers to top event "CR." Top events "RT" and "OAMG" also play a role in success or failure of control rod insertion. In particular, on page 2-3 it states that the UET is determined based on the "... success or failure of control rod insertion (CRI) ... In this case, CRI is equated to 72 steps insertion of the lead bank." However, on page 8-3 the first bullet states "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)." Since these top events represent different conditions, it is important to make sure that the text is clear. Please revise WCAP-15831-P accordingly.

Response: Throughout the WCAP, CRI is defined as the action (manual or automatic) to drive the control rods into the core. Success of this action provides 72 steps (negative reactivity) from the lead bank. CR is defined as a sufficient number of control rods fall into the core to shut down the reactor. On page 2-3 of the WCAP, the phrase "success or failure of control rod insertion (CRI)" is first introduced. In the following sentence it is stated that successful CRI is equated to 72 steps insertion from the lead bank. This is consistent with the above definitions. For clarification purposes, the words "via the rod control system" have been added to the phrase "success or failure of control rod insertion (CRI) via the rod control system" on page 2-3. Similar changes will be made at other places in the WCAP to distinguish between CRI and CR.

Editorial Clarification 7.b. It is noted in the TR that "... it is not necessary to address CR 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)." The latter sentence may be true, but that does not make the former sentence true. This logic infers that there are no means of the rods failing to insert, if the actions identified in CRI are successful. However, CRI success is only dependent on the mode of the rod control system and, if it is in manual, the successful actions of the operators. It does not include the potential for the rods to fail to insert even though the system is in automatic or the operators take the correct actions. If actions related to CRI are successful, there is still the chance that the control rods will not insert. Please revise WCAP-15831-P accordingly.

Response: The latter sentence, "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)," explains why it is not necessary to address CR in the event tree (the former sentence, "... it is not necessary to address CR following success of CRI."). It is not stating that the control rods cannot fail to move into the core if the operator takes the action or if the rod control system starts to drive the control rods into the core. Since the value for CR failing is 1.2E-06/d and the value for CRI failing is 0.1 adding CR into CRI still gives 0.1. Therefore, CR is not explicitly addressed in the model. This is explained in the WCAP in the 3rd paragraph of Section 5.1.1.5 and the last paragraph of Section 5.1.1.6.

The sentence identified above in the Editorial Clarification is at the end of Section 8.2.5. The text will be changed to the following:

> Note that it is not necessary to explicitly address CR following success of CRI. It is understood that the control rods still need to move into the core, but the probability of the rods failing to insert is assumed to be included in the probability of CRI failing (CR is very small compared to CRI).

<u>**Editorial Clarification 7.c.**</u> In Section 8.2.5 of the TR, it is stated that even "[if CR fails, it is assumed that sufficient rods have inserted to be equivalent to 72 steps of D-bank insertion..." It is also stated that failing to get this amount of insertion "... is not credible." This assumption limits the pressure peak and resulting consequences of the ATWS event. The staff does not accept this assumption without significant supporting justification that there are no failure modes that could effectively result in no insertion. The staff believes, absent additional justification, that if top event CR fails, it should be assumed that no rods insert, instead of crediting some insertion even in failure, and the resulting analyses and configuration management approach should be based on this assumption (i.e., no insertion at all if top event CR fails). Please revise WCAP-15831-P accordingly.

Response: Assuming, as the NRC suggests, that no rods insert if CR fails is extremely conservative and could lead to inappropriate conclusions and decisions based on the model. The failure probability associated with CR is 1.2E-06/demand and corresponds to the failure probability for ten or more of 50 control rods to insert. This is taken from NUREG/CR-5500, Vol. 2 ("Reliability Study: Westinghouse Reactor Protection System, 1984 – 1995," December 1998). In this NUREG, the basic event ROD (the equivalent of CR in the WCAP) is defined in Table 5-2 as "Failure of RCCA/CRDM, resulting in failure of RCCA to insert into the core." In Appendix E (Section E-4) of this NUREG, it is noted that "For most transients, the insertion of a few rods is sufficient to shut down the reactor, e.g., less than ten for a mild transient. For others, it requires more rods to insert." It is further noted in Appendix E, with regard to an ATWS event, "that PWR overpressurization can be prevented by a relatively few control rods successfully inserting...".

Since the failure criteria for CR is ten or more control rods (out of 50) fail to insert, there will be a significant negative reactivity insertion even if CR fails. Since there will be a significant number of control rods inserting, the ATWS overpressurization will be partially or fully mitigated, depending on the time in life, auxiliary feedwater flow, and pressure relief capability. To simplify the model, it is assumed that this negative reactivity insertion achieved following CR failure, although not sufficient to shut down the reactor, is the equivalent of 72 steps of control rod insertion from the lead bank and the UETs associated with 72 steps insertion from the lead bank are used when CR fails. To assume no control rods insert would be much too conservative.

<u>**Editorial Clarification 7.d.**</u> The text and logic modeling would be more clear and concise if top events RT and OAMG were combined into a single top event (RT/OAMG) in the ATWS event tree to address scram success/failure and top event CRI were to address initial/partial control rod insertion success/failure. With this approach, the specific component and action failure combinations would need to be addressed via a fault tree logic model, including current top event CR as a potential failure mechanism of both of these top events. Under this streamlining of the logic model, success of the top event RT/OAMG would result in no ATWS (i.e., success sequence) and failure would lead to the CRI

event. CRI success would mean there would be initially 72 steps of insertion of the lead bank to help mitigate the pressure resulting from the ATWS and the rods would continue to be inserted so that the reactor would be maintained subcritical (i.e., no need to address the long term shutdown (LTS) top event for these sequences). CRI failure would mean that there is no rod insertion and the LTS top event would need to be addressed for these sequences. This approach would seem reasonably realistic and it would not be necessary to provide additional justification for the current model assumption that even with failure of CR there are 72 steps of insertion as requested by Item 7c above. Please revise WCAP-15831-P accordingly.

Response: There are two potential problems with following the suggested approach of combining RT and OAMG (and CR) under a single top event. The first is in regard to crediting ESFAS as a means to start AFW and trip the turbine. If RT is successful and CR fails, then the ESF actuation signal and AMSAC can both can be credited with actuating auxiliary feedwater and tripping the turbine. If RT fails, OAMG is successful, and CR fails, then it is assumed that the ESF actuation signals have also failed, consistent with the standard ATWS event progression, and only AMSAC can be credited for actuating auxiliary feedwater and tripping the turbine. This logic is clear with the approach used in the WCAP.

The second potential problem is related to failure of CR. As discussed in the response to Editorial Clarification 7.c, failure of CR still results in a significant negative reactivity insertion. It appears the proposed approach would not credit this negative reactivity insertion and require that CRI be addressed. This would be a very conservative approach. In addition, it is not clear that if CR failed that CRI (control rod drive system) would be able to insert the control rods into the core. Using the approach in the WCAP, no credit is taken for CRI if CR fails.

Based on the above, no changes are planned for the ATWS model described in the WCAP.

Editorial Clarification 8. Section 10 of the TR discusses actions a licensee must take to demonstrate that transitioning to a high reactivity core is acceptable, given that the plant is "not consistent with the bases for the ATWS rule." Because some licensees currently operate with a positive moderator temperature coefficient and can at times operate in some of the adverse plant configurations analyzed in this TR (e.g., PORVs blocked, AFW train out of service, etc.), this discussion might be interpreted to imply that licensees are currently operating in a manner not consistent with the ATWS rule. Please discuss how licensees will track UET to ensure that the bases for the ATWS rule are maintained and revise WCAP-15831-P accordingly.

Response: Licensees will use the ATWS Configuration Management Program (CMP), discussed in the response to Technical Clarification 1.a, to track the plant operating configuration throughout the cycle and compare the current configuration to a set of plant configurations associated unfavorable exposure times (UETs). The UETs define the time during the cycle when the ATWS pressure transient will exceed 3200 psi. The UETs are dependent on control rod insertion (either 72 steps from the lead bank or none), auxiliary feedwater flow (100% or 50%), and pressure relief capability (number of blocked PORVs). The objective is for the licensee to operate the plant in a configuration which is favorable to the mitigation of the ATWS pressure transient. This is defined by the UETs.

The ATWS CMP will most likely, although not necessarily, be incorporated into the Configuration Risk Management Program licensees use to the address the Maintenance Rule. The CRMP tracks the availability of components and this information can be used in the ATWS CMP. The equipment that needs to be considered is:

- Auxiliary feedwater system
- Pressurizer PORVs
- Pressurizer safety valves
- AMSAC
- Steam stop and control valves
- Rod control system (automatic or manual)

To accomplish this tracking, the ATWS CMP will need the following capabilities:

- Identify plant configurations that do not maintain defense-in-depth to an ATWS event. This will be based on equipment important to mitigation of the ATWS pressure transient and the UETs.

- Track the time for individual occurrences when the plant is in an unfavorable plant configuration.

- Track the cumulative time per cycle when the plant is in an unfavorable plant configuration.

- Provide information on the length of time remaining in the UET for plant configurations.

- Provide compensatory actions to take if the unfavorable condition cannot be exited prior to expiration of the time allowed in the unfavorable configuration.

Not all the time in a UET will be tracked against the allowed cumulative total. As discussed in response to Technical Clarification 1.e, some surveillance requirements are necessary to demonstrate equipment operability for design basis events. These surveillance requirements can impact the availability of equipment to mitigate an ATWS event, but since the design basis events contribute greater to risk than ATWS, these surveillance requirements will be performed on schedule. As shown on Table 2, these surveillances to do not contribute significantly to equipment unavailability, therefore, surveillances required to demonstrate equipment operability will not be counted in the cumulative total. But, if the ATWS mitigation equipment is out of service for preventive or corrective maintenance activities, and this places the plant in a unfavorable configuration, then this time will be counted in the cumulative total.

This will be discussed in the revision to the WCAP.

Response to NRC's Supplementary Request for Further Clarifications Made at the March 16, 2004 NRC/WOG Meeting on WCAP-15831-P

Supplementary Clarification 1: How will Technical Specification Surveillance Requirements be addressed with regard to the time allowed in an unfavorable configuration and is this a significant amount of time?

Response: As discussed in the response to Technical Clarification 1.e, entries into unfavorable configurations to meet Technical Specification surveillance requirements are acceptable. Some of the equipment important to mitigation of an ATWS pressure transient is also important to mitigation of other design basis events. These design basis events typically are larger contributors to plant risk than the ATWS event, therefore, it is important to maintain the equipment operability for design basis event mitigation. The surveillance requirements demonstrate equipment operability, therefore, it is recommended that they continue to be completed at the specified interval.

If equipment inoperability due to surveillance requirements moves the plant into an unfavorable configuration, then simultaneous test and maintenance activities that compromise the reactor trip system availability or that place the plant in a higher trip potential configuration should be rescheduled to when the plant returns to a favorable configuration, i.e., until completion of the surveillance.

The surveillance requirements identified in the Technical Specifications (NUREG-1431) for these systems/components are summarized on Table G-1 along with surveillances required by plant procedures. Similar information is provided on Table G-2 for Braidwood. Table G-2 also includes the times the surveillance requirements cause the system/component to be inoperable with regard to ATWS event mitigation. It is concluded from this that the time these systems/components are unavailable to meet surveillance requirements is small in comparison to the (proposed) 30 day total time allowed in an unfavorable configuration. Therefore, it is proposed that this time does not need to be tracked against the total time allowed in unfavorable configurations, if it does place the plant in an unfavorable configuration.

<u>Supplementary Clarification 2:</u> How much time is available for an operator to trip the reactor by interrupting power to the MG sets? The MG set coast down time needs to be accounted for in this assessment.

Response: The time available for the operators to taken action from the start of the event until the RCS pressure increases to 3200 psig is dependent on the core reactivity (low, high, bounding), the available pressure relief, and the AFW flow. For the low reactivity core, without credit for control rod insertion of 72 steps, the time varies from 103 seconds to 111 seconds. For the bounding reactivity core, without credit for control rod insertion of 72 steps, the time varies from 93 seconds to 100 seconds. The times for the high reactivity core lie between these two ranges of time.

The time required for the coastdown of the motor-generator sets is the time from interruption of power to the MG sets until the power decays sufficiently to allow the CRDMs to release the control rods, and the control rods to drop into the core. There is a minimum specified time of 1 second to eliminate reactor trips on momentary losses of power, but no requirements on the maximum length of time. Therefore, licensees implementing this compensatory action will need to document that sufficient time exists for the operator to take the action to interrupt power to the MG sets and for the MG sets to coast down, and release the control rods, within approximately 90 seconds. Alternate approaches to interrupting power to the CRDMs may be pursued if the MG set coast down time does not meet this requirement.

<u>Supplementary Clarification 3</u>: With regard to the impact on UETs, how will the downtime in the cycle and operation at less than 100% power be addressed.

Response: As stated in the response to Technical Clarification 1.d, effective full power days of operation will be the basis for the ATWS CMP. Downtime and operation at less than 100% power will be accounted for through the use of effective full power days.

Appendix H
Additional Issues Identified in a NRC/WOG Telecon on June 9, 2004 to Clarify NRC Issues Identified during a NRC/WOG Telecon on June 2, 2004

WOG Responses are Provided for Each Issue

Further Clarification of Information Provided to the NRC in Reference 1

As discussed on the phone call on June 9, 2004 between J. Wermiel (NRC), Ted Schiffley (WOG/Exelon), and G. Andre (Westinghouse), additional clarification in the following three areas will assist in the Staff's acceptance of the proposed ATWS Configuration Management Program.

1. **A better understanding of the basis for crediting 72 steps from the lead bank is requested. Why 72 steps and how does this impact or factor into the analysis?**

Clarification: Crediting 72 steps from the lead bank was first presented in WCAP-11992 (Reference 2) which was provided to the NRC in May 1995 via a WOG letter (Reference 3). In WCAP-11992, the top event MRI (manual rod insertion) represents the operators using the rod control system to drive the control rods into the core. This assumes the rod control system will be in the manual operating mode. If the rod control system is in the automatic mode, then the control rods will begin to insert automatically. But assuming it is in the manual mode, then per WCAP-11992, the operator action (MRI) action must be initiated within the first minute from the start of the event which leaves about one minute of time for the control rods to step into the core. There is approximately two minutes of time, from the start of the event, prior to the reactor coolant system pressure reaching 3200 psig. One minute of time stepping the control rods into the core equates to 72 steps of the lead bank. Therefore, the 72 steps of control rod insertion is based on the time available to mitigate the event, and not on the amount of control rod insertion required to maintain the RCS pressure below 3200 psig or some other analytical limit.

The negative reactivity insertion provided by 72 steps of the lead bank is then credited in the calculation of unfavorable exposure times (UETs). UETs are determined for combinations control rod insertion (72 steps or none), auxiliary feedwater flow (100% and 50%) and PORV availability (2, 1, or none). The 72 steps of control rod insertion is important to determining the UETs. As described in WCAP-15831-P, the UETs are then factored into the risk analysis and the ATWS Configuration Management Program.

2. **Why does the WOG feel the operator action to trip the plant via interrupting power to the motor-generator sets (that supply power to the CRDMs) can be successfully completed in the short amount of time available?**

Clarification: If a licensee plans to implement in its plant the compensatory action to trip the reactor by interrupting power to the CRDM's motor-generator sets, the action will need to be 1) upfront in the plant's emergency operating procedures and 2) relatively simple and quick to perform. In addition, the plant operators will need to be trained on the actions. The WOG's Emergency Response Guideline E-0 ("Reactor Trip or Safety Injection") provides the actions to verify proper response of the automatic protection system following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery procedure. Step 1 of E-0 requires that reactor trip be verified by:

- Rod bottom lights – LIT
- Reactor trip and bypass breakers – OPEN
- Rod position indicators – AT ZERO ON DRPI
- Neutron flux – DECREASING

If the response is not obtained, then the action is to manually trip the reactor. If the reactor will not trip, then the operators go to Functional Restoration FR-S.1 ("Response to Nuclear Power Generation/ ATWS"). The "Response not obtained" to the first step in FR-S.1, which is verifying reactor trip, is given as:

Manually trip reactor.

If the reactor will not trip, then perform the following:

- manually insert control rods, and
- dispatch NPO to locally trip reactor trip breakers. If the reactor trip breakers will not trip, then locally trip the MG set output breakers.

As can be seen, there is an action in FR-S.1 to locally trip the MG set output breakers, but this action does not meet the requirements of the proposed compensatory action. The proposed compensatory action will need to be placed before the action to "manually insert control rods" and will need to be implemented from the control room or by a dedicated operator located at the MG sets.

As part of the WOG Emergency Response Guidelines Validation Program (for Revision 1 of the ERGs), an ATWS scenario was included as part of the simulator exercises. This event was initiated from full power by the sequential trip of both main feedwater pumps. The reactor trip breakers did not open in response to a manual trip attempt. The operators correctly transitioned from E-0 to FR-S.1, and began to manually insert the control rods in less than 60 seconds. Although the action was to manually insert the control rods, not to trip the reactor via the MG sets, this does demonstrate the operators began actions to shutdown the reactor in the timeframe required to mitigate the potential high RCS pressures from the ATWS event. As noted in Supplementary Clarification 2 of Reference 1, the time for the RCS pressure to reach 3200 psig ranges from 93 seconds to 111 seconds depending on the core. The simulator exercise demonstrates that the operators have sufficient time to take the proposed compensatory action.

In addition, Millstone Nuclear Power Station, Unit 3, has the capability to trip the reactor by interrupting power to the CRDM's MG sets. The EOPs include an action to de-energize the 480V power supplies to the MG sets. This action is early in the EOPs and is performed in the control room. In addition, the operators are trained on the action. Although logs are not kept on the performance of this action, past experience has shown that operators complete this action within one minute.

Therefore, based on the above information, the WOG feels there is a high probability of success of operators completing this action in the timeframe necessary to prevent the RCS pressure from reaching 3200 psig, and that it is appropriate to credit this operator action in response to an ATWS event.

3. **Confirmation that there is no tie between the reactor protection system (that, due to its failure, may be the cause of the ATWS event) and the rod control system (that is credited in mitigation of the ATWS event).**

Clarification: General Design Criteria 24 addresses separation of protection and control systems. GDC 24 states "The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

The reactor protection system (RPS) and rod control system are two separate systems that share only the T_{avg} signals and power range signals. The T_{avg} signals are developed in the process protection system (7300 system, for example) for each loop. The T_{avg} signals are based on the reactor coolant system loop T_{hot} and T_{cold} signals measured by the resistance temperature detectors. The individual loop T_{avg} signals are provided to the rod control system for control purposes and are further processed in the process protection system for reactor protection purposes. The power range signals are also used for protection and control purposes. Due to the logic requirements of the RPS for developing reactor trip signals (2 of 3 or 2 of 4) and the rod control system auctioneered use of the signals, a single failure will not compromise development of reactor trip signals or the operation of the rod control system.

A common cause failure of all the T_{avg} signals or all power range signals may adversely impact the operation of the rod control system, but not of the RPS. Reactor trip signals are developed from a number of sets of channels for the various transient events, and failure of one complete set will not cause complete failure the RPS and prevent a reactor trip.

References

1. WOG-04-257, "Response to Clarification Request – WCAP-15831-P 'WOG Risk Informed ATWS Assessment and Licensing Implementation Process'," May 17, 2004.

2. WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988.

3. OG-95-41, "Transmittal of Topical Report: WCAP-11992 – 'Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process'," May 2, 1995.

Appendix I
Additional Issues Identified in a NRC/WOG Telecon on June 25, 2004

WOG Responses are Provided for Each Issue

Further Clarification of Information Provided to the NRC in Reference 1

As discussed on the phone call on June 25, 2004 between the NRC and the WOG, additional clarification in the following two areas was requested by the Staff.

1. **What is the reliability of the rod control system, both electrical and mechanical (mechanical refers to the control rods failing to move). What is an appropriate reliability value to use for the rod control system and the justification for it?**

Clarification: The rod control system compensates for reactivity changes caused by fuel burnup. Final compensation for fuel burnup is periodically made with adjustments to boron concentration in the coolant. The rod control system then re-adjusts the control rods in response to changes in the reactor coolant temperature due to changes in the boron concentration.

The rod control system can be placed in automatic or manual operation. Of interest in this study is automatic operation. When in automatic, the rod control system controls the position of the lead control rod bank in respond to reactor coolant system T_{avg} changes. As the core burns, the T_{avg} decreases, and the rod control system steps out the lead bank to maintain T_{avg} at the appropriate level. Slight adjustments are made to the control rod position by the rod control system on a semi-continuous basis. These small changes can occur many times per day. The operators also monitor T_{avg} to ensure the reactor is operating within the correct parameter limits.

Many failures of the rod control system can be detected immediately through either alarms or unexpected control rod movement. In addition, failure of the rod control system due to failures not immediately detected would be detected by either:

- Operators noting the lack of control rod motion on a daily basis
- Operators noting a decreasing T_{avg}

It is expected that the operators will note the lack of rod motion on a very short term basis, that is, within a working shift. If not, then the lack of rod motion would be identified by the downward trend in T_{avg}, probably over the course of several shifts or, at most, several days. Therefore, any failures in the rod control system would be detected in a relatively short time.

The INPO EPIX database was used to identify failures of the rod control system and reactor control system that could lead to failure of the control rods to insert in response to an ATWS event. The database search was limited to Westinghouse NSSS plants for 2002 and 2003. Sixty-nine events were identified. A detailed review of these events indicated 17 may have impacted the ability of the rod control system (electrical and mechanical) to respond to an ATWS event. Of these 17 failures, 6 were immediately detectable via alarms or unexpected control rod movements. Of the remaining 11 failures, it was not always clear from the event description that the component failure would have failed the rod control system in response to an ATWS event. But for simplicity, and to develop a conservative failure history, it was assumed that these 11 failures would fail the response of the rod control system to an ATWS event.

The number of demands over this two year period was determined as follows:

- Number of units: 48
- Number of years: 2
- Assumed average time at power: 80%
- Percent time operating with the rod control system in automatic: 90%
- Time interval to detect failure of rod control system (see above discussion): 3 days
- Assume at least one demand of the rod control system during the detection interval

Number of demands = 48 units x 2 years x 365 days/year x 1 demand/3 days x 0.8 (% time at power) x 0.9 (% time in automatic) = 8410 demands

Failure probability = 11 failues/8410 demands = 1.3E-03/demand

Two sensitivity cases on this value follow:

<u>Sensitivity Case 1:</u> Assume the rod control system is in automatic only 50% of the time.

Number of demands = 48 units x 2 years x 365 days/yr x 1 demand/3 days x 0.8 (% time at power) x 0.5 (% time in automatic) = 4672 demands

Failure probability = 11 failures/4672 demands = 2.4E-03/demand

<u>Sensitivity Case 2:</u> Assume the time interval to detect failure of the rod control system is 1 day.

Number of demands = 48 units x 2 years x 365 days/yr x 1 demand/day x 0.8 (% time at power) x 0.9 (% time in automatic) = 25229 demands

Failure probability = 11 failures/25229 demands = 4.4E-04/demand

From this discussion, the rod control system is demonstrated to be a highly reliably system and failures that do occur will be detected in a short period of time. Therefore, there is a low probability that the rod control system will fail to respond as require to an ATWS event.

It should be noted that the PRA presented in the WCAP assumes an unavailability of 0.1 for the rod control system. Given that the rod control system is in automatic, the WCAP analysis is based on a very conservative value.

2. What are the assumptions and inputs used in the analysis to determine the time for the RCS pressure to reach 3200 psig?

Clarification: The 90 second time is based on the Loss of Load peak pressure cases documented in Tables 4-20 and 4-21 of the topical report. These cases are based on the following key assumptions:

- Pressure limiting Loss of Load event
- Generic ATWS model for 4-loop plant
- Model 51 steam generator

- Core power of 3579 MWt
- Nominal plant operating initial conditions
- AMSAC signal initiates AFW at 60 seconds
- Cases with and without rod control modeled

Cases 1 through 6 of Table 4-20 and cases 13 through 18 of Table 4-21 were reviewed. The times that the RCS pressure reached 3200 psig are now tabulated. It can be seen that the minimum time calculated is about 93 seconds for the case with the bounding reactivity core model, 0 PORVs available, one half of AFW flow available, and no rod control. All of the other cases resulted in longer times to reach 3200 psig. Based on this data, a conservative time of 90 seconds was chosen.

References

1. WOG-04-257, "Response to Clarification Request – WCAP-15831-P 'WOG Risk Informed ATWS Assessment and Licensing Implementation Process'," May 17, 2004.