ADAMS ACCESSION NUMBER ML011200002

ADAMS PACKAGE NUMBER ML011200003

Final Report

Regulatory Effectiveness of the Anticipated Transient Without Scram Rule

April 30, 2001

Prepared by:
William S. Raughley
George F. Lanik

Regulatory Effectiveness Assessment and Human Factors Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

ABSTRACT

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research is reviewing selected regulations to determine if the requirements are achieving the desired outcomes. This initiative is part of an evolving program to make NRC activities and decisions more effective, efficient, and realistic. The goal of this evaluation is to determine whether the anticipated transient without scram (ATWS) rule and the recommendations issued with it were effective. The effectiveness of the ATWS rule was determined by comparing regulatory expectations to outcomes. The report concluded that the ATWS rule was effective in reducing ATWS risk and that the cost of implementing the rule was reasonable. However, uncertainties in reactor protection system reliability and mitigative capability may warrant further attention to ensure the expected levels of safety are maintained.

CONTENTS

TABLES

APPENDICES

TABLES

FIGURES

EXECUTIVE SUMMARY

As part of the Nuclear Regulatory Commission (NRC) program to address regulatory effectiveness, the Office of Nuclear Regulatory Research (RES) is reviewing selected regulations to determine if the requirements are achieving the desired outcomes. SECY-97-180, "Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report," August 6, 1997, describes a plan for the RES staff to assess the effectiveness of several major safety issue resolution efforts.

An anticipated transient without scram (ATWS) is an anticipated operational occurrence followed by failure of the reactor trip portion of the protection system. The likelihood of core damage from an ATWS depends on three factors: (1) the initiating event frequency, (2) the reliability of the reactor protection system (RPS), and (3) the reliability of ATWS mitigation systems.

During ATWS rule development there was considerable disagreement about the reliability of the RPS. Compared to other systems, the RPS is quite reliable – the failure rate is likely less than one in ten thousand demands. However, the strong dependence of ATWS risk on RPS reliability and the uncertainty associated with the value of RPS reliability were major factors in the decision to adopt the ATWS rule.

The ATWS rule required the installation of hardware to improve the nuclear plant's capability to prevent an ATWS and mitigate its consequences. The Commission also issued two recommendations with the ATWS rule to (1) reduce the number of automatic scrams, and (2) improve RPS reliability.

The goal of this assessment is to determine whether the ATWS rule and other relevant Commission recommendations issued with the ATWS rule were effective in achieving the desired outcome and whether certain areas may need attention. For the purposes of this assessment, the regulatory documents are considered effective if the expectations (desired outcomes) are being achieved. The expectations were established from objective measures stated in the ATWS rule and accompanying regulatory documents and compared to the outcomes in the areas of system modifications and operating limitations, risk, and value-impact. The outcomes were obtained from reviews of documents and operating experience after the issuance of the ATWS rule. The value-impact assessment determines if the industry's costs to implement the ATWS rule were reasonable.

The assessment concludes that the ATWS rule has been effective in installing modifications, reducing ATWS risk, and implementing the rule at reasonable cost. However, uncertainties in RPS reliability and mitigative capability warrant continued attention consistent with the NRC performance goals to maintain the expected levels of safety and to improve effectiveness. To elaborate:

- Hardware modifications and operating limitations required by the ATWS rule were implemented. All pressurized-water reactors (PWRs) installed diverse means to trip the turbine and initiate auxiliary feedwater. Combustion Engineering, Inc. (CE) and Babcock & Wilcox Co. (B&W) plants installed a diverse scram system. Westinghouse plants generally maintain an "unfavorable moderator temperature coefficient (MTC)" of one percent.

Boiling-water reactor (BWR) plants implemented diverse recirculation pump trip, alternate rod insertion circuitry, and upgraded emergency operating procedures; or installed high capacity standby liquid control systems.

- The mean frequency of automatic scrams (initiating events for ATWS) decreased from approximately 4/reactor years in 1983 to 0.5/reactor years since 1997. This alone accounts for a reduction of nearly one order of magnitude in the frequency of an ATWS – P(ATWS).

- RPS reliability dominates the risk from an ATWS. There have been no total failures of the RPS system since the ATWS rule was issued in 1984. Point estimates of RPS reliability, based on operating experience since 1984 show that the mean RPS unreliability (one minus RPS reliability) expectations have been met for all four reactor groups and are approximately an order of magnitude better than the RPS reliability estimates before the ATWS rule. These generic estimates were developed using a probabilistic risk assessment (PRA) model for the RPS and failure rates of components.

- PWR scram system reliability is related to reactor trip breaker reliability. As evidenced by NRC generic communications and industry group activities, circuit breaker problems continue to occur. Industry programs to maintain scram system reliability continue to be useful in limiting risk from ATWS.

- During the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05 per reactor year. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected CDF of an unmitigated ATWS. Updating the original generic ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that on a generic basis, all four reactor types achieved the ATWS rule risk goal. A few individual plants had somewhat higher risk as a result of using lower levels of RPS reliability rather than the industry average or assumed additional equipment failures.

- Comparison of the estimated value-impact expectations in the original ATWS regulatory analysis to the corresponding outcome indicates that the costs to implement the ATWS rule were less than expected. This is largely due to fewer than expected spurious scrams caused by ATWS equipment than assumed in the original analysis.

- The higher fuel burnup has resulted in previously unpredicted oxide growth and fuel assembly distortion. In some cases this has resulted in slow or incomplete control rod insertion. These failures and degradations are new phenomena and were not considered during the development of the ATWS rule.

Although past data indicates that the risk from ATWS is in the range foreseen when the ATWS rule was issued, several issues have the potential to erode past achievements. Attention to these issues and regulatory actions that maintain compliance with current regulations can assure that the risk from ATWS remains acceptable. These issues are:

- RPS reliability estimates are subject to large uncertainties. Current point estimates developed using RPS probabilistic risk assessment models show upper and lower bounds of unreliability ranging from 1.8E-6 to 5.7E-5. RPS reliability requirements are so high and

ATWS events are so rare that many more years of operating experience are needed to generate sufficient system demands to reduce current estimates of the uncertainty. Licensee's risk calculations in support of licensing actions that could affect ATWS risk should address these uncertainties.

- ATWS mitigation capability on a PWR is highly dependent on the moderator temperature coefficient (MTC). Mitigative functions are considered by the ATWS rule regulatory basis to be non-viable if the ATWS peak pressure exceeds 3200 psig; and a sufficiently negative MTC will limit the ATWS peak pressure. Fuel design to achieve longer cycles and higher power ratings may result in less negative MTCs at full power for a larger fraction of the cycle time, during which time ATWS mitigation may be less effective. Combustion Engineering, Inc. and Babcock & Wilcox Co. reactors installed a diverse scram system to compensate for large exposure times. Further fuel cycle changes and power upgrades that could affect the ATWS risk may require compensatory measures (e.g., hardware or procedural), consistent with the underlying regulatory basis behind the ATWS rule.

- ATWS mitigation on a boiling-water reactor (BWR) is highly dependent on operator actions. Although improvements in design, procedures, and training since the ATWS rule was issued should have contributed to improved mitigative response capability, BWR operator response to an ATWS continues to be a challenge. Probabilistic risk assessment/individual plant examinations for BWRs indicate large variations in the assumptions for reliability of human actions in response to an ATWS. Similarities in design, procedures, and training argue against such variability. Licensee's risk analyses in support of licensing actions should be supported by technical justification of operator performance assumptions.

ABBREVIATIONS

AFW	auxiliary feedwater
AMSAC	ATWS mitigating system actuation circuitry
ARI	alternate rod injection
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox Co.
BWR	boiling-water reactor
CCF	common-cause failure
CDF	core damage frequency
CE	Combustion Engineering, Inc.
CFR	*Code of Federal Regulations*
DSS	diverse scram system
EOP	emergency operating procedure
FR	*Federal Register*
GE	General Electric Co.
HEP	human error probability
IN	information notice
IPE	individual plant examination
LER	licensee event report
MSIV	main steam isolation valve
MTC	moderator temperature coefficient
NRC	Nuclear Regulatory Commission, U.S.
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RCS	reactor coolant system
RES	Nuclear Regulatory Research, Office of (NRC)
RPS	reactor protection system
RPT	recirculation pump trip
RTB	reactor trip breaker
RTS	reactor trip system
SLC	standby liquid control
TS	technical specification
UET	unfavorable exposure time
USI	Unresolved Safety Issue
W	Westinghouse

1 INTRODUCTION

As part of the Nuclear Regulatory Commission (NRC) program to assess regulatory effectiveness, the Office of Nuclear Regulatory Research (RES) is reviewing selected regulations to determine if the requirements are achieving the desired outcomes. SECY-97-180, "Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report," August 6, 1997 [Ref. 1], describes a plan for the RES staff to assess the effectiveness of several major safety issue resolution efforts.

The work described in this report is an assessment of the anticipated transient without scram (ATWS) rule. An ATWS is an anticipated operational occurrence followed by failure of the reactor trip system (RTS) portion of the protection system. The requirements and Commission recommendations for addressing ATWS were published on June 26, 1984, in the Federal Register (FR) (49FR26036) as Section 50.62 of Title 10 of the *Code of Federal Regulations* (CFR) (10 CFR 50.62), "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water-Cooled Nuclear Power Plants" [Ref. 2]. The Commission intended the ATWS rule requirements to provide further assurance that failure of the reactor to scram following an anticipated operational transient would not adversely affect the public health and safety.

2 BACKGROUND

A number of vendor, industry, and NRC staff studies during the 1970's gave conflicting results on the reliability of the reactor protection system (RPS) system, the probability of an ATWS, and the core damage frequency (CDF) from an ATWS. During ATWS rule development there was considerable disagreement about the "correct" or "appropriate" value of RTS unavailability[1] which is pivotal to the ATWS issue; NUREG-460,Volume 1, "Anticipated Transient with Scram for Light Water Reactors," April 1978 [Ref. 3], indicates RPS unreliability ranged from 3.0E-06 to 1.1E-04 depending on the type of operating experience considered (naval or commercial nuclear), inclusion or exclusion of failure data, and whether the differences in pressurized-water reactor (PWR) or boiling-water reactor (BWR) RPS designs were considered in the reliability estimates. The Commission designated ATWS as Unresolved Safety Issue (USI) A-9, "Anticipated Transient Without Scram," to determine whether ATWS, as a potentially significant contributor to the CDF, called for additional safety requirements.

Following precursor ATWS events at a BWR (Brown's Ferry Unit 2) in 1980 and a PWR (Salem Unit 1) in 1983, the NRC staff completed a regulatory analysis in SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," July 19, 1983 [Ref. 4], which concluded that additional ATWS safety requirements were justified. SECY-83-293 states that a pivotal aspect of the ATWS issue is the reliability of the RPS and provides estimates of RPS unreliability on the order of 2E-4, with large uncertainty. The ATWS rule required the installation of hardware to improve the nuclear plant's capability to prevent an ATWS and mitigate its consequences. The uncertainty associated with the RPS reliability was

[1] Note, for the purpose of this assessment, the reliability of the RPS is represented numerically by it's complement – unreliability. This is standard practice when describing systems of high reliability, (i.e. unreliability of 1E-4 rather than reliability of .9999). Industry and NRC studies often use RPS unavailability, unreliability, and reliability interchangeably; however, regardless of the measurement, the units are almost always failures per demand.

a major factor in the decision to adopt the ATWS rule. USI A-9 was resolved with the publication of the ATWS rule.

The ATWS rule states that PWRs designed by Westinghouse Electric Corp. (W), Babcock & Wilcox Co. (B&W), and Combustion Engineering, Inc. (CE) are required to have equipment that is diverse, reliable, and independent from the RTS to automatically initiate the auxiliary feedwater (AFW) system and initiate a turbine trip under conditions indicative of ATWS. This equipment is called ATWS mitigating system actuation circuitry (AMSAC). In addition to AMSAC, B&W and CE reactors are required to have a diverse scram system (DSS) that is reliable and independent from the RPS as compensatory measures for higher unfavorable MTCs to prevent potentially excessive reactor coolant system (RCS) over-pressure.

The ATWS rule also states that BWRs, which are manufactured by General Electric Co. (GE), are required to have a diverse alternate rod injection (ARI) system and a diverse recirculation pump trip (RPT) that are reliable and independent from the RPS. BWRs have a standby liquid control (SLC) system to inject borated water into the reactor vessel and the ATWS rule also specified a minimum SLC system injection rate and boron concentration.

The ATWS rule required licensees to submit a proposed schedule for compliance with the rule and information sufficient to demonstrate compliance. In responding to the ATWS rule, the owner's groups for each of the four U.S. reactor vendors developed generic design packages that were then tailored to individual plants for implementation. The NRC staff evaluated licensees' implementations on the basis of the design information they submitted. Compliance with the ATWS rule was typically verified by subsequent inspections.

2.1 ATWS Risk

The likelihood of core damage from an ATWS depends on three factors: (1) the initiating event frequency (anticipated transients requiring scrams), (2) the reliability of the RPS, and (3) the reliability of ATWS mitigation systems.

During ATWS rule development, P(ATWS) was the measure of risk proposed by the industry and adopted by the NRC staff in SECY 83-293. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity. P(ATWS) can be viewed as the expected CDF of an unmitigated ATWS. Values of P(ATWS) shown in SECY-83-293 for the GE, W, B&W, and CE reactor groups using the simplified event trees and data shown in Appendix A, "ATWS Rule Event Trees."

The following three subsections summarize BWR and PWR ATWS sequences and event trees, and operation action that were used in the ATWS rule development to evaluate P(ATWS).

2.1.1 Boiling-Water Reactor ATWS Sequence and Event Trees

The BWR ATWS sequence starts with an anticipated transient and the electrical or mechanical failure of the RPS. This is followed by a RPT to reduce power and by a turbine trip. Increased RCS pressure results in the safety relief valves discharging steam to the suppression pool. The residual heat removal (RHR) system is aligned to remove heat from the suppression pool. The steam flow heats up the suppression pool until the reactor power level can be reduced. A

suppression pool temperature of 200 °F was identified as the unacceptable plant condition used in the ATWS rule development. The suppression pool temperature limit was set by the resolution USI A-39, "Determination of Safety/Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments," in NUREG-0783, "Suppression pool Temperature Limits for BWR Containments," November 1981 [Ref. 5]. In some cases, the NRC staff has accepted higher temperature limits based on plant specific design features. As the suppression pool temperature increases to near the boiling point, steam condensation is less effective, resulting in higher containment pressure; and the RHR pumps would eventually fail due to inadequate net positive suction pressure. During an unmitigated ATWS, with continued heat input to the suppression pool, the containment could fail while the core was still intact.

The regulatory analysis considered three BWR sequences modeled in separate event trees. One sequence is an "isolation transient", where the main steam isolation valves (MSIVs) isolate the bypass to the condenser and divert the steam flow to the suppression pool. The second sequence is a "nonisolation transient", where the MSIVs do not isolate the condenser, allowing steam flow to both the condenser and the suppression pool. The isolation transient is more severe than the nonisolation transient because the suppression pool heats up quicker. It was assumed in SECY-83-293 that 30 percent of transients would be isolation transients and 70 percent nonisolation transients. The third sequence applies to plants with automatically initiated SLC systems.

Low frequency events, such as MSIV closure, which would result in core damage unless the control rods insert immediately are not included in the risk analysis.

Operator actions are required to mitigate the isolation transient and the nonisolation transient. The operator must enter the emergency operating procedures (EOPs) to initiate SLC and lower the reactor water level to decrease power before the suppression pool temperature reaches the 200 °F limit. It was estimated in SECY-83-293 that the operator must start SLC in approximately 2 minutes for the isolation transient and 17 minutes for nonisolation transients to prevent the suppression pool from heating up above its 200 °F limit. The human error probabilities (HEPs) for the operator actions were considered in SECY-83-293 based on estimates of the time the operators had to react, shorter reaction times requiring higher HEPs. The HEPs levels considered were 0.005-1.0 for a low-stress situation, 0.2-1.0 for high-stress situation (both levels based on NUREG/CR-1273, "Handbook of Human Reliability With Emphasis on Nuclear Power Plant Applications," August 1983 [Ref. 6]), and 0.01-1 proposed for the utility group. ATWS is generally regarded as a high-stress situation because the BWR emergency procedure guideline/EOPs for an ATWS event typically require the operating crew to execute several difficult steps and contingency actions simultaneously; HEPs used in SECY-83-293 were 0.05-0.5. Although the ATWS rule required installation of RPT and ARI, and defined levels of SLC injection, mitigation remains dependent on operator action.

Appendix A contains the simplified event trees as shown in Figure A-1, "ATWS Rule Event Tree for GE Reactor Group," and Figure A-2, "ATWS Rule Event Tree for GE Reactor Group With Automatic SLC Initiation." Appendix A also shows data used to calculate P(ATWS) in each event tree and a summary of the calculation.

2.1.2 Pressurized-Water Reactor ATWS Sequence and Event Trees

The PWR ATWS sequence also starts with an anticipated transient and the electrical or mechanical failure of the RPS. In a PWR, the ATWS transient results in a RCS pressure rise, the magnitude and timing of which is dependent on the moderator temperature coefficient (MTC), the relief capacity, and the energy removal capacity of the steam generators. The MTC is a measure of the reduction in the core reactivity as the water temperature increases. The energy removal capacity of the steam generators depends largely on the secondary side inventory. Loss of main feedwater can result in the steam generator quickly going dry. Turbine trip, which reduces steam flow and preserves steam generator inventory, can delay steam generator dry-out and the consequent ATWS pressure pulse.

During an ATWS, the primary coolant temperature increases, since with the turbine trip or loss of feedwater, heat removal is diminished while the reactor continues to generate power. For a PWR with a negative MTC, an increase in the primary coolant temperature provides negative reactivity feedback to reduce reactor power as the primary coolant temperature increases. The fuel temperature coefficient, also known as the Doppler coefficient, which is always negative, also works to reduce reactor power. However, the fuel temperature coefficient only comes into play after the primary coolant temperature has increased because of the delay caused by the thermal time constant of the fuel; this can be an important factor in a fast moving transient such as ATWS. Since, as a whole, the combined steady state fuel coefficient and MTC are always negative, reactor power tends to come into balance with energy removal, but at a higher temperature and pressure. Higher temperature expands the primary coolant, with the potential for causing the pressurizer to go water solid.

Primary system pressure does not increase rapidly until the steam generators near dry-out. With loss of main feedwater, steam generator dryout would generally occur in less than two minutes. Without loss of main feedwater as initiating or consequential event, energy removal from the steam generator could continue until condensate was depleted and the main feedwater system tripped.

Once the steam generators near dry-out, the pressure increase is very rapid. In this part of the transient, since the MTC acts first, it has a major impact on the peak pressure of the fast-moving ATWS event. However, unless core power can be reduced by inserting the control rods or injecting boron, reactor pressure would remain high and coolant would continue to be lost through the primary system safety valves.

Two aspects of system response need be considered: (1) the integrity of primary pressure boundary under peak ATWS pressure and 2) the capability to inject borated water to shut down the reactor and maintain core coverage. The peak ATWS pressure is primarily a function of the MTC and the primary system relief capacity. The capability to inject borated water could be affected if the initial peak pressure deformed or disabled valves in the injection path or if the primary system pressure remained higher than the shutoff head of the high-head injection pumps for an extended time.

In SECY-83-293, the ASME Service Level C pressure of 3200 psig was assumed be to an unacceptable plant condition during ATWS rule development. A higher ASME service level was considered for B&W and CE plants but rejected on the basis that the RCS pressure boundary could deform to the point of inoperability. Also steam generator tubes might fail before other

primary coolant system components and bypass containment. Even if the peak pressure is limited to acceptable levels, the pressure must be reduced to successfully inject borated water from the high-pressure injection system. The pressure is reduced by removing heat through the steam generators, the AFW system, and steam generator relief valves. If the pressure cannot be reduced, reactor coolant will be lost through the primary safety and relief valves and the core uncovered. W plants generally have a larger relief capacity than the B&W and CE plants and are more tolerant of an ATWS. For PWRs, it is likely that for an unmitigated ATWS, the core would melt with the containment intact.

The MTC becomes more negative (less positive) later in the fuel cycle; and the MTC is more negative (less positive) at 100 percent power than at lower power. During the first part of the fuel cycle below 100 percent power, the MTC can be positive or insufficiently negative. If an ATWS occurs when the MTC is either positive or insufficiently negative to limit reactor power and the ATWS pressure increase, all subsequent mitigative functions are likely to be ineffective. The period of the fuel cycle time when the MTC is insufficiently negative to maintain the RCS pressure below 3200 psig during an ATWS is designated "unfavorable MTC."

Appendix A contains the simplified generic event trees as shown in Figure A-3, "ATWS Rule Event Tree For Westinghouse Reactor Group," and Figure A-4, "ATWS Rule Event Tree For B&W/CE Reactor Group." Appendix A also shows data used to calculate P(ATWS) in each event tree and summary of the calculation.

At the time of the ATWS rulemaking, the estimated impact of periods of “unfavorable MTC” was included in the event tree as the “MTC Overpressure” factor. For Westinghouse plants, the original ATWS expected risk approach estimated an “MTC Overpressure” factor of 0.01 for turbine trip scenarios and 0.1 for non-turbine trip scenarios; and the expected outcome of the ATWS rule with installation of AMSAC assumed that Westinghouse plants would be in the turbine trip scenario 70 percent of the time (i.e., 70 percent of the ATWS transients would be the lower risk scenario). As stated above, the turbine trip provides the operator additional time to respond.

For CE/B&W plants, a higher “MTC Overpressure” factor of 0.5 was used regardless of turbine trip status. The reason was that the MTC was expected to be insufficiently negative to limit peak pressure below 3200 psig for up to 50 percent of the cycle. For CE/B&W plants, the expected outcome of the ATWS rule included a factor of 10 improvement in the electrical portion of the scram system based on installation of the DSS. These same assumptions were used to determine the current outcomes.

Like the BWR risk analysis, this risk model does not include low frequency “anticipated operational occurrences” such as loss of primary system flow for which core damage occurs, regardless of the MTC or other factors, unless the rods insert immediately.

2.1.3 Operator Action

In developing the ATWS rule, the NRC staff view was that (1) operator action should not be relied upon during the first ten minutes of an accident including a manual scram and (2) operator actions should be relied upon later in the course of an accident if the condition in the reactor and mitigating systems is available to the operator, that sufficient time is available to assess the condition and take action, and that the operator has been trained in the action.

In practice, operators are trained to identify an ATWS condition, initiate a manual scram, manually insert the control rods, manually trip the RPS motor-generators, and begin borating. After the Salem events and before the ATWS rule was issued, there was discussion in NUREG-1000, “Generic Implications of ATWS Events at the Salem Nuclear Power Plant”, April 1983, regarding credit for operator action to scram the reactor following an ATWS, since the Salem operator scrammed the reactor 30 seconds after the RTS failed. NUREG-1000 stated that the Salem ATWS events were not as serious as they could have been as they took place at low power, the operators were at the correct control panel to support power ascension, and there was sufficient relief capacity and steam generator mass.

Operator recognition that an ATWS condition exists can be difficult, especially for events involving malfunctions of the reactor trip system which do not activate the trip indicator lights. The fast moving ATWS event does not provide much time for the operator to recognize what has occurred. For example, NUREG-1000 provides peak pressure versus time curves that show that if the operator takes no action following an ATWS, the ASME service level C stress would be exceeded in 47, 70, and 100 seconds for CE, B&W, and W PWRs, respectively. In addition, the pressure ramps up from a normal operating values of 2250 psia to its peaks of approximately 3500, 4000, and 3500 psia in 47, 20, and 10 seconds for CE, B&W, and W PWRs, respectively.

2.2 Commission Recommendations for Reducing the Risk From An ATWS

When the Commission issued the ATWS rule in 1984, it made two recommendations to reduce the risk from an ATWS: (1) the industry develop a reliability assurance program to achieve and maintain the desired high level of RTS reliability and (2) licensees reduce the frequency of challenges to plant safety systems by reducing the initiating event frequency. Each of these recommendations is discussed in more detail below.

In 49FR26036 the Commission stated that a pivotal aspect of the ATWS issue is the reliability of the RTS and the difficulty associated with assessing the reliability of a system designed for very high reliability. The Commission noted that despite perceived high reliability, two precursor events had occurred.

One of the principal findings from the Salem ATWS event was of lack of attention to RTS reliability. Accordingly, the staff recommended to the Commission that a reliability assurance program be included in the final ATWS rule (NUREG-1000, Volume 1, "Generic Implications of the ATWS Events at the Salem Nuclear Power Plant," April 1983 [Ref. 7]). While the ATWS rule does not require such a program the Commission recommended a RTS reliability program with the following elements: (1) an analysis of the challenges to and failure modes of the RTS system, considering independent and common-cause failures; (2) a numerical performance standard for the RTS challenges and RTS unavailability to aid in the initial and continuing evaluation of the adequacy of the system; (3) a process for evaluating operating experience to provide feedback to assess whether the RTS is performing reliably enough; and (4) procedures within quality assurance programs to ensure that the RTS performs satisfactorily as the frequency of challenges to the RTS should be as low as practicable.

In 49FR26036 the Commission also observed that licensees did not have a formal program to reduce the number of automatic scrams and urged licensees to analyze challenges to safety systems to determine where improvements could be made. In a Commission briefing on

June 24, 1984, key industry representatives stated that each utility would adopt practices and policies to reduce the number of automatic reactor trips, including identifying and correcting the root cause of all automatic trips; that one representative had agreed to collect, analyze, and trend industry data; and that licensees of plants operated more than 3 years, would strive to reduce the average number of automatic trips in 1985 to three. They also stated that the average number of automatic scrams per reactor had fallen from six in 1980 to four in 1983.

2.3 NRC Regulatory Analysis for the ATWS Rule

SECY-83-293 contains an analytical "baseline" value of P(ATWS) based on operating data and assumed expected values of P(ATWS) and other performance expectations. This value credits installed design features relevant to ATWS or planned modifications to meet regulatory requirements to carry out initiatives unrelated to the ATWS rule. For example, after the Salem event the baseline for the RPS reliability for W PWRs was found to be 2E-04 per demand. However, SECY-83-293 assumed a smaller failure-to-scram rate of 3E-05 per demand based on credit for the reactor trip breaker (RTB) modifications and attention to RTB test and maintenance as required by GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983 [Ref. 8]. SECY-83-293 shows the potential alternate modifications for the ATWS rule requirements and calculated a value of P(ATWS) for each alternative. The difference between the P(ATWS) for each alternative and the baseline values of P(ATWS) established the risk reductions that were used in the value-impact analysis (discussed in Section 3.3).

3 ASSESSMENT OF THE ATWS RULE

The goal of this assessment is to determine whether the ATWS rule and the other relevant Commission recommendations issued with the ATWS rule were effective in achieving the desired outcomes and whether certain areas may need attention. The assessment reviews and uses plant specific operating experience, and risk and reliability information to make conclusions about the adequacy of the regulatory documents; the assessment does not address plant-specific issues as these continue to be addressed elsewhere in the regulatory process.

3.1 Method for Assessing Regulatory Effectiveness of the ATWS Rule

For the purposes of this assessment, the regulatory documents present the regulatory expectations (desired outcomes) as specific objectives, requirements, and guidance. The regulatory documents are considered effective if the expectations are being achieved. Expectations were established from objective measures stated in the ATWS rule, the accompanying 49FR26036, and SECY-83-293 for comparison to the outcomes in the areas of modifications, risk reduction, and value-impact. The value-impact assessment determined if the industry's costs to implement the ATWS rule were reasonable.

The risk quantification method used to calculate the expected risk in the original ATWS analysis was used for this assessment in order to maintain a consistent approach. In so doing, the improvement in risk that was identified would more obviously be the result of less frequent initiating events and improved equipment performance rather than changes in the risk model or

assumptions. This approach provides a reasonable framework for understanding the basic elements of the ATWS risk – initiating event frequency, scram system reliability, and mitigative reliability.

The ATWS rule generic risk expectations were calculated assuming that the ATWS rule requirements were implemented. Thus, the original analysis assumed that AMSAC, DSS, ARI, RPT trip, etc., were installed and operable.

For PWRs, the original generic risk analysis and this assessment assume that the following equipment is available: emergency feedwater, primary power-operated relief valves and safeties, and secondary power-operated relief valves and safeties – this tends toward lower peak pressures and lower ATWS risk. On the other hand, the original generic risk analysis assumes that the rod control system is in manual control – this tends toward higher peak pressures and higher ATWS risk. Since the affects are offsetting, the total impact may be minimal. Again, to determine whether the ATWS rule was effective in reducing risk from ATWS, we looked to see if it met or exceeded the goal based on the original method of calculation.

The original generic risk analysis event trees did not distinguish those initiating events accompanied by loss of main feedwater. And again, for consistency, neither did this assessment, since the goal of this assessment was to determine if the ATWS rule resulted in improved performance, rather than "improved modeling." However, for W plants, the likelihood of loss of feedwater events was a consideration in adoption of the values of the "MTC Overpressure" factor and separate event trees for turbine trip and non-turbine trip scenarios.

The original generic event trees were developed after much deliberation to focus on what were the most important sequences. The deliberations which went into determination of the type of event trees to use considered aspects of the ATWS transient such as the status of the main feedwater system. The current assessment was not meant to be a total reevaluation of the approach to ATWS risk analysis, but an attempt to determine if the ATWS rule goals were met. In order to make a valid comparison of the improvements due to less frequent initiating events and improved equipment performance rather than changes in the risk model or assumptions, the simplified event tree approach was used. Additionally, we believe that this approach provides a reasonable estimate of ATWS risk.

Operator action to scram the plant was not credited in the original risk estimates of the ATWS rule. To be consistent, the current assessment used the same approach. The goal of this assessment was to compare the ATWS rule expectations and outcomes. Consequently, as long as operator performance has not changed greatly since the ATWS rule was enacted or the impact of including operator action to scram is small, this approach is valid.

> First, regarding changes in operator performance to scram the plant which have occurred since the ATWS rule was implemented, there is no basis to support major improvements. At the time of the ATWS rule, operating procedures and training had already been greatly improved following the Three Mile Island accident; and the Salem ATWS event implications had been included as previously discussed. The implicit pressure on operators to minimize the number of scrams may not have been as strong as it is now.

Secondly, recent work supports the position that the impact of considering operator action is relatively small. “Reliability Study: Westinghouse Reactor Protection System, 1984–1995," NUREG/CR-5500 Volume 2, shows values of RPS unavailability of 2.1E-5 without operator action and 4.9E-6 with operator action – the operator provides reduced risk by a factor of about 4. This improvement in unavailability assumes that the operator initiates a manual scram 99 percent of the time when a scram signal is present and 50 percent of the time even when a scram signal is not present. In other words, these values already assume excellent operator performance. The reliability study did not include the potential for operator action to manually trip the RPS motor generator sets. The motor generator breaker controls are not always accessible from the control room, which would negate the possibility of timely operator action.

Thirdly, the initiating event frequency is assumed to be the rate of automatic scrams. Since the time of the ATWS rule, the fraction of all scrams which are automatic has decreased and the fraction of operator initiated scrams has increased. Manual scrams were not included in the current ATWS initiating event frequency; their inclusion in initiating event frequency would about double the initiating event frequency.

The outcomes were obtained from operating experience as documented by the NRC performance indicator program and licensing event reports (LERs); completed and draft NRC reliability studies for each reactor group; NRC survey of PWR MTCs; and vendor information presented to the NRC. The NRC 1997–1999 performance indicator data used for counting automatic scrams are given in Appendix B, "Plant-Specific SBO Information by Reactor Type and Operating Status." Insights on the initiating event frequency were obtained from NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995," December 1998 [Ref. 9]. NRC RPS reliability studies completed based on operating experience since the ATWS rule are: NUREG/CR-5500, Volume 2, “Reliability Study: General Electric Reactor Protection System 1984–1995,” December 1998 [Ref. 10]; NUREG/CR-5500, Volume 3, “Westinghouse Reactor Protection System 1984–1995,” December 1998 [Ref. 11], the initial draft of NUREG/CR 5500, Volume 4, “ Reliability Study: Babcock & Wilcox Reactor Protection System 1984–1998", March 2000 [Ref. 12]; and the initial draft of NUREG/CR 5500 Volume 10, “Reliability Study: Combustion Engineering Reactor Protection System 1984–1998", March 2000 [Ref. 13]. The RPS reliability data were obtained from NRC studies that model the components of the RPS system as recommended by the Commission in issuing the ATWS rule. The AFW system reliability was obtained from NUREG/CR 5500, Volume 1, “Reliability Study: Auxiliary/Emergency Feedwater Study System, 1987–1995,” August 1998 [Ref. 14]. Information on the plant and reactor group MTCs was obtained from a 1994 NRC survey [Ref. 15], and owners’ group presentations to the NRC [Ref. 16], [Ref. 17], [Ref. 18].

Appendix B gives plant-specific data on the actual outcomes of the ATWS rule regarding modifications and data from probabilistic risk assessment/individual plant examinations (PRA/IPEs). The data were collected from NRC licensee correspondence, particularly licensee correspondence committing to modifications in response to the ATWS rule, and from licensee PRA/IPEs dated from November 30, 1991, to July 27, 1994, as recorded in the NRC PRA/IPE databases. NUREG-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” October 1997 [Ref. 19], was used to obtain ATWS insights based on licensee PRA/IPEs. When using these PRA/IPEs, it was recognized that their data did not always reflect the current design or operating performance of safety systems.

3.2 Comparison of Expectations and Outcomes

Table 1, “Summary of ATWS Rule Expectations and Outcomes,” compares the ATWS rule expectations and outcomes in the areas of system modifications and operating limitations, risk reduction, and value-impact. The expectations and outcomes are discussed in the sections that follow.

Table 1 Summary of ATWS Rule Expectations and Outcomes

ATWS Rule Expectations		Current Outcomes	Observations
Area	**Expected Result**		
System Modifications and Operating Limitations	W: AMSAC, unfavorable MTC 1–10% CE: AMSAC, DSS, unfavorable MTC <50% B&W: AMSAC, DSS,unfavorable MTC<50% GE: ARI, RPT, and SLC capacity; some automatic SLC initiation	All required modifications implemented; trend to increase technical specification MTC limits	Expectation met
Risk Expectations	Average ATWS risk <1.0E-05. W: 5.8E-06 CE: 2.2E-05 B&W: 2.2E-05 GE: 1.2E-05	All plants < 1.0E-05 W: 5.0E-07 CE: 2.8E-07 B&W: 1.3E-07 GE: 1.0E-06	Expectation met but large uncertainties in RPS reliability ATWS mitigating capability sensitive to unfavorable MTC for PWRs and HEP for BWRs.
Value-Impact	Industry cost to implement the ATWS rule: $354M	Industry cost to implement the ATWS rule: $166M	Expectations met Significant economic benefit from reduction of the number of scrams

3.2.1 Modifications and Operating Limitations

Appendix B, Table B-1," Pressurized Water Reactor Data," and Table B-2, "Boiling-Water Reactor Data," summarize the modifications licensees committed to install in response to the ATWS rule. A review of licensing correspondence found that the expected modifications from the ATWS rule were installed, typically between 1986 and 1990.

Table 2, “ATWS Rule Modifications,” was prepared to show the degree of defense in depth provided by the ATWS rule modifications that were intended to prevent an ATWS; to obtain a decrease in potential common-cause failure (CCF) in the RTS; and for the B&W and CE reactors to ensure a reactor trip in view of relatively high Unfavorable MTCs. To assess the degree of defense in depth, Table 2 shows the number of total RPS system failures; RPS trip system design redundancy; the modification expectations (and for PWRs, the Unfavorable MTC); and a summary statement of the outcome.

Table 2 indicates:

(1) The GE ARI with the RPS trip system redundancy appear to provide additional measure to ensure prevention of an ATWS and or ensure a reactor trip from a CCF in the RTS.

(2) The B&W and CE RPS trip system design redundancy and the DSS appear to provide a compensation for the higher unfavorable MTC (up to 50 percent), provide diverse means to prevent an ATWS, and provide protection against a CCF in the RTS. In comparison to the B&W and CE plants, W plants have no need for these compensatory measures because of unfavorable MTC limited to 1 percent. However, 49FR26036 states the requirement for a DSS for W plants would be published separately to provide opportunity for public comments regarding marginal value-impact, accident prevention, and common mode failures. On December 3, 1984 the NRC determined the DSS was not required for W plants.

Table 2 ATWS Rule Modifications

	Reactor Protection System			Expectations		Current Outcomes
	RPS Failures	RPS Trip System		Mods	PWR Fuel Strategy	
GE	1	all plants	primary and backup trip system, one-out-of-two-twice logic	ARI, RPT, SLC		Modifications installed
W	2	all plants	½ RTB, shunt and undervoltage trips	AMSAC	unfavorable MTC 1–10%	Modifications installed
B&W	0	6 plants	1/4 RTB (2 ac, 2 dc), shunt and undervoltage trips SCRs provide diverse trip of regulating rods	AMSAC, DSS	unfavorable MTC<50%	Modifications installed
		1 plant	1/4 RTB, shunt and undervoltage trips, SCRs provide diverse trip of safety rods			
CE	0	2 plants	1/4 contactors	AMSAC, DSS	unfavorable MTC<50%	Modifications installed
		9 plants	1/8 RTB, shunt undervoltage trips			
		3 plants	1/4 RTB (2 W and 2 GE) with automated shunt and undervoltage			

3.2.2 Comparison of ATWS Rule Risk Expectations and Outcomes

SECY-83-293 set the goal that P(ATWS) be no more than 1.0E-05/RY. SECY-83-293 did not treat this goal as a requirement as this goal might be too costly for some plants to achieve. SECY-83-293 established more reasonable risk expectations for each reactor group in terms of P(ATWS), using estimates for the (1) the initiating event frequency, (2) the reliability of the RPS, and (3) the reliability of ATWS mitigation systems. The SECY-83-293 expectations in these three areas that were used to estimate P(ATWS) are given in Table 3, "Summary of ATWS Rule Risk Expectations and Outcomes," under the Expected column for each reactor

group. Under the Outcomes column, Table 3 also lists the corresponding outcomes for each reactor group based on current operating experience data gathered since the ATWS rule was issued as summarized in Appendix A. In a few cases noted in Appendix A, there was no readily available operating experience and the expectations were used to estimate the P(ATWS) outcome. The Appendix A event trees provide a consistent comparison of the regulatory risk expectations to the outcomes.

Table 3 Summary of ATWS Rule Risk Expectations and Outcomes

Risk Factor Means	General Electric		Westinghouse		Babcock & Wilcox		Combustion Engineering	
	Expected	Outcome	Expected	Outcome	Expected	Outcome	Expected	Outcome
Initiating Event Frequency (1/RY)	4.3	0.5	4.0	0.5	4.0	0.5	4.0	0.5
RPS Unreliability	1.2E-05	5.8E-06	3.0E-05	2.1E-05	1.2E-05	5.0E-07	1.2E-05	1.1E-06
Mitigation Unreliability	2.3E-01	3.6E-01	4.8E-02	4.7E-02	4.6E-01	5.1E-01	4.6E-01	5.1E-01
P(ATWS) (unmitigated)	1.2E-05	1.0E-06	5.8E-06	5.0E-07	2.2 E-05	1.3E-07	2.2E-05	2.8E-07

Table 3 indicates that P(ATWS) for each reactor group is < 1.0E-05/reactor year (RY); the industry has met the SECY-83-293 risk goal. Comparison of Table 3 P(ATWS) expectations to outcomes shows that the P(ATWS) outcomes are better than expected by factors of approximately 10, 10, 170, and 80 for the GE, W, B&W, and CE reactor groups, respectively. To elaborate on the reductions in P(ATWS):

(1) The initiating event frequency has been reduced by a factor of 8 demonstrating that the Commission's recommendation to reduce the number of automatic reactor scrams has been very effective in reducing P(ATWS). The NRC performance indicator data in Appendix A was used to develop the initiating event frequency which shows that in 1997, 1998, and 1999, respectively, 70, 67, and 59 reactors, had zero automatic scrams. However the same data indicate that in 1997, 1998, and 1999, respectively indicates 9, 10, and 17 reactors, respectively, had automatic scram rates of 2-4/RY so outliers could increase plant specific ATWS risk.

(2) The data used to develop the RPS reliability indicates there have been no total system failures of the RPS for any reactor group since issuing the ATWS rule. The RPS unreliabilities are by far the smallest P(ATWS) factor (on the order E-04 to E-06) indicating that RPS system failures should be very rare events. Table 3 values indicate that the all reactor groups achieved better than expected improvements in RPS reliability. These numbers were developed using a fault tree model of the RPS system that may not include all failure modes, a question of completeness for all PRA calculations. For comparison, alternate RPS reliability expectations and outcomes shown in Table 4 were developed using other techniques as discussed in Section 3.2.3.

Appendix B, Table B-3, "RPS Unreliability Uncertainties," summarizes the uncertainties that accompany the RPS unreliabilities in Table 3. The RPS unreliabilities in Table B-3 for the B&W and CE reactors do not include the effect of the DSS on RPS electrical unreliability. Table B-3 shows that the outcomes of upper and lower bounds of unreliability range from 1.8-06 to 5.7E-05 and this range is smaller than that noted Section 2.0 before the ATWS rule was issued. Consideration of the lower values of RPS unreliability in Table B-3 would result in smaller values of P(ATWS), while consideration of the upper values in Table B-3 would increase P(ATWS) for each reactor type by a factor of approximately 2.5. Table B-3 also shows the upper bound of uncertainty analysis associated with current estimates are 5.7E-05, 1.4E-05, 2.5E-05, and 0.78E-05 for the W, GE, CE and B&W reactors, respectively[2].

(3) The reliability outcomes for the mitigation systems were about as expected and did not heavily influence the risk reduction[2]. From this, it could be deduced that the mitigation functions described in the ATWS rule (ARI and RPT on BWRs; AMSAC on PWRs) achieved less risk reduction than the Commission's recommendations in 49FR26036 to reduce the rate of reactor scrams.

As previously discussed, ATWS mitigation on a BWR is highly dependent on operator actions. This is discussed further in Section 3.2.4 below

For the W plant mitigation systems unreliability, the value (~5E-2) is largely determined by the unfavorable MTC. Increasing MTC can have a major adverse impact on the unfavorable MTC. This trend is discussed further in Section 3.2.6.

3.2.3 Reactor Protection System Reliability Validity

Table 4, "Alternate Estimates of Reactor Protection System Reliability," was developed to show how reliability estimates vary depending on the type of analysis. SECY-83-293 "baseline" estimates of RPS unreliabilities were obtained by using classical statistics, modeling the RPS as a component, estimating the number of RPS demands from periodic plant test and operating experience at all operating U.S. nuclear plants from initial date of commercial operation through 1983, and failures which include Salem and Browns Ferry. SECY-83-293 stated that counting RPS system failures and system demands was the best way to account for CCF effects.

Table 4 also shows the corresponding updated calculation using the SECY-83-293 methodology and data plus the data since the ATWS rule was issued (1984-1995) from NRC RPS reliability studies (references 11 and 12). These are based on the same failures (Salem and Brown's Ferry) but a much larger number of demands.

For comparison, Table 4 shows the results of NRC RPS reliability studies. The NRC RPS reliability studies results were obtained by modeling of the RPS components (as recommended by the Commission), operating data since 1984 (scram failures at Browns Ferry in 1980 and Salem in 1983 not included), and Bayesian statistics.

---

[2] The DSS is not relevant to the discussion here – the RPS reliability values for CE and B&W do not include the contribution of the DSS as the DSS was not modeled in the RPS reliability studies.

Comparisons of the W and GE reliability expectation and outcomes calculated by the methods used to establish the “baseline” values of SECY-83-293 show only modest improvements in scram system reliability. This is expected since the calculation is based largely in the number of demands – that number increased by a factor of about 2. These results are close to the lower bound reliability numbers in the NRC reliability studies.

The range of values of RPS reliability illustrates the difficulty of estimating reliability values in highly reliable systems. The significance is that the uncertainty of the values of RPS reliability argue for maintaining defense in depth regarding ATWS.

Table 4 Alternate Estimates of Reactor Protection System Reliability

Assumptions and Data	W RPS Unreliability	BWR RPS Unreliability
SECY-83-293 baseline – classical statistics RPS modeled as a component; data prior to ATWS rule W PWR – one RPS failure (Salem) in 4975 demands BWR – one RPS failure (Brown’s Ferry) in 5258 demands	20E-05	19E-05
SECY-83-293 expectation – estimated improvement Assumed improvement in the baseline RPS unreliability	3E-05	1.2E-05
SECY-83-293 baseline update to 1995 – classical statistics RPS modeled as a component; all data to 1995 W PWR – one RPS failure (Salem) in 10182 demands BWR – one RPS failure (Brown’s Ferry) in 8119 demands	9.8E-05	12E-05
NUREG-5500 (Ref 12) – Bayesian statistics with non-informative prior RPS modeled to component level; component failure data from 1984 to 1995 (Salem and Brown’s Ferry events not included); demands estimated based on number of tests and unplanned demands	2.1E-05	0.58E-05

3.2.4 Risk Insights From Licensee Probabilistic Risk Assessment/Individual Plant Examinations

NUREG-1560 provides common ATWS risk perspectives for each reactor group gained from the NRC staff review of 75 of the IPEs submitted to the NRC for 108 nuclear power plants. NUREG-1560 noted that licensee IPEs show that ATWS was relatively unimportant from a risk perspective regardless of the reactor group. However, in some cases the ATWS contribution to core melt was more than 10 percent of the total. NUREG-1560 qualified its conclusion about ATWS CDF, noting the variability in the PRA/IPE modeling of ATWS events for both BWRs and PWRs.

Comparison of the ATWS CDF for all the plants in each reactor group in Appendix B to the P(ATWS) expectation for each reactor group found that six licensees do not meet the ATWS risk goals. Five of those licensees assumed RPS reliability levels lower than industry average and in the lower range of the RPS reliability uncertainties discussed in Section 3.2.2. One PWR licensee assumed an inoperable power-operated relief valve; relief capacity impacts ATWS peak RCS pressure and consequently ATWS risk. Several plants routinely operate with blocked power-operated relief valves and their IPEs may underestimate ATWS risk.

PRA/IPEs for BWRs, discussed in NUREG-1560, indicate large variations in the assumptions about the reliability of human actions in response to an ATWS. NUREG-1560 sampled 33 plants and found the HEP for SLC initiation ranged from 0.0001 to 0.5. NUREG-1560 also sampled 25 plants and found the HEP for automatic depressurization system inhibition ranged from 0.00001 to 0.5. Usually, a low HEP is associated with low stress events. Similarities in BWR design, procedures, and training would seem to indicate that more consistent HEP assumptions should be used in the IPE analyses.

Operating experience supports the view that ATWS is a high stress event. A review of LERs in the last 10 years found only one instance where the ATWS EOPs had been entered. This was the focus of a special inspection documented in NUREG-1455, "Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1994," October 1991 [Ref. 20]. This event was due to the loss of control rod position indication so the operator was unable to verify that the reactor had shut down. Although the rods were actually inserted and the human error was of no consequence, NUREG-1455 indicated that in early steps of the ATWS EOP to control injection of coolant, the operators did not anticipate that depressurizing the reactor would cause the condensate pumps to inject cold water, even though the EOP cautioned that injecting water would induce a power excursion that would could lead to core damage. This event illustrates that BWR EOP implementation is challenging for the operators in an ATWS situation.

ATWS mitigation on a BWR is highly dependent on operator actions. Although improvements in design, procedures, and training since the ATWS rule was issued should have contributed to improved mitigative response capability, BWR operator response to an ATWS continues to be a challenge. PRA/IPEs for BWRs indicate large variations in the assumptions for reliability of human actions in response to an ATWS. Similarities in design, procedures, and training argue against such variability.

3.2.5 Insights From NRC Reliability Studies and Operating Experience

NUREG/CR-5750 analyzed the data for all unexpected reactor trips and revealed that most risk significant initiating frequencies with respect to ATWS have decreased over time. Those events include: total loss of feedwater; loss of instrument or control air; inadvertent closure of MSIVs; and loss of the BWR condenser heat sink. Some observations based on the NRC RPS reliability studies (References 11, 12, 13, and 14) and operating experience are as follows:

NRC RPS reliability studies have identified potential important CCFs. Appendix B, Tables B-4 through B-7 list CCF data from the NRC RPS reliability studies. They identify the RPS components of greatest importance to RPS reliability and risk for each reactor group. Tables B-4 through B-7 show the components of greatest importance, as indicated by the Fusel-Vesely importance measure, the risk ratio increase, and the probability of failure. These tables show that the important common-mode failures, though of low probability, are the train bistables, trip logic relays, undervoltage cards, RTB, and rod and rod mechanisms. In addition, the PWR channel process module and the BWR solenoid-operated valve scram pilot values are also important.

Review of recent operating experience in areas found to be dominant CCFs found the industry is still addressing RTB maintenance/reliability issues that were identified from the Salem ATWS

events, and the phenomena related to control rod insertion was not considered during development of the ATWS rule.

NRC Information Notice (IN) 99-13, "Insights From NRC Inspection of Low-and Medium-Voltage Circuit Breaker Maintenance Programs," April 29, 1999 [Ref. 21], summarizes NRC inspections of licensee circuit breaker maintenance programs at eight nuclear plant sites in 1998. The inspections followed NRC Temporary Inspection Procedure TI 2515/137, "Inspection of Medium-voltage and Low-voltage Power Circuit Breakers," Revision 1, March 9, 1998 [Ref. 22]. TI 2515/67 lists more than 60 INs notifying the industry of circuit breaker problems including INs that address low voltage power circuit breakers of the type used in RPS circuit breaker applications. Although the inspections concluded that the programs were generally adequate, the inspections found some licensees' circuit breaker maintenance programs may need attention. The inspectors reported these concerns to the industry. It appears the industry is resolving these problems through Electric Power Research Institute/Nuclear Maintenance Assistance Center workshops, an Nuclear Energy Institute circuit breaker task force, and circuit breaker users groups all of which have helped plants to fix their circuit breaker problems. However, information from recent circuit breaker users group meetings, indicates that not all the plants have yet initiated effective circuit breaker maintenance programs, as advocated by NRC and the industry groups.

The higher fuel burnup has resulted in previously unpredicted oxide growth and fuel assembly distortion. In some cases this has resulted in slow or incomplete control rod insertion. Table 5, "Control Rod Insertion Events," summarizes recent events and NRC INs involving control rod insertion. While the number of control rods affected is only a portion of the total available rods actually needed to obtain reactor control and shutdown, these failures and degradations are new phenomena and were not considered during the development of the ATWS rule. Although these conditions do not affect present ATWS analysis assumptions directly, they cannot be dismissed as precursor events.

3.2.6 Changes in Fuel Management May Affect Pressurized-Water Reactor ATWS Mitigating Capability

Since the ATWS rule was developed, fuel cycle lengths have been extended. This requires higher enrichment fuel and often results in less negative (more positive) MTCs. As previously discussed in Section 2.1.2, the MTC strongly influences the peak RCS pressure and the ability to mitigate an ATWS; the less negative (more positive) the MTC, the higher the pressure peak.

Pressures for economic efficiency are also prompting licensees to increase the power of the reactor. The higher power rating leads to faster dry-out of the steam generator for PWRs and faster heatup of the suppression pool for BWRs. The combination of less negative MTC and higher reactor power has a greater effect on the plant response to ATWS than either one alone. Less negative MTC and higher power result in the ATWS pressure spike occurring earlier, allowing less time for operator actions, quicker, less time for MTC to limit the pressure increase, and higher, increasing the likelihood of equipment damage. These fuel management changes potentially impact both the deterministic and probabilistic analyses for ATWS and other reactor transients.

Table 5 Control Rod Insertion Events

Reference	Description of Event
LER 289/97-008 Three Mile Island	On 7/21/97an LER after a scram found that 8 of 61 four control rods exhibited slower than normal scram times (4 of 61 were not within technical specification limits) because of a hydraulically induced effect from reduced clearances in the thermal barriers because of deposits on the internal check valves and between the thermal barrier parts. The LER noted that plants conditions during the event are different than during control rod trip insertion time testing performed at hot shutdown. In LERS 95-002, 94-004, and 94-002, the licensee also reported excessive control rod drops times.
Bulletin 96-01, "Control Rod Insertion Problems" (12/8/96)	After three licensees reported that control rods in high-burnup fuel assemblies had insertion times greater than expected, licensees were requested to take prescribed actions to ensure the required shutdown margins are maintained during reactor trip.
IN 96-12, "Control Rod Insertion Problems" (2/16/96)	Three licensees reported that control rods in high burnup fuel assemblies had insertion times greater than expected.
IN 94-40,Supplement 1, " Failure of a Rod Control Cluster Assembly to Fully Insert Following a Reactor Trip at Braidwood Unit 2. (12/15/94)	This informed licensees that after five nuclear plant licensees found that loosened pins have caused control rod(s) to jam, <u>W</u> recommended an inspection at the next outage.
IN 94-72, "Increased Control Rod Droptime From Crud Buildup" (10/5/94)	Two licensees reported increased rod drop times, because crud deposits caused the thermal barrier ball check valves to stick and reduced clearances in the thermal barrier bushing.

Table 6, "ATWS Moderator Temperature Coefficient and Peak Pressure for Pressurized-Water Reactors," summarizes the MTC and corresponding limiting peak pressures of 1979 and 1988 ATWS analyses by the PWR manufacturers. The units of reactivity in Table 6 are pcm/°F (E-05 ΔK/K/°F).

Table 6 ATWS Moderator Temperature Coefficient and Peak Pressure for Pressurized-Water Reactors

Parameters		1979 ATWS Analysis	1988 Update of ATWS Analysis	1994 NRC Technical Specification Survey
CE	MTC	-2.0 to -6.8	-2.6 to -5.7	0 to +3 above 70% power
	Peak Pressure	4290 psia	4153 psia	
B&W	MTC	-10.5	18 month cycle: -11.0 24 month cycle: -4.3	0 above 95% power
	Peak Pressure	3464 psia	3764psia 18 month cycle: > 3200 24 month cycle: > 3200	
<u>W</u>	MTC	-8.0	-8.0 average range -5 to -15	Linear to 0 from 70% to 100% power One plant at + 2 at 100% power
	Peak Pressure	3197 psia	3497 psia (-5pcm/°F)	

In 1987, the NRC staff requested the PWR owners groups to quantitatively reassess the initial MTC analyses that support the ATWS position for PWRs and specifically address reload core designs that have less conservative initial MTCs (a typical letter is given in [Ref. 23]). Table 6 summarizes the MTCs and peak pressure obtained from the owners group responses [Refs. 16, 17, and 18]. The responses generally concluded that no significant changes were required in ATWS analyses. The peak pressure of 3497 psia update in Table 6 for W was calculated based on values obtained from sensitivity analysis in NUREG-460. The sensitivity analysis shows a pressure increase of approximately 100 psi for a change in MTC of 1 pcm/ °F. Thus, the change from the 1979 value of -8 pcm/ °F to -5pcm/ °F in Table 6 corresponds to a 300 psi increase.

The plant technical specifications (TSs) limit the MTC value and require MTC surveillance. Table 6 also summarizes the range of PWR TS MTC limits from a 1994 NRC survey [Ref. 16]. The survey results are listed in Appendix B. Table 6 indicates that the 1994 PWR TSs limit the MTCs to positive or zero levels at full power. These are less negative (more positive) than the 1979 and 1988 MTC values. Based on NUREG-460 sensitivity analysis that indicates a 1 pcm/ °F less negative (more positive) MTC increases the RCS peak pressure approximately 100 psi calculations based on the limiting TS MTC at full power could lead to higher peak ATWS pressures and longer unfavorable MTCs. CE and B&W reactors installed the DSS to counteract risk and peak pressure effects of unfavorable MTC. The B&W and CE DSS the reactor trips independent of the RPS, improving RPS electrical reliability by a factor of 10. The DSS trips the reactor at approximately 2450 psia to prevent the RCS pressure following an ATWS from reaching the unacceptable condition-3200 psia ASME service level C limit.

W WCAP-11992, “Joint Westinghouse Owners Group/Westinghouse Program: Anticipated Transient Without Scram (ATWS) Rule Administration Process,” was formally submitted to the staff in May, 1995. WCAP-11992 introduced the concept of an unfavorable exposure time (UET) which is the percentage of the fuel cycle time the pressure is in excess of 3200 psia; in this respect UET is fundamentally the same as the unfavorable MTC percentage used in the ATWS rule development. However the UET is calculated differently than the unfavorable MTC percentage and in this respect they are different. WCAP-11992 provides a risk based approach to justify UETs up to 38 percent for 18 and 24 month fuel cycles. The NRC staff review [Ref. 24] did not find WCAP-11992 acceptable for use in licensing or other regulatory matters based on several issues. The Westinghouse Owners Group has been working with the NRC staff to address those issues.

Higher peak pressures and longer unfavorable MTCs or UETs lessen the effectiveness of the mitigative functions required by the ATWS rule. If the W assessment were done using the same method as B&W and CE with an “unfavorable MTC” changing from 1 percent to 40 percent of the fuel cycle, the ATWS risk would increase by a factor of about 20 to 4.3E-6. CE and B&W reactors installed the DSS to counteract the risk and peak pressure effects of unfavorable MTC. The effectiveness of the ATWS rule would be compromised if unfavorable MTCs or UETs are increased. In particular, large percentage increases in UET being considered by W plants, without compensating DSS or mitigative capability, challenge the intent of the ATWS rule.

3.3 Value-Impact

SECY-83-293, Appendix C, “Regulatory Analysis for Amendments Related To ATWS” provides a generic value-impact analysis for each reactor group. The value-impact analysis ranked the ATWS rule alternatives for each reactor group by value-impact ratio, the highest value-impact ratio being the favored alternative. Appendix B, Table B-8, “Value Impact Data for Each Reactor Group,” summarizes the industry value-impact baseline information for each reactor group in SECY-83-293. SECY-83-293 determined that the total impact on the industry was $525 million (131 operating plants).

Table 7, “ATWS Rule Value-Impact Summary,” is an update of similar value-impact calculations based on current data. Table 7 shows the calculations of expectations and outcomes normalized for 102 plants so the expected values are not the same as those in SECY-83-293. A comparison of the estimated value-impact expectations in the original ATWS regulatory analysis to the corresponding current outcome indicates that the costs to implement the ATWS rule were approximately $166 million, which is over 50 percent less than the expected $354 million. This difference is largely due to the lower monetized value of releases (in keeping with the reduced probability of ATWS events) and fewer than expected spurious scrams caused by ATWS mitigation equipment.

The SECY 83-293 value-impact analysis did not consider the effects of the industry commitment to reduce the number of scrams in response the Commission’s recommendation in 49FR 26036. The reduction in the number of automatic scrams accounts for a factor of 8 reduction in P(ATWS) and corresponding significant portion of the outcome value. In addition, although not included in the this value-impact analysis, the 3.5 reduction in the average number of automatic scrams for 102 operating reactors has significant monetary value due to replacement power costs.

Table 7 ATWS Rule Value-Impact Summary

Value-Impact Factors	Expectation	Outcome
Value:		
102 x (P(ATWS) expected – P(ATWS) baseline)(30 years)($10 billion)	$1238M	
102 x (P(ATWS) outcome – P(ATWS) baseline)(30 years)($10 billion)		$1521M
Impact		
Design, installation, operation and maintenance	$142M	$142M
Replacement power from plant trip due to spurious actuation of ATWS hardware costs $500,000/day, for 2 to 6 days	$212M	$ 24M
Total	$354M	$166M

In summary, compared to the expectations when the ATWS rule was issued, the value (savings) was greater than expected and the impact (cost) was less than expected. Thus from a value-impact perspective, the ATWS rule was effective.

4 CONCLUSIONS

The assessment concludes that the ATWS rule has been effective in installing modifications, reducing ATWS risk, and implementing the rule at reasonable cost. However, uncertainties in RPS reliability and mitigative capability warrant continued attention consistent with the NRC performance goals to maintain the expected levels of safety and to improve effectiveness. To elaborate:

- Hardware modifications and operating limitations required by the ATWS rule were implemented. All pressurized-water reactors (PWRs) installed diverse means to trip the turbine and initiate auxiliary feedwater. Combustion Engineering, Inc. (CE) and Babcock & Wilcox Co. (B&W) plants installed a diverse scram system. Westinghouse plants generally maintain an "unfavorable moderator temperature coefficient (MTC)" of one percent. Boiling-water reactor (BWR) plants implemented diverse recirculation pump trip, alternate rod insertion circuitry, and upgraded emergency operating procedures; or installed high capacity standby liquid control systems.

- The mean frequency of automatic scrams (initiating events for ATWS) decreased from approximately 4/reactor years in 1983 to 0.5/reactor years since 1997. This alone accounts for a reduction of nearly one order of magnitude in the frequency of an ATWS – P(ATWS).

- RPS reliability dominates the risk from an ATWS. There have been no total failures of the RPS system since the ATWS rule was issued in 1984. Point estimates of RPS reliability, based on operating experience since 1984 show that the mean RPS unreliability (one minus RPS reliability) expectations have been met for all four reactor groups and are approximately an order of magnitude better than the RPS reliability estimates before the ATWS rule. These generic estimates were developed using a probabilistic risk assessment (PRA) model for the RPS and failure rates of components.

- PWR scram system reliability is related to reactor trip breaker reliability. As evidenced by NRC generic communications and industry group activities, circuit breaker problems continue to occur. Industry programs to maintain scram system reliability continue to be useful in limiting risk from ATWS.

- During the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05 per reactor year. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected CDF of an unmitigated ATWS. Updating the original generic ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that on a generic basis, all four reactor types achieved the ATWS rule risk goal. A few individual plants had somewhat higher risk as a result of using lower levels of RPS reliability rather than the industry average or assumed additional equipment failures.

- Comparison of the estimated value-impact expectations in the original ATWS regulatory analysis to the corresponding outcome indicates that the costs to implement the ATWS

rule were less than expected. This is largely due to fewer than expected spurious scrams caused by ATWS equipment than assumed in the original analysis.

- The higher fuel burnup has resulted in previously unpredicted oxide growth and fuel assembly distortion. In some cases this has resulted in slow or incomplete control rod insertion. These failures and degradations are new phenomena and were not considered during the development of the ATWS rule.

Although past data indicates that the risk from ATWS is in the range foreseen when the ATWS rule was issued, several issues have the potential to erode past achievements. Attention to these issues and regulatory actions that maintain compliance with current regulations can assure that the risk from ATWS remains acceptable. These issues are:

- RPS reliability estimates are subject to large uncertainties. Current point estimates developed using RPS probabilistic risk assessment models show upper and lower bounds of unreliability ranging from 1.8E-6 to 5.7E-5. RPS reliability requirements are so high and ATWS events are so rare that many more years of operating experience are needed to generate sufficient system demands to reduce current estimates of the uncertainty. Licensee's risk calculations in support of licensing actions that could affect ATWS risk should address these uncertainties.

- ATWS mitigation capability on a PWR is highly dependent on the moderator temperature coefficient (MTC). Mitigative functions are considered by the ATWS rule regulatory basis to be non-viable if the ATWS peak pressure exceeds 3200 psig; and a sufficiently negative MTC will limit the ATWS peak pressure. Fuel design to achieve longer cycles and higher power ratings may result in less negative MTCs at full power for a larger fraction of the cycle time, during which time ATWS mitigation may be less effective. Combustion Engineering, Inc. and Babcock & Wilcox Co. reactors installed a diverse scram system to compensate for large exposure times. Further fuel cycle changes and power upgrades that could affect the ATWS risk may require compensatory measures (e.g., hardware or procedural), consistent with the underlying regulatory basis behind the ATWS rule.

- ATWS mitigation on a boiling-water reactor (BWR) is highly dependent on operator actions. Although improvements in design, procedures, and training since the ATWS rule was issued should have contributed to improved mitigative response capability, BWR operator response to an ATWS continues to be a challenge. Probabilistic risk assessment/individual plant examinations for BWRs indicate large variations in the assumptions for reliability of human actions in response to an ATWS. Similarities in design, procedures, and training argue against such variability. Licensee's risk analyses in support of licensing actions should be supported by technical justification of operator performance assumptions.

5 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report," SECY-97-180, August 6, 1997.

2. U.S. *Code of Federal Regulations*, Title 10, Part 50, “Requirements for reduction of risk from anticipated transients without scram ATWS) events for light-water-cooled nuclear power plant,” (*Federal Register*, Volume 49, No. 124, page 26036, June 26, 1984).

3. U.S. Nuclear Regulatory Commission, “Anticipated Transient Without Scram for Light Water Reactors,” NUREG-460, Volume 1, April 1978.

4. U.S. Nuclear Regulatory Commission, “Amendments To 10CFR50 Related to Anticipated Transients Without Scram (ATWS) Events,” SECY-83-293, July 19, 1983.

5. U.S. Nuclear Regulatory Commission, “Suppression pool Temperature Limits for BWR Containments,” NUREG-0783, November 1981.

6. Sandia National Laboratories, “Handbook of Human Reliability With Emphasis on Nuclear Power Plant Applications,” NUREG/CR-1273, August 1983.

7. U.S. Nuclear Regulatory Commission, “Generic Implications of the ATWS Events at the Salem Nuclear Power Plant,” NUREG-1000, Volume 1, April 1983.

8. U.S. Nuclear Regulatory Commission, “Required Actions Based on Generic Implications of Salem ATWS Events,“ Generic Letter 83-28, July 8, 1983.

9. U.S. Nuclear Regulatory Commission, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995,” NUREG/CR-5750, December 1998.

10. U.S. Nuclear Regulatory Commission, “Reliability Study: General Electric Reactor Protection System 1984–1995,” NUREG/CR-5500, Volume 2, December 1998.

11. U.S. Nuclear Regulatory Commission, “Reliability Study: Westinghouse Reactor Protection System 1984–1995,” NUREG/CR-5500, Volume 3, December 1998.

12. U.S. Nuclear Regulatory Commission, “Reliability Study: Babcock & Wilcox Reactor Protection System 1984–1998,” initial draft of NUREG/CR-5500, Volume 4, December 1998.

13. U.S. Nuclear Regulatory Commission, “Reliability Study: Combustion Engineering Reactor Protection System 1984–1998,” initial draft of NUREG/CR-5500 Volume 10, March 2000.

14. U.S. Nuclear Regulatory Commission, “Reliability Study: Auxiliary/Emergency Feedwater System, 1987–1995,” NUREG/CR-5500 Volume 1, August 1998.

15. U.S. Nuclear Regulatory Commission, note from G. Holahan to J. Blaha, “Requested Material,” May 13, 1997.

16. Babcock & Wilcox Owners Group ATWS Committee, “Effects of Plant and Fuel Changes on ATWS Basis Presentation to the NRC,” February 18, 1988.

17. Combustion Engineering Owners Group, “Combustion Engineering Owners Group Meeting With the NRC Concerning Trends in MTCs,” January 11, 1988.

18. Westinghouse Owners Group, “Westinghouse program ATWS Rule Administration, Combined Core Performance/Scram Systems Reliability, ACRS Subcommittee Meeting,” February 19, 1988.

19. U.S. Nuclear Regulatory Commission, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” NUREG-1560, October 1997.

20. U.S. Nuclear Regulatory Commission, “Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1994,” NUREG-1455, October 1991.

21. U.S. Nuclear Regulatory Commission, Insights From NRC Inspection of Low- and Medium-Voltage Circuit Breaker Maintenance Programs,” NRC Information Notice 99-13, April 29, 1999.

22. U.S. Nuclear Regulatory Commission, “Inspection of Medium-voltage and Low-voltage Power Circuit Breakers,” Revision 1, NRC Temporary Inspection Procedure TI 2515/137 March 9, 1998.

23. U.S. Nuclear Regulatory Commission, letter from A. Thadani, Office of Nuclear Regulatory Regulation, to Roger Newton, Chairman of Westinghouse Owners Group, “ATWS Moderator Temperature Coefficient,” June 12, 1987.

APPENDICES

APPENDIX A

ANTICIPATED TRANSIENT WITHOUT SCRAM RULE EVENT TREES

DATA FOR BOILING-WATER REACTOR EVENT TREE

SECY-83-293 calculated P(ATWS) for a BWR using the results from the event tree in Figure A-1 (using the data in Table A-1.1 for the isolation transient, assuming 30 percent of transients were isolation transients) and the results from the event tree in Figure A-1 using the data in Table A-1.2 for the nonisolation transient, (assuming 70 percent of transients were nonisolation transients). RPS electrical unreliability values include the channel and trip system. RPS mechanical unreliability values include the hydraulic control unit and the rods. The event tree values for the RPS unreliability expectations reflect the addition of the ARI system under the ATWS rule. SECY-83-293 used the results from the event tree in Figure A-2 and the data in Table A-2.1 to evaluate P(ATWS) for a BWR with automatic SLC initiation.

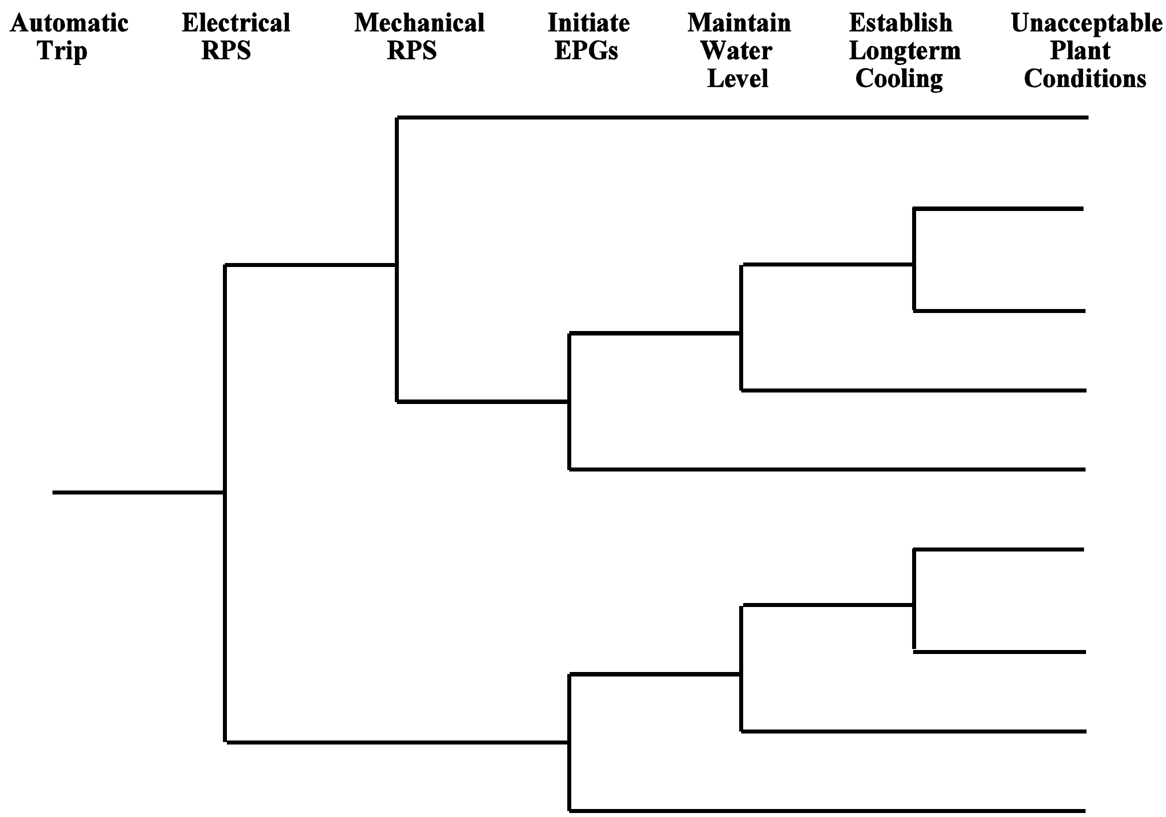

Figure A-1: ATWS Rule Event Tree for General Electric Reactor Group

Table A-1.1 Data for Boiling-Water Reactor ATWS Isolation Transient

Figure A-1 Event	Figure A-1 Likelihood Estimates	
	Expected (SECY-83-293)E	Outcome (reference)
AT	4.3 events/RY (30% of transients)	0.5 events/RY (note 1)
Electrical RPS Reliability	<2E-06	3.7E-06 (Ref. 1)
Mechanical RPS Reliability	1.0E-05	2.1E-06 (Ref. 1)
Initiate EPGs	0.5	0.5 (assumed, no update available)
Maintain Water Level	0.1	0.1 (assumed, no update available)
Establish Long-term Cooling	0.05	0.05 (assumed, no update available)

Table A-1.2 Data for Boiling-Water Reactor ATWS Nonisolation Transient

Figure A-1 Event	Figure A-1 Likelihood Estimates	
	Expected (SECY-83-293)	Outcome (reference)
AT	4.3 events/RY (70% of transients)	0.5 events/RY (Note 1)
Electrical RPS Reliability	<2E-06	3.7E-06 (Ref. 1)
Mechanical RPS Reliability	1.0E-05	2.1E-06 (Ref. 1)
Initiate EPGs	0.5	0.5 (assumed, no update available)
Maintain Water Level	0.05	0.05 (assumed, no update available)
Establish Long-term Cooling	0.05	0.05 (assumed, no update available)

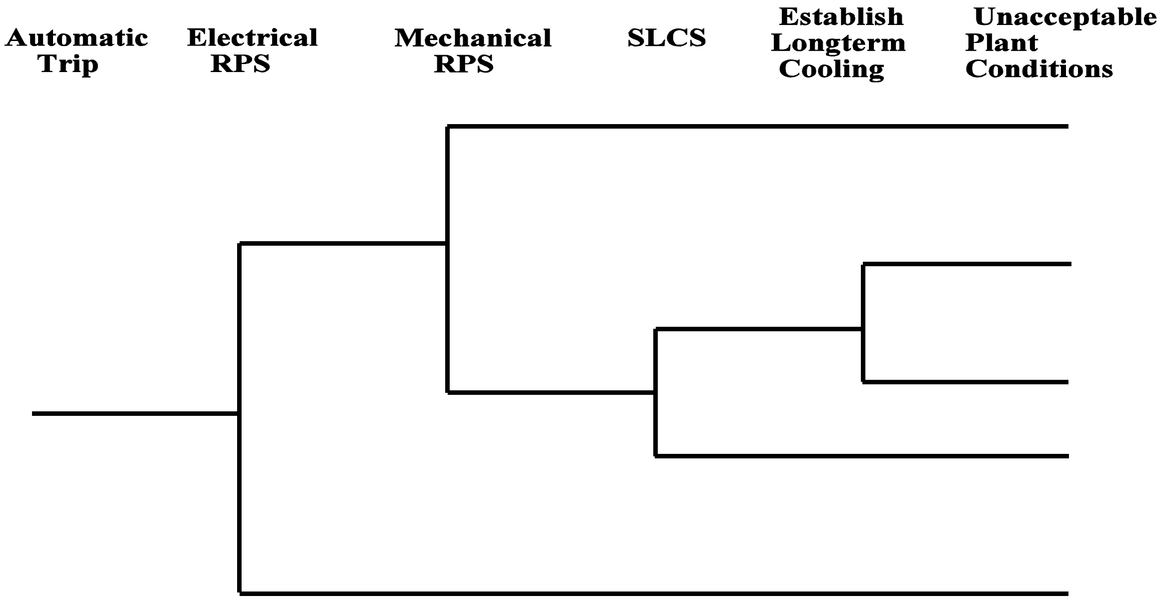

Figure A-2: ATWS Rule Event Tree for General Electric Reactor Group Automatic SLCS

Table A-2.1 Data for Boiling-Water Reactor ATWS With Automatic SLCS Initiation

Figure A-2 Event	Figure A-2 Likelihood Estimates	
	Expected (SECY-83-293)	Outcome (reference)
AT	4.3 events/RY	0.5 events/RY(Note 1)
Electrical RPS Reliability	<2E-06	3.7E-06 (Ref. 1)
Mechanical RPS Reliability	1.0E-05	2.1E-06 (Ref. 1)
SLCS	0.01	1.0E-05
Establish Long-term Cooling	0.05	0.05 (assumed, no update available)

DATA FOR PRESSURIZED-WATER REACTOR EVENT TREE

SECY-83-293 evaluated P(ATWS) for the Westinghouse PWRs using the results from the event tree in Figure A-3 using the data in Table A-3.1 for the turbine trip transient (assuming it occurred in 70 percent of transients) summed with the results from event tree in Figure A-3 using the data in Table A-3.2 for the nonturbine trip transient (assuming it occurred in 30 percent of transients). RPS electrical unreliability values include the train channel, and the trip breakers. RPS mechanical unreliability values include the rods. The event tree values for the RPS unreliability expectations reflect the addition of the shunt trip for the Westinghouse reactor group. SECY-83-293 used the results from the event tree in Figure A-4 and the data in Table A-4.1 to evaluate P(ATWS) for the CE/B&W PWRs.

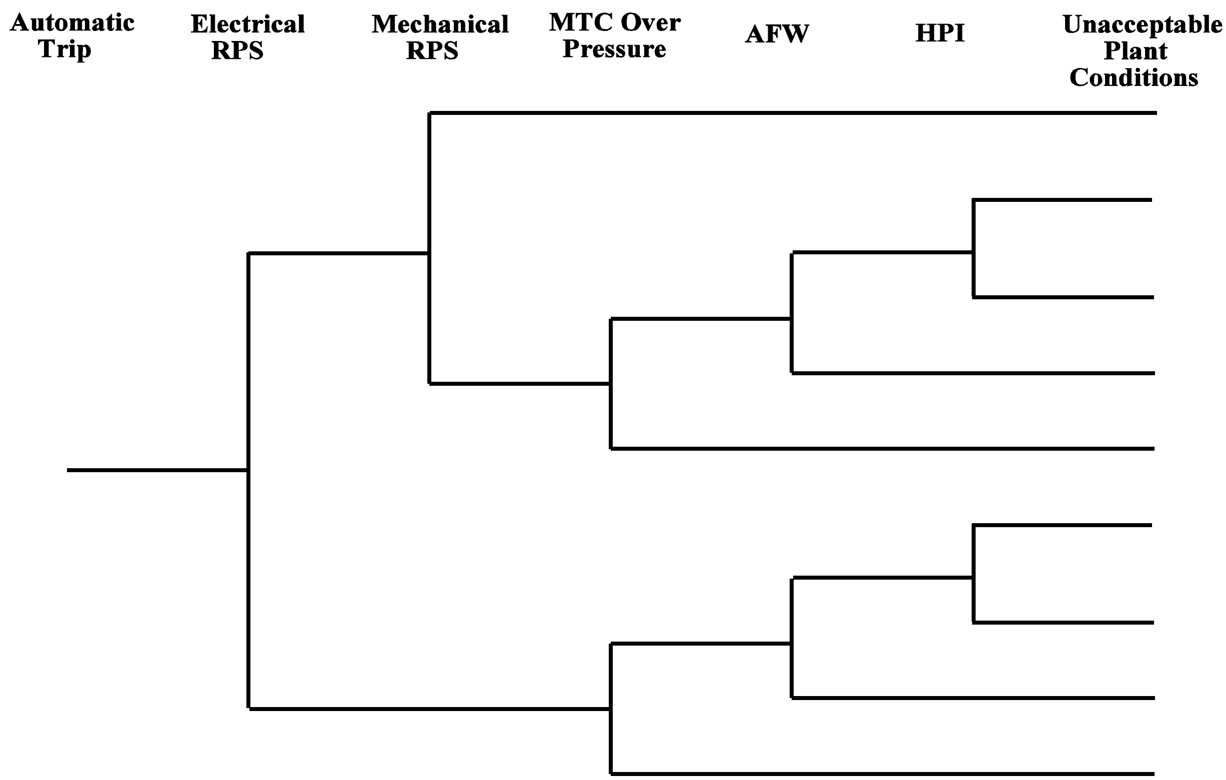

Figure A-3: ATWS Rule Event Tree for Westinghouse Reactor Group

Table A-3.1. Data for Westinghouse Pressurized-Water Reactor ATWS Turbine Trip Transient

Figure A-2 Event	Figure A-3 Likelihood Estimates	
	Expected (SECY-83-293)	Outcome (reference)
AT	4.0 events/RY (70 percent of transients)	0.5 events/RY (note 1)
RPS Electrical	2E-05	2E-05 (Ref. 2 and note 2)
RPS Mechanical	1E-05	1.2E-06 (Ref. 2 and note 2)
MTC Overpressure	0.01	0.01 (Ref. 5)
ATWS Reliability	0.001	0.00045 (Ref. 8)
HPI	0.01	0.01 (assumed, no current data available)

Table A-3.2 Data for Westinghouse Pressurized-Water Reactor ATWS Non-Turbine Trip Transient

Figure A-3 Event	Figure A-3 Likelihood Estimates	
	Expected (SECY-83-293)	Outcome (reference)
AT	4.0 events/RY (30 percent of transients)	0.5 events/RY (note 1)
RPS Electrical	2E-05	2E-05 (Ref. 2 and note 2)
RPS Mechanical	1E-05	1.2E-06 (Ref. 2 and note 2)
MTC Overpressure	0.01	0.01 (Ref. 5)
ATWS Reliability	0.001	1E-05
HPI	0.01	0.01

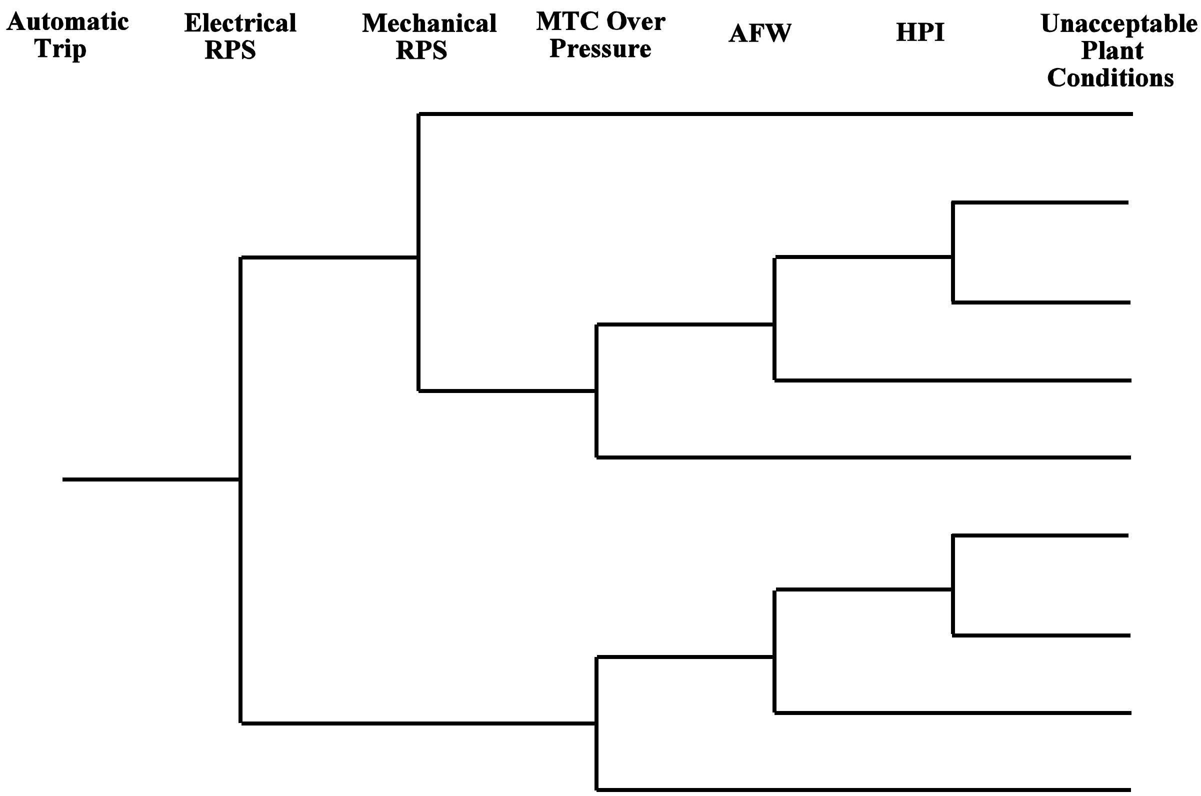

Figure A-4: ATWS Rule Event Tree for Combustion Engineering/ Babcock & Wilcox Reactor Group

Table A-4.1 Data for Combustion Engineering/ Babcock & Wilcox ATWS Transient

Figure A-4 Event	Figure A-4 Likelihood Estimates	
	Expected (SECY-83-293)	Outcome (reference)
AT	4.0 events /RY	0.5 events/RY (note 1)
RPS Electrical*	2E-05	CE: 1.0E-05 (Ref. 3, note 3) B&W: 4.0E-06 (Ref. 4, note 4)
RPS Mechanical	1E-05	CE: 5.8E-08 (Ref. 3, note 3) B&W: 1.0E-07 (Ref. 4, note 4)
MTC Overpressure	0.5	0.5 (Ref. 6, 7)
ATWS Unreliability	0.001	0.00045 (Ref. 8)
HPI	0.01	0.01

* Values used in risk calculation are a factor of 10 less to account for DSS.

NOTES AND REFERENCES FOR APPENDIX A

Note 1: The mean of the 1997, 1998, 1999 BWR and PWR initiating event frequencies from Appendix B, Table B-2, was 0.50/RY and 0.52/RY. Used 0.50 overall to simplify.

Note 2: These values are approximately the same as the RPS unreliabilities for the Westinghouse Analog 7300 and Eagle 21 systems, which were shown to be similar in Reference 2.

Note 3: This reflects the average plant RPS unreliabilities for the four representative CE RPS reliability groups in Reference 3.

Note 4: This reflects the average plant RPS unreliabilities for the two representative B&W RPS reliability groups in Reference 4.

1. U.S. Nuclear Regulatory Commission, “Reliability Study: General Electric Reactor Protection System 1984–1995,” NUREG/CR-5500, Volume 2, December 1998.

2. U.S. Nuclear Regulatory Commission, “Westinghouse Reactor Protection System 1984–1995,” NUREG/CR-5500, Volume 3, December 1998.

3. U.S. Nuclear Regulatory Commission, “Reliability Study: Combustion Engineering Reactor Protection System 1984–1998,” initial draft of NUREG/CR-5500, Volume 10, March 2000.

4. U.S. Nuclear Regulatory Commission, “ Reliability Study: Babcock & Wilcox Reactor Protection System 1984–1998,” the initial draft of NUREG/CR-5500, Volume 4, March 2000.

5. Westinghouse Owners Group, “Westinghouse program ATWS Rule Administration, Combined Core Performance/Scram Systems Reliability, ACRS Subcommittee Meeting,” February 19, 1988.

6. Babcock & Wilcox Owners Group ATWS Committee, “Effects of Plant and Fuel Changes on ATWS Basis Presentation to the NRC,” February 18, 1988.

7. Combustion Engineering Owners Group, “Combustion Engineering Owners Group Meeting With the NRC Concerning Trends in MTCs,” January 11, 198.

8. U.S. Nuclear Regulatory Commission, “Reliability Study: Auxiliary Feedwater System NUREG/CR-5500, Volume 1,

APPENDIX B

PLANT-SPECIFIC AND GENERAL ATWS INFORMATION BY REACTOR GROUP

Plant Specific ATWS and General Information by Reactor Type

Table B-1 Operating Pressurized-Water Reactor Data

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
Arkansas Nuclear One Unit 1	4.67E-05	9.93E-07	2.13	DSS AMSAC	0,0,0	+9E-05 (0 above 95% RTP)
Arkansas Nuclear One Unit 2	3.40E-05	1.02E-06	3.00	DSS DEFAS	0,0,0	+5E-05 (0 above 70% RTP)
Beaver Valley Unit 1	2.14E-04	4.30E-05	20.1	AMSAC	2,1,0	0
Beaver Valley Unit 2	1.92E-04	8.06E-06	4.20	AMSAC	2,0,0	0
Braidwood Unit 1	2.74E-05	3.70E-07	1.35	AMSAC	0,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Braidwood Unit 2	2.74E-05	3.70E-07	1.35	AMSAC	0,1,2	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Byron Units 1&2	3.09E-05	4.20E-07	1.36	AMSAC	Unit 1: 0,0,1 Unit 2: 1,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Callaway	5.85E-05	4.80E-07	0.821	AMSAC	0,0,1	

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
Calvert Cliffs Unit 1	2.40E-04	2.40E-05	10.0	DTT DSS DAFAS	1,0,1	+7E-05 below 70% RTP (linear to +3E-05 from 70 to 100%)
Calvert Cliffs Unit 2	2.40E-04	2.40E-05	10.0	DTT DSS DAFAS	0,0,0	+7E-05 below 70% RTP (linear to +3E-05 from 70 to 100%)
Catawba Unit 1	5.80E-05	1.00E-06	1.72	AMSAC	0,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Catawba Unit 2	5.80E-05	1.00E-06	1.72	AMSAC	1,0,1	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Comanche Peak Unit 1	5.72E-05	5.00E-06	8.74	AMSAC	1,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100%)
Comanche Peak Unit 2	5.72E-05	5.00E-06	8.74	AMSAC	0,1,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100%)
Crystal River Unit 3	1.53E-05	1.00E-10	0.000655	DSS AMSAC	0,1,0	+9E-05 (0 above 95% RTP)
Davis-Besse	6.60E-05	3.54E-07	0.536	DSS AMSAC	1,2,0	+9E-05 (0 above 95% RTP)

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
DC Cook Unit 1	6.26E-05	2.85E-06	4.55	AMSAC	0,0,0	5E-05 below 70% RTP (linear to 0 from 70 to 100%)
DC Cook Unit 2	6.26E-05	2.85E-06	4.55	AMSAC	0,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100%)
Diablo Canyon Unit 1	8.80E-05	7.00E-07	0.795	AMSAC	0,0,1	+5E-05 below 70% RTP (linear to 0 from 70 to 100%)
Diablo Canyon Unit 2	8.80E-05	7.00E-07	0.795	AMSAC	2,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100%)
Farley Unit 1	1.30E-04	7.30E-08	0.0562	AMSAC	0,1,1	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Farley Unit 2	1.30E-04	7.30E-08	0.0562	AMSAC	0,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Fort Calhoun	1.36E-05	2.86E-07	2.10	DTT DSS DAFAS	0,0,0	+5E-05 below 80% RTP (+2E-05 above 80%)
Ginna	8.74E-05	1.60E-07	0.183	AMSAC	0,0,2	+5E-05 below 70% RTP (0 above 70%)

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
Harris	7.00E-05	5.00E-06	7.14	AMSAC	3,0,2	+5E-05 below 70% RTP (linear to from 70 to 100%)
Indian Point Unit 2	3.13E-05	1.81E-06	5.78	AMSAC	3,0,1	0
Indian Point Unit 3	4.40E-05	8.70E-06	19.80	AMSAC	2,1,2	0
Kewaunee	6.65E-05	6.85E-08	0.103	AMSAC w/o C-20 permissive	0,1,0	0
McGuire Unit 1	4.00E-05	1.50E-06	3.75	AMSAC	1,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
McGuire Unit 2	4.00E-05	1.50E-06	3.75	AMSAC	2,1,1	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Millstone Unit 2	3.42E-05	1.83E-08	0.0535	DTT DSS DAFAS	0,0,0	+7E-05 (4 above 70% RTP)
Millstone Unit 3	5.61E-05	3.40E-06	6.06	AMSAC	0,1,0	

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
North Anna Units 1&2	7.16E-05	4.20E-07	0.60	AMSAC	Unit 1: 0,0,0 Unit 2: 0,1,0	+6E-05 (0 above 70% RTP)
Oconee Unit 1	2.30E-05	1.00E-07	0.435	DSS AMSAC	0,0,2	+9E-05 (0 above 95% RTP)
Oconee Unit 2	2.30E-05	1.00E-07	0.435	DSS AMSAC	1,2,4	+9E-05 (0 above 95% RTP)
Oconee Unit 3	2.30E-05	1.00E-07	0.435	DSS AMSAC	1,1,0	+9E-05 (0 above 95% RTP)
Palisades	5.07E-05	4.30E-06	8.48	DTT DSS DAFAS	0,0,0	+5E-05
Palo Verde Units 1 & 2	9.00E-05	3.08E-06	3.42	DTT DSS DAFAS	Unit 1: 1,1,1 Unit 2: 0,0,1	Linear from +5E-05 at 0 to 0 at 100% RTP
Palo Verde Unit 3	9.00E-05	3.08E-06	3.42	DTT DSS DAFAS	1,0,0	Linear from +5E-05 at 0 to 0 at 100% RTP
Point Beach Unit 1	1.15E-04	2.72E-07	0.237	AMSAC w/o C-20 permissive	0,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
Point Beach Unit 2	1.15E-04	2.72E-07	0.237	AMSAC w/o C-20 permissive	0,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)
Prairie Island Unit 1	5.05E-05	3.20E-07	0.634	AMSAC	1,2,1	+7E-05 ITC less than +5E-05 (ITC 0 above 70% RTP)
Prairie Island Unit 2	5.05E-05	3.20E-07	0.634	AMSAC	0,1,0	+7E-05 ITC less than +5E-05 (ITC 0 above 70% RTP)
Robinson Unit 2	3.20E-04	5.70E-06	1.78	AMSAC	1,2,0	+5E-05 below 50% RTP 0 above 50% RTP
Salem Unit 1	5.20E-05	1.40E-06	2.69	AMSAC	0,0,2	0
Salem Unit 2	5.50E-05	1.30E-06	2.436	AMSAC	0,0,0	0
San Onofre Units 2 & 3	3.00E-05	2.70E-06	9.00	DTT DSS DEFAS	Unit 1: 0,0,0 Unit 2: 0,0,0	+5E-05 (above 70% RTP)
St. Lucie Unit 1	2.30E-05	4.13E-07	1.80	DTT DSS DAFAS	1,0,1	+7E-05 below 70% RTP (+2E-05 above 70% RTP)

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
St. Lucie Unit 2	2.62E-05	1.76E-06	6.72	DTT DSS DAFAS	0,0,0	+7E-05 below 70% RTP (+2E-05 above 70% RTP)
Seabrook	6.60E-05	6.63E-06	10.1	AMSAC	1,1,0	0
Sequoyah Unit 1	1.70E-04	7.10E-06	4.18	AMSAC	0,2,0	0
Sequoyah Unit 2	1.70E-04	7.10E-06	4.18	AMSAC	0,2,0	0
South Texas Unit 1	4.30E-5	3.00E-07	0.698	AMSAC	1,0,3	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
South Texas Unit 2	4.30E-5	3.00E-07	0.698	AMSAC	1,1,1	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
V.C. Summer	2.00E-04	2.03E-06	1.02	AMSAC	1,0,1	+7E-05 below 70% RTP (linear to 0 from 70 to 100%)
Surry Unit 1	1.25E-04	3.20E-07	0.256	AMSAC	0,2,0	+3E-05 below 50% RTP (linear to 0 from 50 to 100%)
Surry Unit 2	1.25E-04	3.20E-07	0.256	AMSAC	0,0,1	+3E-05 below 50% RTP (linear to 0 from 50 t o 100%)

Plant Specific ATWS and General Information By Reactor Group

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	Miscellaneous Data	
					1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)	Summary of NRC Survey of 1994–1996 PWR MTC Technical Specification Limits
Three Mile Island Unit 1	4.49E-05	1.00E-10	0.00022	DSS AMSAC	1,0,0	+9E-05 below 95% RTP
Turkey Point Unit 3	3.73E-04	4.40E-06	1.18	AMSAC	1,1,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)
Turkey Point Unit 4	3.73E-04	4.40E-06	1.18	AMSAC	1,0,0	+5E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)
Vogtle Unit 1	4.90E-05	1.13E-07	0.23	AMSAC	0,0,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)
Vogtle Unit 2	4.90E-05	1.13E-07	0.23	AMSAC	0,2,0	+7E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)
Waterford Unit 3	1.80E-05	1.30E-07	0.722	DTT DSS DAFAS	0,0,1	Not surveyed
Watts Bar Unit 1	8.00E-05	3.80E-06	4.75	AMSAC	3,1,0	0
Wolf Creek	4.20E-05	3.10E-08	0.0738	AMSAC	1,0,1	+6E-05 below 70% RTP (linear to 0 from 70 to 100% RTP)

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
Browns Ferry Unit 1	4.80E-05	1.30E-06	2.71	ARI RPT Increased SLCS boron concentration	Not available
Browns Ferry Unit 2	4.80E-05	1.30E-06	2.71	ARI RPT Increased SLCS boron concentration	2.1.2
Browns Ferry Unit 3	4.80E-05	1.30E-06	2.71	ARI RPT Increased SLCS boron concentration	0,0,0
Brunswick Unit 1	2.70E-05	7.00E-07	2.59	ARI RPT Increased SLCS addition rate (2-pump operation)	0,0,0
Brunswick Unit 2	2.70E-05	7.00E-07	2.59	ARI RPT Increased SLCS addition rate (2-pump operation)	0,0,2
Clinton	2.66E-05	1.40E-07	0.526	ARI RPT Increased SLCS boron concentration	0,0,0

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
Cooper	7.97E-05	3.90E-06	4.89	ARI RPT Increased SLCS addition rate (2-pump operation) and suction piping mods	0,0,0
Dresden Unit 2	1.85E-05	5.34E-07	2.89	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	1,3,0
Dresden Unit 3	1.85E-05	5.34E-07	2.89	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,1,1
Duane Arnold	7.84E-06	1.90E-06	24.2	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,0,0
Fermi	5.70E-06	1.80E-06	31.6	ARI RPT Increased SLCS boron concentration	0,1,0

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
FitzPatrick	1.92E-06	1.20E-08	0.625	ARI RPT Increased SLCS boron concentration	0,1,2
Grand Gulf	1.72E-05	5.56E-08	0.323	ARI RPT Increased SLCS boron concentration	0,0,0
Hatch Unit 1	2.23E-05	3.84E-07	1.72	ARI RPT Increased SLCS boron concentration	0,0,1
Hatch Unit 2	2.36E-05	4.78E-07	2.03	ARI RPT Increased SLCS boron concentration	1,0,2
Hope Creek	4.63E-05	7.45E-07	1.61	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,1,0

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
LaSalle Unit 1	4.74E-05	1.87E-07	0.395	ARI RPT Increased SLCS boron concentration	0,0,1
LaSalle Unit 2	4.74E-05	1.87E-07	0.395	ARI RPT Increased SLCS boron concentration	0,0,1
Limerick Unit 1	4.30E-06	9.30E-07	21.6	ARI RPT Auto start SLC	0,0,2
Limerick Unit 2	4.30E-06	9.30E-07	21.6	ARI RPT Auto start SLC	0,0,1
Monticello	2.60E-05	2.50E-06	9.62	ARI RPT Increased SLCS boron concentration	0,0,1
Nine Mile Point Unit 1	5.50E-06	5.40E-07	9.82	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,0,2

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
Nine Mile Point Unit 2	3.10E-05	1.10E-06	3.55	ARI RPT Increased SLCS boron concentration Auto start SLC	0,0,2
Oyster Creek	3.90E-06	2.40E-07	6.15	ARI RPT Increased SLCS boron concentration	0,0,0
Peach Bottom Unit 2	5.53E-06	1.44E-06	26.0	ARI RPT Increased SLCS boron concentration	1,0,1
Peach Bottom Unit 3	5.53E-06	1.44E-06	26.0	ARI RPT Increased SLCS boron concentration	0,0,0
Perry	1.30E-05	4.74E-06	36.5	ARI RPT	3,1,0
Pilgrim	5.80E-05	4.10E-6	7.07	ARI RPT Increased SLCS boron concentration	1,0,1

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
Quad Cities Unit 1	1.20E-06	7.61E-08	6.34	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,2,1
Quad Cities Unit 2	1.20E-06	7.61E-08	6.34	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,1,0
River Bend	1.55E-05	1.00E-10	0.000645	ARI RPT Increased SLCS boron concentration	1,0,1
Susquehanna Unit 1	1.70E-05	3.20E-07	1.88	ARI RPT Increased SLCS addition rate (2-pump operation) and suction piping mods	0,1,1
Susquehanna Unit 2	1.70E-05	3.20E-07	1.88	ARI RPT Increased SLCS addition rate (2-pump operation) and suction piping mods	0,2,1

Plant Specific ATWS and General Information by Reactor Type

Table B-2 Operating Boiling-Water Reactor Data (Cont.)

Plant	Plant CDF	ATWS CDF	Percent ATWS CDF of Plant CDF	Modification Summary	1997, 1998, 1999 ATWS Initiating Event Frequency (per RY)
Vermont Yankee	4.30E-06	8.85E-07	20.6	ARI RPT Increased SLCS boron concentration	2,1,0
Washington Nuclear Plant Unit 2	1.75E-05	6.25E-07	3.57	ARI RPT Increased SLCS addition rate (2-pump operation) and boron concentration	0,1,0

Table B-3 Reactor Protection System Unreliability Uncertainties

		General Electric	Westinghouse	Babcock & Wilcox	Combustion Engineering
Before (data through 1983)	Mean	1.9E-04	2.0E-04	0	0
	Lower 5%–Upper 95%	0.0098-9.0E-04	0.1-9.5E-04	0-2.6E-03	0-2.5E-03
Expected (improvement based on ATWS regulatory actions)	Mean	1.2E-05	3.0E-05	1.2E-05	1.2E-05
Outcome (data since 1984)	Mean	5.8E-06	2.1E-05	4.0E-06	1E-05
	Lower 5%–Upper 95%	1.8-14E-06	0.58-5.7E-05	2.2-7.8E-06	0.35-2.5E-05

Table B-4 General Electric Reactor Protection System Common-cause Failure Insights (NUREG-5500, Volume 3)

RPS Components of Greatest Importance to RPS Reliability and Risk			Greatest CCF Failure Probability
Importance Measures	Important to RPS Reliability FV Importance	Important to CDF Risk Increase Ratio	
Channel Bistable	6.5E-02	1.7E+5	3.8E-07
Solenoid-operated Valve Scram Pilot Value	2.9E-01	1.7E+5	1.7E-06
Rod	4.8E-02	1.7E+5	2.5E-07
HCU AOV scram inlet & outlet valves	1.2E-03	1.7E+5	6.9E-09
HCU Accumulator	1.9E-02	1.7E+5	1.1E-07
Train Relays	4.7E-02	1.7E+5	2.8E-07
Channel Bistable	5.3E-01	1.7E+5	3.1E-06

Table B-5 Westinghouse Reactor Protection System Common-cause Failure Insights (NUREG-5500, Volume 2)

RPS Components of Greatest Importance to RPS Reliability and Risk			Greatest CCF Probability
Importance Measures	Important to RPS Reliability FV Importance	Important to CDF Risk Increase Ratio	
RCCA/CRDM	5.6E-02	4.6E+4	1.2E-06
RTB	7.4E-02	4.6E+4	1.6E-06
SSPS Universal Card	9.7E-02	4.6E+4	2.1E-06
Undervoltage Trip Driver Card	4.8E-01	4.6E+4	1E-05
Channel Bistables	2.2-12E-02	.27-4.3E+4	1.2-4.2E-05
Channel Processing Module (CCP,CDT,CMM)	1.7-8.0E-02	.27-4.3E+4	CCP: 1.5E-05 CDT: 2.5E-04
Train Bistables	3.5E-03	4.3E+4	8.2E-06

Table B-6 Babcock and Wilcox Reactor Protection System Common-cause Failure Insights (NUREG-5500, Volume 4)

RPS components of greatest importance to RPS reliability and Risk			CCF Failure Probability
Importance Measures	Important to RPS reliability FV Importance	Important to CDF Risk Increase Ratio	
Rod	2.3E-02	2.3E+5	1E-07
Reactor Trip Breaker (Mechanically)	5.2E-01	2.3E+5	2.3E-06
Trip Logic-Trip Relay	9.0E-02	2.3E+5	4.0E-07
Channel Bistables	2.7E-01	2.3E+5	1.2E-06
Trip Logic-Logic Relay	6.0E-02	2.3E+5	2.7E-06

Table B-7 Combustion Reactor Protection System Common-cause Failure Insights (NUREG-5500, Volume 10)

RPS Components of Greatest Importance to RPS Reliability and Risk			CCF Failure Probability
Importance Measures	Important to RPS Reliability FV Importance	Important to CDF Risk Increase Ratio	
Reactor Trip Breaker (Mechanically)	3.1E-01	9.0E+4	3.4E-06
Rod/Assembly	5.0E-03	9.0E+4	5.5E-08
Trip Logic-Trip Relay	5.8E-01	9.0E+4	6.4E-06
Channel Bistables	8.0E-02	8.9E+4	9.0E-07
Trip Logic-Logic Relay	1.9E-02	8.9E+4	2.1E-07

Table B-8 Value Impact Data for Each Reactor Group

Modifications for the ATWS Rule	P(ATWS) (frequency of an ATWS event/year)	Value $ Million	Impact $ Million	V-I ratio
GE				
0 Baseline	5.3E-05			
1. Increase SLCS to 86 GPM and ARI	1.2E-05	12.3	**3.5**	3.5
2. Increase SLCs capacity and automatic initiation (new plants)	2.6E-06	2.8	5.0	0.56
W				
0 Baseline	3.7E-05			
1 a. Diverse auxiliary feedwater automatic actuation and turbine trip (AMSAC)	5.8E-06	9.4	2.8	3.3
CE/B&W				
0 Baseline	**8E-05**			
1. **DSS and diverse turbine trip and auxiliary feedwater initiation**	**2.2E-05**	17.4	**5.5**	3.2

APPENDIX C

RESOLUTION OF COMMENTS

Appendix C
Resolution of Comments

On October 18, 2000, a letter entitled "Draft Report, Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," was sent to David Modeen, Director of Engineering, Nuclear Energy Institute, from Farouk Eltawila, Acting Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, for review prior to finalizing the report. The letter requested comments regarding (1) the reasonableness of the approach, (2) the appropriateness of the conclusions, (3) and other regulations, accompanying regulatory guides, and inspection documents that should be assessed to make U.S. Nuclear Regulatory Commission activities more effective, efficient and realistic. Letters were also sent to other members of the industry, the NRC staff, and made publically available.

In response, letters with comments were received from the Union of Concerned Scientists, General Electric, Combustion Engineering, and the Westinghouse House Owners Group. These letters are entered into ADAMS, Package Accession Number ML010220103, and are publicly available. Comments were received from the NRC staff of the Reactor Systems Branch (SRXB) and the Probabilistic Safety Assessment Branch (SPSB), Division of Systems Safety and Analysis (DSSA), Office of Nuclear Reactor Regulation (NRR), and Region II. The NRC staff also met with the Advisory Committee on Reactor Safety (ACRS) on February 2, 2001. The ACRS wrote a letter to the Executive Director of Operations (EDO), March 8, 2001 (ML010730375), with their comments and a response from the EDO to the ACRS was signed April 9, 2001 (ML010950412).

The resolution to each of the comments is addressed below. The comments are presented by organization, in the order of the date received, followed by a restatement of their comments verbatim, and the resolution of each comment. The resolution is stated specifically in terms of "no change required," with an explanation, if appropriate, or a revision to the report that is shown in quotations and italics. Conforming changes were made throughout the report but are not listed in the resolution of the comment.

UNION OF CONCERNED SCIENTISTS (UCS) provided the following comments in a letter to the NRC dated October 30, 2000.

UCS COMMENT 1: On page x of the Executive Summary, the fourth bullet states that "all four reactor types achieved the risk goal of P(ATWS)<1.0E-5 per reactor year." However, data in the report suggests that this conclusion may not be accurate. For example:

a) The last bullet on page x continuing onto page xi points out that "some BWR risk analyses may underestimate the risk of ATWS" because "Probabilistic risk assessment/individual plant examinations for BWRs indicate large variations in the assumptions for reliability of human actions in response to an ATWS. Similarities in design, procedures, and training argue against such variability." According to Table I on page 7, the outcome for BWRs after implementation of the ATWS rule modifications was P(ATWS) of 1.0E-6, or one order of magnitude below the stated risk goal of 1.0E-5. **Recommendation:** Perform a perturbation study to determine if the least conservative assumption for human action reliability could cause the BWR P(ATWS) to increase above 1.0E-5. If not, revise the report to include the results of the study to support the conclusion. Otherwise, revise the conclusion accordingly.

RESOLUTION: Section 3 emphasizes that this is a generic assessment, not plant specific – "the assessment does not address plant-specific issues as these continue to be addressed elsewhere in the regulatory process."

Changes were made in several sections of the report to emphasize that this is a generic assessment, not plant specific.

This assessment is intended to determine if the ATWS rule achieved its expected results on a generic basis, not plant specific. The generic risk calculation in this assessment used reasonable human performance assumptions, not the low human error probabilities assumed in some of the licensee risk assessments. With those values, the generic P(ATWS) met the ATWS rule goal.

The risk quantification method used to calculate the expected risk in the original ATWS analysis was used for this assessment in order to maintain a consistent approach. In so doing, the improvement in risk that was identified would more obviously be the result of less frequent initiating events and improved equipment performance rather than changes in the risk model or assumptions. Since most of the relevant operator training and procedure improvements were made prior to the ATWS rule, the same assumptions regarding operator performance were used in this assessment as in the original ATWS risk calculations. This approach provides a reasonable framework for understanding the basic elements of the ATWS risk – initiating event frequency, scram system reliability, and mitigative reliability.

b) The next-to-last bullet on page x states that "ATWS mitigation capability on a PWR is highly dependent on the moderator temperature coefficient (MTC). Mitigative functions are considered non-viable if the ATWS peak pressure exceeds 3200 psig; and a sufficiently negative MTC will limit the ATWS peak pressure." On page 4, the report states "During the first part of the fuel cycle below 100 percent power, the MTC can be positive. If an ATWS occurs when the MTC is either positive or insufficiently negative to limit reactor power and the ATWS pressure increase, all subsequent mitigative functions are likely to be ineffective. The percentage of the fuel cycle time when the MTC is insufficient to maintain the RCS pressure below 3200 psig during an ATWS is designated the

'unfavorable exposure time.' At the time of the ATWS rulemaking, the UET was assumed to be 1 percent for W [Westinghouse] and 50 percent for B&W/CE reactors...". Table 1 on page 7 reports the "UET greater than expected for a few plants." Table 6 on page 14 reports that a peak pressure of 3200 psia is exceeded 1 to 10 percent of the time for Westinghouse plants. **Recommendation:** Perform a perturbation study to determine if the plants exceeding the expected unfavorable exposure time causes the P(ATWS) to increase above 1.0E-5. If not, revise the report to include the results of the study to support the conclusion. Otherwise, revise the conclusion accordingly.

RESOLUTION: Based on a comment from Westinghouse, the phrase "unfavorable exposure time" was replaced by "unfavorable MTC" which has a similar meaning. See response to WOG General Comment 1.

Section 3.2.6 revised to read:

... *"If the W assessment were done using the same method as B&W and CE with an "unfavorable MTC" changing from 1 percent to 40 percent of the fuel cycle, the ATWS risk would increase by a factor of about 20 to 4.3E-6.*"

(c) The next-to-last bullet on page x states that "ATWS mitigation capability on a PWR is highly dependent on the moderator temperature coefficient (MTC). Mitigative functions are considered non-viable if the ATWS peak pressure exceeds 3200 psig; and a sufficiently negative MTC will limit the ATWS peak pressure." On page 11, Section 3.2.4 of the report states "Comparison of the ATWS CDF for all the plants in each reactor group in Appendix B to the P(ATWS) expectation for each reactor group found that six licensees do not meet the ATWS risk goals." This statement contradicts, or at least seriously undermines, the conclusion that all reactor types achieved the ATWS goal. **Recommendation:** Revise the overall conclusion to indicate that some reactors did not achieve the ATWS risk goal.

RESOLUTION: Conclusions sections revised to read:

During the ATWS...."on a *generic* basis, all four reactor types achieved the ATWS rule risk goal. *A few individual plants had somewhat higher risk as a result of using lower levels of RPS reliability rather than the industry average or assumed additional equipment failures.*"

d)The next-to-last paragraph on page 11 reports "Several [PWR] plants routinely operate with blocked power-operated relief valves and their IPEs may underestimate ATWS risk." **Recommendation:** Perform a perturbation study to determine if routine operation with blocked power-operated relief valves causes the P(ATWS) to increase above 1.0E-5. If not, revise the report to include the results of the study to support the conclusion. Otherwise, revise the conclusion accordingly.

RESOLUTION: The report conclusion was revised. See resolution to UCS Comment 1(c).

UCS COMMENT 2: The report indicates that the effectiveness of the ATWS rule for PWRs is highly dependent on the MTC issue. The last sentence on page 6 states "Information on the plant and reactor group MTCs was obtained from a 1994 NRC survey (Ref. 15), and owners' group presentations to the NRC (Ref. 16), (Ref. 17), (Ref. 18)." The latter three references all date back to 1988. Thus, information on this important parameter is at least six year old. **Recommendation:**

Determine the current plant and reactor group MTCs. Revise the report as appropriate. As a minimum, the report should be revised to reference contemporary sources for the MTCs.

RESOLUTION: No change required. The readily available information serves to make the point and support the conclusions without the cost of an additional survey.

UCS COMMENT 3: The third paragraph on page 4 states "In SECY-83-293 it was assumed that a peak pressure of above 3200 psig was unacceptable for ASME Service Level C. ... Also steam generator tubes might fail before other primary coolant system components and bypass containment." Table 6 on page 14 reports peak pressures of 3962 psia for CE plants, 3600–4000 psia for B&W plants, and >3200 psia up to 10 percent of the time for Westinghouse plants. Page 1 defines an ATWS as "an anticipated operational occurrence followed by failure of the reactor trip system (RTS) portion of the reactor protection system (RPS)." On October 11, 2000, a subcommittee of the Advisory Committee on Reactor Safeguards conducted a meeting regarding the differing professional opinion (DPO) initiated by Dr. Joram Hopenfeld of the NRC staff. The meeting transcript is available on the Internet at http://www.nrc.gov/ACRS/rrs1/Trans_Let/index_top/ACRS_sub_tran/adhoc001011. In addition to the DPO, the NRC also created Generic Safety Issue 163 (GSI-163) about Dr. Hopenfeld's concerns. Basically, Dr. Hopenfeld is concerned that allowing nuclear power plants to operate with cracked steam generator tubes increases the potential for unacceptable consequences from design bases events. During his presentation to the ACRS subcommittee, Dr. Hopenfeld referred to a May 2000 memo issued by Westinghouse to plant owners informing them that the support plates inside steam generators were designed for a differential pressure of 1,500 psid. If an ATWS can result in a reactor coolant system pressure in excess of 3,200 psia, it seems certain that a differential pressure of at least 2,000 psid will be experienced across the steam generator tube walls. If the operational occurrence that triggered the ATWS is one that causes secondary side pressure to decrease, then the resulting differential pressure across the steam generator tube walls may approach the reactor coolant system pressure. The ATWS rule might not be adequate by accepting peak pressures of the reactor coolant system in excess of 3,200 psia if those pressures cause degraded steam generator tubes to rupture. **Recommendation:** Evaluate the impact of primary side pressures exceeding 3,200 psia on steam generator tube integrity. If tube integrity is assured even under those extreme conditions, revise the report to include the results of the evaluation. Otherwise, revise the report to reflect that the ATWS rule as currently implemented compromises steam generator tube integrity.

RESOLUTION: No change required. It was not within the scope of the ATWS rule making or this report to determine what components will fail as a result of exceeding 3200 psia. The steam general tube issues of concern are being addressed as Generic Safety Issue 188, "Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass."

On February 1, 2001, Dr. Dana Powers, Chairman of the Advisory Committee for Reactor Safeguards (ACRS) Ad Hoc Subcommittee on a Differing Professional Opinion provided the Executive Director for Operations (EDO) a copy of its summary report on matters pertaining to a differing professional opinion on steam generator tube integrity. In its draft report, "Voltage-Based Alternative Repair Criteria," the Ad Hoc Subcommittee discussed its conclusions and recommendations. This draft report was issued in final form as NUREG-1740 dated February 2001. On March 5, 2001, the EDO directed that the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) review the ACRS Ad Hoc Subcommittee report and develop a joint action plan to address the conclusions and recommendations (ADAMS Accession Number ML010670217).

UCS COMMENT 4: The NRC is to be commended for evaluating the effectiveness of the ATWS rule. My first impression was that this evaluation was long overdue, but after having reviewed the draft report, I realize that the delay was necessary in order to compile sufficient operating experience to permit a meaningful evaluation.

RESOLUTION: No change required.

UCS COMMENT 5: It is a great idea to provide the ADAMS accession number (ML003753154) for the draft report both on the report's cover page and in the transmittal letter. This practice is currently not the norm, but it makes document retrieval much easier. **Recommendation:** Encourage the rest of the NRC staff to adopt this extremely useful practice.

RESOLUTION: No change required. The suggestion has been made as requested.

UCS COMMENT 6: The paragraph immediately preceding Table 6 on page 14 reports that the MTC values summarized in the table came from References 17, 18, and 19. It appears that the information really came from References 16, 17, and 18 instead. **Recommendation:** Check the proper references and revise the report if necessary.

RESOLUTION: The report was revised as indicated in the comment.

GENERAL ELECTRIC provided the following comments in a letter to the NRC dated October 30, 2000.

GE COMMENT 1: GE has received the subject report and completed a review of the document. In this report, the NRC states that ATWS risk is comprised of three elements: a) frequency of scrams, b) reliability of the reactor protection system (RPS), and c) reliability of ATWS mitigation systems. These items are discussed separately below:

a. The NRC recognizes that the scram frequency has come down by a factor of ten since the ATWS rule was issued and this by itself has greatly reduced the risk associated with ATWS. GE concurs with this conclusion.

RESOLUTION: No change required.

b. When the ATWS rule was issued, the NRC estimated that RPS reliability was about IE-5 per demand, while GE estimated it was an order of magnitude better, about 1E-6 per demand. At this very low failure rate, it takes many years of data collection to demonstrate the failure rate is accurate. With several ensuing years of additional operation without a major RPS failure in the industry, the NRC has reduced their failure rate estimate, and though not as low as the GE estimate, there is a smaller difference between the GE and NRC estimates.

RESOLUTION: Section 2.0 indicates the wide range of RPS unreliability (3.0E-06 to 1.1E-04) before the ATWS rule. Section 3.2.2 is revised to "Table B-3 shows that the outcomes of upper and lower bounds of unreliability range from 1.8E-06 to 5.7E-05 *and this range is smaller than that noted Section 2.0 before the ATWS rule was issued*." Section 3.2.3 and the report conclusions makes the remaining GE point by discussing that the values of RPS unreliability illustrates the difficulty of estimating reliability values in highly reliable systems.

c. Reliability of mitigation systems is dominated by short-term operator action reliability for ATWS. The report states, "... examinations for BWRs indicate large variations in the assumptions for reliability of human actions in response to an ATWS. Similarities in design, procedures, and training argue against such variability. Consequently, some BWR risk analysis may underestimate the risk of ATWS." Operator action reliability has long been open to uncertainty and disagreement. In the absence of concrete specifications, different assessments are likely to have different assumed values. However, it is clear that improvements made in design, procedures, and training since the ATWS rule was issued have all contributed to improving operator action reliability. GE does not necessarily agree that "BWR risk analysis may underestimate the risk of ATWS."

RESOLUTION: The report conclusion was revised to delete "BWR risk analysis may underestimate the risk of ATWS" and add *"Risk analyses in support of licensing actions would need to be justified on a more consistent basis."* The original generic BWR risk analysis event trees used to develop the ATWS rule expectations were based on the completed implementation of the ATWS rule. For BWRs, that included ARI, RPT, and improved SLC. Most improvements in operating procedures and training had been implemented earlier as a result of the TMI and responses to the Brown's Ferry partial ATWS. To maintain consistency with the original approach, the same operator response assumptions were used in this assessment.

GE COMMENT 2: The Background section of the report states that the Commission designated ATWS as Unresolved Safety Issue (USI) A-9. It might be beneficial to the reader to state how and when the USI was resolved, otherwise, one might assume that USI A-9 was still open.

> **RESOLUTION:** Section 2.0 was revised after the discussion of USI A-9 to add "*USI A-9 was resolved with the publication of the ATWS rule.*"

GE COMMENT 3: Section 2.1.1 of the draft NRC report discusses the basis and application of the 200 °F suppression pool temperature limit.

(a) In NEDO-30832-A, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," May 1995, the NRC accepted elimination of the 200 °F local pool temperature limit for T-type and X-type SRV discharge quenchers. This is based on the fact that as suppression pool temperature approaches saturation, condensation loads decrease. Therefore, the statement, *"As the suppression pool temperature increases, the potential for unstable condensation of the discharge to the pool may overload the containment structure,"* is not factual.

> **RESOLUTION:** Section 2.1.1 revised to read:
>
> "*In some cases, the NRC staff has accepted higher temperature limits based on plant specific design features. As the suppression pool temperature increases to near the boiling point, steam condensation is less effective, resulting in higher containment pressure; and the RHR pumps would eventually fail due to inadequate net positive suction pressure. During an unmitigated ATWS, with continued heat input to the suppression pool, the containment could fail while the core was still intact.*"

(b) The statement, *"During an ATWS at a BWR, the containment would probably fail prior to core damage,"* is true, but should be clarified to note that failure only is postulated after a long series of other failures of ATWS mitigation features such as ARI, RPT, boron injection, manual insertion of control rods, and additional failures in containment mitigation features such as pool cooling and containment pressure relief

> **RESOLUTION:** See resolution to GE Comment 3(a) above.

(c) As discussed above, the 200 °F local pool temperature limit has been removed for plants which meet the requirements of NEDO-30832-A. Therefore, plant specific limits are used as the limiting suppression pool temperature for ATWS. It is suggested that "the 200 °F limit" be replaced with "the plant-specific pool temperature limit for ATWS evaluations."

> **RESOLUTION:** See resolution to GE Comment 3(a) above.

REGION 2 provided comments in a note dated November 30, 2000.

REGION 2 COMMENT: The report was well organized, and discussed the major cost and impact issues associated with the NRC's ATWS rule. Risk information was included, but was primarily based on the work performed in 1983 to support the original paper recommending the rule. Information from the licensee's IPE submittals was also included. Both of these sources of risk information are already dated, and many licensees are using more recent information in their current risk models. Section 3.2 could be expanded to also include a new subsection to show the new risk estimates, event trees and associated fault trees for ATWS that have been developed for the NRCs Standardized Plant Analysis Risk (SPAR) Models for RES. Revision 3 SPAR models are already available for many plants, and the ATWS risk is available in the solution to these models. The event trees reflect the current owners group guidelines for coping with ATWS, and would be more accurate than the trees used in the original 1983 work. The old event trees should stay in their own section in the report, as they were used as the basis for comparing the current risk to the projected risk improvement. The updated event trees and fault trees would provide a new baseline to judge future proposed improvements to the rule.

With the exception of the recommended improvement noted above, Region 2 has no comments. The report is well written, and conveys the necessary information in a concise manner.

> **RESOLUTION:** No change required. Given that the goal of this assessment was to determine if real changes in plant and equipment performance were achieved with the ATWS rule, rather than model improvements and changed assumptions, the simplified event trees provide a consistent and reasonable approach for this assessment.
>
> As a matter of interest, we looked at the SPAR models (which are not yet approved) for ATWS and they approximate those used for the ATWS rule development except for the operator action to scram the reactor. The HEP for operator action to scram the reactor was not significantly different that the information already in the ATWS report from the RPS reliability studies.

NRC, Office of Nuclear Reactor Regulation, Division of Systems Safety Analysis, Reactor Systems Branch (NRR/DSSA/SRXB) provided the following comments in a note NRC dated December 8, 2000.

NRR/DSSA/SRXB COMMENT 1: The staff of SRXB considers the approach taken by the Office of Research (RES) to assess the “effectiveness” of the ATWS Rule to be very reasonable and logical. The “Background” provides the reader with an excellent history of the ATWS issue, which is important for understanding the complex regulatory history.

RESOLUTION: No response required.

NRR/DSSA/SRXB COMMENT 2: The “Assessment” Section is accurate and complete. Sub-section 3.2.6 deals with the subject of fuel management. DSSA believes that a change in fuel management has a significant effect on ATWS. Fuel management is at the heart of the ATWS issue. When a vendor or a licensee designs a core for a particular cycle, fuel enrichment, burnable absorber, and boron concentration are parameters that play a major role in determining the length of the fuel cycle and such Technical Specifications (TS) as those associated with the shutdown margin. The interplay between the cycle length and the shutdown margin is a crucial one. For longer fuel cycles (say 18 months instead of 12 months) with higher enriched fuel, increased boron concentration or fuel with integral burnable absorber are necessary to maintain the shutdown margin to a TS value at beginning of cycle (BOC). However, fabrication of fuel with integral burnable absorber is very costly, and it is far more expensive than increasing the boron concentration. Should a vendor or licensee substitute soluble boron for the burnable absorber, the increased boron concentration will make the moderator coefficient become less negative, and under certain circumstances even positive.

RESOLUTION: No change required.

NRR/DSSA/SRXB COMMENT 3: The “Conclusion” Section is very well written in terms of clarity and decisiveness, and provides the reader with unbiased results. The staff is in agreement with the RES conclusion that the ATWS Rule has been effective in implementing and modifying system hardware at a reasonable cost while reducing the occurrence of an ATWS event.

RESOLUTION: No change required

NRR/DSSA/SRXB COMMENT 4: The staff has no recommendations for additional reviews of regulatory effectiveness.

RESOLUTION: No change required

NRR/DSSA/SRXB COMMENT 5: Finally, DSSA notes that recent submittals related to ATWS depart from the traditional bases supporting the rule making. Specifically, licensees or vendors are proposing to rely more on operator performance during unfavorable exposure time (UET) as the primary means for ATWS mitigation. The RES effectiveness study does not speak to the issues or concerns regarding these approaches as it relates to the technical bases of the present rule. It is our view that the study should be modified to discuss the impacts of the new industry strategies.

RESOLUTION: The impacts of new industry strategies regarding the UET are briefly discussed in 3.2.6; further review of new industry strategies regard ATWS are beyond the scope of the report as explained in DSSA/PSAB Comment 4.

The technical basis regarding MTC is addressed in Section 2.1.2 and 3.2.6. The report was revised regrading the technical bases of the present rule regarding operator performance as follows:

"*2.1.3 Operator Action*

In developing the ATWS rule, the NRC staff view was that (1) operator action should not be relied upon during the first ten minutes of an accident including a manual scram and (2) operator actions should be relied upon later in the course of an accident if the condition in the reactor and mitigating systems is available to the operator, that sufficient time is available to assess the condition and take action, and that the operator has been trained in the action.

In practice, operators are trained to identify an ATWS condition, initiate a manual scram, manually insert the control rods, manually trip the RPS motor-generators, and begin borating. After the Salem events and before the ATWS rule was issued, there was discussion in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant", April 1983, regarding credit for operator action to scram the reactor following an ATWS, since the Salem operator scrammed the reactor 30 seconds after the RTS failed. NUREG-1000 stated that the Salem ATWS events were not as serious as they could have been as they took place at low power, the operators were at the correct control panel to support power ascension, and there was sufficient relief capacity and steam generator mass.

Operator recognition that an ATWS condition exists can be difficult, especially for events involving malfunctions of the reactor trip system which do not activate the trip indicator lights. The fast moving ATWS event does not provide much time for the operator to recognize what has occurred. For example, NUREG-1000 provides peak pressure versus time curves that show that if the operator takes no action following an ATWS, the ASME service level C stress would be exceeded in 47, 70, and 100 seconds for CE, B&W, and W PWRs, respectively. In addition, the pressure ramps up from a normal operating values of 2250 psia to its peaks of approximately 3500, 4000, and 3500 psia in 47, 20, and 10 seconds for CE, B&W, and W PWRs, respectively."

NRR/RES Coordinator provided comments on December 8, 2000, based on the Reactor Systems Branch (SPXB) comments.

COMMENT 1. Sub-section 3.2.6 deals with the subject of fuel management, and states that changes in fuel management "may" affect the ATWS mitigating capability. Fuel management is at the heart of the ATWS issue, and we recommend that "may" be replaced with a stronger word, such as "will" or "can."

RESOLUTION: No change required. Since some of the material relates to proposed industry methods we do not wish to use wording that would pre-disposition staff reviews.

COMMENT 2. In regard to the risk associated with the ATWS event, the study should identify concerns/ issues associated with current Vendors suggestions to substitute man-made efforts (such as plant operational procedures) for natural controls such as negative moderator coefficient.

RESOLUTION: See resolution to DSSA/SPXB Comment 5.

NRC/Office of Nuclear Reactor Regulation/Division of Systems Safety and Analysis/Probabilistic Safety Assessment Branch (NRR/DSSA/PSAB) provided the following comments in a note dated December 11, 2000.

NRR/DSSA/PSAB GENERAL COMMENT: Overall, the draft report "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule" did a good job in demonstrating that the Anticipated Transient Without Scram (ATWS) rule was effective in making hardware modifications and imposing operating limits, reducing ATWS risk, and implementing the ATWS rule at reasonable cost. In particular, the sections on boiling water reactor (BWR) and pressurized water reactor (PWR) ATWS sequence and event trees provided a thorough discussion of how the ATWS rule evaluated the expected core damage frequency (CDF) of an unmitigated ATWS. The value-impact analysis also provided a good discussion of how the ATWS rule resulted in savings to the nuclear industry due to a reduction in the number of automatic scrams.

RESOLUTION: No change required

NRR/DSSA/PSAB COMMENT 1: A few terms are used in the executive summary that are not defined until the main report. It would be helpful to the reader if these terms were defined in the executive summary. In particular, the term "Unfavorable Exposure Time" (UET) is first used on page ix, but not defined until page 4, section 2.1.2 of the main report. Similarly, P(ATWS) is first used on page x, but not defined until page 2 of the main report.

RESOLUTION: The UET concept was deleted on from the report (See response to the resolution to WOG General Comment 1). P(ATWS) was defined in the executive summary consistent with its definition in the report.

NRR/DSSA/PSAB COMMENT 2: Some terms and phrases need to be defined and/or explained when they are first used. Specifically, "ASME Service Level C" on page 4 needs to be explained (i.e., what is ASME Service Level C and how does that impact the ATWS analysis?) and the acronym "SBO" on page 6 needs to be defined.

RESOLUTION: The referenced sentence was revised to read "*In SECY-83-293, the ASME Service Level C pressure of 3200 psig was assumed be to an unacceptable plant condition during ATWS rule development.*"

NRR/DSSA/PSAB COMMENT 3: Were any other reports issued regarding the ATWS rule? It may be useful to contrast the results of this study to the results of other studies.

RESOLUTION: There were no other ATWS related reports with the same scope. Section 3.1 discusses the use of several ATWS related reports (references 9 through 19) that were used for this assessment.

NRR/DSSA/PSAB COMMENT 4: The second numbered item on page 13 of Section 3.2.5 is a very important finding that needs to be highlighted more. New phenomena, resulting from using higher fuel burn-up, that may affect the operability and reliability of the reactor protection system (RPS) (i.e., control rod insertion) needs to be investigated further and considered by the staff in reviewing related licensee submittals and/or in discussions with the owners groups

(e.g.,WOG). This observation needs to be presented in the executive summary and the conclusions, Section 4, to ensure it is properly raised as potentially eroding the effectiveness of the ATWS rule.

RESOLUTION: The conclusions of the report were revised to add:

"*The higher fuel burnup has resulted in previously unpredicted oxide growth and fuel assembly distortion. In some cases this has resulted in slow or incomplete control rod insertion. These failures and degradations are new phenomena and were not considered during the development of the ATWS rule*."

NRR/DSSA/PSAB COMMENT 5: It should also be pointed out that the individual issues identified in the executive summary and conclusions, Section 4, that could erode the effectiveness of the ATWS rule are not independent of each other, but rather, have synergistic effects. For example, the use of higher burn-up fuel to achieve longer cycles, may be creating new phenomena that would impact the reliability of the RPS, which typically drives the low values and contributions of ATWS sequences due to its postulated extremely high reliability though it has a large uncertainty associated with it (this new phenomena would tend to increase the uncertainty or at least skew it towards higher unreliability values), and would also increase the unfavorable exposure time (UET) and possibly increase the peak pressure reached during an ATWS event. The report needs to conclude that the identified issues, especially those associated with the use of higher burnup fuel, needs to be carefully watched, considered, and investigated by the staff.

RESOLUTION: See response to NRR/DSSA/PSAB 4 and WOG General Comment 4.

NRR/DSSA/PSAB COMMENT 6: On page 15, the first paragraph states that the WOG met with the NRC staff on August 23, 2000, "to obtain NRC concurrence that the WOG approach to ATWS for licensing issues, such as MTC, is acceptable ..." At this meeting, it was made clear to the WOG that the NRC staff at the meeting would not be "concurring" or approving their approach, but rather, the meeting allowed the WOG to present how they were addressing issues identified during a 1998 meeting with the staff and allowed the WOG and staff to discuss (and refine) these issues further. Discussions with the WOG regarding their proposed approach and its implications will continue. The RES staff involved in the ATWS rule's effectiveness study should remain engaged and provide input in these discussions and with the other staff addressing these issues (e.g., NRR SRXB and SPSB staff).

RESOLUTION: The report was revised to delete "On August 23, 2000....by the staff."

Westinghouse Owners Group (WOG) provided the following general and detailed comments in a letter to the NRC dated December 20, 2000.

CLARIFICATION: WOG provided comments that address traditional ATWS rule issues that are the subject of the report. The WOG also provided comments presenting material that potentially supports a new licensing basis for ATWS and this material is not within the scope of the report. Some of the new material is supported by WCAP-11992 that was submitted to the staff in May 1995, and found not to be acceptable for use in licensing or other regulator matters, and that the WOG and the NRC continue to work to address the concerns as mentioned in Section 3.2.6 of the report. WOG comments not within the scope of this report will state "see the Clarification."

WOG GENERAL COMMENT 1: From a Westinghouse/WOG perspective, the term "Unfavorable Exposure Time" or UET represents the duration of a given fuel cycle, for a specific plant configuration, in which the total core reactivity feedback is insufficient to preclude exceeding a peak RCS pressure of 3200 psig following an ATWS event. UET was defined by the Westinghouse Owners Group (WOG) for use in a more detailed ATWS PRA model developed as part of the "Westinghouse / WOG ATWS Rule Administration Process," WCAP-11992. The concept of UET is also being applied in a revised Risk-Informed ATWS PRA model supporting an ongoing WOG ATWS PRA program.

To determine the UET values used in this ATWS PRA model, the reactivity feedback required to just yield a peak RCS pressure of 3200 psig is first determined by specific ATWS transient analyses. The reactivity feedback conditions for a given reload core model are then compared to the transient reactivity feedback models to determine the value of UET for a given plant configuration. For the Westinghouse / WOG ATWS PRA model, a total of 12 UET values are typically determined and used.

The term UET as defined and applied above was not a term directly used in the basis of the Final ATWS Rule as documented in SECY-83-293. In SECY-83-293, the terms "favorable MTC" and unfavorable MTC" are applied in the discussion of the simplified ATWS PRA model. These terms are not the same as UET defined above. Hence, while Westinghouse and the WOG concur with and support the industry's use and correct application of this term in ATWS related discussions and documentation, it is requested that the use of the term UET in this Draft Report be replaced with "unfavorable MTC" to be consistent with the reference to SECY-83-293.

> **RESOLUTION:** The report was revised to delete use of the term UET and use unfavorable or favorable MTC, except in Section 3.2.6 which acknowledges WCAP-11992 and the UET concept. See the Clarification above.

WOG GENERAL COMMENT 2: For Westinghouse PWRs, the discussions of ATWS events as they relate to RCS pressure, unfavorable MTC, and ATWS mitigation should be clearly defined and limited to only anticipated transients that result in a direct or consequential loss of main feedwater. The only ATWS events that potentially lead to high RCS pressures that challenge the ASME Service Level C Stress Limit (i.e., 3200 psig) are those events that result in a direct (i.e., Loss or Normal Feedwater ATWS) or consequential (i.e., Loss of Load ATWS) loss of main feedwater. The AMSAC system required by the Final ATWS Rule and installed at Westinghouse PWRs is designed to actuate a turbine trip and initiate the AFWS based only on detection of a condition representative of a loss of main feedwater. Other non-loss of feedwater

related anticipated transients considered in the generic ATWS analyses supporting the basis of the Final ATWS Rule do not lead to RCS overpressure concerns and, hence, have no ATWS related mitigation Systems.

RESOLUTION: Not just Westinghouse PWRs, but all PWRs are subject to high pressures during an ATWS only if they result in direct or consequential loss of main feedwater, since steam generator relief capacity is generally capable of relieving 100 percent steam flow. However, the following considerations apply:

First of all, given there is no reactor trip, no turbine trip, and no loss of feedwater, but there is some condition requiring a reactor scram (hence, an ATWS), then the operator may be unlikely to recognize that an ATWS condition exists quickly enough to take action. There may be no pressure spike, but core damage from departure from nucleate boiling or other cause.

Secondly, for the condition without loss of feedwater, reactor power would remain high since there would be no negative reactivity feedback to reduce reactor power. Then, if the turbine were tripped, condensate would be depleted through the steam generator safeties and the main feedwater system would eventually trip on loss of suction pressure. Thus, even with no immediate loss of feedwater, condensate is eventually depleted out the steam generator safeties; and then the steam generators would dry-out, only on a longer time scale than two minutes.

Thirdly, the turbine will almost always trip on loss of main feedwater; and conversely, the main feedwater system will usually not trip on turbine trip. This logic was part of the "MTC Overpressure" factor where turbine trip is the lower risk scenario.

The point is that given an ATWS with main feedwater operating doesn't assure that the outcome will be benign. Complex operator actions are still required to achieve safe shutdown.

Section 3.1 was revised to add:

The ATWS rule original risk expectations were calculated assuming that the ATWS rule requirements were implemented. Thus, the original analysis assumed that AMSAC, DSS, ARI, RPT trip, etc., were installed and operable.

.....

"*The original generic risk analysis event trees did not distinguish those initiating events accompanied by loss of main feedwater. And again, for consistency, neither did this assessment, since the goal of this assessment was to determine if the ATWS rule resulted in improved performance, rather than "improved modeling." However, for <u>W</u> plants, the likelihood of loss of feedwater events was a consideration in adoption of the values of the "MTC Overpressure" factor and separate event trees for turbine trip and non-turbine trip scenarios.*"

WOG GENERAL COMMENT 3: Drawing conclusions based on the continued use of the SECY-83-293 PRA approach is inappropriate and may result in misleading conclusions and recommendations. The SECY-83-293 model is too simplistic to assess the risk from ATWS events and does not account for a number of important considerations. One such

consideration is the probability that the plant is operating in a configuration that corresponds to the UET (i.e., unfavorable MTC) continually cited in this report. The conditions for this referenced UET are no rod insertion, all auxiliary feedwater available, and no PORVs blocked, but this could be a low probability configuration.

RESOLUTION: Section 3.1 was revised to read:

"*For PWRs, the original generic risk analysis and this assessment assume that the following equipment is available: emergency feedwater, primary power-operated relief valves and safeties, and secondary power-operated relief valves and safeties – this tends toward lower peak pressures and lower ATWS risk. On the other hand, the original generic risk analysis assumes that the rod control system is in manual control – this tends toward higher peak pressures and higher ATWS risk. Since the affects are offsetting, the total impact may be minimal. Again, to determine whether the ATWS rule was effective in reducing risk from ATWS, we looked to see if it met or exceeded the goal based on the original method of calculation.*"

.....

"*The original generic event trees were developed after much deliberation to focus on what were the most important sequences. The deliberations which went into determination of the type of event trees to use considered aspects of the ATWS transient such as the status of the main feedwater system. The current assessment was not meant to be a total reevaluation of the approach to ATWS risk analysis, but an attempt to determine if the ATWS rule goals were met. In order to make a valid comparison of the improvements due to less frequent initiating events and improved equipment performance rather than changes in the risk model or assumptions, the simplified event tree approach was used. Additionally, we believe that this approach provides a reasonable estimate of ATWS risk.*"

WOG GENERAL COMMENT 4: The draft report frequently specifies and refers to a 1% UET (i.e., unfavorable MTC) as being applicable to Westinghouse PWRs. This is incorrect. A 1% unfavorable MTC was modeled in one of the SECY-83-293 simplified PRA models (Figure 11 in SECY-83-293) that represented what is called turbine trip transients in the PRA evaluation. As discussed in SECY-83-293, Section 5.5, Item 2 for Westinghouse plants, a 1% unfavorable MTC was assumed to exist for this turbine trip event since the pressure transient for this type ATWS is relatively mild. As earlier discussed in the same section of SECY-83-293, for non-turbine trip events that challenge the ASME Service Level C limit, a 10% unfavorable MTC was assumed (Figure 12 in SECY-83-293).

Neither of these unfavorable MTC values is considered to be applicable for Westinghouse PWRs. The Westinghouse generic ATWS analyses performed in response to NUREG-0460 and which form the deterministic analysis basis for the Final ATWS Rule are documented in Westinghouse letter NS-TMA-2182, "ATWS Submittal," dated December 30, 1979. In these analyses, a full power MTC of -8 pcm/°F was used in the analyses and a sensitivity to a change in MTC to -7 pcm/°F was included. In 1979, these MTCs represented the values of MTC that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively, as specified by NUREG-0460. The peak RCS pressures using these MTC values (without changes in any other assumptions) are shown in NS-TMA-2182 to be less than 3200 psig.

RESOLUTION: With regard to the first paragraph of the comment, see resolution to WOG General Comment 1 above which clarifies the use of the 1, 5, and 10 percent unfavorable MTCs.

Also, peak pressures are in excess of 3200 psia for the MTCs noted in the comment. Section 3.2.6 of the report was revised as follows:

Section 2.1.2 revised to read:

"*At the time of the ATWS rulemaking, the estimated impact of periods of "unfavorable MTC" was included in the event tree as the "MTC Overpressure" factor. For Westinghouse plants, the original ATWS expected risk approach estimated an "MTC Overpressure" factor of 0.01 for turbine trip scenarios and 0.1 for non-turbine trip scenarios; and the expected outcome of the ATWS rule with installation of AMSAC assumed that Westinghouse plants would be in the turbine trip scenario 70 percent of the time (i.e., 70 percent of the ATWS transients would be the lower risk scenario). As stated above, the turbine trip provides the operator additional time to respond.*

For CE/B&W plants, a higher "MTC Overpressure" factor of 0.5 was used regardless of turbine trip status. The reason was that the MTC was expected to be insufficiently negative to limit peak pressure below 3200 psig for up to 50 percent of the cycle. For CE/B&W plants, the expected outcome of the ATWS rule included a factor of 10 improvement in the electrical portion of the scram system based on installation of the DSS. These same assumptions were used to determine the current outcomes.

Like the BWR risk analysis, this risk model does not include low frequency "anticipated operational occurrences" such as loss of primary system flow for which core damage occurs, regardless of the MTC or other factors, unless the rods insert immediately."

Section 3.2.6 was revised to read:

"Since the ATWS rule was developed, fuel cycle lengths have been extended. This requires higher enrichment fuel and often results in less negative (more positive) MTCs. As previously discussed in Section 2.1.2....

Pressures for economic efficiency are also prompting licensees to increase the power of the reactor. The higher power rating leads to faster dry-out of the steam generator for PWRs and faster heatup of the suppression pool for BWRs. The combination of less negative MTC and higher reactor power has a greater effect on the plant response to ATWS than either one alone. Less negative MTC and higher power result in the ATWS pressure spike occurring earlier, allowing less time for operator actions, quicker, less time for MTC to limit the pressure increase, and higher, increasing the likelihood of equipment damage. These fuel management changes potentially impact both the deterministic and probabilistic analyses for ATWS and other reactor transients.

In 1987.... no significant changes were required in ATWS analyses. *The peak pressure of 3497 psia update in Table 6 for W was calculated based on values obtained from sensitivity analysis in NUREG-460. The sensitivity analysis shows a pressure increase of approximately 100 psi for a change in MTC of 1 pcm/ °F. Thus, the change from the 1979 value of -8 pcm/ °F to -5pcm/ °F in Table 6 corresponds to a 300 psi increase.*

Table 6 ATWS MTC and Peak Pressure for PWRs

Parameters		1979 ATWS Analysis	1988 Update of ATWS Analysis	1994 NRC Technical Specification Survey
CE	MTC	-2.0 to -6.8	-2.6 to -5.7	0 to +3 above 70% power
	Peak Pressure	4290 psia	4153 psia	
B&W	MTC	-10.5	18 month cycle: -11.0 24 month cycle: -4.3	0 above 95% power
	Peak Pressure	3464 psia	3764psia 18 month cycle: > 3200 24 month cycle: > 3200	
<u>W</u>	MTC	-8.0	-8.0 average range -5 to -15	Linear to 0 from 70% to 100% power One plant at + 2 at 100% power
	Peak Pressure	3197 psia	3497 psia (-5pcm/°F)	

Table 6 also summarizes ...1979 and 1988 MTC values. *Based on NUREG-460 sensitivity analysis that indicates a 1 pcm/ °F less negative (more positive) MTC increases the RCS peak pressure approximately 100 psi calculations based on the limiting TS MTC at full power could lead to higher peak ATWS pressures and longer unfavorable MTCs. CE and B&W reactors installed the DSS to counteract risk and peak pressure effects of unfavorable MTC. The B&W and CE DSS the reactor trips independent of the RPS, improving RPS electrical reliability by a factor of 10. The DSS trips the reactor at approximately 2450 psia to prevent the RCS pressure following an ATWS from reaching the unacceptable condition-3200 psia ASME service level C limit."*

Westinghouse WCAP-11992.... staff in May, 1995. *WCAP-11992 introduced the concept of an unfavorable exposure time (UET) which is the percentage of the fuel cycle time the pressure is in excess of 3200 psia; in this respect UET is fundamentally the same as the unfavorable MTC percentage used in the ATWS rule development. However the UET is calculated differently than the unfavorable MTC percentage and in this respect they are different. ...."*

Higher peak pressures...ATWS rule. *"If the <u>W</u> risk assessment in SECY 83-293 were done using the same method as B&W and CE with an "unfavorable MTC" changing from 1 percent to 40 percent of the fuel cycle, the ATWS risk would increase by a factor of about 20 to 4.3E-6.*"...

WOG GENERAL COMMENT 5: Operator actions to trip the plant are an important part of protecting the reactor and need to be considered in this report. The reactor protection system is a highly reliable system that is backed up by operator actions to trip the reactor. Plant operators are highly trained in this action and there is a high probability of success. Such operator actions need to be considered when drawing conclusions related the reactor protection system reliability and the risk from ATWS events. In fact, in the one ATWS event for a Westinghouse PWR (i.e., Salem), the reactor was tripped manually within 30 seconds after the reactor trip breakers did not open when demanded by the solid-state protection system.

RESOLUTION: See resolution to NRR/DSSS/PSAB Comment 4. Section 3.1 was revised to read:

"*Operator action to scram the plant was not credited in the original risk estimates of the ATWS rule. To be consistent, the current assessment used the same approach. The goal of this assessment was to compare the ATWS rule expectations and outcomes. Consequently, as long as operator performance has not changed greatly since the ATWS rule was enacted or the impact of including operator action to scram is small, this approach is valid.*

First, regarding changes in operator performance to scram the plant which have occurred since the ATWS rule was implemented, there is no basis to support major improvements. At the time of the ATWS rule, operating procedures and training had already been greatly improved following the TMI accident; and the Salem ATWS event implications had been included as previously discussed. The implicit pressure on operators to minimize the number of scrams may not have been as strong as it is now.

Secondly, recent work supports the position that the impact of considering operator action is relatively small. "Reliability Study: Westinghouse Reactor Protection System, 1984–1995" NUREG/CR-5500 Volume 2, shows values of RPS unavailability of 2.1E-5 without operator action and 4.9E-6 with operator action – the operator provides reduced risk by a factor of about 4. This improvement in unavailability assumes that the operator initiates a manual scram 99 percent of the time when a scram signal is present and 50 percent of the time even when a scram signal is not present. In other words, these values already assume excellent operator performance. The reliability study did not include the potential for operator action to manually trip the RPS motor generator sets. The motor generator breaker controls are not always accessible from the control room, which would negate the possibility of timely operator action.

Thirdly, the initiating event frequency is assumed to be the rate of automatic scrams. Since the time of the ATWS rule, the fraction of all scrams which are automatic has decreased and the fraction of operator initiated scrams has increased. Manual scrams were not included in the current ATWS initiating event frequency; their inclusion in initiating event frequency would about double the initiating event frequency."

WOG GENERAL COMMENT 6: In several locations within this draft report, it is stated that B&W and CE reactors are required by the ATWS Rule to have DSS as compensatory measures for higher UETs and Westinghouse reactors do not have this requirement due to a 1% UET. For this comment, the discussion above for General Comments I and 4 first apply. Furthermore, the concept that B&W and CE require DSS due to higher UETs, i.e., higher unfavorable MTC, is misleading. In the 1979 baseline ATWS analyses, all NSSS vendor were subject to the use of the same MTC specifications as given in NUREG-0460 (see General Comment No.4 above). The fact of the matter is that the pressure relief capacity of Westinghouse PWRs is significantly higher, and thus, significantly more favorable than that for the B&W and CE PWRs. The results of the generic deterministic ATWS analyses for Westinghouse PWRs showed a peak RCS pressure of less than 3200 psig when using the base case MTC of -8 pcm/°F as specified by NUREG-0460. The peak RCS pressures for the B&W and CE PWRs were significantly higher. As indicated in Table 6 of this draft report, the 1979 baseline peak RCS pressures for B&W and CE PWRs were 3464 psia and 4290 psia, respectively. It is recognized in SECY-83-293 that with the 1979 baseline MTCs for the B&W and CE plants, a peak pressure in excess of 3200 psig results 50% of the time. Hence, DSS was required for these plants since a change to more favorable MTC conditions was not a truly

viable option from a plant Operation versus reactor physics perspective. It should also be noted that on December 3, 1984, the Commission decided not to issue the proposed rule on ATWS that would have required a diverse scram system for Westinghouse plants.

RESOLUTION: See resolution to WOG General Comments 1 to 4.

The resolution to WOG General Comment 4 clarifies how the CE and B&W DSS provides compensation and that the Westinghouse peak pressures is in excess of 3200 psia for MTC of -8pcm/°F.

To the contrary, the relief capacity of the CE -3800 plants is larger than the W plants.

Section 3.2.1 and Table 2 were revised to acknowledge that December 3, 1984, Commission decision not to issue a DSS rule for W plants.

WOG GENERAL COMMENT 7: Portions of this report and in particular, Section 3.2.5, include discussion on partial rod insertion events. The original concern identified and associated with ATWS is the potential for a common-mode failure of the reactor trip system following anticipated transients that rely on a reactor trip for event mitigation. For such a condition, the real concern is the potential for no automatic or timely rod insertion. In comparison with this potential scenario, the occurrence and consequences of the partial rod insertion events specified in Table5 are far from being comparable to a common-mode failure of the entire reactor trip system. All plants' Technical Specifications have reactor trip scram time requirements and associated testing and surveillance requirements. The purposes of these tests are to specifically confirm reactor trip system and rod insertion characteristics to those assumed in plants' licensing basis safety analyses. The detection of slower than required rod insertion times as a result of these tests indicates a problem. Such problems are appropriately identified and addressed. In addition, only a portion of the total available rods is actually needed to obtain reactor control and shutdown. Hence, such instances of slow or partial rod insertion as listed in Table 5 should not be considered and classified as a precursor to ATWS events.

RESOLUTION: Slow or partial insertion of the rods are within the scope of the ATWS rule. An ATWS is an anticipated operational occurrence followed by failure of the reactor trip portion of the protection system. The reactor trip system function includes the reactivity control systems. 49FR26036 that issued the ATWS rule specifically states the reactor trip system consists of the control rods and the control rod mechanisms.

Section 3.2.5 was revised to read: "*While the number of controls rods affected is only a portion of the total available rods actually needed to obtain reactor control and shutdown*, these failures and degradations are new phenomena and were not considered during the development of the ATWS rule. *Although* these conditions do not affect *present* ATWS analysis assumptions directly, they cannot be dismissed as precursor events."

Also see NRR/DSSA/PSAB Comment 4 which takes a different view than the WOG.

WOG GENERAL COMMENT 8: The report summarized its conclusions in the following manner: "The ATWS Rule was effective in reducing ATWS risk and that the cost of implementing the rule was reasonable. However, uncertainties in reactor protection system reliability and mitigative capability may warrant further attention to ensure the expected levels of safety are maintained."This assessment was based on several of factors for a PWR. One factor is that the RPS system has

been shown to be extremely reliable. This contributes to the inability to quantify the PRA numbers for failure and was used in the justification for further attention. The inability to quantify the numbers for RPS reliability due to the extremely reliable nature of the system is not justification for additional rulemaking. Since the system is so reliable that the numbers can not be accurately quantified, this is justification that rulemaking is not required. Another factor cited is that the cost of modification implementation was less than expected due to reduction in the number of trips per year, which was a recommendation of the rule. It is not appropriate to include industry efforts to improve reliability and availability in the cost benefit of the rulemaking unless the modifications imposed by the rule can be shown to have resulted in that improved reliability. Efforts to reduce the number of trips were in place prior to the rulemaking. The actual cost of the modifications and their continuing costs for maintenance are not effected by the efforts of the utility to improve availability times. The imposed modifications clearly have not increased the availability times of any plant. Therefore, there is no "reduction"in cost as a result of the rulemaking.

The executive summary of the draft itself appropriately frames the current industry situation in that reliability requirements are so high and ATWS events are so rare that many years of operating experience are needed to generate sufficient system demands to reduce current estimates of the uncertainty. The frequency and number of industry events support that useful predictions can be made regarding future occurrences. Increasing the size of the sample or the number of observations may reduce the sampling error, but not any biases. In the absence of a large number of observations, related knowledge of the subject and scientific judgment must be relied upon in framing a course of action. Statistical "significance" or the absence of it by itself is not a rational basis for additional action.

RESOLUTION: No change is required. The report does not call for additional rulemaking. The original ATWS rule value-impact expectation estimated the costs for the ATWS rule modifications and the annual replacement power due to reactor trips from spurious actuation of protective features that were required by the ATWS rule. A like comparison found the actual impact was less than expected due fewer spurious reactor trips from spurious actuation of protective features that were required by the ATWS rule. The industry effort reduce the number of trips was discussed but not included value-impact comparison of the expectations and outcomes even though the Commission instrumental in formalizing the industry scram reduction initiative. Also see resolution to the CE comment.

The report concluded that some issues (based on operating experience) have the potential to erode past achievements and that attention to these issues and regulatory actions that maintain compliance with current regulations can assure the risk from ATWS remains acceptable; specific actions were not mentioned as stated in the comment.

Three issues were identified including RPS reliability uncertainty, PWR moderator temperature coefficient levels and BWR human error probability assumptions for operator action. Regarding RPS reliability, the report indicates the ATWS rule evolved as a result of the statistical uncertainty in RPS reliability due to low amounts of data and (three) ATWS events that statistically should not have occurred. The report assessed the full range of statistics and accompanying uncertainties provided by the inclusion and exclusion of failure data and the application of ATWS rule and present day methodologies.

Specific Comments

WOG COMMENT 1: Section 2, Third Paragraph. See General Comment No. 6 as it relates to the last sentence of this paragraph.

RESOLUTION: See resolution to WOG General Comment 6.

WOG COMMENT 2: Section 2.1, First Paragraph. This paragraph defines three factors that impact the likelihood of damage from an ATWS event. Left out of these factors is the ability of the operator to take actions to trip the reactor if the RPS fails (see WOG General Comment 5). The operators are trained to trip the reactor from the control room within the time available. The operators can also take an action to trip the reactor by interrupting power from the motor-generators to the CRDMs (not all plants can do this from the control room, so all plants do not credit this action in their PRA model). It is important to include these operator actions since they are highly reliable and act as a diverse mechanism for reactor trip.

RESOLUTION: See resolution to WOG General Comment 3 and 5.

WOG COMMENT 3: Section 2.1, Second Paragraph. The statement "P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity" is misleading. An ATWS event that results in core damage does not necessarily continue to containment failure and a release of radioactivity. Additional component failures are required, independent of the ATWS event, for containment failure to occur. This should be clarified in the report.

RESOLUTION: P(ATWS) as defined in the report exactly as defined by the industry and adopted by the NRC during ATWS rulemaking. See resolution to WOG Comment 4

WOG COMMENT 4: Section 2.1.2, Paragraphs 1 and 2. The last sentence in each paragraph is "For PWRs, it is likely that for an unmitigated ATWS, the core would melt prior to containment failure." This doesn't seem to fit at the end of the first paragraph. In addition, please recognize that exceeding the ASME Service Level C stress limit, i.e., 3200 psig, has conservatively been equated to resulting in core damage for the purposes of the ATWS PRA. It is not clear what mechanism is implicitly implied that would lead to containment failure. If the sentence is maintained, it is recommended that the sentence be revised to read"... Prior to any potential for containment failure."

RESOLUTION: The report was revised to read "For PWRs, it is likely that, for an unmitigated ATWS, the core would melt *with the containment intact.*"

WOG COMMENT 5: Section 2.1.2, Paragraph 2. The discussion here describing the ATWS transient conditions only pertains to the condition with a loss of main feedwater. See General Comment 2 above. Also, the water temperature increase described in the analysis can not occur without the fuel temperature increase. Therefore, it is inappropriate to discount the fuel temperature increase and thus the effects of Doppler. It is also inaccurate to state that the water must heat prior to the fuel.

RESOLUTION: See resolution of WOG General Comment 2 on main feedwater. Regarding the doppler coefficient, the fuel thermal time constant can delay the impact of the doppler coefficient in a fast moving transient.

Section 2.1.2 revised to read:

> ... However, the fuel temperature coefficient only comes into play after *the primary coolant temperature has increased because of the delay caused by the thermal time constant of the fuel; this can be an important factor in a fast moving transient such as ATWS*. ...

WOG COMMENT 6: Section 2.1.2, Paragraph 4. A higher ASME service level was only considered for B&W and CE plants due the higher peak RCS pressures for these plants.

> **RESOLUTION:** The report was revised to read "....*for B&W and CE plants"*

WOG COMMENT 7: Section 2.1.2, Paragraph 5. The discussion in this paragraph regarding positive MTC below 100% power is only applicable to plants that have requested and received licensing amendments to permit operation under these conditions. Not all plants have licensed these conditions. The third sentence in this paragraph states that "...all subsequent mitigative functions are likely to be ineffective." This is not totally accurate. Manual rod insertion is a viable mitigative function that would he effective. Regarding the definition of "unfavorable exposure time" given in this paragraph, please see General Comment No. 1 above.

> **RESOLUTION:** The discussion regarding the MTC below 100 percent is supported by plant specific data in Appendix B, Table B-1, that shows most plants have a positive MTC below 100 percent power. See resolution to WOG General Comments 3 and 5 regarding operator action and WOG General Comment regarding the UET.

WOG COMMENT 8: Section 2.2, Paragraph 2. The last sentence should read "... despite perceived high reliability, ..."

> **RESOLUTION:** The sentence was revised as indicated.

WOG COMMENT 9: Section 2.2, Paragraph 4. The discussion in the last part of this paragraph appears open ended. It states an objective for 1985 and states values for 1980 and 1983. Was the 1985 objective met? Page 17 states an average of 0.5 trips / reactor year since 1997.

> **RESOLUTION:** No change required. The report clearly and closely states what the industry stated in the Commission briefing on June 24, 1984. Deleting proportions of what the industry said could imply this solely an NRC initiative. The industry did not meet the 1985 objective as the average number of scrams increased to 5.28

WOG COMMENT 10: Section 3.1, Last paragraph. A review of the information in Appendix B indicates the need for the last sentence in this paragraph. Comments received from WOG utility representatives question the basis for some of the plant-specific numbers in Table B-1. It is recommended that a reference basis be included in the table for the plant- specific values. Otherwise, the validity of the conclusions drawn from the use of these values may be challenged.

> **RESOLUTION:** No change required. The references for all also for the data used in the report information are provided in Section 3.1 and the reference section of the report a bibliography for each reference. In addition, the first sentence of the paragraph being discussed above explicitly indicates where the data came from as follows: "Appendix B gives plant-specific data on the actual outcomes of the ATWS rule regarding modifications and data from probabilistic risk assessment/individual plant examinations (PRA/IPEs). The data were collected from...."

WOG COMMENT 11: Section 3.2, Table I. The "UET<1%" in the "Expected Result" block for Westinghouse plants is not correct. See General Comment No. 4 above.

RESOLUTION: See resolution to WOG General Comment 1.

WOG COMMENT 12: Section 3.2.1, Second paragraph and Table 2. The second paragraph states that Table 2 "was prepared to show the degree of defense-in-depth provided by the ATWS rule modifications that were intended to prevent an ATWS;....". It should also be noted on this table that the operator could trip the plant via the reactor trip switch in the control room. This provides a diverse means of reactor trip that is effective if the reactor trip signal failed due to failure of the analog channels or components in the logic cabinets. It should also be noted that at some plants the operators could also trip the reactor by interrupting power from the motor-generators to the CRDMs. This action is also taken from the control room and is effective if the reactor trip signal failed due to failure of the analog channels, logic cabinet components, or reactor trip breakers. The "UET<1 %" in the PWR Fuel Strategy block of Table 2 for Westinghouse plants is not correct. See General Comment No. 4 above. Also note that a December 3, 1984, letter from the Commission to Westinghouse clearly states that DSS is not required for Westinghouse PWRs. Hence, the "Current Outcomes" block of Table 2 for Westinghouse plants should state "DSS not required".

RESOLUTION: See resolution to WOG General Comment 1, 3, 4, and 5. Table 2 was revised delete to note "DSS not proposed to date." The supporting text was revised accordingly to include acknowledgment of the December 3, 1984, letter.

WOG COMMENT 13: Section 3.2.1, Third paragraph, Item (2). It is not obvious from the information provided in Table 2 how it is can be concluded that DSS appears to provide a compensation for higher UETs (up to 50%) for B&W and CE plants. Also, the discussion on 1% UET for Westinghouse plants is incorrect.

RESOLUTION: See response to WOG General Comment 4.

WOG COMMENT 14: Section 3.2.2, Table 3. The "Outcome" RPS reliability for Westinghouse plants is given as 2.1E-05. This does not account for the backup operator actions as discussed in Comments 2 and 12. Including the operator action to trip the plant from the control room via the reactor trip switch reduces this value to 5.5E-06 (NUREG/CR-5500, Vol.2, "Reliability Study: Westinghouse Reactor Protection System, 1984-1995", December 1998). This also does not account for tripping the reactor via power interruption from the motor-generators to the CRDMs. It is misleading to provide the reliability as 2.1E-05 without stating this does not include credit for the operator actions to trip the plant. Also, the paragraph following Table 3, second sentence, should read "Comparison of Table 3

RESOLUTION: See resolution to resolution to WOG General Comment 5. The second sentence in the paragraph following Table 3 was revised as suggested.

WOG COMMENT 15: Section 3.2.2, First paragraph under Item #2. The statement "These numbers were developed using a fault tree model of the RPS system that may not include all failure modes, a question of completeness for all PRA calculations" can be misleading. This may be true for a complex system with little or no operating history, but there is a significant

amount of operating history on the RPS and operation of the RPS is well understood. It is suggested that the phrase "that may not include all failure modes" is either dropped from the report, or a discussion of why this statement is applicable to the RPS evaluation is included in the report. This discussion should address the quality of the fault trees and data used to determine the RPS reliability, and also how the fault trees and data were developed. This statement should make it clear that these fault trees were developed for an NRC program assessing the reliability of the Westinghouse RPS by NRC contractors, and that their development was done by personnel familiar with the RPS design and operation, and considered the past operating experience of the RPS.

RESOLUTION: The phrase was removed.

WOG COMMENT 16: Section 3.2.2, Item #3. This indicates that the unreliability of the mitigation systems for Westinghouse plants is controlled by the UET and that as the UET increases the mitigation unreliability increases proportionally. This is not true. The 5% UET referred to in the NRC report is associated with a plant operating configuration that will result in, given that an ATWS event occurs, 100% (all) auxiliary feedwater available, relief from all pressurizer safety valves and PORVs (no PORVs are blocked), and no control rod insertion. The statement in the report does not account for the probability, which may be very low, of the plant being in an operating configuration that results in these conditions. If a plant is operating with the rod control system in automatic, no PORVs blocked, and all auxiliary feedwater available, then the UET is 0, and significant increases to the UET for the first set of plant conditions may have no impact on the LIFT for the second set of plant conditions. The unreliability of the mitigation systems (as defined in this report) and the UET is dependent on the plant operating configuration. The probability of being in each plant configuration needs to be taken into account.

It is also noted that the values for mitigation unreliability provided in Table 3 refer only to the situation when there is no credit for any control insertion, both PORVs are available, and all auxiliary feedwater is available. In addition to this configuration, there are eleven other configurations or conditions a plant can be operating in depending on the availability of the PORVs, auxiliary feedwater systems, and the ability for a limited amount of control rod insertion. There are UETs associated with each set of conditions. Referencing a UET for only one set of conditions does not provide the complete picture. Also see General Comment No. 1.

RESOLUTION: See resolution to WOG General Comments 1 and 4, and the Clarification.

WOG COMMENT 17: Section 3.2.3, Paragraph 5. The statement "The range of values of RPS reliability illustrates the difficulty of estimating reliability values in highly reliable systems" is true if you want to base the system reliability on total system failures. Fault tree methods that break the system down to the basic component level at which reliability data is collected is another approach that can be used to assess the reliability of highly reliable Systems. The best reliability assessment from this table is that based on NUREG-5500 which uses the fault tree approach. This approach is a well-established and acceptable method of determining the reliability of highly reliable systems.

RESOLUTION: No change required. Section 3.2.3 including Table 4 explicitly provides the NUREG-5500 results along with other the methods each used in the ATWS rulemaking to be consistent with the report methodology. We cannot revise the report to reflect Westinghouse preferences or to label a methodology as the best.

WOG COMMENT 18: Section 3.2.3, Table 4. In the assessment of "W RPS Unreliability" corresponding to SECY-83-293 baseline update to 1995" it is not appropriate to include the one RPS failure (Salem). Following the Salem RPS failure the reactor trip breakers were redesigned such that an automatic trip occurs by either an under-voltage trip or shunt coil trip. With the redesigned breaker, the automatic signal would have tripped the reactor at Salem.

RESOLUTION: No change required. There were actually two failures at Salem 3-days apart. See resolution to Comment 17 above. Statistically, there are two perspectives and both are presented. As stated in Section 3.2.5, the RTBs are an Important common-mode failure.

WOG COMMENT 19: Section 3.2.5, Item 1. With regard to the importance of common cause failures, it should be noted in this section that these importances are based on an RPS model that does not credit the operator action to trip the reactor. To get a more realistic assessment of the importance of common cause failures it is necessary to include the operator action to trip the reactor from the control room.

RESOLUTION: No change required. See resolution to WOG General Comment 3 and 5.

WOG COMMENT 20: Section 3.2.5, Item 2. This is a very important point. A highly reliable operator action is available to back up a highly reliable system and this needs to be considered when drawing conclusions about the risk of ATWS events at Westinghouse plants. Also, see General Comment No. 7 regarding control rod insertion events.

RESOLUTION: See resolution to WOG General Comment 7.

WOG COMMENT 21: Section 3.2.6, Paragraph 3. References 17, 18, and 19 should be References 16, 17, and 18.

RESOLUTION: Revised accordingly.

WOG COMMENT 22: Section 3.2.6, Table 6. The 1979 Baseline information for Westinghouse plants presented in Table 6 is misstated. As discussed in General Comment No. 4 above, in 1979 the 99% MTC value for Westinghouse plants was -7 pcm/°F (-0.7 10E-4 ΔK/°F) and the corresponding peak RCS pressure was below 3200 psig. Hence, the 1979 Baseline data block in Table 6 for Peak Pressure should read ">3200 psig < 1% of the time." Also, no Westinghouse plants have a positive MTC at 100% power and not all Westinghouse plants are licensed to operate with positive MTC at part-power conditions. This should he reflected in the 1994 MTC survey block in Table 6.

RESOLUTION: See resolution to WOG General Comment 4. The 1994 Survey block reflects the Tech Spec limit, not the operating point.

WOG COMMENT 23: Section 3.2.6, Paragraph 4. Reference 16 should be Reference 15. Also, the statement that HFP MTC values are less negative at HFP conditions than they were in 1979 is correct. These less negative MTC values at HFP are the result of changes in licensed allowable operation at part-power conditions associated with the implementation of positive MTC Technical Specifications. The need for positive MTC at part-power conditions is to support longer operating cycles. Not only are longer cycles an economic benefit, the reduction in the number of startups and shutdowns serve to reduce the potential number of reactor trip demands. In 1979 the

TS MTC at full power was 0 pcm/°F, the same as it is today. Hence, the peak ATWS pressure at the HFP MTC limit remains unchanged.

RESOLUTION: The reference numbers were revised accordingly. See resolution to WOG General Comment 4 and the Clarification.

WOG COMMENT 24: Section 3.2.6, Paragraph 5. The statement that "WCAP-11992 described a risk based approach to justify increasing UETs from 1 percent to 37 percent" is incorrect. The intent of WCAP-11992 was to present a PRA methodology for demonstrating continued compliance with the risk-based PRA approach used in the basis of the Final ATWS Rule and to show that the value of P(ATWS) of 1.0E-05 for Westinghouse plants was still being met when considering changes in plant conditions (e.g., MTC, AFW, PORV availability) important to ATWS. As indicated in Comment No. 16 above, there are UETs associated with different sets of conditions. Referencing a UET for only one set of conditions does not provide the complete picture.

RESOLUTION: See Resolution to WOG General Comment 4.

WOG COMMENT 25: Section 3.2.6, Paragraph 6. The statement "Since the UET is the percentage of the fuel cycle during which an ATWS is unmitigated, increasing the UET from 1 percent to 37 percent results in increasing P(ATWS) by a factor of 37 as calculated by the techniques of SECY-83-293" may be correct if the techniques of SECY-83-293 are correct. But SECY-83-293 is no longer an appropriate approach for this calculation. It does not account for the various configurations in which the plant may be operating and does not credit partial rod insertion. Increasing the UET by a factor of 37 may have no impact on ATWS risk if the plant is operating in a configuration consistent with a 0 UET. The 37% UET referred to here is associated with a plant operating configuration that may not be the most probable operating configuration. As discussed above in General Comment No. 6, CE and B&W plants were required to install DSS and AMSAC due to limited pressure relief capacity relative to Westinghouse plants. An option of adding relief valves was also considered to reduce the peak pressure in CE and B&W plants but was eliminated due to the high cost associated with adding valves and the resulting low valve-impact ratio. Finally, the last two sentences in this paragraph appear to imply that the implementation of DSS would resolve any ATWS concerns for Westinghouse plants seeking significantly higher UETs. If this is true, this option should be more clearly described in the corresponding text for this section and in the Executive Summary and Conclusions sections of the report.

RESOLUTION: See Clarification and resolution to WOG General Comment 4.

WOG COMMENT 26: Section 3.3 , Paragraph 3. The first sentence should read ".... effects of the industry.....in 49FR26036." The last sentence should reflect 7140 trips. Also this sentence uses 20 years in determining the total number of trips but Table 7 reflects 30 years.

RESOLUTION: Sentence edited as suggested. See Resolution to CE comment.

WOG COMMENT 27: Section 3.3, Paragraph 4. The last sentence should read "....the ATWS rule was cost effective."

RESOLUTION: No change required. This is a conclusion that can be found in conclusions, Section 4.0 of the report as stated in this comment.

WOG COMMENT 28: Section 3.3 and Section 4, Second Bullet. It is not reasonable to claim the reactor trip frequency reduction is due entirely to the Commission's recommendation to reduce the number of scrams. A good percentage of this reduction would most likely have occurred anyway as utilities gained experience in plant operation, from changes to test and maintenance strategies, and due to economic pressures to keep the plants online.

RESOLUTION: No change required. There is no such claim. The Section 2.2 acknowledges this was an industry initiative and that the Commission involvement served to obtain an industry commitment to formal the program.

WOG COMMENT 29: Section 4, First and Eighth Bullets. See earlier discussions on the definition of UET and on the reference to 1% UET in lieu of DSS for Westinghouse plants. These apply here also.

RESOLUTION: See resolution to WOG General Comment 1.

WOG COMMENT 30: Table B-I. As previously discussed in Comment No. 10, comments received from WOG utility representatives question the basis for some of the plant-specific numbers in Table B-1. It is recommended that a reference basis be included in the table for the plant-specific values to allow a cross-reference with the data source. Otherwise, the validity of the conclusions drawn from the use of these values may be questionable. Since so much of the discussion in this report involves the plant type by NSSS vendor, it is suggested that a column be added to identify the NSSS vendor / plant type.

The units for the MTC should be provided in the last column title block. Many of the values given in this column are incorrect or misrepresented. For example, the TS MTC for the Byron, Braidwood, and V.C. Summer units is not just +7E-05. The TS MTC for these plants are the same as given for the Catawba units. The Robinson Unit 2 TS MTC (per TS 3.1.3 in Amendment #176) is +5E-05 below 50% RTh and 0 above 50% RTh. The South Texas units show a 0 MTC in Table B-1but the TS MTC is the same as given for the Comanche Peak units.

RESOLUTION: Table B-1 was updated.

COMBUSTION ENGINEERING (CE) provided the following comments in a letter to the NRC dated December 29, 2000.

CE COMMENT: After review of the reference letter, the CEOG finds the basis for the ATWS rule reasonable. However, it is difficult to attribute to the ATWS rule the economic benefits achieved by extensive industry efforts to reduce plant trips and challenges to safety systems. The CEOG applauds staff efforts to improve the cost effectiveness of NRC activities and decisions, and welcomes opportunities to promote regulatory efficiency and realism.

> **RESOLUTION:** The Section 3.3 of the report is revised as follows: "In addition, although not included in the this value impact analysis, the 3.5 reduction in the average number of automatic scrams for 102 operating reactors *has significant monetary value due to replacement power costs."*