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Draft Report

Regulatory Effectiveness of the Anticipated Transient Without Scram Rule

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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.

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ABBREVIATIONS

AMSAC	ATWS mitigating system actuation circuitry
ARI	alternate rod injection
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox Co.
CCF	common-cause failure
CDF	core damage frequency
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
DSS	diverse scram system
EPG	emergency procedure guideline
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
RCS	reactor coolant system
RES	Nuclear Regulatory Research, Office of (NRC)
RPS	reactor protection system
RPT	recirculation pump trip
RTB	reactor trip breaker
RY	reactor-year
SLC	standby liquid control
TS	technical specification
UET	unfavorable exposure time

W	Westinghouse

EXECUTIVE SUMMARY

As part of the Nuclear Regulatory Commission 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 reactor protection system (RPS). The likelihood of core damage from an ATWS depends on three factors: (1) the initiating event frequency, (2) the reliability of the 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. and Babcock & Wilcox Co. plants installed a diverse scram system (DSS). Westinghouse plants generally maintain an "unfavorable exposure time" of one percent in lieu of a DSS. 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 (<1E-5 per reactor year) 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. The 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. Updating the original ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that all four reactor types achieved the risk goal of P(ATWS)<1.0E-5 per reactor year .

- 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.

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 PRA 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. The current uncertainty associated with RPS reliability argues for continued application of the requirements of the ATWS rule.

- 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. Fuel design to achieve longer cycles result in less negative MTCs at full power for a larger fraction of the cycle time, during which time ATWS mitigation is rendered ineffective. CE and B&W reactors installed a DSS to compensate for large exposure times.

Further fuel cycle changes that significantly increase the ATWS risk due to longer exposure times may require compensatory measures consistent with the ATWS rule.

- ATWS mitigation on a BWR is highly dependent on operator actions. 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. Consequently, some BWR risk analyses may underestimate the risk of ATWS.

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 reactor protection system (RPS). The requirements and Commission recommendations for addressing ATWS were published on June 26, 1984, in the *Federal Register* (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 anticipated operational transients 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 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 PWR or 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 boiling-water reactor (BWR) (Brown's Ferry Unit 2) in 1980 and a pressurized-water reactor (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

[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.

uncertainty associated with the RPS reliability was a major factor in the decision to adopt the ATWS rule.

The ATWS rule states that PWRs designed by Westinghouse Electric Corp. (W), Babcock and Wilcox Co. (B&W) and Combustion Engineering, Inc. (CE) are required to have equipment that is diverse, reliable, and independent from the reactor trip system to automatically initiate the auxiliary feedwater (AFW) system and initiate a turbine trip under conditions indicative of ATWS. This equipment is called ATWS mitigating systems 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 exposure times (UETs) to 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. The Commission also issued two recommendations with the ATWS rule: (1) reduce the number of automatic scrams and (2) improve RPS reliability.

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, and B&W and CE reactor groups using the simplified event trees and data shown in Appendix A, "ATWS Rule Event Trees."

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

2.1.1 BWR 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 degrees Fahrenheit 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]. As the suppression pool temperature increases, the potential unstable condensation of the discharge to the pool may overload the containment structure. During an ATWS at a BWR, the containment would probably fail prior to core damage.

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.

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 degrees Fahrenheit 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 degrees Fahrenheit 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 (EPG)/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 PWR 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 of which is dependent on the moderator temperature coefficient (MTC), the relief capacity, and how much energy the steam generators can remove. The MTC is a measure of the reduction in the core reactivity as the water temperature increases. In SECY 83-293 it was assumed that 70 percent of the transients involve a turbine trip for which the bypass to the condenser is available and 30 percent of the transients were treated as a non-turbine trip, loss of feedwater transient for which the bypass is not available. For PWRs, it is likely that for an unmitigated ATWS, the core would melt prior to containment failure.

During an ATWS, the primary coolant temperature increases, since 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 limit the power increase. The fuel temperature coefficient, otherwise known as the Doppler coefficient, which is always negative, also works to limit the power increase. However, the fuel temperature coefficient only comes into play after the fuel temperature increases, (i.e., after the primary coolant temperature has already increased). Thus, since the MTC acts first, it has a major impact on the peak pressure of the fast-moving ATWS event. Since, as a whole, the combined steady state fuel coefficient and MTC are always negative, reactor power eventually comes into balance with energy removal. However, unless core power can be reduced by injecting boron or inserting the control rods, reactor pressure would remain high and coolant would continue to be lost through the primary system safety valves. For PWRs, it is likely that, for an unmitigated ATWS, the core would melt prior to containment failure.

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 it was assumed that a peak pressure of above 3200 psig was unacceptable for ASME Service Level C. A higher ASME service level was considered 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 (HPI) system. The pressure is reduced by removing heat through the steam generators, the auxiliary feedwater (AFW) system, and steam generator relief valves. If the pressure cannot be reduced, reactor coolant will be lost through the 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.

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. 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 and 50 percent for B&W/CE reactors

on the basis of an analysis by the three reactor manufacturers; B&W and CE installed DSS to ensure a reactor trip and compensate for the longer UET. At the time the ATWS rule was issued, it was believed that the UET for B&W and CE reactors would become more favorable because of increased use of burnable poisons with high-burnup fuel. SECY-83-293 also stated that relief valves could be installed as additional compensatory measures to increased UETs.

Appendix A contains the simplified 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.

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 high perceived 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, the Nuclear Management and Resources Council (now the Nuclear Energy Institute) and Institute of Nuclear Power Operators (INPO) 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 INPO 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 Westinghouse 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 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 auxiliary feedwater 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, UET<1% CE: AMSAC, DSS, UET<50% B&W: AMSAC, DSS, UET<50% GE: ARI, RPT, and SLC capacity; some automatic SLC initiation	All required modifications implemented; UET greater than expected for a few plants	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: 6.4E-07 CE: 2.6E-06 B&W: 9.4E-07 GE: 1.0E-06	Expectation met but large uncertainties in RPS reliability ATWS mitigating capability sensitive to UET 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 UETs. 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 UETs); and a summary statement of the outcome.

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 Propose rule for DSS	UET<1%	Modifications installed, DSS rule not proposed to date.
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	UET<50%	Met intent of the ATWS rule as modifications installed , UET<50%.
		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	UET<50%	
		9 plants	1/8 RTB, shunt undervoltage trips			
		3 plants	1/4 RTB (2 W and 2 GE) with automated shunt and undervoltage			

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 UETs (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 UETs 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.

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	3.7E-06	1.2E-05	1.0E-05
Mitigation Unreliability	2.3E-01	3.6E-01	4.8E-02	6.0E-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	6.4E-07	2.2 E-05	9.4E-07	2.2E-05	2.6E-06.

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 2 P(ATWS) expectations to outcomes shows that the P(ATWS) outcomes are better than expected by factors of approximately 12, 9, 24, and 8 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. Table B-3 shows that the outcomes of upper and lower bounds of unreliability range from 1.8-06 to 5.7E-05. 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 factors of approximately 2.5-8. Table B-3 also shows the upper bound of uncertainty analysis associated with current estimates are 5.7E-05, 2.5E-05, 1.4E-05, and 0.78E-05 for the W, GE, CE and B&W reactors, respectively.

(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 low UET of approximately 5 percent. The impact of increasing UET is proportionate. Increasing MTC can have a major adverse impact on the UET. This trend is discussed further in Section 3.2.6.

3.2.3 Reactor Protection System Reliability Validity

Table 4, "Alternate Estimates of RPS 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. The authors of 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

[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.

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.

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	2.2E-05	0.58E-05

3.2.4 Risk Insights From Licensee PRA/IPEs

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. 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. Consequently, some BWR risk analyses may underestimate the risk of ATWS.

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) are as follows:

(1) 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 dominant common-mode failures, though of low probability, are the train bistables, trip logic relays, undervoltage cards, reactor trip circuit breaker (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.

(2) The ATWS rule development did not credit a manual scram. NRC RPS reliability studies indicate that credit for operator actions to scram improves the RPS reliability by a factor of approximately 2 for the GE, W, B&W, and CE reactor groups. Further, in the W design, manual scram bypasses a driver card which is a dominant but not probable common-cause failure mode for the automatic scram.

Review of recent operating experience in areas found to be dominant CCFs found (1) the industry is still addressing RTB maintenance/reliability issues that were identified from the Salem ATWS events, and (2) the phenomena related to control rod insertion was not considered during development of the ATWS rule. To elaborate:

(1) 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 EPRI/NMAC workshops, an NEI 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.

(2) 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. These failures and degradations are new phenomena and were not considered during the development of the ATWS rule nor the recent NRC reliability studies. While these conditions do not affect ATWS analysis assumptions directly, they cannot be dismissed as precursor events.

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, Westinghouse 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.

3.2.6 Changes in Fuel Management May Affect PWR ATWS Mitigating Capability

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. Since the ATWS rule was developed, there has been a trend to use higher burnup fuels to extend the fuel cycle which often leads to a less negative (more positive) MTC. The plant technical specifications (TSs) limit the MTC value and require MTC surveillance at the beginning of the fuel cycle.

Table 6, “ATWS MTC and Peak Pressure for PWRs,” 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 10E-4 ($\triangle K/^{\circ}F$).

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. 17, 18, and 19]. The responses generally concluded that no significant changes were required in ATWS analyses.

Table 6 ATWS MTC and Peak Pressure for PWRs

Parameters		1979 Baseline	1988 Update	1994 NRC Technical Specification Survey
CE	MTC	-0.20 to -0.68	-0.26 to -0.57	0 to +0.3 above 70% power
	Peak Pressure	4290 psia	3962 psia	Analysis not requested by NRC
B&W	MTC	-1.05	18 month cycle: -1.10 24 month cycle: -0.43	0 above 95% power
	Peak Pressure	3464 psia	3600-4000 18 month cycle: > 3200, UET=31% 24 month cycle: > 3200, UET=40%	Analysis not requested by NRC

W	MTC	-1.0 average	-1.0 average range -0.5 to -1.5	Linear to 0 from 70% to 100% power One plant at + 0.2 at 100% power
	Peak Pressure	>3200 psia 1%-10% of time	>3200 psia 1%-10% of time	Analysis not requested by NRC

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. Calculations based on the limiting TS MTC at full power could lead to higher peak ATWS pressures and longer UETs.

Westinghouse 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 described a risk based approach to justify increasing UETs from 1 percent to 37 percent. 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 WOG has been working with the NRC staff to address those issues. On August 23, 2000, the WOG met with the staff to obtain NRC concurrence that the WOG approach to ATWS for licensing issues, such as MTC, is acceptable and to continue discussions of the WOG responses to issues raised by the staff.

Higher peak pressures and longer UETs lessen the effectiveness of the mitigative functions required by the ATWS rule. 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. CE and B&W reactors installed the DSS to increase the reliability of the scram system to counteract those factors. The effectiveness of the ATWS rule would be compromised if UETs are increased. In particular, large percentage increases in UET being considered by Westinghouse 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 (V-I) ratio, the highest V-I 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 V-I 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 an industry commitment to reduce the number of scrams in response the Commission's recommendation in FR 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 this V-I analysis, the 3.5 reduction in the average number of automatic scrams for 102 plants for 20 years, if maintained at this level, has future value of more than $10 billion for averting 7104 trips.

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 V-I 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. and Babcock & Wilcox Co. plants installed a diverse scram system (DSS). Westinghouse plants generally maintain an "unfavorable exposure time" of one percent in lieu of a DSS. 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 (<1E-5 per reactor year) 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. The 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. Updating the original ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that all four reactor types achieved the risk goal of P(ATWS)<1.0E-5 per reactor year

- 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.

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 PRA 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. The current uncertainty associated with RPS reliability argues for continued application of the requirements of the ATWS rule.

- 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. Fuel design to achieve longer cycles result in less negative MTCs at full power for a larger fraction of the cycle time, during which time ATWS mitigation is rendered ineffective. CE and B&W reactors installed a DSS to compensate for large exposure times. Further fuel cycle changes that significantly increase the ATWS risk due to longer exposure times may require compensatory measures consistent with the ATWS rule.

- ATWS mitigation on a BWR is highly dependent on operator actions. 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. Consequently, some BWR risk analyses may underestimate the risk of ATWS.

5 REFERENCES

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March 2000.

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

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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.

24. U.S. Nuclear Regulatory Commission, letter from M. Slosson, Office of Nuclear Regulatory Regulation, to V.D. VanderBerg, Chairman of Westinghouse Owners Group, “Review of WCAP-11992,” July 1, 1997.

APPENDICES

APPENDIX A

ANTICIPATED TRANSIENT WITHOUT SCRAM RULE EVENT TREES

DATA FOR BWR 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). 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.

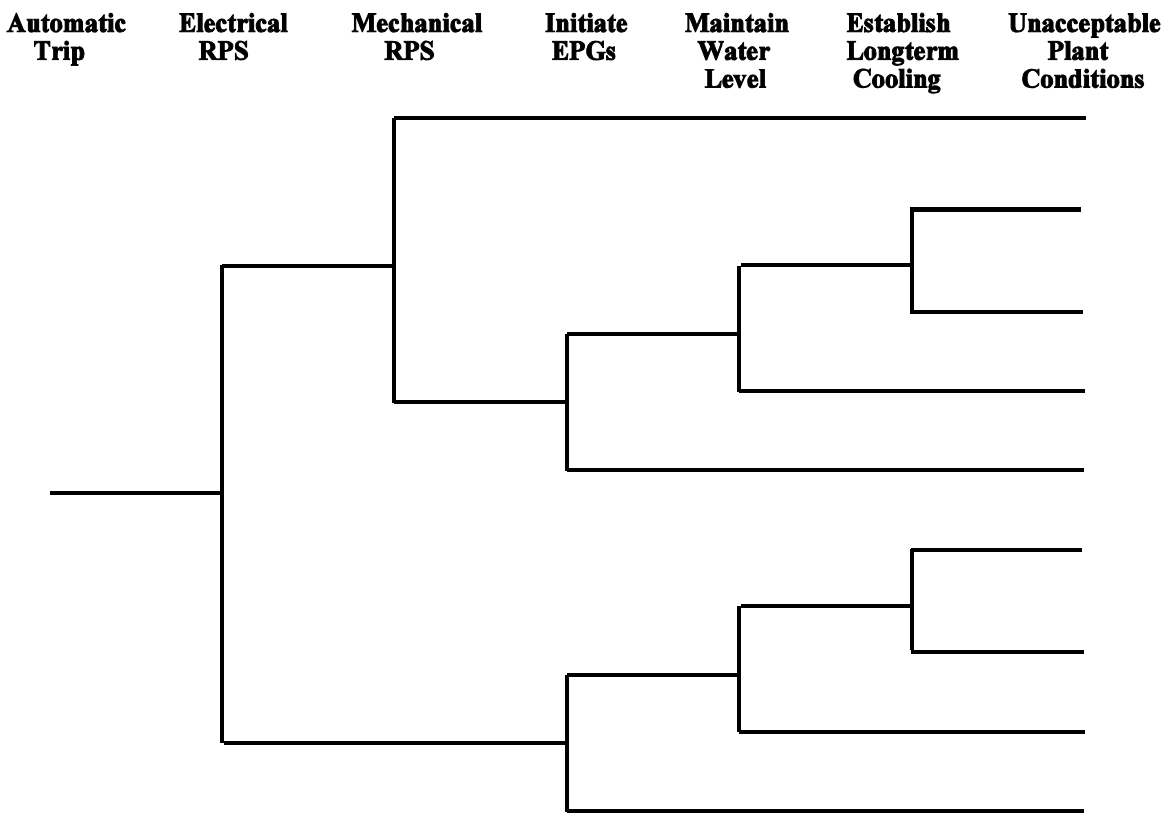

Figure A-1: ATWS Rule Event Tree for GE Reactor Group

Table A-1.1 Data for BWR 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	1.0E-05	2.1E-06 (Ref. 1)
Mechanical RPS Reliability	<2E-06	3.7E-06
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 BWR 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	1.0E-05	2.1E-06 (Reference 1)
Mechanical RPS Reliability	<2E-06	3.7E-06 (Reference 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)

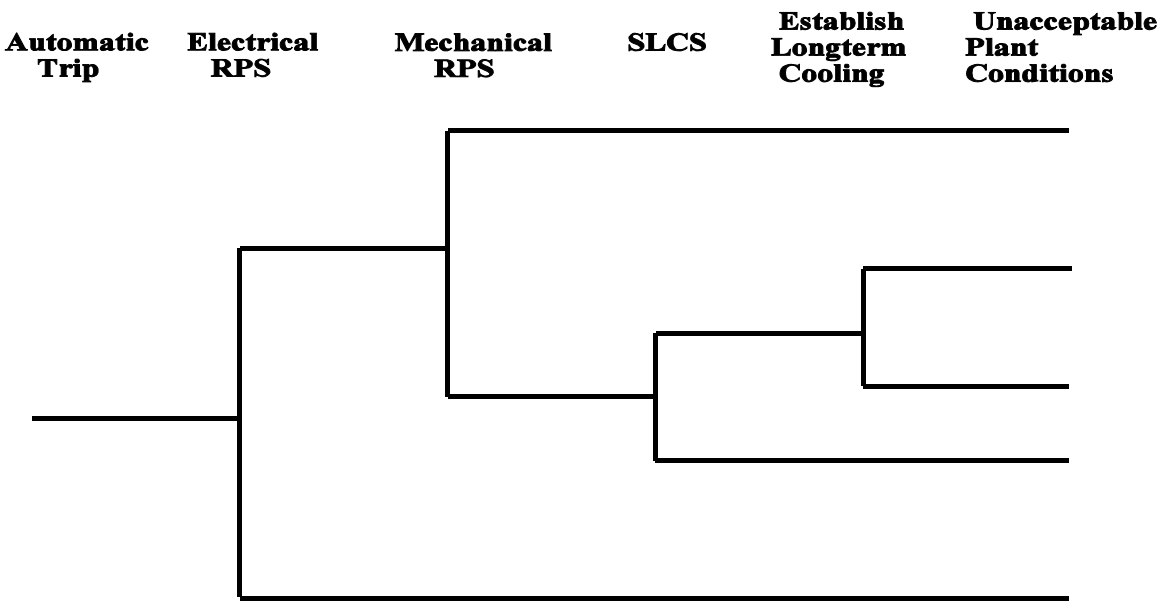

Figure A-2: ATWS Rule Event Tree for GE Reactor Group Automatic SLCS

Table A-2.1 Data for BWR ATWS Transient With Automatic SLC System 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	1.0E-05	2.1E-06 (Ref. 1)
Mechanical RPS Reliability	<2E-06	3.7E-06 (Ref. 1)
SLCs	0.01	0.01 (assumed, no update available)
Establish Long-term Cooling	0.05	0.05 (assumed, no update available)

DATA FOR PWR 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). 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.

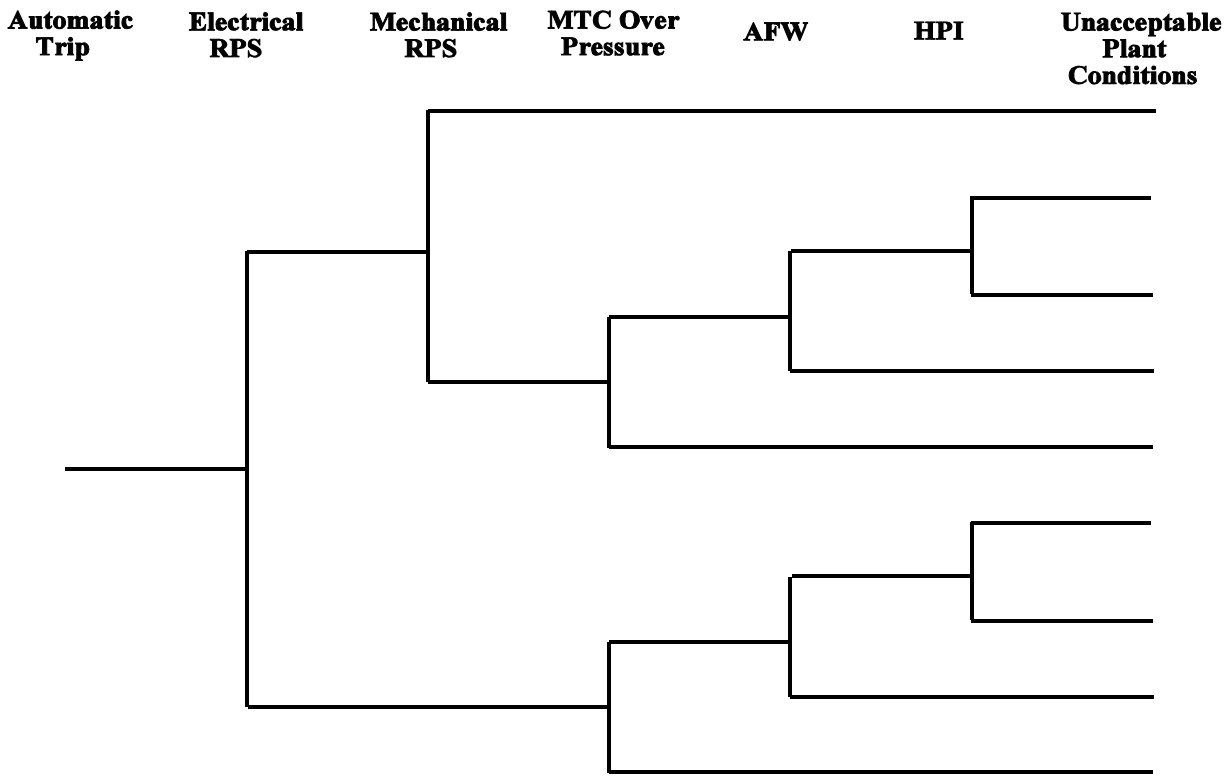

Figure A-3: ATWS Rule Event Tree for Westinghouse Reactor Group

Table A-3.1. Data for Westinghouse PWR 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	1E-05	1.2E-06 (Ref. 2 and note 2)
RPS Mechanical	2E-05	2E-05 (Ref. 2 and note 2)
MTC Overpressure	0.01	0.05 (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 PWR 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	1E-05	1.2E-06 (Ref. 2 and note 2)
RPS Mechanical	2E-05	2E-05 (Ref. 2 and note 2)
MTC Overpressure	0.01	0.05 (Ref. 5)
ATWS Reliability	0.001	0.00045 (Ref. 8)
HPI	0.01	0.01 (assumed, no current data available)

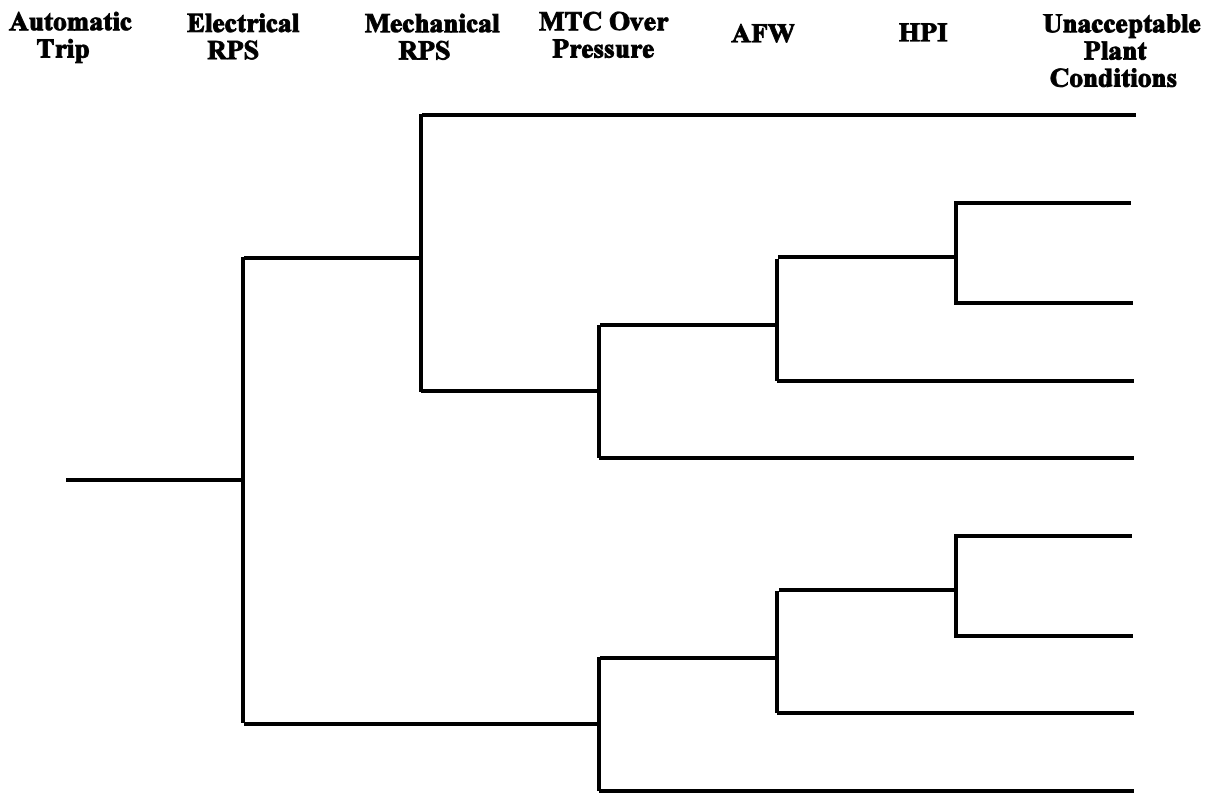

Figure A-4: ATWS Rule Event Tree for CE / B&W Reactor Group

Table A-4.1 Data for CE/B&W 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	1E-05	CE: 5.8E-08 (Ref. 3, note 3) B&W: 1.0E-07 (Ref. 4, note 4)
RPS Mechanical	2E-05	CE: 1.0E-06 Ref. 3, note 3) B&W: 2.0E-06 (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

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.

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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.

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

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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
Braidwood Unit 2	2.74E-05	3.70E-07	1.35	AMSAC	0,1,2	+7E-05
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
Callaway	5.85E-05	4.80E-07	0.821	AMSAC	0,0,1	
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%)

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 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)

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
Davis-Besse	6.60E-05	3.54E-07	0.536	DSS AMSAC	1,2,0	+9E-05 (0 above 95% RTP)
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%)

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
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%)
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)

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
Millstone Unit 3	5.61E-05	3.40E-06	6.06	AMSAC	0,1,0	
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

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 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)
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	+7 to +2 at 100% 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)

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 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)
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	0
South Texas Unit 2	4.30E-5	3.00E-07	0.698	AMSAC	1,1,1	0
V.C. Summer	2.00E-04	2.03E-06	1.02	AMSAC	1,0,1	+7E-05
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

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
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

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
LaSalle Unit 1	4.74E-05	1.87E-07	0.395	ARI RPT Increased SLCS boron concentration	0,0,1

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 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	0,0,2
Limerick Unit 2	4.30E-06	9.30E-07	21.6	ARI RPT	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
Nine Mile Point Unit 2	3.10E-05	1.10E-06	3.55	ARI RPT Increased SLCS boron concentration	0,0,2
Oyster Creek	3.90E-06	2.40E-07	6.15	ARI RPT Increased SLCS boron concentration	0,0,0

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)
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
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

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)
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
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 RPS 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	3.7E-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 RPS 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 RPS 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 RPS 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 RPS 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+04	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