UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 5 1983

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

Enclosed for your consideration is a Commission paper which recommends issuance of a final amendment and a related proposed amendment to 10 CFR Part 50. The amendments would require design improvements in systems for shutting down a reactor in the event of anticipated transients, or for mitigating the effects of not shutting down the reactor. We believe the paper fairly presents the ATWS problem and recommends a reasonable course of action to the Commission.

The enclosed paper was developed with the assistance and guidance of the ATWS Task Force and Steering Group and the Generic Implications Task Force on Salem. The ATWS Task Force and Steering Group was established in July 1982, to resolve the ATWS rule. The Generic Implications Task Force on Salem was established immediately after the event on February 25, 1983. The Salem incidents significantly increased interest in the ATWS issue and caused a concerted effort to expedite action on the rulemaking.

The rule changes recommended in the enclosed paper are compatible with actions likely to be taken as a result of recommendations of the Generic Implications Task Force on Salem.

The ACRS's Subcommittee on ATWS has been briefed on this rulemaking, and a briefing of the ACRS Committee is scheduled for July 7, 1983. The rulemaking actions were discussed on three occasions with the CRGR, and we believe the enclosed paper is consistent with suggestions provided by the CRGR.

Concurring in recommendations of this paper are the Offices of Nuclear Reactor Regulation and Inspection and Enforcement, and the Division of Rules and Records. The Executive Legal Director has no legal objection. The Office of Public Affairs prepared the draft public announcement. Representatives from Regions I and II were on the ATWS Steering Group and copies of the paper have been furnished to all the Regional Administrators.

Robert B. Minogue

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosure:
Commission Paper on ATWS Rule Changes

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

Purpose: To obtain Commission approval for publication of a final rule and a related proposed rule in the Federal Register.

Category: This paper covers a major policy question.

Issue: What improvements, if any, should be required by regulation in the design of light-water-cooled nuclear power plants to reduce the risk of ATWS events to an acceptable level.

Discussion: An anticipated transient without scram (ATWS) is an expected operational occurrence (such as a loss of feed water, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor protection system to shut down the reactor. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to core melt and release of radioactivity to the environment. Anticipated transients occur on the average of ten times per year. The ATWS question involves the need, if any, to provide backup safety systems for shutting down the reactor or mitigating the effects of an ATWS, if there is an anticipated transient and a concurrent failure of the reactor protection system.

In 1980 the staff completed review and evaluation of information developed over the preceeding 10 years on ATWS events and the manner in which such events should be considered in the design and safety evaluation of nuclear power plants. This evaluation, reported in "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, suggested that the frequency of a severe ATWS event may be unacceptably high.

CONTACT:
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443-5960

In SECY 80-409, dated September 4, 1980, the staff recommended publication of a proposed rule to require improvements in the design of reactors to reduce the likelihood of ATWS events and to mitigate the consequences of such events. Following extensive consideration of SECY 80-409 by the Commission and of an alternative approach proposed by then Chairman Joseph Hendrie, two alternative proposed rules were published for public comment on November 24, 1981. In the November 24, 1981, Federal Register notice the Commission stated that it also was considering a third alternative proposed rule that was contained in a petition for rulemaking (PRM-50-29) received from a Utility Group on ATWS (which now represents 22 electric utilities) that had been published for comment on November 4, 1980. When the two alternative proposed rules were published, the comment period for the Utility Group's petition was reopened and the public was invited to comment on the three proposed rules.

While each of the three alternative proposed rules had the objective of reduction of risk from ATWS, each featured a different approach to achieving that objective. The staff-developed alternative emphasized individual reactor evaluation to identify needed improvements. The Hendrie alternative emphasized reliability assurance and would have also required certain hardware modifications. The Utility Group alternative prescribed specific changes keyed to the type of reactor and its manufacturer.

As noted in the June 28, 1982, status report to the Commission on this rulemaking (SECY 82-275), 39 public comments were received on the three alternative proposed rules. Thirty-one of the comments were from utilities. The most frequent comment was that no rule on ATWS was needed, but if one were to be adopted the preferred rule was the alternative proposed by the Utility Group.

The staff has reviewed all the comments and now recommends publication of a final rule and a related proposed rule. The final rule, set out in Enclosure "A", adopts the same approach that is used in the Utility Group's petition for rulemaking in that the prescribed changes are keyed to the reactor's type and manufacturer. With three exceptions the prescribed changes in the final rule were proposed in the Utility Group's petition. Those exceptions are the new requirements for the standby liquid control system (SLCS) used on boiling water reactors and the inclusion of reactor trip breakers for the diverse scram system on pressurized water reactors manufactured by Combustion Engineering and Babcock and Wilcox.

The final rule requires an increased capacity for the SLCS on all BWRs. The petitioner has proposed the increased capacity for SLCS only for BWRs that are yet to be constructed. The final rule also requires that SLCS initiation be automatic (in lieu of manual initiation by the operator) for new plants and for other plants that have already been designed to include this feature. Automatic initiation of SLCS was not proposed by the petitioner. The staff believes the SLCS is sufficiently important to reactor safety to require the increased capacity for both present and future reactors and that it would be cost effective to require automatic initiation of SLCS for future reactors. The staff believes that the inclusion of a diverse method of interrupting power to the diverse scram system is warranted based on the potential of common cause failure.

The proposed rule, set out in Enclosure "B", is a consequence of the recent ATWS events at the Salem Nuclear Power Plant. Salem uses pressurized water reactors (PWRs) manufactured by Westinghouse. The Salem events caused reconsideration of the need for diversity in scram systems in Westinghouse PWRs. The staff believes that the attached final rule's diversity requirement for CE and B&W PWRs should also apply to Westinghouse PWRs and should also include the reactor trip breakers.

The estimated cost and reduction in the probability of a core melt accident attributable to the requirements in the attached rules and of other requirements that were considered are discussed in Enclosure "D".

In evaluating comments on the three proposed rules and developing the attached final rule and related proposed rule, the staff used a combination of probabilistic risk assessment (PRA) techniques and engineering judgment. These PRA techniques were useful in comparing cost/benefit ratios for candidate regulatory changes and in developing a set of changes which should ensure an acceptable low risk from ATWS events. The results of the PRA are consistent with the staff's engineering judgment that the rules are needed to reduce the risk from ATWS events.

<u>Alternatives</u>: The staff is proposing that the Commission consider two alternative courses of action:

<u>Alternative 1</u> - Withdraw the proposed rules that were published in the Federal Register on November 24, 1981, and maintain present regulations (thereby adopting the position that operating plants as designed are safe enough or, if changes in certain plants are needed, those changes can be required by order on a case-by-case basis).

Pro: Existing Reactor Protection Systems (RPS) were designed, manufactured, inspected, installed, operated, and tested to specified and approved codes, standards, guides, and procedures. In addition, this system incorporates independence, redundancy, and some diversity. These considerations suggest that the RPS design provides adequate protection and the costs of additional requirements should not be imposed on licensees. If certain licensees have reliability problems such as occurred at the Salem plant, those licensees can be given case-by-case consideration.

Con: While the RPS may provide adequate protection from multiple random independent failures, the design is susceptible to common mode failures. Common mode failures of reactor protection systems have occurred, e.g., at the Salem Plant in February 1983. The RPS is challenged on the average of 10 times per year and there is need for the high reliability that would be provided by the final and proposed rules. Cost/benefit analyses indicate the requirements in the rules would be cost effective. Remedy of the generic ATWS problem by rulemaking is administratively more efficient than licensee-by-licensee consideration.

Alternative 2 - Issue a final rule that prescribes generic safety improvements keyed to the reactor's type and manufacturer, and issue a proposed rule to include Westinghouse plants among PWRs that would be subject to the scram system diversity requirement.

Pro: Information gained over the past several years has led to a full understanding of all of the implications of the various prescriptive requirements. Therefore, the prescriptive rule approach avoids unnecessary extensive individual case analyses by licensees and the staff. Staff evaluations indicate that the prescribed changes would be cost effective in reducing the risk of ATWS events. The proposed rule for Westinghouse plants* would alleviate the potential for common cause failure of the Westinghouse trip system. The two rules could lead to completion of the rulemaking proceeding and to maintaining an acceptable level of risk from ATWS events without the expense to the NRC and the industry of plant-specific evaluations.

Con: Generic treatment of the plants causes difficulty in performing Regulatory analyses because of risk

*A proposed rule rather than a final rule is needed because the 1981 proposed rules did not explicitly include the scram diversity requirement for Westinghouse plants. Diversity was proposed for the scram systems for Combustion Engineering and Babcock & Wilcox PWRs.

important variations in design and operations among the plants. A prescriptive rule discourages consideration of alternative (and perhaps better) ways of achieving the desired safety for individual plants. Variations among the plants may result in differences from plant to plant in the effectiveness of the prescriptive design changes which will not be quantified.

Recommended Alternative - The staff recommends that Alternative 2 be implemented. The staff believes that the final rule set out in Enclosure "A" and the proposed rule set out in Enclosure "B", if made effective, would substantially reduce the ATWS risk in a cost effective manner and assure an acceptable level of risk from ATWS events.

Resource Requirements - Implementation of the rules would have only a minor impact on NRC resources. ATWS considerations are already being reviewed by the staff on a case-by-case basis for applicants, and since ATWS has never been specifically addressed in the Commission's regulations, it has become a contested issue by intervenor groups. The Regulatory treatment of ATWS provided in the attached rules should reduce the amount of staff time currently required to communicate with applicants and intervenors on this issue.

Initially, little effort would be required to implement the rules. NRR will need to agree with vendors on acceptable design modifications and implementation schedules will be worked out with licensees. The Regional Offices will be responsible for assuring that the modifications are properly installed.

Implementation of the rules will cost licensees an estimated \$3.5M to \$5.5M per plant. The estimated total cost to industry is on the order of \$525M. Enclosure "C" provides further information on the estimated costs.

Recommendations: That the Commission:

1. Approve publication in the Federal Register of the notices of:

   a. final amendment of 10 CFR 50 to add a new §50.62 regarding design improvements to reduce risk from ATWS events (Enclosure "A",) and

   b. proposed amendment of 10 CFR 50 which proposes change of the new §50.62 (c)(2) to include Westinghouse PWRs among the PWRs required to have a diverse scram system (Enclosure "B").

2. Note

a. The notice of proposed rulemaking in Enclosure "B" will be published in the Federal Register with a 45 day comment period.

b. No environmental impact statement, negative declaration, or environmental impact appraisal need be prepared in connection with the amendments because the actions taken or proposed will not significantly affect the quality of the human environment.

c. That pursuant to the Regulatory Flexibility Act of 1980 the notices contain statements that the Commission certifies that the final rule will not, and the proposed rule if promulgated will not, have a significant economic impact upon a substantial number of small entities. The rules would affect only nuclear power plants and companies that own these plants do not fall within the definition of "small entities." The Chief Counsel for Advocacy of the Small Business Administration will be informed by DRR of the certification regarding economic impact on small entities together with the reason for it.

d. That the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works, the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy Conservation and Power of the House Committee on Energy and Commerce, and the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations will be informed of the Commission's action by letter such as Enclosure "E".

e. The reporting requirements in connection with the implementation schedule required by the rule impose information collection requirements that are subject to the Paperwork Reduction Act. The requirements were approved by OMB.

f. That a Regulatory Analysis is attached as Enclosure "C".

g. That a public announcement will be issued (Enclosure "F".)

h. That copies of the Notices will be distributed by TIDC, ADM to affected licensees and other interested parties.

i. That the staff recommends the paper be placed in the PDR.

Scheduling: It is recommended that this paper be considered at an open session.

William J. Dircks
Executive Director for Operations

Enclosures:
"A" - Notice of Final Rulemaking
"B" - Notice of Proposed Rulemaking
"C" - Regulatory Analysis
"D" - Task Force Report
"EE - Draft Congressional Letter
"F" - Draft Public Announcement

ENCLOSURE "A"

FINAL AMENDMENT

OF

10 CFR 50.62

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Commission is amending its regulations to require improvements in the design and operation of light-water-cooled nuclear power plants to reduce the likelihood of failure of the reactor protection system to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of anticipated transients without scram (ATWS) events. The final rule requires the installation of certain equipment in nuclear power plants. It also encourages the development of a reliability assurance program for the reactor trip system on a voluntary basis. This will significantly reduce the risk of nuclear power plant operation.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (301) 443-5960.

SUPPLEMENTARY INFORMATION: An anticipated transient without scram (ATWS) is an expected operational transient (such as a loss of feedwater, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor trip system (RTS), a part of the protection system, to shut down the reactor. The reactor trip system consists of those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices that are required to initiate reactor shutdown; this includes circuit breakers, the control rods and control rod mechanisms. That portion of the RTS exclusive of the control rods and control rod mechanisms is here referred to as the scram system. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to severe core damage and release of radioactivity to the environment. The ATWS question involves safe shutdown of the reactor during a transient, if there is a failure of the RTS. There have been precursors to an ATWS; the latest was a failure of the automatic portion of the RTS at the Salem 1 nuclear generating station on February 25, 1983. In that incident, manual shutdown was accomplished after 30 seconds, and no core damage or release of radioactivity occurred.

On November 24, 1981, the Commission invited comments on three alternative proposed rules relating to ATWS (46 FR 57521). Each of the three alternative proposed rules had the objective of reduction of risk from ATWS and each featured a different approach to achieve that objective. One alternative (the staff rule) emphasized individual reactor evaluation to identify needed improvements. The second alternative (the Hendrie rule) emphasized reliability assurance and would have also required certain hardware modifications. The third alternative, proposed by the Utility Group on ATWS in petition for rulemaking PRM 50-29, prescribed specific changes that were keyed to the type of reactor and its manufacturer.

Thirty-nine public comments were received at or close to the April 23, 1982, deadline for submission of comments. An additional comment was received on

June 24, 1982. Copies of the comments may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. The following organizations and individuals provided comments:

1. M. I. Lewis, Philadelphia, Pennsylvania (private citizen)
2. S. L. Hiatt, Mentor, Ohio (private citizen)
3. Washington Public Power Supply System (WPPSS)
4. Standardized Nuclear Unit Power Plant System (SNUPPS)
5. South Carolina Electric and Gas Company (South Carolina)
6. General Electric Company (GE)
7. Duke Power Company (Duke)
8. Atomic Industrial Forum (AIF)
9. Detroit Edison (DE)
10. Mississippi Power and Light Company (MP&L)
11. Texas Utilities Generating Company (TUGC)
12. Commonwealth Edison Company
13. Combustion Engineering, Incorporated (CE)
14. The Utility Group on ATWS, representing 22 Utilities
15. Combustion Engineering Owners Group
16. Houston Lighting and Power (HL&P)
17. Portland General Electric Company (PGEC)
18. GPU Nuclear (GPU)
19. Babcock and Wilcox Company (B&W)
20. Ebasco Services, Incorporated (Ebasco)
21. Public Service Electric and Gas Company (PSE&G)
22. Carolina Power and Light Company (CP&L), first comment
23. Stone and Webster Engineering Corporation (S&W)
24. Florida Power Corporation (FPL)
25. Gulf States Utilities Company (Gulf)
26. Duquesne Light Company
27. Wisconsin Public Service Corporation (WPSC)
28. Pacific Gas and Electric Company (PG&E)
29. Tennessee Valley Authority (TVA)
30. Pennsylvania Power and Light Company (PP&L)
31. Virginia Electric and Power Company (VEPCO)
32. Arkansas Power and Light Company (AP&L)
33. Alabama Power Company (Alabama)
34. Wisconsin Electric Power Company (WEPC)
35. Power Authority of the State of New York (PASNY)
36. Yankee Atomic Electric Company (Yankee)
37. Public Service Company of Indiana (Indiana)
38. Northeast Utilities Service Company (NUSCO)
39. Carolina Power and Light Company (CP&L), second comment
40. American Electric Power Service Corporation (received June 24, 1982)

Following are members of the Utility Group on ATWS, the petitioner in the PRM-50-29.

- Arkansas Power and Light Company
- Boston Edison Company
- Connecticut Yankee Power Company
- The Detroit Edison Company
- Florida Power Corporation
- Gulf States Utilities Company
- Maine Yankee Atomic Power Company
- Northeast Nuclear Energy Company
- Pacific Gas and Electric Company
- Public Service Electric and Gas Co.
- Washington Public Power Supply System
- Baltimore Gas and Electric Company
- Commonwealth Edison Company
- Consumers Power Company
- Duke Power Company
- Florida Power and Light Company
- Long Island Lighting Company
- Nebraska Public Power District
- Omaha Public Power District
- Pennsylvania Power and Light Company
- Vermont Yankee Nuclear Power Corp.

The breakdown by preference among commenters for the three alternative proposed rule approaches is as follows:

<u>Support "Utility Rule" (PRM-50-29)</u>

- WPPSS
- DE
- Commonwealth Edison
- The Utility Group on ATWS
- HL&P
- Ebasco
- PSE&G
- FPL
- Gulf
- PP&L
- Yankee

<u>Support "Hendrie Rule"</u> (Most support for this option is tentative with many reservations)

- South Carolina
- Duquesne
- CP&L, first comment (could also be considered a "No Rule" choice)
- WPSC
- VEPCO
- S&W

<u>Favor No Rule</u>

- SNUPPS
- GE
- Duke
- AIF
- MP&L
- TUGC
- CE
- CE Owners Group
- PGEC
- GPU
- B&W
- PG&E
- AP&L
- Alabama
- WEPC
- Indiana
- CP&L, second comment
- NUSCO
- American Electric

The Staff Rule option was favored by Ms. S. L. Hiatt who commented that it was the most stringent of the three proposals, but that it would be better to return to the implementation of specific hardware changes than to require evaluation models. Commenters TVA and PASNY stated a preference for "Alternative 2A" of NUREG-0460[1], Vol. 4, which is very similar to the Utility rule. The comments from Mr. M. I. Lewis did not favor any of the alternatives, but he pointed out limitations of both NRC proposed rules (limitations of modeling) and felt that the Commission was not fully addressing ATWS.

Most of the utility commenters preferred that the Commission promulgate no rule on ATWS. However, many commenters chose either the Utility Rule or the Hendrie Rule as the more favorable of the alternatives presented (including some commenters within the Utility Group). The No Rule category described above includes those who felt that the risks from ATWS are already sufficiently low, plus those who recommended combining the ATWS rulemaking with other Commission activities such as the Severe Accident Program or the development of a Safety Goal.

The comments provided by the Utility Group on ATWS consisted of a three-volume technical report which includes a review and evaluation of past NRC and industry studies, a generic but substantial probabilistic risk assessment of the issue for each NSSS vendor, and a value-impact analysis of all three proposed rules. Their conclusions are:

[1] Copies of all NUREG reports are available for purchase through the Division of Technical Information and Document Control, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

1. The Staff and Hendrie Rules fail the value-impact test.

2. Only the Utility Rule is consistent with current NRC policies.

3. The record and notice for the Staff and Hendrie Rules are inadequate.

In order to resolve the ATWS rule issue, it was necessary for the NRC staff to evaluate the Utility Group report. This was done by a technical assistance contract.

A report which provided a critique of the Utility Group comments was prepared by Energy Incorporated through Sandia National Laboratories and may be examined at the Commission's Public Document Room (PDR) at 1717 H Street, Washington, D.C. Also, a summary of 39 public comments, as well as a plan to resolve the ATWS rule, is available in SECY-82-275 at the PDR.

As proposed in SECY-82-275 and the Commission briefing on July 13, 1982, a Task Force and Steering Group of NRC personnel from several offices was formed to consider the following alternatives:

1. Promulgation of no ATWS rule or including ATWS under the Severe Accident Program;

2. Adoption of the proposed or a modified version of the Utility Group Rule (PRM-50-29);

3. Adoption of the Staff Rule or a modification of it; or

4. Adoption of those portions of the Hendrie Rule for which there exists a technical basis.

The Commission has given careful consideration to all the comments and is now publishing a final rule. This final rule uses in part the same approach that is used in the Utility Group's petition for rulemaking. Prescribed changes, keyed to the reactor's type and manufacturer, are set out in the final rule. The costs and values of these changes and of other considered changes are discussed in a document on file in the Commission's Public Document Room, entitled "Recommendations of the ATWS Task Force".

Summary of Staff, Hendrie, and Utility Rules

The Staff Rule (46 FR 57521) would have resolved ATWS by establishing performance criteria (e.g., there would be analyses to verify that Service Level C of the ASME Boiler and Pressure vessel code would not be exceeded, fuel integrity would be maintained, there would be no excessive radioactivity release, the containment would not fail, and long-term shutdown and cooling would be assured). The Hendrie Rule (46 FR 57521), while using much of the same information base as the Staff Rule, proposed to resolve ATWS by establishing a reliability assurance program for systems that prevent or mitigate ATWS accidents and prescribing certain hardware modifications which would allow for: (1) automatically tripping recirculation pump of a BWR under conditions indicative of an ATWS, (2) automatically actuating the standby liquid control system (SLCS) for BWRs, (3) providing a reliable scram discharge volume for BWRs, (4) providing for the prompt, automatic initiation of the auxiliary feedwater system for conditions indicative of an ATWS, and (5) assuring that the instruments necessary for the diagnosis of and recovery from ATWS accident sequences will not be disabled. Finally, the Utility Rule proposed specific design modifications for each reactor manufacturer. It contained proposals that: (a) all Westinghouse reactors have initiation of the auxiliary feedwater system and turbine trip diverse from the reactor protection system; (b) all Combustion Engineering and Babcock and Wilcox reactors have diverse initiation of auxiliary feedwater and turbine trip (similar to Westinghouse) and a diverse scram system; and (c) existing boiling water reactors manufactured by General Electric have (1) a means to trip the recirculation pumps upon receipt of a signal indicative of an ATWS, (2) a diverse scram system, and (3) a modification of the scram discharge volume. Also, new (three years after the rule becomes effective) General Electric plants would have a standby liquid control system increased to 86 gpm and all reactor licensees would institute training for operators.

Basis for Final Rule as Promulgated by the Commission

The vast majority of the commenters felt that the approach of the Staff Rule was too open-ended in terms of costs to resolve ATWS (e.g., the analyses could be very costly and time consuming). The Hendrie Rule was found difficult to interpret by most commenters. The ATWS Steering Group opted to evaluate generic plants, in a fashion similar to the Utility Group approach, and define the various fixes and estimate the reduction in probability for ATWS sequences as each additional requirement was added. This would then give a value (reduction in risk) that could be compared to the impact (cost in dollars) of each incremental requirement. There are large uncertaintites in these analyses, and the detailed results of the analyses can be found in the report entitled "Recommendations of the ATWS Task Force" (discussed above). A brief discussion of the final rule's provisions, including value/impact evaluations, is given next:

Diverse and Independent Auxiliary Feedwater Initiation and Turbine Trip for PWRs: Paragraph 50.62 (c)(1)

This was proposed by the Utility Group on ATWS. It consists of equipment to trip the turbine and initiate auxiliary feedwater independent of the reactor trip system. It has the acronym AMSAC, which stands for Auxiliary (or ATWS) Mitigating Systems Actuation Circuitry. It has a highly favorable value/impact for Westinghouse plants and a marginally favorable value/impact for Combustion Engineering and Babcock and Wilcox plants. Since it has the potential for a spurious trip of the reactor which reduces its value/impact, it should be designed to minimize these trips.

Diverse Scram System: Paragraphs 50.62 (c)(2) and (c)(3)

This was proposed by the Utility Group on ATWS for General Electric, Combustion Engineering, and Babcock and Wilcox plants. It has a favorable value/impact from the Staff's analysis. However, the principal reasons for requiring the feature are to assure emphasis on accident prevention and to obtain the resultant decrease in potential common cause failure paths in the trip system. It also has the potential for a spurious trip of the reactor; therefore it should be designed to minimize spurious trips. For General Electric plants, installation may extend by one or two days the downtime during a refueling outage.

A diverse scram system for Westinghouse plants was not a recommendation of the Utility Group on ATWS and was not a clear requirement of the Staff Rule or the Hendrie Rule, although the Utility Group on ATWS interpreted the Staff Rule to include it. The system does, however, have a marginally favorable value/impact for Westinghouse plants. To assure full opportunity for public comment, the requirement for a diverse scram system for Westinghouse plants will be published separately as a proposed rule.

Increased Standby Liquid Control System (SLCS): Paragraph 50.62 (c)(4)

The SLCS is a system for injecting borated water into the reactor primary coolant system. The neutron absorbtion by the boron causes shutdown of the reactor. Addition of this system was proposed by the Utility Group on ATWS for new plants (those receiving an operating license three years after the effective date of the final rule). The Commission believes that, with the use of the Emergency Procedure Guidelines proposed by the BWR Owners Group and General Electric that are being implemented at operating BWRs, increasing the SLCS capacity for operating plants may insure an intact containment for isolation transients, although there is uncertainty in containment failure modes. Because of the vulnerability of BWR containments to ATWS sequences, the Commission has determined that this enhanced mitigation feature is warranted. The high pressure portion of the ECCS of BWR/5 and BWR/6 licensees (HPSC) is injected into spray spargers in the core exit plenum. For these plants, the preferred location for the injection of the borated water from the SLCS is the HPCS line just external to the reactor vessel instead of the standpipe at the core inlet plenum. A similar location is preferred for those BWR/4 licensees with HPCI injection into spargers in the core exit plenum. This injection location provides significant improvement in mixing of the borated water, particularly under low vessel water level conditions such as encountered when the EPGs are followed. This injection location is also preferred, since it could prevent local power increases and possible power excursions during the recovery phase of an ATWS when cold unborated ECCS water could be added above the core. Some BWR/5 and BWR/6 licensees already have this injection location and have designed the SLCS accordingly.

Automatic Recirculation Pump Trip for BWRs: Paragraphs 50.62 (c)(5)

Recirculation pump trip (RPT) was proposed as a rule requirement by the Utility Group on ATWS. This safety feature will result in a reduction of reactor power from 100 percent to about 30 percent following a transient (and failure to scram) within a minute or so. This proposed requirement has already been implemented on all operational BWRs in response to a show cause order dated February 21, 1980. The BWR owners generally agree that this is a necessary requirement, and it is being included in the final rule for completeness.

Automatic Initiation of Standby Liquid Control System

One of the alternatives considered by the Task Force was an automatically initiated standby liquid control system with a capacity of greater than 86 gpm (such as 150-200 gpm). This would have resulted in a considerable risk reduction (about a factor of seven) after the ARI is installed for operating plants. Unfortunately, the cost to do this (based on information supplied by the Utility Group on ATWS) is on the order of $24 million per plant and is significantly impacted by the costs of downtime from an inadvertent trip which would inject boron into the reactor water and by the costs of downtime for installation in existing plants. The value/impact does not favor this alternative for existing plants.

New plants (those which will receive construction permits after the effective date of this rule) will be required to have equipment for automatic initiation of the SLCS. Most of those plants already have been designed for this feature. Also, other plants that have been designed and built to include this feature must utilize the feature. The equipment for automatic SLCS actuation should be designed to perform its function in a reliable manner and to provide high reliability against spurious actuation.

Adding Extra Safety Valves or Burnable Poisons

One of the alternatives considered by the Task Force was adding more safety valves to plants manufactured by Combustion Engineering (CE) and Babcock and Wilcox (B&W). This would reduce the peak pressure in the reactor vessel and yield a higher probability of the plant surviving an ATWS with no core damage. The peak overpressure could also be reduced by modifying the core behavior (the fraction of the time the moderator temperature coefficient is unfavorable) by adding burnable poisons. The Utility Group on ATWS estimated that installing larger valve capacity could cost up to $10 million per plant. A large fraction of this cost is the downtime for installation of the valves. While the probability of ATWS can be reduced about a factor of three or more, the value/impact is unfavorable for this alternative for existing plants. These plants all have large dry containments and will be most able to mitigate the radiological consequences from an ATWS. This rule does not cover enhanced pressure relief capacity for new CE and B&W plants. However, the Commission expects that this issue would be addressed during the NRC's design review of any specific new plant or standard plant application.

Need for All Control Rods to be Inserted for PWRs

By using soluble boron for burnup and xenon control, PWRs normally operate at or near 100 percent power with control rods nearly out (except for some Babcock and Wilcox "rodded" reactors which keep one bank inserted for xenon control). Thus, nearly all rods are available to participate in a scram.

Insertion of only about 20 percent (approximately 10) of the control rods is needed to achieve hot, zero power provided that the inserted rods are suitably uniformly distributed. What is important is the uniform spacing of the rods. In installing a diverse scram system, the licensee can allow for partial scram failures if it is demonstrated that the rod insertion pattern is sufficiently uniformly spaced such that a hot, zero power is achieved.

Considerations Regarding Reliability Assurance

As a result of the failure of the Salem Unit 1 reactor to scram automatically on February 25, 1983, the NRC conducted an investigation of the events (see NUREG-0977, "NRC Fact-Finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station, Unit 1, on February 25, 1983"[1]). One of the principal findings was the lack of adequate attention being paid to the reliability of the reactor trip system. The Salem Generic Issues Task Force recommended to the Commission that a reliability assurance program be included in the final ATWS rule (NUREG-1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant"[1]). While this rule does not require such a program, the Commission urges the voluntary development of one.

The reliability assurance program should have the following elements:

1. An analysis of the failure modes of the RTS system, considering independent failures quantitatively and common cause failures qualitatively. An estimate of the reliability of the RTS.

2. A numerical performance standard for the RTS unavailability to use as an aid in the initial and continuing evaluation of the adequacy of the system.

3. A process for evaluating plant-specific and industry-wide operating experience to provide feedback to assess whether the RTS is performing reliably enough.

4. Procedures to augment existing quality assurance programs to ensure that quality assurance is being implemented properly.

A pivotal aspect of the ATWS issue is the reliability of the reactor trip system (RTS), including the control rods, and the difficulty associated with assessing the impact of common cause failures on the availability of the system to function when required. All RTS systems are designed for high availability, yet ATWS precursors at Kahl and Browns Ferry 3, and the ATWS event at Salem 1 did occur and were the result of common cause failures of the RTS. The Kahl and Browns Ferry 3 incidents were described in the Federal Register notice containing the proposed rules which was published on November 24, 1981 (46 FR 57521). The Salem 1 incident occurred after the proposed rules were published.

An analysis of the RTS should be performed using existing methodologies for quantitative evaluation of system reliability (e.g., unavailability). A fault tree and qualitative common cause failure analysis should be performed to identify the potential important faults of the RTS. Examples of quantitative analysis for the RTS are: WASH-1400 (the Reactor Safety Study)[2], the Indian Point Probabilistic Safety Study[3], the Zion Probabilistic Safety Study[3], and other probabilistic safety studies performed by Industry at their own initiative or at the request of the Commission. There are an estimated 15-20 probabilistic studies of plants that have been performed or are being performed, although some of these do not include detailed RTS analyses.

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[2]Microfiche copies are available for purchase from the Division of Technical Information and Documentation Control, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555.

[3]These may be examined at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland.

Additional methodological guidance is given in the PRA Procedures Guide, NUREG/CR-2300, January 1983. This Guide was developed jointly by the Commission, the American Nuclear Society and the Institute of Electrical and Electronic Engineers.

Each licensee should establish a goal or benchmark to assess the performance of the trip system. The Commission and the Industry have had considerable disagreement about the "correct" or "appropriate" value of RTS unavailability. It would be more fruitful for each licensee to have a benchmark for comparison as the plant operates and generates new data. The treatment of common cause failures will be analyzed in a qualitative fashion to determine if there are any significant failure modes previously unidentified. The cost of doing this can be minimized by forming or using existing Owners Groups, since there is much commonality in RTS designs.

Each licensee, as part of the RTS unavailability analysis, should examine its maintenance, surveillance, and testing requirements. The testing frequency would be examined to determine if testing is done too often or not often enough. The type of testing, e.g., completeness and sequencing of component verification for operability, would be thoroughly reviewed. The nature and frequency of maintenance, e.g., lubrication, cleaning, calibration, dimensional verification, physical movement, would be reviewed. Recordkeeping procedures should be reviewed.

The Commission believes that a reliability assurance program for the reactor trip systems should be developed and implemented, with clear objective of providing additional assurance that the desired high reliability of the RTS is indeed achieved and maintained. Operating experience in the United States appears to demonstrate, in some instances, that implementation of Appendix A (particularly General Design Criterion 21) and Appendix B to 10 CFR 50, and other NRC regulatory requirements may not have yielded the degree of reliability that is possible to achieve with available technology in a cost effective manner. One reason for this failure might be that a reliability standard has not been quantitatively set. Another reason might be a failure to understand fully the dominant role played by common cause failures.

The techniques for a reliability assurance program are in existence. They have been applied in an orderly, structured fashion in defense and aerospace applications since at least the 1960s. However, details of its application to a commercial nuclear power plant have not been worked out. Therefore, it is strongly recommended that the development of a voluntary reliability assurance program, focusing on the reactor trip system, be performed jointly by the NRC and Industry, appropriately coordinated with INPO, EPRI, and the various owners groups. If this program is not voluntarily implemented in an effective manner within the next two years, the Commission will reconsider the question of rule-making in this area.

The development of industry programs on a voluntary basis has precedence in the evaluation of operating data for commercial nuclear power plants. The industry has developed the Nuclear Plant Reliability Data (NPRD) System as a voluntary program for the reporting of reliability data. The success of this program has yet to be demonstrated. The NRC previously evaluated that system and published an Advanced Notice of Proposed Rulemaking inviting public comments on a plan to make that program mandatory.

Considerations Regarding System and Equipment Criteria

The additional equipment required by this amendment to implement diversity for auxiliary feedwater system initiation, turbine trip, recirculation pump trip, and reactor trip, while required to be reasonably reliable, will not have to meet all of the stringent requirements normally applied to safety related equipment. The equipment required by this amendment is for the purpose of reducing the probability of unacceptable consequences following anticipated operational occurrences. Since the combination of an anticipated operational occurrence, failure of the existing reactor trip system, and a seismic event or an event which results in significant plant physical damage has a low probability, seismic qualification and physical separation criteria need not be applied to the equipment required by this rule. In view of the redundancy provided in existing Reactor Trip Systems, the equipment required by this amendment does not have to be redundant within itself. Of course, the diverse system should be designed in such a way to reduce the likelihood of spurious trips; e.g., use of a two-out-of-two logic system for the diverse trip signal would be acceptable.

The amendment is to require diversity to those portions of existing Reactor Trip Systems where only minimal diversity is currently provided. The logic circuits and actuation devices (e.g., circuit breakers on Pressurized Water Reactors) in existing Reactor Trip Systems utilize redundant, but in general identical, components and, thus, are subject to potential common cause failures. Existing Reactor Trip Systems, however, measure a variety of plant parameters and utilize a variety of sensor types. Common cause failures in the diverse sensors of existing Reactor Trip Systems are considered sufficiently unlikely that additional sensor diversity is not necessary. Even though sensor diversity is not necessary, sensors in the existing Reactor Trip System should not be used to provide the signals for the diverse equipment required by this amendment. Use of the same sensor for the existing Reactor Trip System and the diverse equipment would result in a design where a single electrical failure of the shared sensor or single electrical failure in the signal conditioning equipment of the shared sensor could simultaneously prevent operation of one redundant channel in the existing Reactor Trip System and the diverse equipment. In addition, this could increase the potential for common cause failures affecting both systems. This is considered unacceptable.

The equipment required by this amendment must be implemented such that it does not degrade the existing protection system. This is to be accomplished by making the diverse equipment electrically independent from the existing protection system and by insuring that the existing protection system will continue to meet all applicable safety related criteria after installation of the diverse equipment.

Finally, even though the diverse systems are not deemed to be safety related, they will be subject to the requirements of Appendix B to 10 CFR Part 50. Quality Assurance is generally recognized as a disciplined method of insuring that equipment will perform satisfactorily in service. See the following table for a more detailed itemization of guidance regarding the system specifications that the Staff would find acceptable for the diverse trip and mitigating systems.

GUIDANCE REGARDING SYSTEM AND EQUIPMENT SPECIFICATIONS

Guidance \ System	Diverse Reactor Trip System	Mitigating Systems (Recirculation Pump Trip and Automatic SLCS actuation for BWRs: Auxiliary Feedwater Actuation and Turbine Trip for PWRs)
Safety Related (IEEE-279)	*Not required, but the implementation must be such that the existing protection system continues to meet all applicable safety related criteria	*Not required, but the implementation must be such that the existing protection system continues to meet all applicable safety related criteria
Redundancy	Not Required	Not Required
Diversity from existing Reactor Trip System	Equipment diversity is required from the sensors to and including the components used to interrupt control rod power or vent the scram air header. Circuit breakers from different manufacturers alone is not sufficient to provide the required diversity for interruption of control rod power. The sensors need not be of a diverse design or manufacturer, but must be electrically independent from those in the existing protection system. Existing protection system instrument sensing lines may be used.	Equipment diversity is required from the sensors to, but not including, the final actuation device--e.g., existing circuit breakers may be used for auxiliary feedwater initiation. The sensors need not be of a diverse design or manufacturer, but must be electrically independent from those in the existing protection system. Existing protection system instrument sensing lines may be used.

*Although equipment provided in this amendment is not required to be safety related, this equipment is in the broader class of structures, systems, and components important to safety defined in 10 CFR Part 50, Appendix A (General Design Criteria), "Introduction". GDC-1 requires that "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed".

Electrical Independence from existing protection system	Required up to the final actuation device at which point non-safety related circuits must be isolated from safety related circuits.	Required up to final actuation device at which point non-safety related circuits must be isolated from safety related circuits.
Physical Separation from existing Reactor Trip System	Not required, unless redundant divisions and channels in the existing Reactor Trip System are not physically separated. The implementation must be such that separation criteria applied to the existing protection system are not violated.	Not required, unless redundant divisions and channels in the existing Reactor Trip System are not physically separated. The implementation must be such that separation criteria applied to the existing protection system are not violated.
Environmental Qualification	For anticipated operational occurrences only, not for accidents	For anticipated operational occurrences only, not for accidents
Seismic Qualification	Not required	Not required
Quality Assurance	Appendix B to 10 CFR Part 50	Appendix B to 10 CFR Part 50
Safety Related (1E) Power Supply	Not required, but must be capable of performing safety functions with loss of offsite power. Sensor, channel, logic, and actuation device power must be from an instrument power supply independent from the power supplies for the existing Reactor Trip System.	Not required, but must be capable of performing safety functions with loss of offsite power. Sensor, channel, and logic power must be from an instrument power supply independent from the power supplies for the existing Reactor Trip System.
Testability at Power	Required	Required
Inadvertent Actuation	The design should be such that the frequency of inadvertent reactor trips is minimized.	The design should be such that the frequency of inadvertent actuations is minimized.

Exemptions

Some of the older operating nuclear power plants (e.g., those licensed to operate prior to August 22, 1969) may be granted an exemption from these amendments, if they can demonstrate that their risk from ATWS is sufficiently low. Factors important to this demonstration could be power level, unique design features that could prevent or mitigate the consequences of an ATWS, remaining plant lifetime, or remote siting.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. Single copies of the analysis may be obtained from David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 443-5960.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et. req.). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that the rule will not have a significant economic impact on a substantial number of small entities. This rule affects only licensees that own and operate nuclear utilitzation facilities licensed under sections 103 and 104 of the Atomic Energy Act of 1954, as amended. These licensees do not fall within the definition of small businesses set forth in section 3 of the Small Business Act, 15 U.S.C. 632, or within the Small Business Size Standards set forth in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 552 and 553 of Title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The Authority citation for Part 50 continues to read as follows:

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c) 50.44, 50.46, 50.48, 50.54 and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. A new §50.62 is added to read as follows:

§50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) Applicability. The requirements of this section apply to all commercial light-water-cooled nuclear power plants

(b) Definition. For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) Requirements.

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

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensors to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse from the reactor trip system. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after [insert the effective date of this rule], and for plants granted a construction permit prior to [insert the effective date of this rule] that have already been designed and built to include this feature.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Director, Office of Nuclear Reactor Regulation.

(d) Implementation. By [insert a date 6 months after the effective date of the amendment] each licensee shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each licensee shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after the effective date of the amendment, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

Dated at ____________________ this day of ______________, 1983

For the Nuclear Regulatory Commission.

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Samuel J. Chilk
Secretary of the Commission

ENCLOSURE "B"

PROPOSED AMENDMENT

OF

10 CFR 50.62

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Additional Scram System Requirement for Westinghouse Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Commission is considering amending its regulations to require an improvement in the design of light-water-cooled nuclear power plants manufactured by Westinghouse. The proposed amendment would require a diverse scram system from sensor output to interruption of power to the control rods. This system would reduce the likelihood of an accident if the existing reactor protection system fails to shut down the reactor following an anticipated transient.

The proposed requirement for Westinghouse reactors is the same requirement that is imposed on Combustion Engineering and Babcock and Wilcox reactors by a final rule that is being published concurrently with this proposed rule. That final rule also should be reviewed, since it contains important supporting information that is not repeated here.

DATES: Comment period expires ( ). Comments received after this date will be considered if practical to do so, but only those comments received on or before this date can be assured of consideration.

ADDRESSES: Comments should be submitted in writing to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. All comments received will be available for public inspection in the Commission's Public Document Room at 1717 H. Street NW., Washington, D.C.

FOR FURTHER INFORMATION CONTACT: David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (301) 443-5960.

BACKGROUND: On November 24, 1981, the Commission invited comments on three alternative proposed rules relating to backup systems and procedures to shut down (scram) the reactor if there is a failure of the reactor protection system.

One of the requirements in the proposed rules concerned an additional diverse scram system for pressurized water reactors (PWRs) designed by Combustion Engineering and Babcock & Wilcox. The proposed rules would not have required an additional diverse scram system for PWRs manufactured by Westinghouse, because the mitigative measures of diverse turbine trip and automatic auxiliary feedwater actuation for this design were believed to be sufficient to reduce the risk of an Anticipated Transient Without Scram (ATWS), principally based on the value/impact analysis.

Recent experience at a Westinghouse PWR, reported in NUREG-1000, Vol. 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant,"[1] has caused the Commission to reconsider the need for a diverse scram system for Westinghouse PWRs. The Salem events increased the Commission's concern over potential, unidentified common-cause failure modes.

A requirement for a diverse reactor scram system is now proposed for Westinghouse reactors as a means of improving scram system reliability. The scram system is challenged on the average of ten times per year; hence, there is a need for high reliability, as defined in GDC 21, Appendix A, 10 CFR 50. Highly reliable accident prevention <u>and</u> accident mitigation systems are

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[1]Copies of this report are available for purchase from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

important to safety. To require both preventive and mitigative systems is fully consistent with the safety philosophy of defense in depth.

While the value/impact analysis indicates the diversity requirement is reasonable for Westinghouse PWRs, it is recognized that there are substantial uncertainties in both the risk and cost factors; and, therefore, value/impact analyses of this type should only be one consideration in any final decision with regard to this issue. The principal factors favoring this decision are: (1) the scram system is a very important safety system; (2) this system must function with a high degree of reliability; (3) experience has shown that previously unidentified common-cause failure paths have been the dominant contributor to estimates of scram system unavailability; (4) diversity is an important way to reduce the likelihood of occurrence of other significant common-cause failure paths presently not identified; (5) the costs of adding diversity to the Westinghouse scram system should be relatively small, considering that this added system could be incorporated into the modifications already required for diverse turbine trip and automatic auxiliary feedwater actuation; and (6) the value/impact analysis appears to be favorable, and it certainly does not indicate that such a modification should not be required.

Concurrently with publication of this proposed rule, the Commission is publishing a final rule which requires a diverse scram system for Combustion Engineering and Babcock and Wilcox PWRs. The following proposed amendment proposes that the final rule's requirement for a diverse scram system also be applied to Westinghouse PWRs. Discussion of how the Commission intends to implement the diversity requirement is set out in the Preamble for the final rule.

Regulatory Analysis

The Commission has prepared a regulatory analysis for the final rule and for this proposed amendment to § 50.62. The analysis examines the costs and benefits of the alternatives and the decision criteria considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.

Paperwork Reduction Act Statement

The proposed rule would establish information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et. seq.). These requirements were approved by the Office of Management and Budget: Approval No. 3150-0011.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, U.S.C. 605(b), the Commission hereby certifies that this proposed regulation will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed regulation affects licensees that own and operate nuclear utilization facilities licensed under sections 103 and 104 of the Atomic Energy Act of 1954, as amended. These licensees do not fall within the definition of small businesses set forth in section 3 of the Small Business Act, 15 U.S.C. 632, or within the Small Business Size Standards set forth in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of Title 5 of the United States Code, notice is hereby given that adoption of the following amendment to 10 CFR Part 50 is contemplated.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The Authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 22[illegible]4). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§50.10(a), (b), and (c) 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. Paragraph (c)(2) of §50.62 is revised to read as follows:

§50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

* * * * *

(c) ***

(2) Each pressurized water reactor must have a diverse scram system from the sensors to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

Dated at ____________ this ___________ day of ___________, 1983.

For the Nuclear Regulatory Commission.

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Samuel J. Chilk
Secretary of the Commission

ENCLOSURE "C"

REGULATORY ANALYSIS

FOR

AMENDMENTS RELATED TO ANTICIPATED

TRANSIENTS WITHOUT SCRAM

REGULATORY ANALYSIS
FOR
AMENDMENTS RELATED TO ANTICIPATED
TRANSIENTS WITHOUT SCRAM (ATWS)

1. STATEMENT OF THE PROBLEM

1.1 Background

An anticipated transient without scram (ATWS) is an expected operational transient (such as a loss of feedwater, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor protection system to shut down the reactor. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to core melt and to the release of radioactivity to the environment. The ATWS issue involves backup safety systems and procedures for shutting down the reactor, if there is a failure of the reactor protection system. The issue of ATWS has been under consideration since the late 1960s. Precursors have occurred at the Kahl reactor (1963), the Monticello reactor prior to startup, and at the Browns Ferry Unit 3 reactor (1980). An ATWS event did occur at the Salem 1 nuclear generating station on February 25, 1983, although no core damage or release of radioactivity occurred. The reactor was scrammed manually 30 seconds after the automatic trip system failed.

On November 24, 1981, the Commission invited comments on three alternative proposed rules relating to ATWS (46 FR 57521). Each of the three alternative proposed rules had the objective of reduction of risk from ATWS, and each featured a different approach to achieving that objective. One alternative emphasized individual reactor evaluation to identify needed improvements (the Staff Rule). The second alternative emphasized reliability assurance and would have also required certain hardware modifications (the Hendrie Rule). The third alternative, proposed by the Utility Group on ATWS in petition for rulemaking PRM-50-29, prescribed specific changes that were keyed to the type of reactor and its manufacturer.

The Staff Rule (46 FR 57521) would have resolved ATWS by establishing performance criteria (e.g., there would be analyses to verify that Service Level C of the ASME Boiler and Pressure vessel code would not be exceeded, fuel integrity would be maintained, there would be no excessive radioactivity release, the containment would not fail, and long-term shutdown and cooling would be assured). The Hendrie Rule (46 FR 57521), while using much of the same information base as the Staff Rule, proposed to resolve ATWS by establishing a reliability assurance program for systems that prevent or mitigate ATWS accidents and to prescribe certain hardware modifications which would allow for: (1) automatically tripping the recirculation pump of a BWR under conditions indicative of an ATWS, (2) automatically actuating the standby liquid control system (SLCS) for BWRs, (3) providing a reliable scram discharge volume for BWRs, (4) providing for the prompt, automatic initiation of the auxiliary feedwater system for conditions indicative of an ATWS, and (5) assuring that the instruments necessary for the diagnosis of and recovery from ATWS accident sequences will not be disabled.

The Utility Group on ATWS Rule (PRM-50-29) specified equipment modifications by reactor manufacturer. It contained proposals that: (a) all Westinghouse reactors have initiation of the auxiliary feedwater system and turbine trip diverse from the reactor protection system; (b) all Combustion Engineering and Babcock and Wilcox reactors have diverse initiation of auxiliary feedwater and turbine trip (similar to Westinghouse) and a diverse scram system; and (c) existing boiling water reactors manufactured by General Electric have (1) a means to trip the recirculation pumps upon receipt of a signal indicative of an ATWS, (2) a diverse scram system, and (3) a modification of the scram discharge volume. Also, new (within three years after the rule became effective) General Electric plants would have a standby liquid control system increased to 86 gpm and all reactor licensees would institute training for operators.

Forty public comments were received on the three alternative proposed rules (46 FR 57521). A Task Force and Steering Group of NRC staff from the Offices of Inspection and Enforcement, Nuclear Reactor Regulation, and Nuclear Regulatory Research was established to evaluate the components and recent related information and to develop a single recommendation for an ATWS rule.

1.2 Description of Rulemaking

A § 50.62 is being added to 10 CFR Part 50 to reduce risk from ATWS. The rule change would apply to most nuclear power plants. Older plants may apply for an exemption from this amendment. The rule would require that:

a. Each pressurized water reactor must have equipment, that is diverse and independent from the reactor protection system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS.

b. Each pressurized water reactor* must have a diverse scram system from sensor output to interruption of power to the control rods. This scram system must be reliable and independent from the reactor protection system.

c. Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse from the reactor protection system.

d. Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 percent sodium pentaborate solution. The SLCS must be automatically initiated under conditions indicative of an ATWS for new reactors seeking a construction permit as well as for other reactors that have already been designed and built to provide for automatic initiation.

*This requirement for Westinghouse reactors is the subject of a separate proposed rulemaking because the three alternative proposed rules which were previously published (46 FR 57521) did not give clear notice that this requirement was being considered for Westinghouse reactors.

e. Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS.

The rule will require submittal of proposed designs to the Commission for approval.

2. OBJECTIVES

The objective of the rule is to provide requirements to address what was previously an unresolved safety issue; was the subject of three proposed rules; and which, as a result of the Salem 1 incident on February 25, 1983, was again shown to be a significant safety problem that requires hardware and procedural modifications at existing nuclear power plants.

3. ALTERNATIVES

In Section 1 three proposed approaches for ATWS rulemaking (46 FR 57521) were identified. As proposed in SECY-82-275 and the Commission briefing on July 13, 1982, a Task Force and Steering Group of NRC personnel from several offices was formed to consider the following alternatives:

1. Promulgation of no ATWS rule or including ATWS under the Severe Accident Research Program;

2. Adoption of the proposed or a modified version of the Utility Group Rule (PRM-50-29);

3. Adoption of the Staff Rule or a modification of it; or

4. Adoption of those protions of the Hendrie Rule for which there exists a technical basis.

These alternatives are discussed below.

Alternative 1 - No ATWS rule (or include under the Severe Accident Research Program)

The Task Force concluded that the risk from ATWS is already unacceptably high and that a rule should be implemented before the results of the Severe Accident Research Program are available.

Alternative 2 - Adopt the proposed or a modified version of the Utility Group rule

The Task Force evaluated generic plants in a manner similar to the Utility Group on a value/impact basis. The recommendations of the Task Force are close to the Utility Group recommendations. However, the Task Force evaluation showed that additional risk reduction is warranted. The additional recommended ATWS requirements were to increase the standby liquid control system (SLCS) capacity for existing as well as for new plants of the boiling water reactor design, to automate SLCS for new plants, and to add a diverse scram system (DSS) for Westinghouse plants.

Alternative 3 - Adopt the Staff rule or a modification of it

The Task Force concluded that while this Alternative may yield the largest risk reduction, the program requirements needed to be much better defined. Moreover, it was concluded that a rule which involves open-ended evaluation model analyses as opposed to prescriptive requirements would result in a lengthier resolution of this issue and ultimately risk may not be reduced.

Alternative 4 - Adopt those portions of the Hendrie rule for which there exists a technical basis

The Task Force favors the implementation of a reliability assurance program as a matter of principal. However, the content and implementation of such a concept is still in the early stages of development for nuclear power reactor safety application. The Task Force has recommended that safety research be conducted, with cooperation from the nuclear industry, to develop this alternative reactor safety concept in a timely manner.

The value of each alternative is based on the estimated reduction in annual core damage frequency and the resultant release of radioactive fission products to the environment for each of the generic reactor types. Estimates of core damage frequency for the various fixes are based on event tree analyses, and the results are presented in a separate report, "Recommendations of the ATWS Task Force". This report should be a part of this Regulatory Analysis.

4. CONSEQUENCES (COSTS)

4.1 Plants Affected

There are currently 72 reactors licensed to operate, including one HTGR.* The breakdown by reactor manufacturer is Westinghouse: 31 reactors, General Electric: 24 reactors, Combustion Engineering: 8 reactors, Babcock and Wilcox: 8 reactors. There are an additional 60 reactors under construction ("World List of Nuclear Power Plants," Nuclear News, February 1983). The breakdown by reactor manufacturers is Westinghouse: 27 reactors, General Electric: 17 reactors, Combustion Engineering: 11 reactors, Babcock and Wilcox: 5 reactors.

*NUREG-0020, "Licensed Operating Reactors," Vol. 6, No. 9

4.2 Costs of ATWS Hardware Installation

The costs to implement hardware modifications were provided by the Utility Group on ATWS and were done on a reactor-manufacturer basis. The cost to design, construct, and install the equipment is approximately 30 percent of the total cost, with the remaining 70 percent of the cost being due to downtime for installation and spurious trips of the newly installed equipment. The Utility Group used an average cost of $500,000 per day for replacement power. If spurious trips indeed do occur, the total cost for new plants will be close to the cost for backfits with the exception of BWRs. For the hardware modifications in the rule, the cost breakdown by generic plant is provided below.

Westinghouse

a. Diverse Auxiliary Feedwater Automatic Actuation and Turbine Trip (AMSAC)*

- Hardware, engineering, installation	$1.0 Million
- Operation and maintenance	$0.8M
- Inadvertent trip (2 days downtime)	$1.0M
Subtotal	$2.8M

b. Diverse Scram System

- Hardware, engineering, installation (if installed with AMSAC)	$1.0M
Total Westinghouse	$3.8M

*Based on conversations with the Utility Group on ATWS, many Westinghouse PWRs have already installed AMSAC (ATWS Mitigating System Actuation Circuitry).

CE/B&W

a. and b. Diverse Scram System and Diverse Turbine Trip and Auxiliary Feedwater Initiation

-	Hardware, engineering, installation	$2.0 Million
-	Operation and maintenance	$0.5M
-	Inadvertent trip (total 6 days downtime)	$3.0M
	Total CE/B&W	$5.5M

General Electric

c. Installing ARI

-	Hardware, engineering, installation	$0.86 Million
-	Inadvertent trip (2 days downtime)	$1.0M
-	Allowance for funds used during construction (AFUDC)	$0.4M
-	Downtime for installation (2 days)	$1.0M
	Subtotal	$3.3M

d. Increase Standby Liquid Control System Capacity

-	Total cost	$0.2M

Automatic SLCS

-	Hardware, engineering, installation	$3.35 Million
-	Downtime for installation (operating plants only)	$10.0M

- Inadvertent Trip (10 days downtime)	$5.0M
- AFUDC, Operation and Maintenance	$4.2M
Subtotal	$22.5M
Total GE operating plants	$3.5M
Total GE new plants	$16.0M

The total industry impact would be as follows:

58 Westinghouse reactors operating* and under construction

58 x $3.8M = $220M

41 General Electric reactors operating and under construction**

Operating Reactors - 24 x 3.5M = $84M

Reactors Under Construction - 17 x $2.3M = $39M

19 Combustion Engineering reactors operating and under construction

19 x $5.5M = $110M

13 Babcock and Wilcox reactors operating and under construction

13 x $5.5M = $72M

*Many operating Westinghouse plants have installed automatic turbine trip and auxiliary feedwater initiation. The numbers presented here for cost are conservatively high.

**These costs do not include the impact of an automatic initiation requirement for the SLCS and the resultant downtime for cleanup, since this requirement for automatic SLCS would apply to only a few plants under construction.

The total cost of the ATWS modifications is estimated to be $525M for 131 plants. A perspective on this cost can be achieved in comparison with the total revenue generating capacity. Using a gross assumption of $500,000 per day for 200 days per year for 131 plants over an estimated 30 years brings the total revenue generation up to approximately 400 billion dollars. The ATWS modifications thus would represent approximately 0.1 of one percent of the estimated revenue generation of these plants.

4.3 Sources of Cost Estimates

Most of the estimates for hardware modifications were based on "Comments on the Utility Group on ATWS," submitted as a comment on the three alternative proposed rules (45 FR 57521), dated April 23, 1982. Additional costs were based on a discussion on April 1, 1983 at a public meeting with the Utility Group.

4.4 Occupational Exposures

There should be no undue occupation exposures (virtually none) for installing the modifications required by this rule.

5. DECISION RATIONALE

As part of the decision process, a value-impact analysis was performed and was done on a reactor manufacturer generic basis. The annual frequency of an ATWS leading to unacceptable plant conditions was estimated for various options. Also computed were the costs of implementing these options and the estimated value due to the reduction of ATWS frequency. The details of the risk reduction and of incremental value impact analyses are provided in a separate report, "Recommendations of the ATWS Task Force". The pertinent results of the value/impact analyses are provided in Tables 1, 2 and 3.

The bases for each specific hardware modifications recommended in this rule are provided below:

5.1 PWR Diverse and Independent Auxiliary Feedwater Initiation

The Utility Group rule proposes this for all PWRs. The value/impact analysis supports this requirement.

5.2 Diverse Scram System

The Utility Group rule proposes this for GE, CE and B&W. The staff value/impact supports a diverse scram system for all PWRs. With regard to Westinghouse plants, value/impact is not strongly favorable. The decision to require diversity for Westinghouse plants rests largely on arguments other than value/impact. The principal arguments are threefold:

(1) The RTS is an extremely important safety system and should be highly reliable, as required by General Design Criterion (GDC) 21 of Appendix A of 10 CFR Part 50.

(2) Experience has shown that by far the largest contribution to unreliability of the RTS is common cause failures. Even though one could argue that the Salem problem has been fixed, and that trip breaker and RTS reliability should be at the level which was estimated pre-Salem, the staff is not convinced that such is the case or that there are no other common cause failure modes

that could dominate the unavailability of the RTS. The staff's best estimates of RTS unavailability (without considering statistical variance) is in the range of $3x10^{-5}$ to $2x10^{-4}$. Diversity of the RTS will go a long way towards reducing the impact of other common cause failure modes that have not yet been experienced.

(3) There must be a balance maintained between accident prevention and mitigation (this is consistent with the concept of defense in depth), and prevention must not be short changed. Considering this point, the staff sees little justification for permitting the reliability of the Westinghouse RTS to be demonstrably (by experience and analysis) less reliable than other systems.

5.3 Increased Flow Capacity for BWR SLCS

By using the emergency procedure guidelines (EPGs) as developed by the BWR Owner's Group, the increased effective capacity of the SLCS allows a reasonable chance of mitigating an ATWS in which the reactor is isolated by MSIV closure from the condenser. The capacity of SLCS of currently operating reactors is inadequate to reduce reactor power level during an isolation transient ATWS prior to reaching unacceptable plant conditions (e.g., exceeding 200^{o}F limit in the suppression pool). With the increased SLCS, if initiated manually within 2 to 5 minutes of an ATWS initiation with reactor isolation, the peak suppression pool temperature can be kept in the range of of about 200^{o}F-220^{o}F. However, the staff believes that slightly exceeding this pool temperature limit would probably not cause containment failure. In addition, the staff does not wish to rely wholly on the EPGs. The increased SLCS capacity allows more opportunity for successful mitigation if the operator does not follow the EPGs for non isolation reactor transients. If the cost is on the order of $1 million or less, (it could be as low as $200,000 per plant), which is the preliminary Utility Group estimate, this is an effective risk reduction option.

5.4 Automatic BWR SLCS

Automatic SLCS would appear to be reasonable for new plants from a value/impact standpoint, particularly if one considers the apparent vulnerability of BWRs to ATWS sequences potentially having severe consequences. The staff believes that this system could be designed to greatly minimize the likelihood of spurious actuation, which is by far the biggest contributor to the cost estimations for new plants.

5.5 Recirculation Pump Trip

This is already required for existing plants via an order and is included in the amendment for completeness. It very quickly reduces the reactor power to approximately 30 percent of steady state power.

6. IMPLEMENTATION

The requirements of this amendment would be implemented within approximately two years after the effective date of the rule. Actual schedules will be agreed on by the Commission and each licensee.

TABLE 1

GENERAL ELECTRIC GENERIC OPTIONS

Generic Option	Description	P_{ATWS}/yr	Value (Millions)	Impact (Millions)	V/I
0. Current Commitment	Recirculation Pump Trip (RPT) Scram Discharge Modifications (SDV) Emergency Procedure Guidelines (EPG) L_1 MSIV Closure	5.3×10^{-5}	--	--	--
1. Rule for Operating, CP Plants	Diverse Scram System (ARI) Augment SLCS (86 gpm of equivalent, manual operation)	1.2×10^{-5}	$12.3M	$3.5M	3.5
2. Automatic SLCS					
o Operating, CP Plants	Automatic SLCS and increasing capacity >86gpm	2.6×10^{-6}	$2.8M	$22.5M	0.13
o New Plants and OL applicants with Auto SLCS already installed	Automatic SLCS and increasing capacity >86gpm	2.6×10^{-6}	$2.8M	$5.0M	0.56

TABLE 2

WESTINGHOUSE GENERIC OPTIONS

Generic Option	Description	P_{ATWS}/yr	Value (Millions)	Impact (Millions)	V/I
0.	Base Case; Assume Shunt Automatic	3.7×10^{-5}	--	--	--
1.	Diverse Auxiliary Feedwater and Turbine Trip	5.8×10^{-6}	$9.4M	$2.8M	3.3
2.	Diverse AFW and T/T Diverse Scram System	2×10^{-6}	$1.1M	$1.0M	1.1
3.	Diverse Scram System Diverse AFW and T/T	2×10^{-6} (same as Option 2)	$10.5M	$3.8M	2.8

TABLE 3

CE/B&W GENERIC OPTIONS

Alternative	Description	P_{ATWS}/yr	Value (Millions)	Impact (Millions)	V/I
0	Base Case	$8x10^{-5}$		--	--
1	- Diverse Scram System - Diverse Auxiliary Feed-water and Turbine Trip	$2.2x10^{-5}$	$17.4M	$5.5M	3.2

ATTACHMENT "D"

RECOMMENDATIONS OF THE ATWS TASK FORCE

RECOMMENDATIONS OF THE

ATWS TASK FORCE

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Preface

This report of the ATWS Task Force was prepared in response to a plan presented in SECY-82-275 to resolve the ATWS rulemaking with a single recommended approach. A Task Force of senior NRC technical personnel and a Steering Group of NRC management was established. The Steering Group was responsible for the approach to the alternatives, and the Task Force was responsible for the technical basis. This report was prepared under the direction of the Task Force and Steering Group and represents a consensus. The approach was based on information prior to the ATWS events at Salem 1 on February 25, 1983 and a similar event on February 22, 1983. Another Task Force (Salem Generic Issues Task Force), under the direction of Roger J. Mattson, Director, Division of Systems Integration, NRR, was established on March 1, 1983 to investigate the generic implications of the Salem 1 incidents. One of the recommendations of the Salem Task Force was to add a reliability assurance program to the rule. This report provides a basis for the value-impact assessments of the generic reactor manufacturer modifications which are required in the final rule for §50.62 to 10 CFR 50 and the proposed rule for a Westinghouse diverse scram system. The description of a reliability assurance program will be provided in the Statement of Consideration to §50.62.

NOTE: This Task Force report is currently under review for clarity and accuracy of technical detail. Therefore, there will be changes made to this report prior to publication of the rule; but these changes will not substantively affect either the evaluations made of the probability or risks of ATWS or the conclusions drawn from these evaluations.

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1.0 Summary

Generic value/impact calculations were performed for a General Electric, Westinghouse, and Combustion Engineering/Babcock and Wilcox plant in order to determine the degree of prescribed requirements for each vendor. The CE/B&W generic plants were grouped together because the proposed ATWS modifications, probability reduction, and costs to implement the modifications were very close for these two plants. In order to obtain the value of each proposed modification, the frequency of unacceptable plant states was estimated and was conservatively assumed to lead to core damage. It was also assumed that the public health cost of an ATWS accident was $10 billion for both PWRs and BWRs based on offsite radiological consequences with a value of $1,000/man-rem averted. The costs (impact) to implement the various prescribed modifications were based on information supplied by the Utility Group on ATWS. The results of the value/impact assessment are given in Table S-1.

For General Electric plants the Task Force recommends Option 1, equivalent to the Utility Proposal for New Plants, which includes a manually initiated 86 gpm Standby Liquid Control System (SLCS) plus an alternate rod injection system to be implemented at operating plants and those under construction. The value impact results of this study are favorable for Option 1. For several plants under construction that have installed an automatically actuated SLCS and for plants that have not yet received a construction permit, the Task Force recommends Generic Option 2, which includes automatic initiation of the SLCS. The value/impact results of

Table S-1

Comparison of Alternatives

GE Generic Options	P_{ATWS}/yr	Impact* (Millions)	V/I*
0. Base Case	$5.3x10^{-5}$	--	--
1. Utility Proposal, Increased SLCS to 86 gpm	$1.2x10^{-5}$	$3.5M	3.5
2. Increased SLCS Capacity and Automatic Initiation, Diverse Scram System	$2.6x10^{-6}$	$22.5M	0.13
- New Plants and OL Applicants with Auto SLCS Already Installed	$2.6x10^{-6}$	$5.0M	0.56
Westinghouse Generic Options			
0. Base Case	$3.7x10^{-5}$	--	--
1. Diverse Auxiliary Feedwater Initiation and Turbine Trip (Utility Proposal)	$5.8x10^{-6}$	$2.8M	3.3
2. Diverse Scram System, Added to Utility Proposal	$2x10^{-6}$	$1.0M	1.1
CE/B&W Generic Options			
0. Base Case	$8x10^{-5}$	--	--
1. Utility Proposal	$2.2x10^{-5}$	$5.5M	3.2
2. Safety Valves or Modifying Core, Added to Utility Proposal	$7.2x10^{-6}$	$10.0M	0.44

*Done on an incremental basis.

Option 2 for new plants is close to 1; however, the Task Force has concluded that for a BWR ATWS the consequences are significant enough to warrant this Option.

The Task Force recommends Option 2 for Westinghouse plants. The Utility Group recommended installing an auxiliary feedwater initiation and diverse turbine trip, which has a value/impact ratio of about 3. The principal argument for requiring a diverse scram system is based on the findings of the Salem Generic Issues Task Force, which concluded that the reliability of Westinghouse reactor trip systems should be substantially improved. In addition, installing a diverse scram system has a value/impact ratio of 1, and it was estimated that it could be installed for between $0.2M and $1.0M.

The Task Force recommends Option 1 for CE/B&W plants, which includes the installation of a diverse auxiliary feedwater initiation and turbine trip and a diverse scram system. Option 2, adding extra safety valve flow capacity to the pressurizer, could reduce ATWS risk but could cost $10.0M for existing plants. The value impact analysis and risk reduction achieved does not favor this additional requirement.

2.0 Introduction

An anticipated transient without scram (ATWS) is an expected operational transient (such as a loss of feedwater, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor protection system to shut down the reactor. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to core melt and to the release of radioactivity to the environment. The ATWS issue involves backup safety systems and procedures for shutting down the reactor, if there is a failure of the reactor protection system. The issue of ATWS has been under consideration since the late 1960s. Precursors have occurred at the Kahl reactor (1963), the Monticello reactor prior to startup, and at the Browns Ferry Unit 3 reactor (1980). An ATWS event did occur at the Salem 1 nuclear generating station on February 25, 1983, although no core damage or release of radioactivity occurred. The reactor was scrammed manually 30 seconds after the automatic trip system failed.

On November 24, 1981, the Commission invited comments on three alternative proposed rules relating to ATWS. Each of the three alternative proposed rules had the objective of reduction of risk from ATWS and each features a different approach to achieving that objective. One alternative emphasized individual reactor evaluation to identify needed improvements (the Staff Rule). The second alternative emphasized reliability assurance and would have also required certain hardware modifications (the Hendrie Rule). The third alternative, proposed by the Utility Group on ATWS in petition for rulemaking PRM-50-29, prescribed specific changes that were keyed to the type of reactor and its manufacturer. The Utility Group on ATWS Rule specified equipment modifications by reactor manufacturer (Westinghouse, General Electric, Babcock and Wilcox, and Combustion Engineering).

Forty public comments were received on the three alternative proposed rules. One of these comments was a probabilistic risk assessment (PRA) submitted by the Utility Group in support of their proposed rule.(1) The study was performed by Science Applications, Incorporated, with substantial assistance from the Utility Group. This report represented a significant advancement in the analysis of ATWS by using detailed event trees to delineate the accident sequences for each type of anticipated transient.

The Utility Group PRA evaluated each generic plant by reactor manufacturer. An event tree was constructed for each initiating event, and plant conditions that would lead to an unacceptable condition, such as exceeding reactor pressure vessel pressure limits or leading to overheating of the containment suppression pool, were evaluated. Finally, the value by reducing the unacceptable plant conditions and the impact (cost of modifications) were evaluated.

The NRC contracted with Energy Incoprorated to review the Utility Group PRA(2) and concluded that the study represented an "appropriate application of probabilistic methods." However, certain specific assumptions which strongly influence the results were identified and evaluated further by the Task Force. A Task Force and Steering Group of NRC staff from the Offices of Inspection and Enforcement, Nuclear Reactor Regulation, and Nuclear Regulatory Research was established to evaluate the coments and recent related information and to develop a single recommendation for an ATWS rule. The ATWS Task Force and Steering Group was formed to provide a basis for a final ATWS rule since there was no consensus from the three proposed rules. Since ATWS has been an unresolved safety issue for close to 15 years, the Task Force was requested to reach a decision as early as possible. The group was chartered to consider the following alternatives to resolve the ATWS issue:

1. No ATWS rule (or include ATWS under the Severe Accident Program).

2. Adopt the proposed or a modified version of the Utility Group rule.

3. Adopt the staff rule or a modification of it.

4. Adopt those portions of the Hendrie rule for which the staff has a technical basis.

The Steering Group provided direction to the Task Force. The Task Force met six times, and the Steering Group met three times between September 3, 1982 and October 12, 1982. A final combined meeting of both groups was held on May 20, 1983. In several of these meetings the form of an ATWS rule was discussed. Both prescriptive and evaluation model rule formats were considered. In light of the substantial technical evaluation performed in the past and in consideration of the potentially large expense of performing evaluation model analyses, the Steering Group directed the Task Force to evaluate prescriptive options for each generic reactor manufacturer. Value/impact analyses were performed on an incremental basis to evaluate each prescriptive option. This report contains the technical basis for the ATWS rule recommendations.

While this is considered to be a consensus report, there was no attempt made to receive concurrences from the individual Task Force members.

3.0 Basis for a Decision

As part of the decision process, a value-impact analysis was performed and was done on a reactor manufacturer generic basis. The annual frequency of an ATWS leading to unacceptable plant conditions was estimated for various options. Also computed were the costs of implementing these options and the estimated value due to the reduction of ATWS frequency.

In consideration of the potential consequences of an ATWS core melt accident, the Task Force set as a goal that the estimated core melt frequency due to ATWS events should probably be no more than about 1×10^{-5} per year. It was realized during this study that this goal might be costly to achieve for some plants and cannot be treated as a requirement. It was also realized that doing value-impact calculations is somewhat subjective in arriving at the optimal level of fix, due to uncertainty in probabilistic assessments and in the cost estimates for the modification. Therefore, the Task Force used value-impact calculations only as an aid to evaluate ATWS rule alternatives. The Utility Group on ATWS(1) performed extensive analyses and arrived at somewhat different estimates for P_{ATWS} from those that were obtained from the event trees used in this study. The Task Force used some of the event tree structure of the Utility Group PRA.

In the Utility Group Study the chosen index of ATWS hazard was the probability of an ATWS sequence leading to unacceptable plant conditions. This hazard index, designated P_{ATWS}, was computed for accident sequence outcomes which result in exceeding certain design parameters such as reactor vessel pressure in a PWR or suppression pool temperature in a BWR. Ultimately, these unacceptable plant conditions can lead to core melt, containment failure, and release of radioactivity to the environment. Recognizing the difficulty in computing the likelihood of core melt following unacceptable plant conditions, the Task Force adopted the P_{ATWS} index to perform value-impact evaluations. The consequences of core melt were then equated with the frequency of P_{ATWS}. Therefore, during Task Force deliberations the value-impact results were given supplementary consideration of the likelihood of core melt and containment failure given the occurrence of the unacceptable plant conditions.

The potential ATWS requirements were formulated in terms of prescriptive hardware and procedural requirements. These potential requirements were identified by reviewing the proposed ATWS rules and the technical studies which were performed in support of the proposed rules, especially NUREG-0460. In this way the hardware and procedural modifications which can reduce P_{ATWS} were selected for the generic V/I evaluation. Potential requirements of known impracticality or those which were technologically undeveloped were not considered further. The potential hardware modifications were limited to those that can reduce P_{ATWS} as opposed to those which could reduce ATWS accident consequences to the public once core melting has begun.

The incremental approach to value impact evaluation was used to evaluate the value-impact of successively more costly plant modifications. Requirements for additional equipment were evaluated separately and in appropriate combinations to arrive at the optimum recommendation. The cost of each modification was assumed to be as estimated by the Utility Group. When value-impact results were borderline, the Task Force relied much more on engineering judgment to determine whether an alternative should or should not be included in the ATWS rule.

4.0 Identification of Options for ATWS Requirements

The identification of options for ATWS requirements was done generically for three LWR groups: (1) General Electric BWRs, (2) Westinghouse PWRs, and (3) Combustion Engineering and Babcock and Wilcox PWRs. For each LWR group a base case denoted as Option 0 was defined to identify currently installed design features relevant to ATWS or those modifications which will be made in the future due to regulatory requirements outside ATWS. For instance, as a result of the Salem Task Force recommendations, Westinghouse plants are assumed to include additional diversity in the RTS circuit breaker actuation--i.e., both the undervoltage relay and shunt coil will trip the breaker. This type of hardware modification was considered to be installed as the baseline case for considering further improvements through rulemaking. Where possible the prescriptive rule options selected for evaluation include requirements similar to the Utility Group proposed for an ATWS rule.

4.1 General Electric ATWS Rule Options

In the base case, or current commitment option, several ATWS preventive and mitigating features were identified which have been or will be installed on

General Electric plants. These features include: (1) recirculation pump trip, (2) scram discharge volume modifications, (3) emergency procedure guidelines applicable to ATWS,[3] and (4) main steam isolation valve setpoint lowered to level 1 where practical. The Utility Group proposal for an ATWS rule identified the installation of an alternate rod insertion (ARI) subsystem to enhance the reliability of the scram system for General Electric plants. The ARI provides a diverse means of automatically actuating control rod insertion. It includes an automatic logic separate from the currently installed scram system which senses high reactor pressure and low water level to actuate the system. The signals are used to actuate separate dump valves which cause the pilot air header to bleed down allowing the insertion of existing control rods to shut the reactor down. Cost estimates for the installation of ARI are about $3.3 million, based on the Utility Group value impact study, and include the power generation losses from extending an outage 2 days for installation.

The Utility Group also proposed an increased Standby Liquid Control System (SLCS) injection flow of 86 gpm for new plants. The SLCS is a manually actuated system which provides the necessary boration of the reactor coolant system to achieve reactor shutdown, if the rodded scram system is unavailable. The Task Force considered upgrading the SLCS injection flow from 43 to 86 gpm on operating plants and automating SLCS actuation as other options. It was recognized that a 43 gpm SLCS could be designed to deliver the equivalent of 86 gpm of 13 percent sodium pentaborate solution by increasing the boron concentration and heat tracing the system to prevent crystalization at higher concentrations of boron. The Utility Group estimate of $200 thousand to increase the SLCS flow rate and $23.6 million to automate the system were used by the Task Force. The potential for one additional spurious actuation of an automated SLCS through the 30 year lifetime of a nuclear plant was judged to be reasonable, and the expected costs in terms of reactor downtime were included in the overall cost figures.

The Task Force also considered the effectiveness of the SLCS with regard to its injection point. The concern here is that the cooler, heavier boron solution would stratify near the bottom of the vessel without natural circulation mixing and that during the recovery phase of an ATWS the injection of cold core spray water on the top of the core might cause an initial power increase, in the worst case causing prompt criticality. BWR/4 and earlier plants inject the boron solution and HPCI through the feedwater sparger for which adequate mixing can be expected. Newer BWR/5 and BWR/6 plants with high pressure core spray systems would need to inject the boron solution into the spray sparger at the core exit to assure adequate mixing.

Table 1 provides a condensed list of the potential ATWS preventive and mitigative features and their cost estimates used by the ATWS Task Force.

4.2 Westinghouse ATWS Rule Options

As a result of the Salem ATWS incident, Westinghouse plants are expected to upgrade the reactor trip system to include diverse actuation of reactor trip breakers by both an undervoltage relay and a shunt. In addition, improved administrative procedures, surveillance, and maintenance practices have been recommended for implementation. It has been assumed that these modifications are requirements which define the current commitment for Westinghouse plants. In addition, the auxiliary feedwater systems of all PWRs were upgraded as necessary, to meet the requirements of NUREG-0737,[4] including automatic initiation.

The Utility Group on ATWS rule proposal for Westinghouse plants included the requirement for installation of ATWS Mitigating System Actuation Circuitry (AMSAC). This control subsystem would provide a diverse means of signaling the auxiliary feedwater system to start and the turbine to trip under plant conditions indicative of an ATWS. The intent of this control subsystem is to minimize the potential for RTS faults which cause the failure to scram during an ATWS, also causing failure of ATWS mitigative actions. The cost of an AMSAC was estimated at $2.8 million by the Utility Group.

A diverse scram system was identified as a potential ATWS preventive requirement for Westinghouse plants. This subsystem would provide a diverse means of interrupting power to the control rod electromagnets allowing them to drop into the core and shutdown the reactor. The system would include separate sensors, diverse logic, and diverse means of interrupting power to the control rod electromagnets. Diverse sensors are not included since significant diversity already exists in the current set of sensors used to detect conditions requiring reactor trip.

The cost of installing a diverse scram system on Westinghouse plants was estimated based on considering the cost for a similar system for Combustion Engineering and Babcock and Wilcox plants supplied by the Utility Group and in consideration of discussions with representatives of nuclear utilities held in public meetings. If a diverse scram system were installed without AMSAC, the cost could be $2.5 million. However, much of the circuitry hardware required for AMSAC can be used in a diverse scram system. In light of this consideration the minimal additional hardware required for a diverse scram system with AMSAC installed was estimated to be around $250 thousand.

4.3 CE and B&W ATWS Rule Options

Combustion Engineering and Babcock and Wilcox plants were treated together since their accident sequence characteristics in terms of plant response and accident likelihood are similar. The potential ATWS modifications and costs of implementing various ATWS rule options are also very similar. As was the case with the Westinghouse plants, past Salem reactor trip breaker improvements are assumed to represent a current regulatory requirement. And, upgrading of auxiliary feedwater systems to meet NUREG-0737 requirements was assumed as a current commitment in the base case design of these plants.

Both AMSAC and a diverse scram system were proposed for CE and B&W plants in the Utility Group ATWS rule proposal. The total cost of implementing these requirements was estimated at $5.5 million for both features.

The ability to cope with an ATWS at CE and B&W plants can be enhanced by increasing pressure relief capacity by adding safety valves. These valves would have to be added to the reactor coolant pressure boundary at the pressurizer. Therefore, the cost considered must also include the significant radiological exposure to workers which is estimated to be in the neighborhood of 40 man-rem. Installation and lost power generation during installation were estimated to be $10 million by the Utility Group.

The Task Force considered an alternative means of reducing the peak pressure of an ATWS event for CE and B&W plants. The use of burnable poisons and fuel load pattern modifications could produce a more negative moderator temperature coefficient than is typical for current CE and B&W plants. With the industry trend toward high burnup fuel and fuel load patterns which limit reactor vessel neutron fluence for pressurized thermal shock, it is expected that the natural course of events will result in a more negative moderator temperature coefficient for these plants in the future. The Task Force did not feel that the technology required to regulate MTC is currently available nor a wise regulatory practice, and therefore did not attempt to cost out and determine the effectiveness of this option.

Table 2 provides a summary of the Westinghouse plant options and costs considered and Table 3 provides the information for CE and B&W plants.

5.0 Probabilistic Analysis

The calculations of P_{ATWS} were performed by using simplified event trees for each generic reactor design. The event trees were evaluated for each prescribed ATWS preventive or mitigative option and for combinations of options to benchmark the "risk" reduction effectiveness of the proposed ATWS rule options. The following assumptions were made for all of the calculations:

1. The failure to scram is $3x10^{-5}$/demand. This point estimate is taken from NUREG-0460.[5] This has been the source of considerable disagreement between NRC and the nuclear industry. The Utility Group has stated

that this value is quite conservative and could be a factor of 5-10 lower. After the Salem 1 events, however, the statistical and analytical bases for the unavailability to scram were reexamined. It was estimated that the failure to scram for Westinghouse plants could be as high as $2x10^{-4}$/demand (based on 1 failure in 250 reactor years). This estimate can be reduced if the actuation of the reactor trip circuit breaker is initiated by a shunt as well as undervoltage relay. Assuming this modification will be made, the ATWS Task Force used $3x10^{-5}$ for value-impact assessments. Appendix A provides an evaluation of scram system unavailability for generic plants.

2. Two-thirds of the failures to scram would arise in the logic or electrical portion. One-third would be considered mechanical. This assumption is based on a review of ATWS precursors and LERs related to reactor trip system malfunctions that have occurred in all reactor protection system designs. It was judged that the ratio of electrical/mechanical failure could be higher, which would enhance the value of a diverse scram system. However, the precursor data was felt to be insufficient to support a higher ratio. With some reservations for BWRs a 2/1 ratio was judged to be a reasonable estimate. A more detailed description of the electrical and mechancial failure modes of the RTS is provided in Appendix A.

5.1 BWR ATWS

In a BWR with reactor recirculation pump trip an ATWS transient will initially result in an increase in reactor coolant system pressure which can be adequately controlled through safety relief valve discharge to the suppression pool. The pressure relief and power-to-feedflow mismatch results in a lowered reactor vessel water level which increases voiding within the reactor core region. This in turn reduces moderator effectiveness and causes a power reduction. It has been estimated that the power will equilibrate at around 20 to 40 percent of full power for transients where the feedwater system remains in operation, depending on the BWR vintage. If the feedwater system fails or is tripped and HPCI or equivalent makeup flow is initiated, the reactor power level is estimated to level out at 8 to 12 percent.

A BWR is typically designed to bypass up to 25 percent of steam flow to the condenser. Thus, if the ATWS transient has not involved MSIV isolation or loss of condenser, a maximum of 15 percent of steam flow will be directed to the suppression pool. With the main steam lines isolated, or the condenser unavailable, all of this steam flow will be directed to the suppression pool. Suppression pool cooling systems are typically designed to reject shutdown decay heat to the ultimate heat sink. This amount of decay heat rejection capability is less than 5 percent of full power steam flow. Therefore, the suppression pool temperature will increase continuously until the reactor power level can be decreased to a level equal to the combination of turbine bypass capacity and suppression pool cooling capability. At high suppression pool temperatures there is a potential for unstable condensation of the relief valve discharge to the pool causing excessive loading on the containment structure. A loss of containment integrity and core cooling degradation could follow. A 200°F temperature limit has been set by the resolution of unresolved safety issue A-39, Determination of Safety/Relief Valve Pool

Dynamic Loads and Temperature Limits for BWR Containments.[6] Emergency procedure guidelines call for the throttling of reactor coolant makeup to control reactor vessel water level and the actuation of the standby liquid control system when the suppression pool temperature reaches 110^{o}F. The operator is to lower the reactor vessel water level to decrease power (increase voiding) and begin to slowly depressurize.

The standby liquid control system is designed to add sufficient boron to the reactor coolant system to reduce the core reactivity to achieve reactor subcriticality. Most SLCS are designed to deliver a 13 percent sodium pentaborate solution at 43 gpm to the reactor coolant system which will cause reactor power shutdown within approximately one-half hour if adequate mixing exists. This system is manually actuated.

The process of depressurization will result in approximately a 45^{o}F increase in suppression pool temperature. For nonisolation transients it has been estimated that if power reduction to less than 25 percent is not successful within 17 minutes, the suppression pool will reach the 200^{o}F limit. In addition to the suppression pool temperature limit for containment integrity, there is a limit of 160^{o}F for operation of HPCI in the recirculation mode due to pump bearing qualification. Limiting the suppression pool temperature to 160^{o}F without depressurization also requires power reduction to about 25 percent in 17 minutes.

For condenser isolation transients emergency procedure guidelines alone are not adequate to maintain a suppression pool temperature below 200°F. If an enhanced capacity SLCS (86 gpm or more) were provided, operator action to initiate the EPGs would be required in about 2 minutes.

5.2 BWR Analysis

A simplified BWR ATWS event tree model was developed to evaluate the likelihood of accident sequences for the various system and procedural modifications identified in Section 3 for General Electric plants. The event tree structure and success criteria were based on a review of dominant ATWS accident sequence characteristics found in NUREG-0460[5] and the Utility Group PRA.[1] The BWR ATWS event tree proceeds from transient initiators through RTS failure to reactor power reduction by reactor vessel water level control, SLCS actuation, and long term subcriticality and cooling. Failure to provide reactor vessel makeup is not included in the simplified event tree since multiple sources must all fail to cause core damage; the likelihood of which is relatively small. In addition, two event trees have been developed to differentiate between MSIV isolation transients and those ATWS sequences where turbine bypass is available.

The principal unacceptable plant condition selected for evaluation of BWRs was exceeding the suppression pool temperature limit of 200°F. There is currently no test data available to determine the magnitude of SRV quencher condensation loads above that limit. Thermodynamic arguments have been advanced by the Utility Group which suggest that stable condensation can be expected at elevated temperatures in excess of 200°F where containment overpressure results in sufficient subcooling at the quencher device outlet. Although these arguments seem reasonable and the 200°F pool temperature limit may be conservative, the current regulatory position was maintained to identify the BWR unacceptable plant condition.

The definition of the event tree headings, their likelihood, and success criteria are described below:

1. Number of reactor transients

   The number of reactor transients to be considered was derived from the Utility Group Study. Based on EPRI data,[7] that study identified 3.8 transients per year at 25-100 percent power and 1.7 transients per year at less than 25 percent power averaged over an expected 40 year reactor lifetime. Since the event tree structure used in the Utility Group study accounted for subtle differences in the type of ATWS initiating transients, it was necessary to synthesize an equivalent transient frequency for the simplified event tree structure used by the ATWS Task Force. In the Utility Group Study the frequency of unacceptable plant conditions, when no credit is given for operator actions, was estimated at 1.3×10^{-4} per year if the RTS unreliability is 3×10^{-5}. The value for "equivalent" transients can be inferred to be $1.3 \times 10^{-4}/3.0 \times 10^{-5} = 4.3$ per year.

   In addition to the number of reactor transients to be considered, the split between MSIV isolation transients and those transients when turbine bypass is available was estimated. The EPRI data was reviewed to identify the fraction of transients which would not result in turbine trip prior to reaching a condition where MSIV actuation would be expected. Approximately 30 percent of the transients were estimated to fall in this category. For the remaining 70 percent, turbine bypass was assumed to be available.

2. Initiating Emergency Procedure Guidelines (EPG)

o Bypass Available (70 percent of transients)

For the case where the turbine bypass is available (70 percent of transients), it was assumed that the difference in core thermal power, minus the bypass flow, would be dissipated by heating the suppression pool. The resulting suppression pool temperature rise would be about $3.75^{o}F$/minute.[8] Using $200^{o}F$ as an upper bound and subtracting $45^{o}F$ for a depressurization, the allowable time for reduction of power to <u>less</u> than 25 percent would be $\frac{(200-45)-90}{3.75}$ = 17 minutes. From Figure 1, the Human Error Probability (HEP) at 17 minutes is about 0.05.

Although it was assumed that the operator will delay his actions to a maximum point for which the containment limit of $200^{o}F$ will not be exceeded, the Task Force believes that the operator will begin to reduce water level as early as five minutes following the transient.

o Isolation transients (30 percent of transients)

For these cases where all of the reactor power is dissipated in the suppression pool, the suppression pool temperature would exceed $200^{o}F$

slightly even if the operator immediately followed the procedures and actuated the SLCS. If SLCS is increased to 86 gpm, the operator must act within two minutes after the transient begins in order not to exceed the 200°F suppression pool limit. Therefore, it was conservatively assumed that all isolation transients will exceeed the 200°F containment suppression pool limit with the current SLCS capacity of 43 gpm. The Utility Group's technical consultant[8] provided a sensitivity analysis for a 43 gpm SLCS and an 86 gpm SLCS. The results are shown in Tables 4 and 5. We assumed a human error probability of 0.5, based on Figure 1, for initiating the EPGs at 2 minutes.

3. Maintain Water Level

In the event trees, operator actions for maintaining the reactor vessel water level at the appropriate height above the core was treated as a separate event. For nonisolation transients the operator must not lower the level below Level 1, or MSIV actuation will result. For isolation transients the operator must maintain the level at the top of the active fuel and not uncover the core. This procedure is difficult because the operator must check both the level indicators (which may be in error) and the average power range monitors (APRMs) in order to verify that the level is being maintained correctly. A human error probability of 0.05 was selected for transients where bypass was available and 0.1 for the 86 gpm isolation transients.

4. Long Term Cooling

Once the water level is maintained low enough to reduce power and the boron is injected (26 minutes at 43 gpm and 13 minutes at 86 gpm), the water level must be raised by adding inventory to the reactor vessel and still maintain the operating pressure. Also, the RHR coolers must be initiated. There is the possibility that the reactor vessel may depressurize during this time and raise the pool temperature as much as 45^{o}F, which could lead to exceeding the suppression pool temperature limit. A human error probability of 0.05 was assumed for this fairly difficult procedure.

5. Automatic SLCS

Based on previous analyses performed by GE[9], the suppression pool temperature limit of 200^{o}F is met if the SLCS flow is automatically initiated within about 2 minutes following an ATWS and the SLCS flow is on the order of 86 gallons per minute. It was assumed that an automatically initiated SLCS with an unavailability of 0.01 can be designed and installed.

5.3 BWR Results of P_{ATWS}

The results of the General Electric generic options analyses are given in Table 6. The base case P_{ATWS} frequency, using a failure to scram of $3x10^{-5}$/demand, is $5.3x10^{-5}$/yr. The event trees are given in Figures 4 and 5. The base case frequency can be reduced to $1.2x10^{-5}$/yr by installing ARI and increasing the SLCS flow to 86 gpm. The event trees are given in Figures 6 and 7. The installation of an automatic SLCS of approximately 86 gpm flow capacity would reduce the P_{ATWS} to $2.6x10^{-6}$. The event tree results for this case are shown in Figure 8. The automatic SLCS actuation would enable the plant to successfully mitigate an ATWS with no operation intervention other than to establish long term, shutdown cooling.

5.4 PWR ATWS

In a PWR an ATWS transient will result in a reactor coolant system pressure rise, the magnitude of which is dependent on reactor core physics parameters and relief valve capacity as well as ability to maintain sufficient heat sink through steam generator feedwater inventory. Westinghouse designed reactors are the least sensitive of PWRs to ATWS sequences, principally due to the larger relief capacity incorporated into this type of plant design. The pressure transient and ATWS mitigative features are similar for CE and B&W design reactors and for this reason they are analyzed as a single generic reactor type.

The pressure rise that occurs in the reactor coolant system is limited to a large extent by the negative reactivity feedback associated with the moderator temperature coefficient causing a reduction in reactor core power. A more negative moderator temperature coefficient will result in a lower pressure rise, all other parameters kept constant. The magnitude of the pressure rise is also strongly affected by the ability to relieve pressure (capacity of relief valves) and the ability to maintain feedwater inventory within the steam generators to remove some of the core power through secondary heat removal.

Even if the peak pressure rise can be limited to acceptable levels, it is necessary to reduce the pressure to the point that high pressure coolant injection of borated water can succeed in reducing reactor power to shutdown cooling levels. If such pressure reduction were not achieved, reactor coolant would continue to be lost through pressure relief eventually resulting in the core uncovering. The mechanisms by which pressure is reduced is by removing residual power through the steam generators by the auxiliary feedwater system. The residual power level settles out at around 8 percent of full power due to continued high temperature. Since reactor coolant has been blown out the relief valves, the pressure of the reactor coolant system can be lowered to a point at which the high pressure injection system can be actuated. Boration through HPI will eventually reduce power to decay heat levels allowing a controlled shutdown.

If the pressure rise in a PWR cannot be successfully limited to acceptable levels, reactor coolant system integrity may be threatened, or, mechanical components in the RCS pressure boundary may be rendered useless. In either case, the ability to cool the core is jeopardized and core melt may result. Therefore, unacceptable plant conditions for a PWR are associated with the pressure rise during an ATWS event exceeding a level at which reactor coolant integrity and component operability can be assured.

5.5 PWR Analysis

The PWR accident sequences progress from the anticipated transient to RTS failure to RCS overpressure and then to secondary heat removal requirements and ultimate shutdown involving subcriticality through RCS boration. Simplified PWR event trees were constructed to include these accident sequence considerations which have been shown to represent the important ATWS accident sequence events from both NUREG-0460 and the Utility Group PRA. The definition of unacceptable plant condition used by the Task Force was exceeding ASME Service Level C pressure, or about 3200 psi. The Utility Group proposed using ASME Service Level D as a measure of unacceptable plant conditions for CE and B&W plants. Because reactor coolant system integrity becomes highly uncertain at that level and because valves in the reactor coolant pressure boundary could deform to the point of inoperability at that level, that proposal was rejected.

The definition of event tree headings, their likelihood and success criteria are provided below:

1. Number of Transients, AT

   We have assumed four significant transients per year, based on a review of EPRI data(7,10). Based on both EPRI data and the Utility Group PRA, we assumed that approximately 70 percent of the transients involve turbine trip for which bypass to the condenser is available. These transients are both turbine trip initiators and those that would lead to a turbine trip. The remaining 30 percent of the transients are conservatively treated as loss of feedwater transients.

2. Fraction of Time for Favorable Moderator Temperature Coefficient

   The moderator temperature coefficient (MTC) is a measure of the reduction in core reactivity as the water temperature increases. It is usually expressed as PCM/^{o}F, which is the incremental change in reactivity per degree fahrenheit of moderator temperature increase divided by 10^5. As the reactor temperature increases during an ATWS, the MTC affects the efficiency of the moderator, causing a reduction in reactor power.

- o <u>Westinghouse Plants</u>

For non-turbine trip cases, MTC was assumed to be at an unfavorable value such that 10 percent of the time an ATWS transient would result in unacceptable plant conditions--exceeding Service Level C. This estimate takes into account a larger turbine bypass, 60 percent, as opposed to 40 percent bypass capability assumed by Westinghouse in their analyses and the staff evaluation in NUREG-0460(5). For turbine trip transients, an unfavorable MTC value was assumed to exist only one percent of the time since the pressure transient for this type of ATWS is relatively mild.

- o <u>CE/B&W Plants</u>

If turbine trip occurs and auxiliary feedwater is initiated, the value of moderator temperature coefficient is such that Service Level C will not be exceeded 50 percent of the time. This value is based on estimates in NUREG-0460, Vol. 4.(5) With high burnup fuel this parameter is expected to become more favorable in the future with the recent trend to include burnable poisons to increase fuel burnup since the once through fuel cycle would almost certainly continue as a practice for some time. On the other hand, extra safety valves could be added to allow operation with a favorable MTC for longer periods during the fuel cycle. Favorable MTC operating cycles comparable to those of Westinghouse could be achieved.

3. Auxiliary Feedwater Availability

Auxiliary feedwater effectiveness is tied to the fraction of time for unfavorable moderator temperature coefficient. With an unfavorable MTC auxiliary feedwater alone cannot result in successful coping with an ATWS. Conversely, auxiliary feedwater actuation in a timely manner is required to limit RCS pressure for most ATWS transients even with a favorable MTC.

o Westinghouse Plants

Based on Westinghouse analyses previously performed and benchmarked by the NRC in NUREG-0460,[5] one auxiliary feedwater (AFW) pump (out of two or three depending on the plant) is needed to limit the peak pressure to less than approximately 3200 psia. Based on AFW reliability studies performed for Westinghouse plants, the unavailability of all two or three trains of AFW was assumed to be about 0.001, if the initiation signal is given. This is based on the assumption of upgrading all plants per NUREG-0737 for automatic actuation of AFW. The auxiliary feedwater system is actuated on low-low water level in the steam generators, by a safety injection signal, by a trip of all the main feedwater pumps, or by a manual start signal. Approximately 70 percent of the transients are turbine trips, for which the resulting pressure rise in the primary vessel is mild compared to a loss of feedwater transient. It was conservatively assumed that an RTS failure in the electrical portion of the system would also cause failure to automatically initiate auxiliary feedwater as the AFW actuation signal is generated in the RTS for Westinghouse plants. On the other hand, main feedwater addition to the steam generator may continue for some time if there is a turbine trip without scram. For this case the unavailability of AFW initiation would be due to human error. It was estimated that AFW would have to be actuated in roughly 12 minutes which from Figure 1 corresponds to a human error probability of 0.16.

For the remaining 30 percent of the transients where the turbine does not trip (analogous to loss of feedwater), if the electrical portion of the RTS fails, auxiliary feedwater cannot be initiated in time to mitigate the reactor vessel overpressure. In order to have successful mitigation of peak overpressure, the AFW must be initiated within about two minutes. This cannot be manually performed in time with a high probability of success.

Since the event trees used in this study are simplified, it should be noted that there is a higher likelihood that auxiliary feedwater will be available than assumed. If a diverse, automatic auxiliary feedwater actuation and turbine trip (AMSAC) is installed, the potential dependence between RPS failure and successful AFW operation is eliminated. The reliability of AFW is then equivalent to the automatic actuation reliability previously indicated for all of the significant ATWS transients.

o <u>CE/B&W Plants</u>

These plants require two out of two auxiliary feedwater trains operating within about one or two minutes. An unavailability of both trains of AFW was estimated as 0.04 for existing plants if automatically actuated and the initiation signal is present. This estimate was used by the Utility Group but it is recognized that it would be very plant-specific. The same logic to illustrate the benefit of diverse auxiliary feedwater initiation and turbine trip as described above for the Westinghouse plant applies here. However, since AFW actuation must be so prompt to represent successful ATWS mitigation, operator actuation in a timely manner was assumed to be of minimal benefit.

5. High Pressure Injection

o <u>Westinghouse Plants</u>

Safety injection is actuated by (1) a manual signal from the operator, (2) a low pressurizer pressure signal, or (3) a high containment pressure signal. If any one of these signals is present and if the reactor coolant system pressure is less than the shutoff head of the SI pumps, borated water can be pumped into the Reactor Coolant System by high pressure injection (HPI). Also, borated water can be added by the Chemical Volume and Control System (CVCS) and is available in the event that HPI fails. However, operator action may be required for successful boration by CVCS. An unavailability of 0.01 for high pressure injection which corresponds to automatic actuation of SIS was assumed. This is consistent with the Utility Group study unavailability estimations.

o <u>CE/B&W</u>

Manual actuation by realignment of charging pumps, in order to inject boron, is most likely required for both CE and B&W plants, but this is dependent on the actual plant. This is also a conclusion of the Utility Group study although it did not account for human error. The actuation can be delayed for at least 10 minutes and probably longer as the moderator temperature increase keeps the power level down. An unavailability of 0.05 was used which corresponds to the human error probability for taking proper corrective action (initiating the boron injection) in roughly 15 minutes.

5.6 PWR Results for P_{ATWS}

The generic calculations of P_{ATWS} are given in Table 7 for the Westinghouse plant and in Table 8 for the CE/B&W plant. These results are discussed separately below.

o Westinghouse

The base case, using a failure to scram of $3x10^{-5}$/demand, results in a P_{ATWS} frequency of $3.7x10^{-5}$/yr. The event trees are shown in Figures 9 and 10. However, it should be noted that the Westinghouse base case results for ATWS could be as high as $2x10^{-4}$/yr if a failure to scram probability is $2x10^{-4}$/demand (which is an estimate for current Westinghouse scram systems per Appendix A). The base case results are reduced to $5.8x10^{-6}$/yr by installing a system for diverse auxiliary feedwater initiation and turbine trip (AMSAC) as shown in the event trees in Figures 11 and 12. This could be reduced further to $2x10^{-6}$/yr by installing a diverse scram system (DSS), which effectively eliminates the electrical contribution of the reactor protection system unavailability to ATWS. The event trees for DSS added are shown in Figures 13 and 14.

o <u>CE/B&W</u>

The base case, using a failure to scram of $3x10^{-5}$/demand, results in a P_{ATWS} frequency of $8x10^{-5}$/yr. The event trees are shown in Figures 15 and 16. The base case results are reduced to $2.2x10^{-5}$/yr by implementing the Utility Group proposed rule of a diverse AFW actuation and turbine trip (AMSAC), plus a diverse scram system (DSS). The event tree results for this case are provided in Figure 17. The next level of P_{ATWS} reduction could be achieved by the installation of extra safety valves (or larger capacity valves) on the pressurizer, or by a reduction of the fraction of time that an unfavorable moderator temperature coefficient would exist by using burnable poisons. The event tree for this case is shown in Figure 18. For the latter case evaluation it was assumed that the MTC is unfavorable 0.1 of the time if the modifications are made. It is quite likely that further reductions are possible depending on the number of valves installed or the amount of burnable poison used. The P_{ATWS} estimate could be as low as $7x10^{-6}$/yr for this option.

6.0 Value Impact Calculations

The probability calculations were used to determine the reduction in risk and cost to the utility and public. It was assumed that the cost of an ATWS is \$10 billion and the plant will operate for 30 years. The cost of \$10 billion is a very gross approximation (and was used by the Utility Group) and represents an estimate of the monetization of offsite radiological consequences, based on \$1,000/man-rem. The maximum offsite radiological consequences from an ATWS were estimated to be approximately 10^7 man-rem.

The value/impact analysis is done on an incremental basis. The alternative that yields the highest V/I ratio is assumed to be installed first. Subsequent alternatives are placed in the order to give the next largest V/I ratio. The costs to the utilities of implementing the various ATWS rule alternatives are derived from the Utility Group report. The value is ΔP_{ATWS}/yr x 1×10^{10} x 30 yrs of operation, or $\Delta P_{ATWS} \times 3\times10^{11}$. The results of the value-impact assessments are given in Tables 6-8.

o <u>General Electric</u>

As seen in Table 6, the generic requirement proposed in the rule is the addition of a Diverse Scram System (ARI) and to increase the SLCS flow capacity to 86 gpm. The ARI, per our event trees, gives the actual reduction of a factor of three by eliminating the electrical portion of the RPS unavailability. The ARI is also the major cost contributor ($3.3M out of a total of $3.5M). The increased SLCS flow is, we believe, justified since it qualitatively reduces the risk and is relatively inexpensive. The V/I is 3.0 for the generic option of the rule.

If SLCS were automated and the capacity increased above 86 gpm, the P_{ATWS} could be reduced to 2.6×10^{-6}/yr with a sizeable value ($4.6M). However, the Utility Group's cost estimates to do this are over twenty million dollars and are dominated by downtime for installation in existing plants and by downtime for for a spurious actuation. The value/impact ratio for this generic option is unfavorable. For new plants, however, the value/impact would be close to 1 if a reliable system could be designed that would have a high reliability when challenged and a high reliability against spurious actuations.

o <u>Westinghouse</u>

As seen in Table 7, the installation of AMSAC, which is estimated to cost $2.8 million per plant, has a value/impact of slightly over 3. The next modification would be installing a diverse scram system, which has a value/impact of just slightly greater than 1.

o <u>CE/B&W</u>

As can be seen in Table 8, the Utility Group proposed rule, Generic Option 1, (and the recommendation of the Task Force), has a value/impact of just over 3. Generic Option 2, the addition of safety valves or burnable poisons, plus a DSS and AMSAC, has a value/impact of 0.4 for existing plants. For new plants, the value/impact would be approximately 1 since replacement power ($5.0M) would be eliminated.

7.0 Sensitivity Evaluations

An evaluation of key issues was made by Energy, Incorporated[1] and are listed below:

- o Operator Actions for Actuating SLCS for BWRs
- o Suppression Pool Temperature Limits for BWRs
- o Ratio of Electrical vs. Mechanical Failures in the RPS
- o Reactor Coolant System Integrity in PWRs
- o Initiation of High Pressure Injection in PWRs
- o Auxiliary Feedwater Reliability in PWRs
- o Generic vs. Plant Specific Analyses
- o Cost Uncertainties

The Task Force performed sensitivity analyses on certain parameters and considered additional parameters, especially the variation of reactor protection system unavailability.

o <u>BWRs</u>

The Task Force did not perform sensitivity studies on the suppression pool temperature limit and the associated affect on the time for operator action since 200°F is the current regulatory limit based on NUREG-0783.[6] A sensitivity study of P_{ATWS} vs. operator human error probability (HEP) for combined SLCS actuation, water level lowering and long term cooling was performed and is given as Figure 2. The sensitivity of P_{ATWS} to the electrical/mechanical ratio of RPS failure is given as Figure 3.

o <u>CE/B&W</u>

The event trees and success criteria used by the Task Force for CE and B&W plants are such that the unacceptable plant conditions are dominated by the fraction of time for an unfavorable moderator temperature coefficient, MTC, (0.5). The Utility Group assumed a much lower fraction of time for an unfavorable MTC. Therefore, we did not do any sensitivity studies other than the scram system reliability (Appendix A) and P_{ATWS} as a function of the electrical/mechanical ratio (Figure 3).

o <u>Westinghouse</u>

The event trees and success criteria, after the rule is implemented, is independent of human interactions (although this is an oversimplification), and we did not do sensitivity studies on human error probability. We did not do sensitivity studies on auxiliary feedwater reliability since the results of the Energy, Incorporated study have shown P_{ATWS} to be insensitive to AFW unavailability. Considerable thermal hydraulic analyses of the fraction of time for an unfavorable MTC have been performed for Westinghouse plants. The Task Force used a value of 0.01 for turbine trip transients and 0.10 for non-turbine trip transients. The latter value is probably conservative. We did not evaluate P_{ATWS} as a function of unfavorable MTC fraction.

The Salem 1 ATWS event and subsequent NRC evaluations, such as NUREG-0977[11] and NUREG-1000,[12] have shown that Westinghouse reactor protection systems have the least redundancy and the higher unavailability on demand. Appendix A concludes that the current RPS unavailability is approximately $2x10^{-4}$/demand (about 6 times higher than our generic estimate) and can be substantially reduced as the post-Salem requirements (shunt as well as undervoltage trip actuation and implementation of inspection, maintenance and surveillance programs) are implemented.

o Variation of Offsite Consequences

The assumption of 10^7 man-rem is reasonable for BWR Mark I containment ATWS events since the containment may fail prior to core melt. This would lead roughly to a "Category 2" release, which has an expected consequence of approximately 10^7 man-rem. Mark II and Mark III containments are less susceptible to large inventory releases than Mark I containments, and the radiological consequences would be lower. Further quantification of the

Mark II and Mark III release categories is being examined by the Commission as part of the ongoing research on source term reduction. Also, mitigating features on Mark I containments to reduce offsite consequences are being evaluated as part of the Severe Accident Research Program.

For PWRs, the expected releases could be significantly lower than 10^7 man-rem for large, dry containments and are very hard to quantify for ice condenser containments although 10^7 man-rem would be a conservative upper limit. If steam generator tubes rupture due to the reactor overpressure, the offsite consequences would be somewhat higher (comparable to a "category 2" release) than for core melt inside containment. Quantification of tube rupture probability is very difficult with the present state-of-the-art.

o <u>Variation of Cost Data</u>

The costs provided by the Utility Group on ATWS and used by the Task Force, we believe, have a large uncertainty since much of the cost is for replacement power due to spurious trips. We have not attempted to quantify this uncertainty, but we recognize the potential for spurious trips.

References

1. "Technical Support for the Utility Group on ATWS," SAI-011-83-SJ, prepared by Science Applications, Inc., December 31, 1981. Also submitted as part of a public comment by the Utility Group on ATWS on April 23, 1982.

2. "Evaluation of the Comments of the Utility Group on ATWS," Energy Incorporated, October 1982.

3. "Emergency Procedure Guidelines," Republication Draft Revision 2, May 20, 1982.

4. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

5. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, Vol. 1-4.

6. "Suppression Pool Temperature Limits for BWR Containments," NUREG-0783, November 1981.

7. "ATWS: A Reappraisal: Part III, Frequency of Anticipated Transients," EPRI NP-801, July 1978.

8. Letter from Donald M. Knuth, KMC, Inc. to Dr. Charles C. Graves, USNRC, December 2, 1982.

9. NEDE-24222, Assessment of BWR Mitigation of ATWS, Vol. I, II, General Electric Coporation, December 1979.

10. "ATWS: A Reappraisal: Part III, Frequency of Anticipated Transients," EPRI NP-2230, January 1982.

11. "NRC Fact-Finding Task Force Report on the ATWS Event at Salem Nuclear Generating Station, Unit 1, on February 25, 1983," NUREG-0977, March 1983.

12. "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," NUREG-1000, Vol. 1, 1983.

TABLE 1

Cost Estimates of General Electric Generic Options

Generic Option	Cost Estimates (Millions): Design, Engineering, Installation	Downtime for Installation	Downtime for Spurious Trip	AFUDC*, Operation, Maintenance
ARI	$0.86M	$1.0M	$1.0M	$0.44M
Increase SLCS to 86 gpm	$0.2M	--	--	--
Automate SLCS	$3.35M	$10.0M	$5.0M	$4.2M

*AFUDC is Allowable Funds Used During Construction and represents the money that could have been invested elsewhere by the Utility.

TABLE 2

Cost Estimates of Westinghouse Generic Options

	Cost Estimates (Millions)			
Generic Option	Design, Engineering, Installation	Downtime for Installation	Downtime for Spurious Trip	AFUDC, Operation, Maintenance
Diverse Auxiliary Feedwater and Turbine Trip (AMSAC)	\$1.0M	--	\$1.0M	\$0.8M
Diverse Scram System	\$1.0M*	--	--	--

*This assumes a coupling of Diverse Scram System with AMSAC. We have lumped Operation and Maintenance costs with AMSAC and assume no additional spurious trip.

TABLE 3

Cost Estimates of CE/B&W Generic Options

Generic Option	Cost Estimates (Millions)			
	Design, Engineering, Installation	Downtime for Installation	Downtime for Spurious Trip	AFUDC, Operation, Maintenance
Utility Proposal; Diverse Auxiliary Feedwater and Turbine Trip; Diverse Scram System	$2.0M	--	$3.0M	$0.5M
Addition of Larger Capacity Relief Valves	$2.6M*	$5.0M	$2.0M	$0.4M

*Includes $1.0M for any reanalysis of required valve capacity.

TABLE 4

MSIV CLOSURE EVENT

CONTAINMENT TEMPERATURE SENSITIVITY

SLCS Flow of 43 GPM

TAF Water Level = 8% Power

	TIME DELAY IN INITIATION		
BWR-4 with HPCI, RCIC	0	2	10
Maintain Normal Level	200	235	283
EPG	211	221	264
EPG No Blowdown	168	176	221
BWR-5 or 6 with HPCS, RCIC			
Maintain Normal Level	202	207	229
EPG	202	209	243
EPG No Blowdown	159	166	200

TABLE 5

MSIV CLOSURE EVENT

CONTAINMENT TEMPERATURE SENSITIVITY

SLCS Flow of 86 GPM

TAF Water Level = 8% Power

	TIME DELAY IN INITIATION		
BWR-4 with HPCI, RCIC	0	2	10
Maintain Normal Level	171	186	234
EPG	186	197	239
EPG No Blowdown	143	154	196
BWR-5 or 6 with HPCS, RCIC			
Maintain Normal Level	162	167	189
EPG	148	184	218
EPG No Blowdown	135	141	175

TABLE 6

General Electric Generic Option P_{ATWS} Results

Generic Option	P_{ATWS}/yr
0 Base Case	Bypass available (Figure 4) $1.4x10^{-5}$
	Isolation transient (Figure 5) $3.9x10^{-5}$
	Total = $5.3x10^{-5}$
1 ARI Increase SLCS to 86 gpm	Bypass available (Figure 6) $4.5x10^{-6}$
	Isolation transient (Figure 7) $7.5x10^{-6}$
	Total = $1.2x10^{-5}$
2 Automatic SLCS	$2.6x10^{-6}$ (Figure 8)

TABLE 7

Westinghouse Generic Option P_{ATWS} Results

Generic Option	P_{ATWS}/yr
0 Base Case	Bypass available (Figure 9) $1.1x10^{-5}$
	Non-turbine trip transient (Figure 10) $2.6x10^{-5}$
	Total = $3.7x10^{-5}$
1 Diverse Auxiliary Feedwater and Turbine Trip (AMSAC)	Bypass available (Figure 11) $1.8x10^{-6}$
	Non-turbine trip transients (Figure 12) $4.0x10^{-6}$
	Total = $5.8x10^{-6}$
2 Diverse Auxiliary Feedwater and Turbine Trip (AMSAC), Diverse Scram System	Bypass available (Figure 13) $5,9x10^{-7}$
	Non-turbine trip transient (Figure 14) $1.3x10^{-6}$
	Total = $1.9x10^{-6}$

TABLE 8

Combustion Engineering/Babcock and Wilcox P_{ATWS} Results

Generic Option		P_{ATWS}/yr
0	Base Case	Bypass available (Figure 15) $4.9x10^{-5}$ Isolation transient (Figure 16) $3.1x10^{-5}$ Total = $8.0x10^{-5}$
1	Diverse Auxiliary Feedwater and Turbine Trip (AMSAC); Diverse Scram System	Figure 17 $2.2x10^{-5}$
2	Addition of Extra Safety Valves to (AMSAC) and Diverse Scram System	Figure 18 $7.2x10^{-6}$

TABLE 9

General Electric Generic Option Value Impact

Generic Option	$\Delta P_{ATWS}/yr$	Value	Impact	V/I
1 ARI Increase SLCS to 86 gpm	4.1×10^{-5}	\$12.3M	\$3.5M	3.5
2 Automatic SLCS	9.4×10^{-6}	\$2.8M	\$22.5M	0.12

TABLE 10

Westinghouse Generic Option Value Impact

Generic Option		$\Delta P_{ATWS}/yr$	Value	Impact	V/I
1	Diverse Auxiliary Feedwater and Turbine Trip (AMSAC)	$3.1x10^{-5}$	$9.4M	$2.8M	3.3
2	Diverse Auxiliary Feedwater and Turbine Trip; Diverse Scram System	$3.9x10^{-6}$	$1.1M	$1.0M	1.1

TABLE 11

CE/B&W Generic Option Value Impact

	Generic Option	$\Delta P_{ATWS}/yr$	Value	Impact	V/I
1	Diverse Auxiliary Feedwater and Turbine Trip; Diverse Scram System	5.8×10^{-5}	$17.4M	$5.5M	3.2
2	Diverse AFW, T/T, Diverse Scram System, Additional Safety Valves	1.5×10^{-5}	$4.4M	$10.0M	0.44

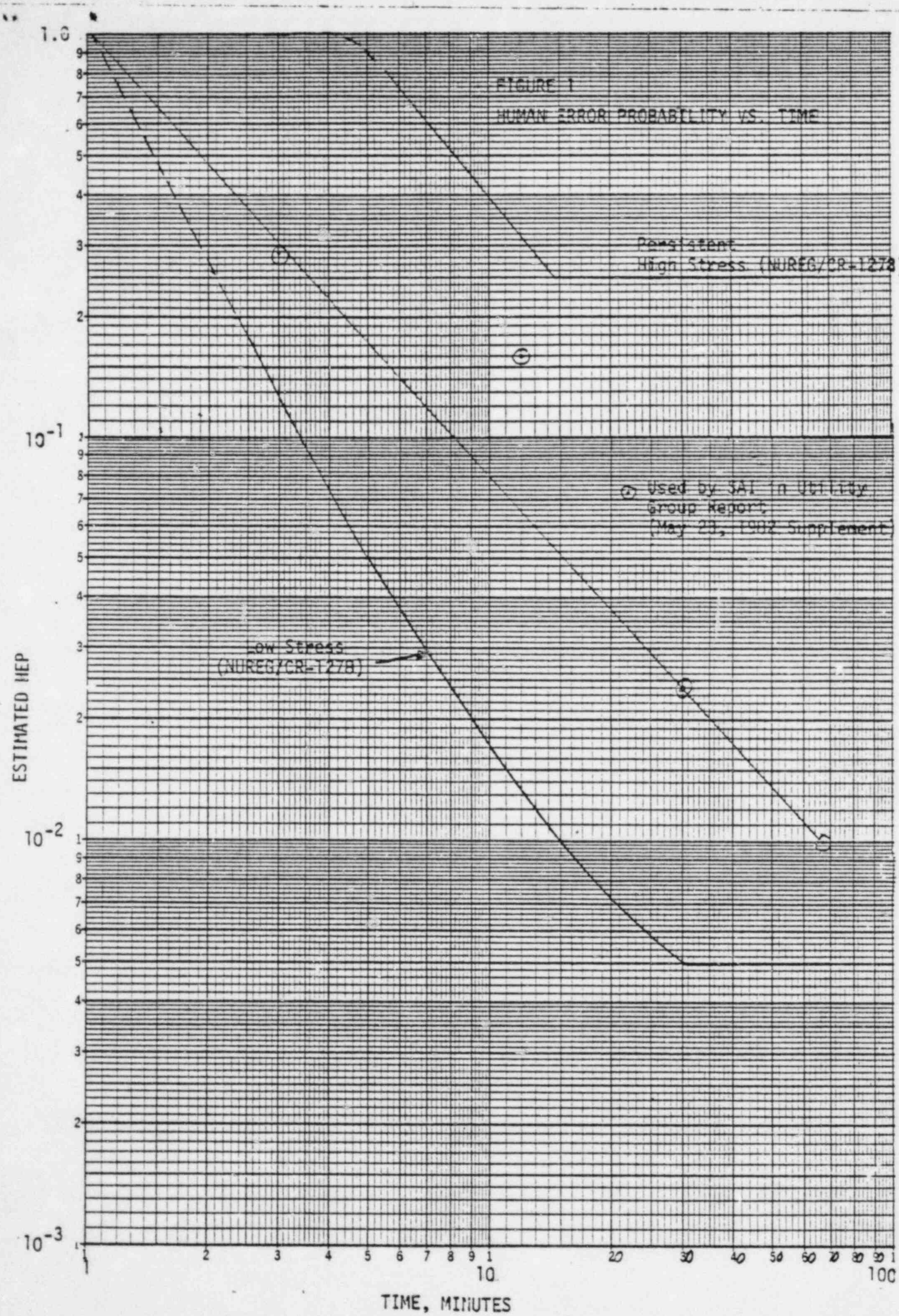
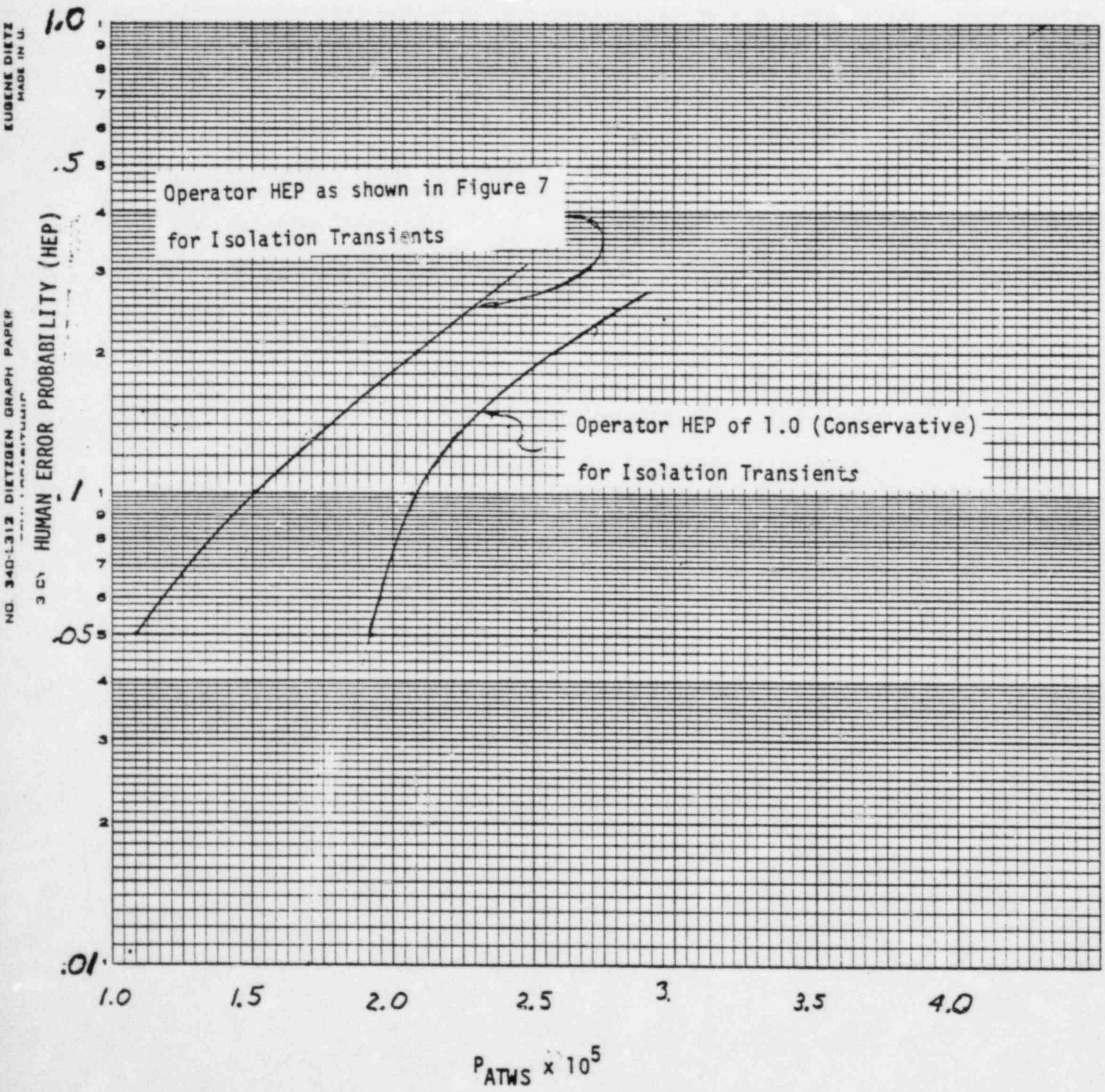


FIGURE 1
HUMAN ERROR PROBABILITY VS. TIME
Persistent
High Stress (NUREG/CR-1278)
Used by SAI in Utility
Group Report
(May 28, 1982 Supplement)
Low Stress
(NUREG/CR-1278)
ESTIMATED HEP
TIME, MINUTES
1.0
10^-1
10^-2
10^-3
1
10
100

FIGURE 2

SENSITIVITY OF BWR OPTION 1 TO ASSUMED HUMAN ERROR PROBABILITY FOR OPERATOR IN BYPASS AVAILABLE TRANSIENTS



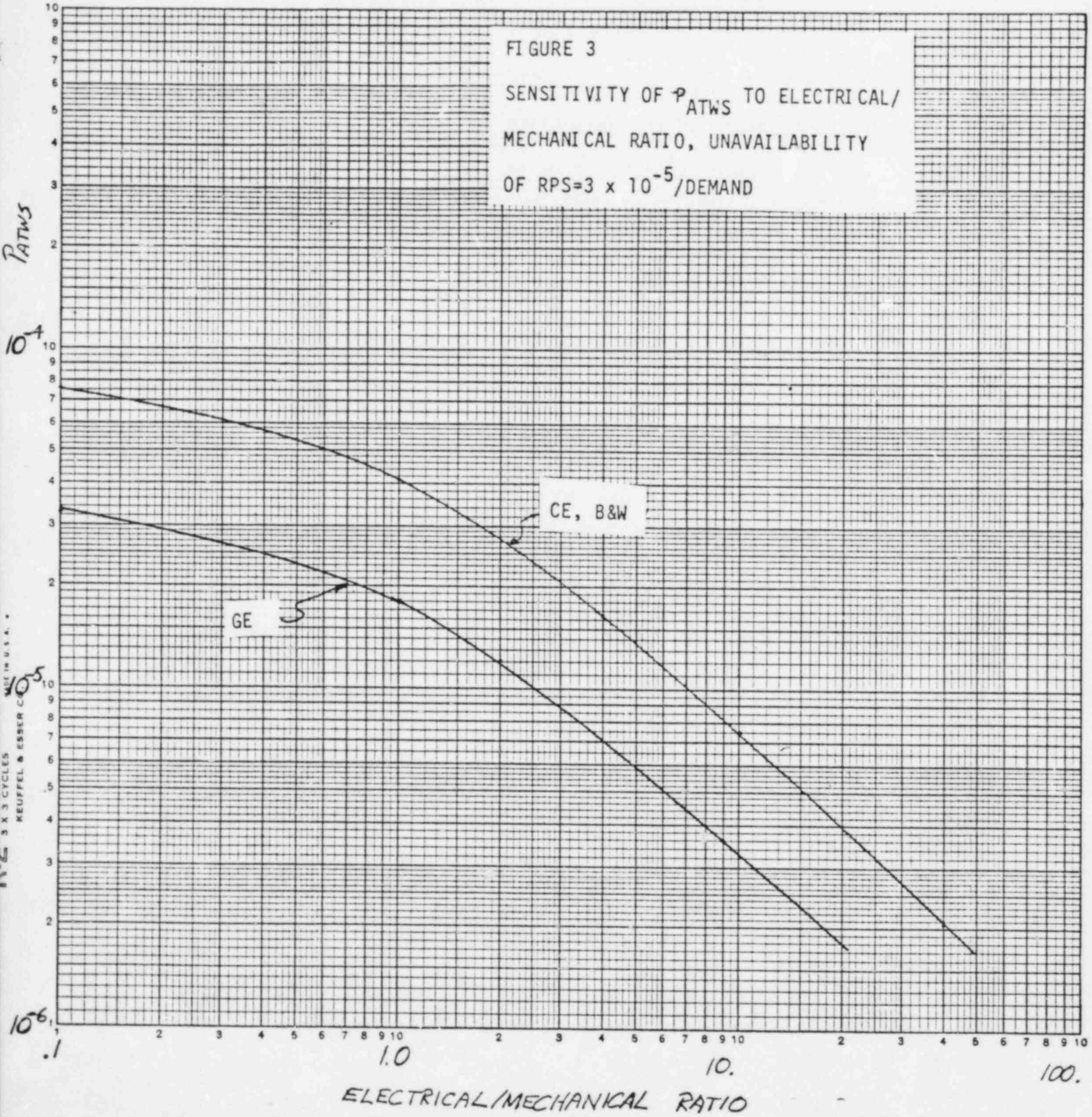


FIGURE 3
SENSITIVITY OF P_{ATWS} TO ELECTRICAL/MECHANICAL RATIO, UNAVAILABILITY OF RPS=3 x 10^{-5}/DEMAND
CE, B&W
GE
P_{ATWS}
10^{-4}
10^{-5}
10^{-6}
.1
1.0
10.
100.
ELECTRICAL/MECHANICAL RATIO
3 X 3 CYCLES
KEUFFEL & ESSER CO.

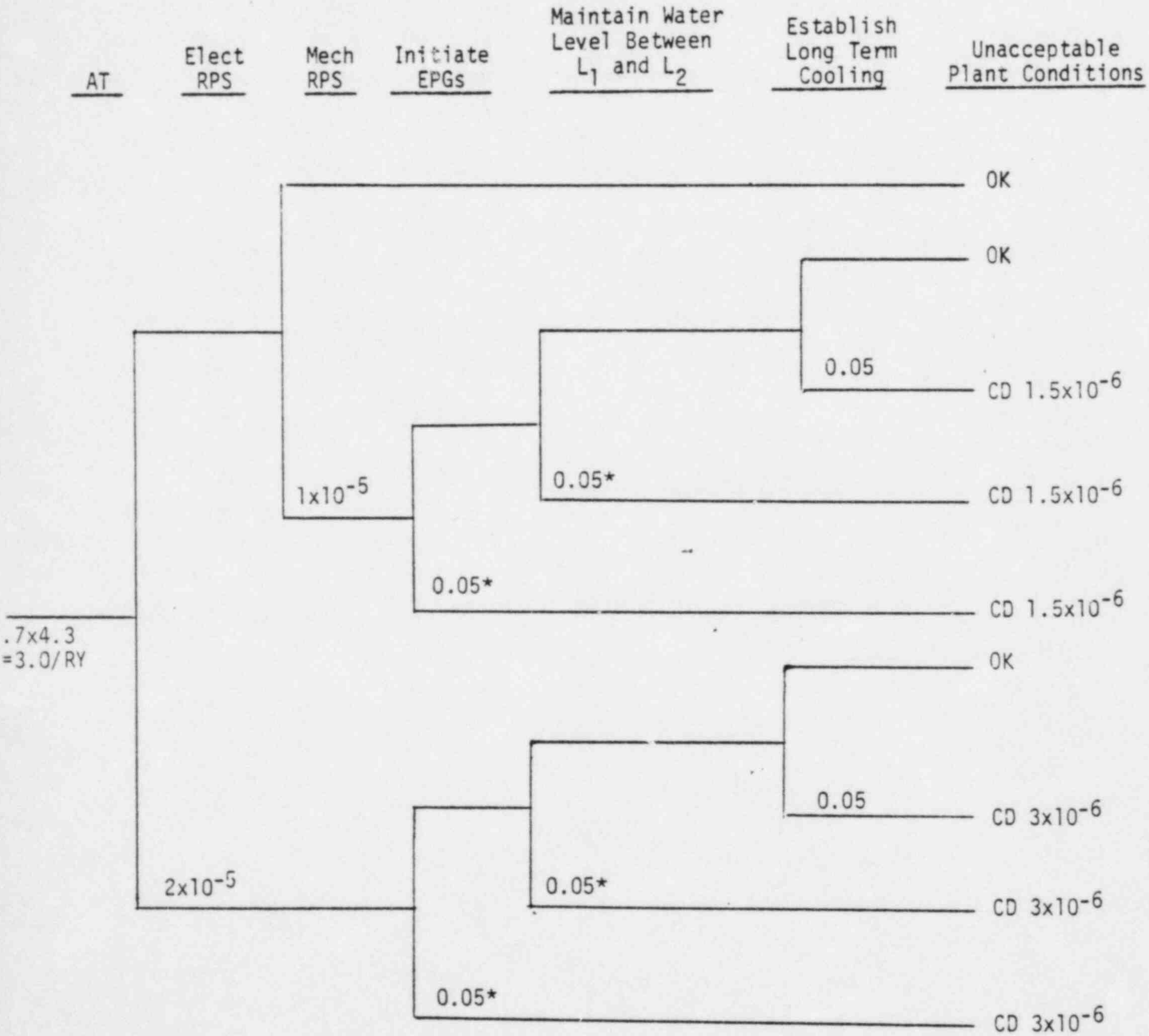
Figure 4

BWR Base Case

Using EPG

L_1 Isolation

Bypass Available



*HEP at approximately 17 minutes

Total $1.4x10^{-5}$

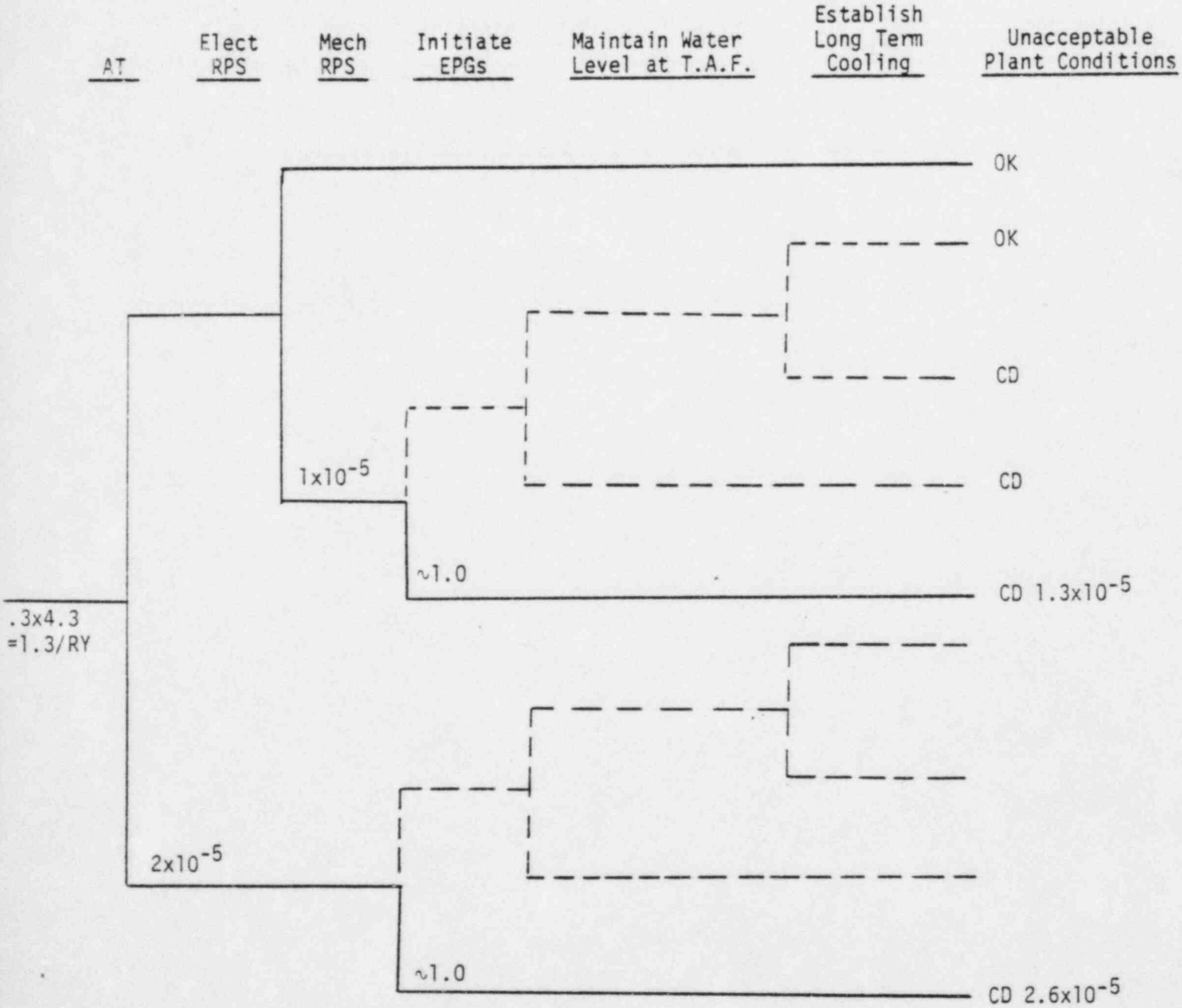
Figure 5

BWR Base Case

Using EPG

L_1 Isolation

Isolation Transients



Total - 3.9×10^{-5}

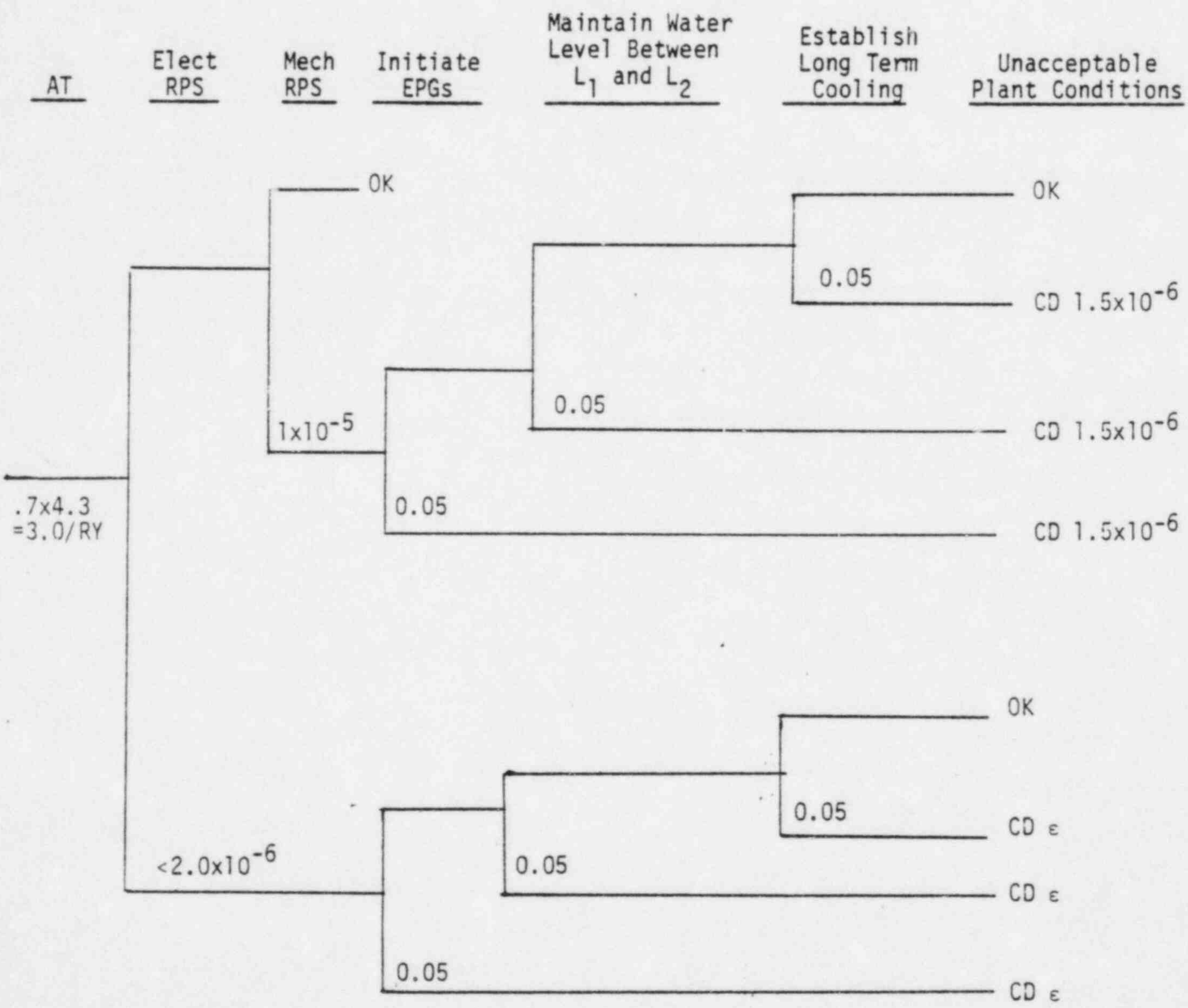
Figure 6

BWR Option 1

Using EPG, ARI, 86 gpm

L_1 Isolation

Bypass Available



Total $4.5x10^{-6}$

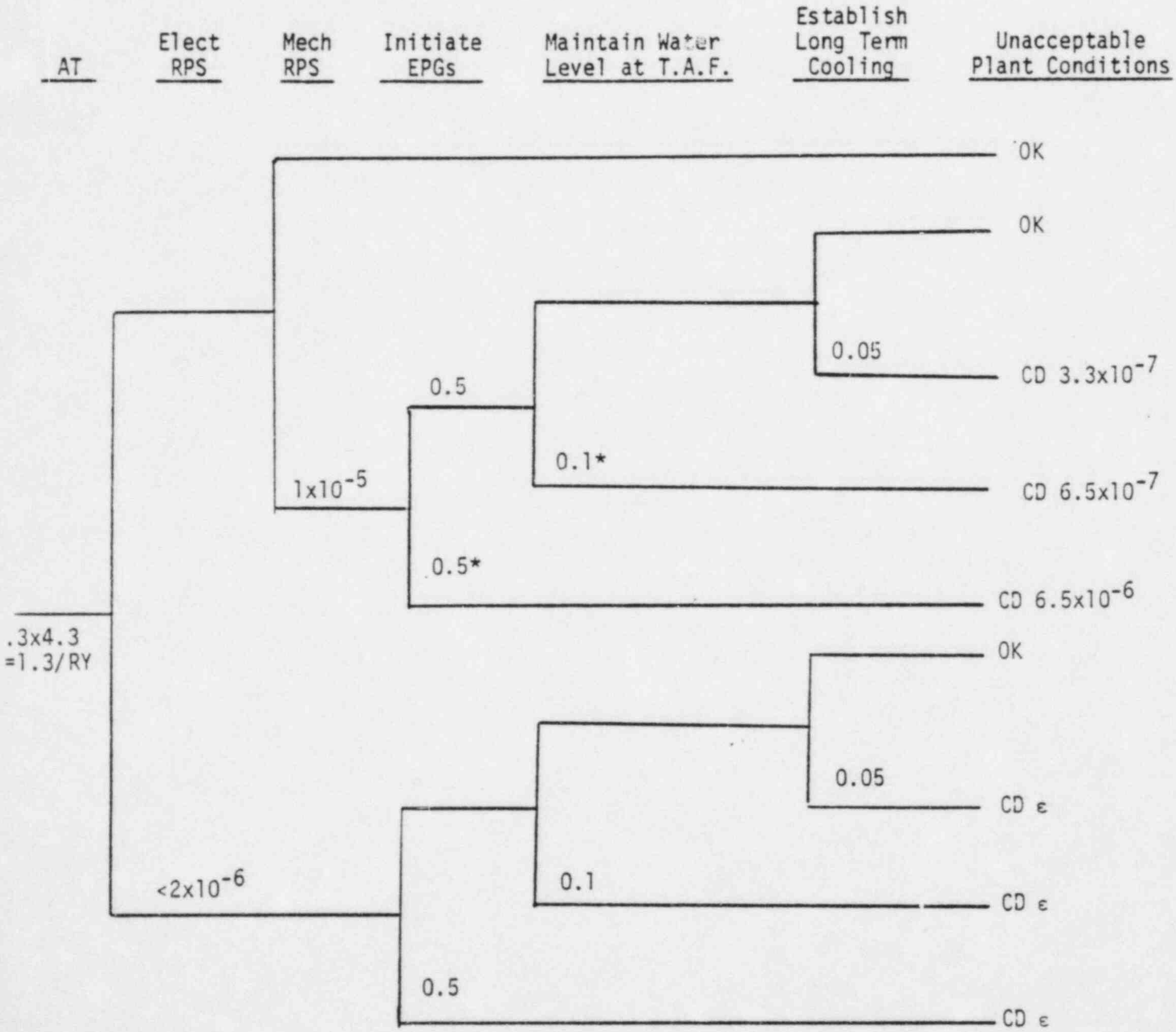
Figure 7

BWR Option 1

ARI, EPG, 86 gpm SLCS

L_1 Isolation

Isolation Transients



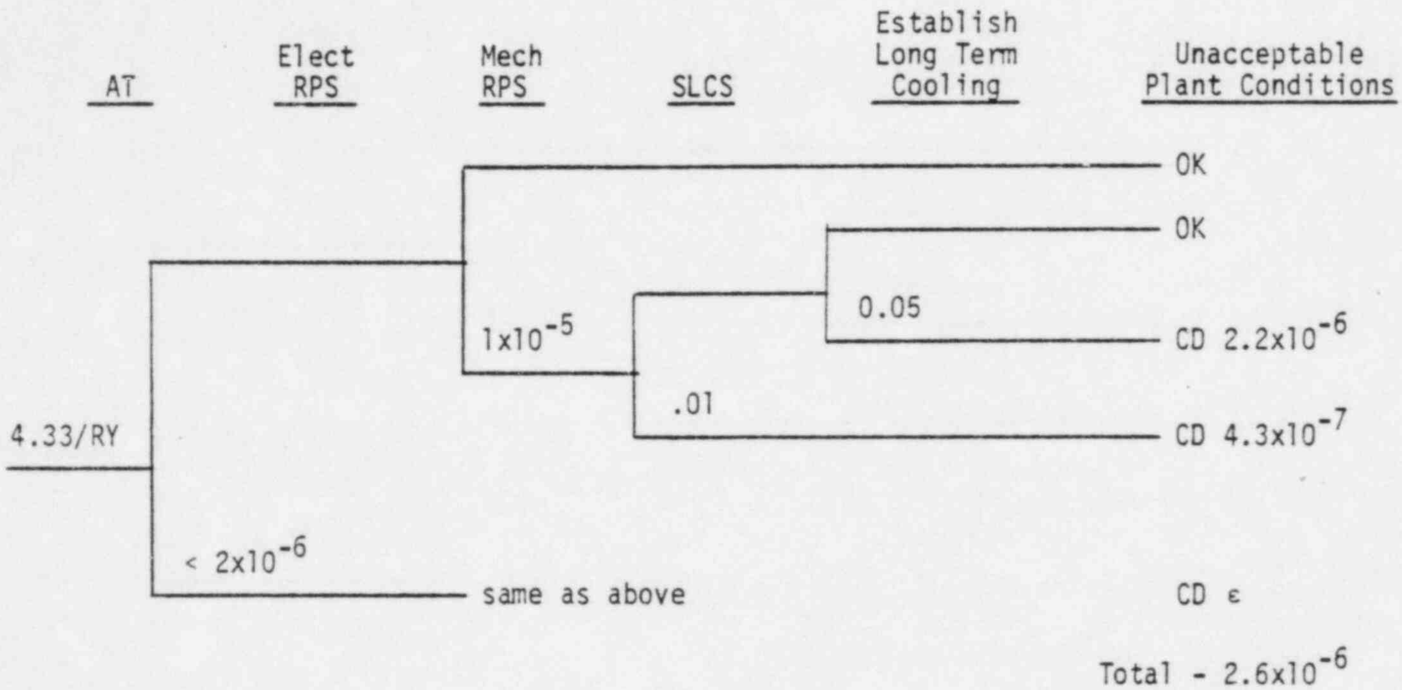
*Human Error Probability Estimates

Total - 7.5×10^{-6}

Figure 8

BWR Option 2

Using EPG, ARI
Automatic 86 gpm* (or equivalent) SLCS



*In order to limit the maximum pool temperature to 200°F, the required SLCS flow could be substantially higher than 86 gpm (up to approximately a factor of 2).

Figure 9

Westinghouse Base Case
Turbine Trip Transients

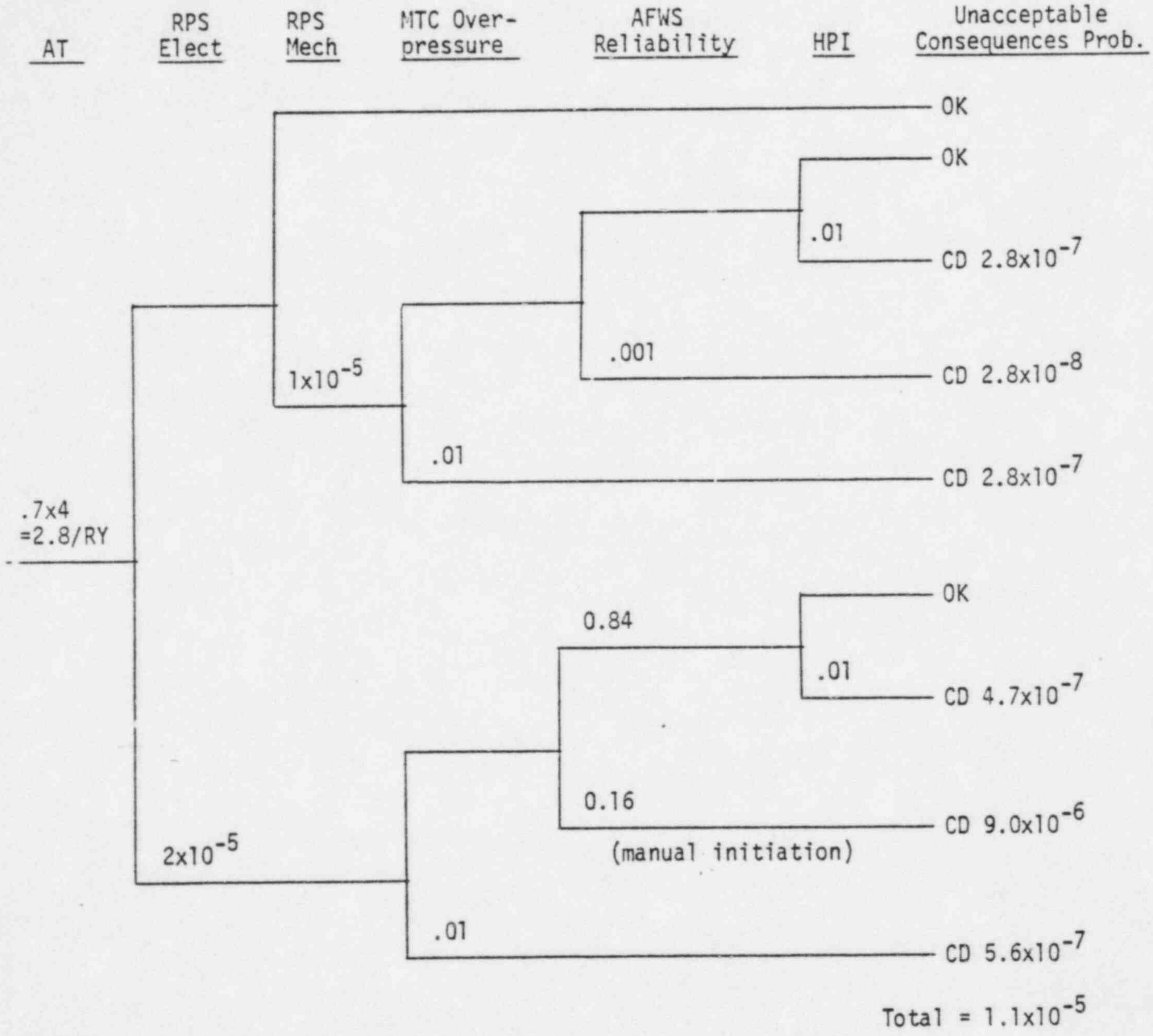


Figure 10

Westinghouse Base Case
Non-Turbine Trip Transients

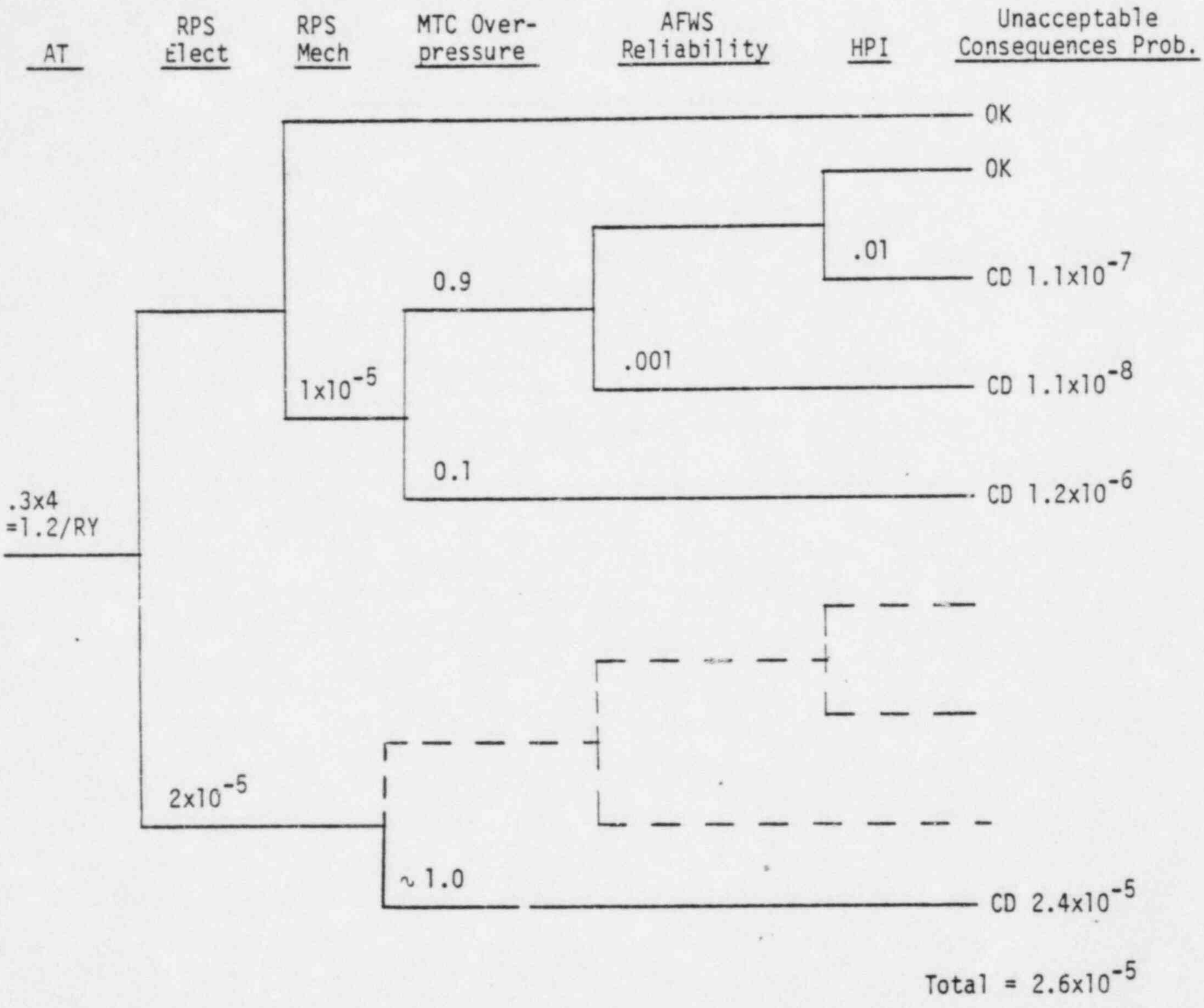


Figure 11

Westinghouse AMSAC (Option 1)
Diverse Initiation of AFW and Turbine Trip
Turbine Trip Transients

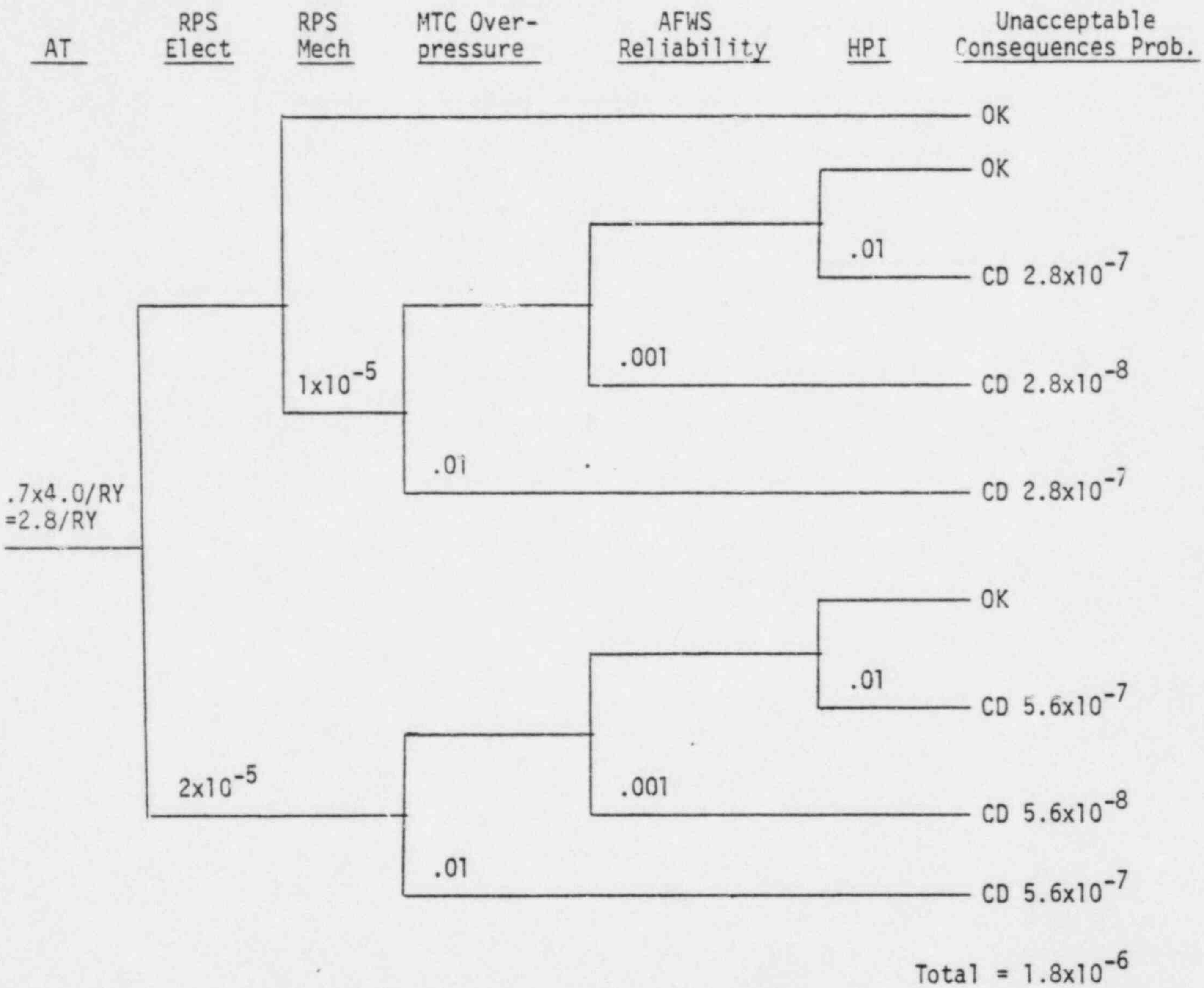


Figure 12

Westinghouse AMSAC (Option 1)
Diverse Initiation of AFW and Turbine Trip
Isolation Transients (Non-Turbine Trip Transients)

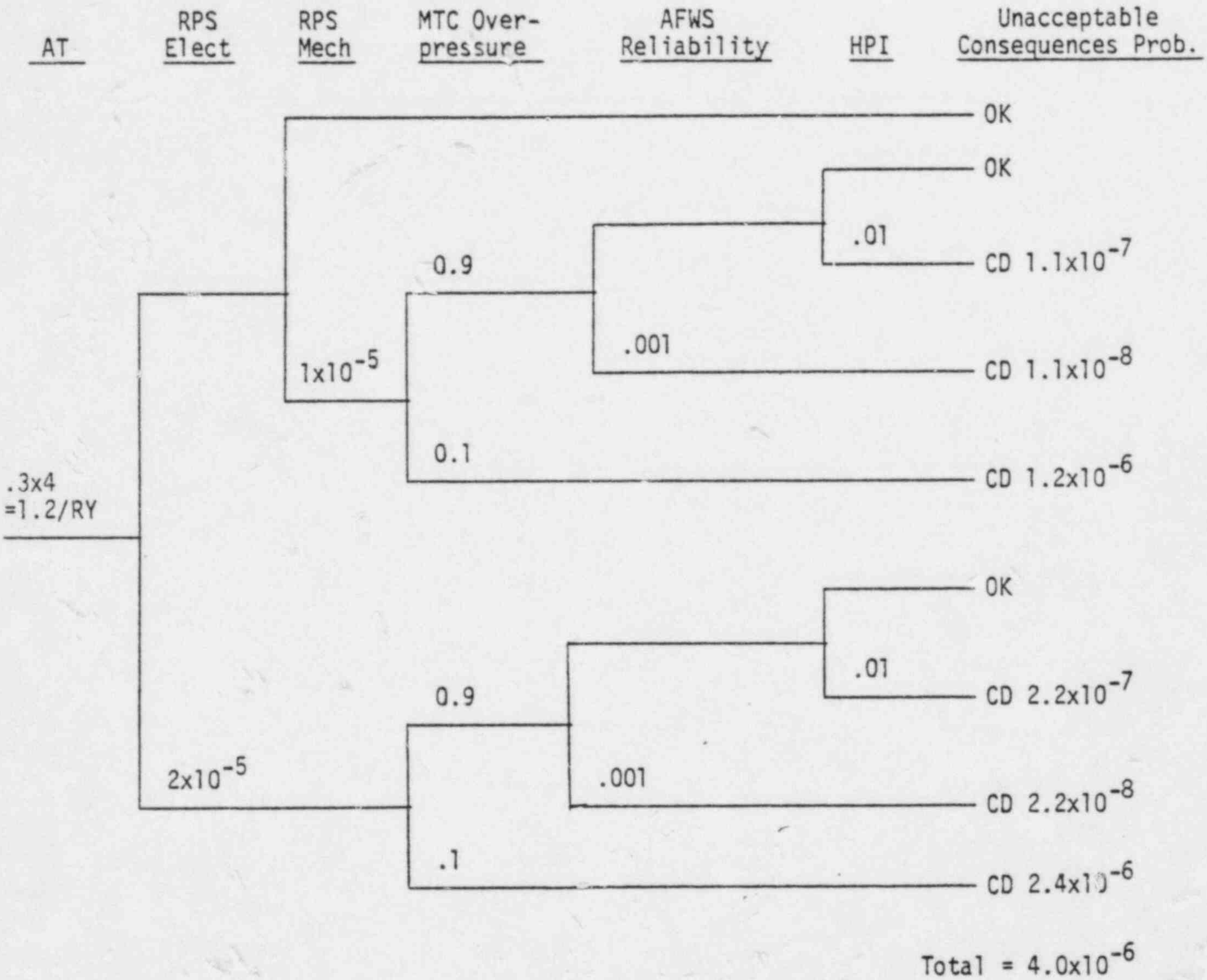
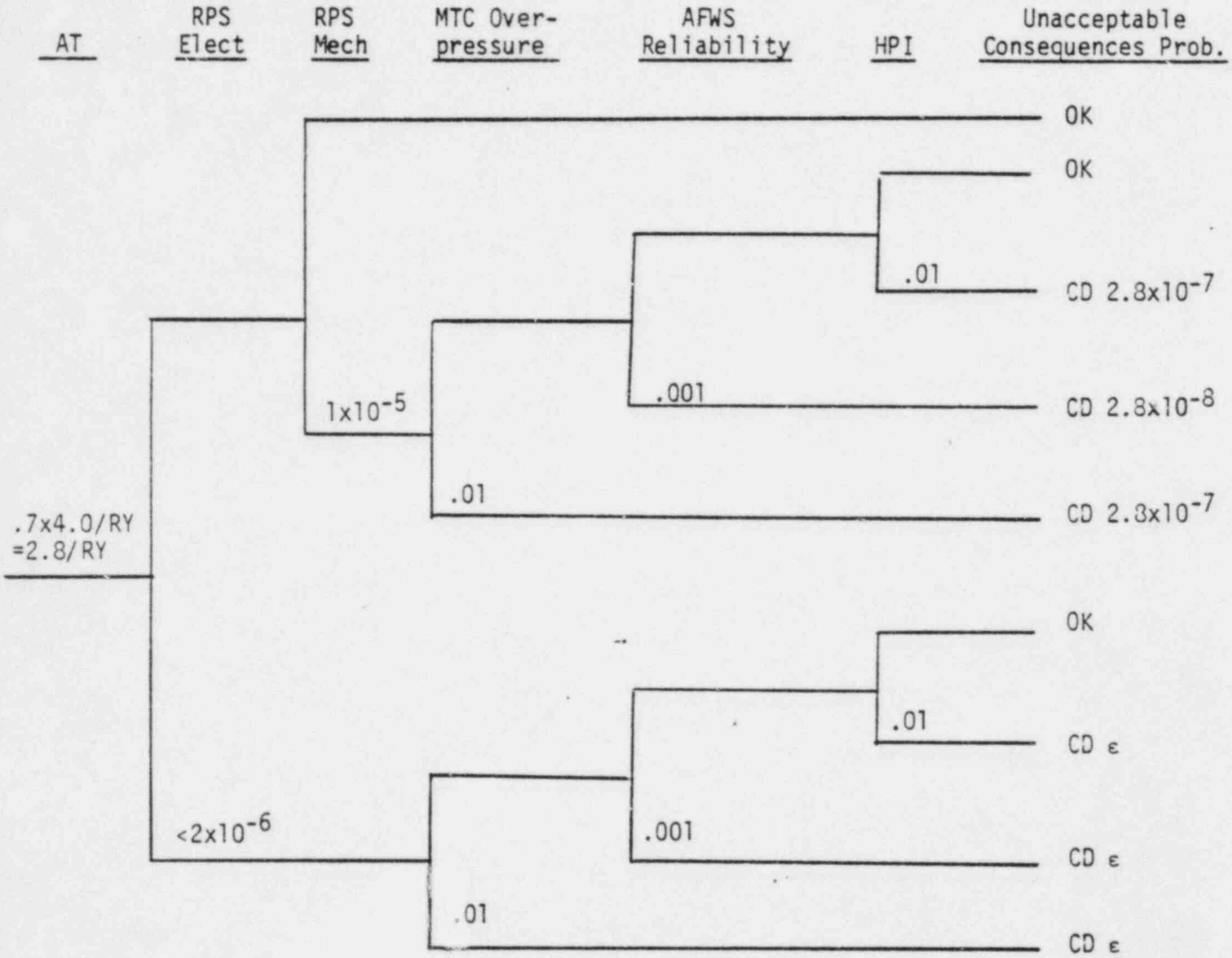


Figure 13

Westinghouse AMSAC (Option 2)
Diverse Initiation of AFW and Turbine Trip
Diverse Scram System
Turbine Trip Transients



Total = 5.9×10^{-7}

Figure 14

Westinghouse AMSAC (Option 2)
Diverse Initiation of AFW and Turbine Trip
Isolation Transients (Non-Turbine Trip Transients)

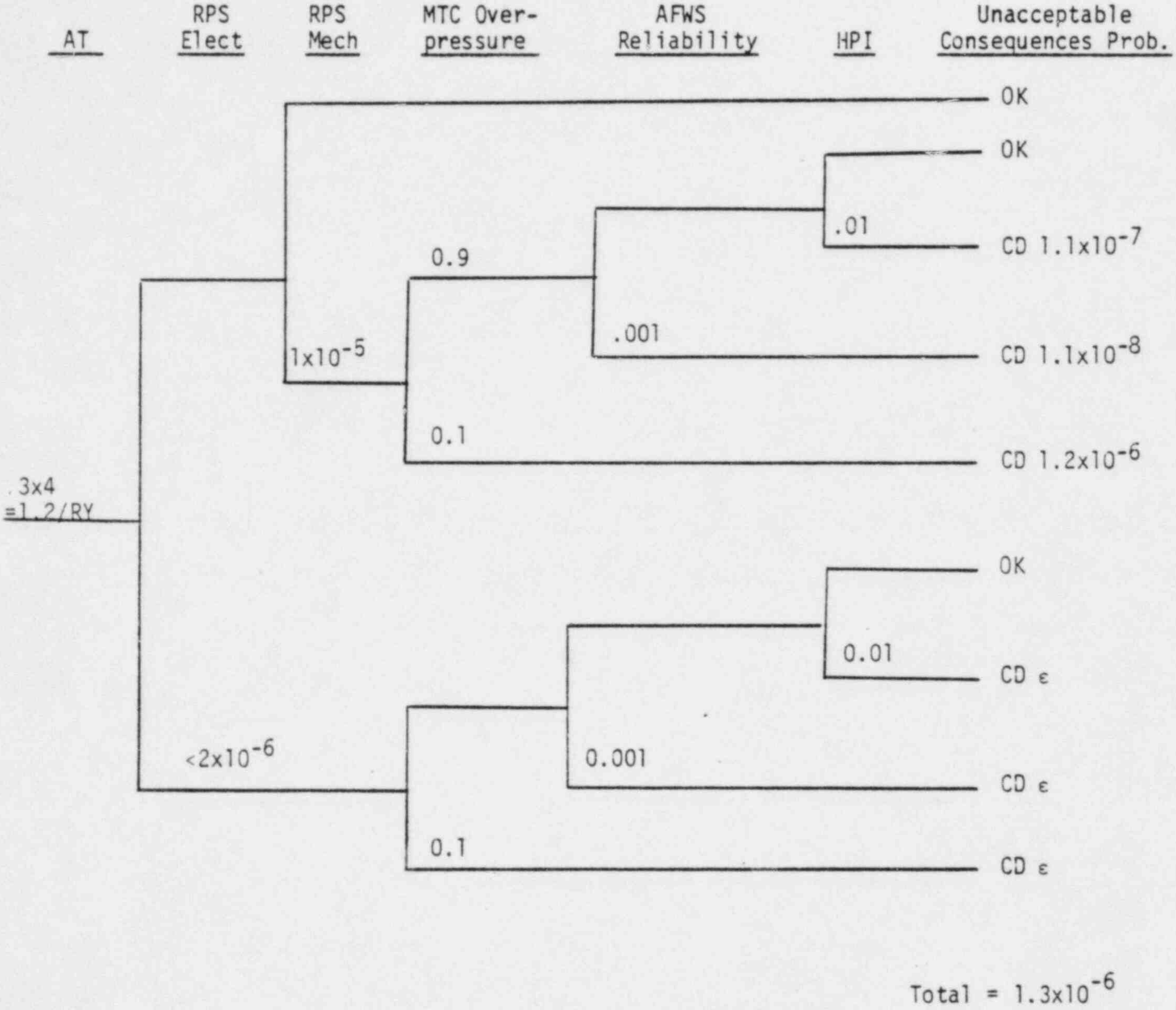


Figure 15

CE/B&W Base Case
Turbine Trip Transients

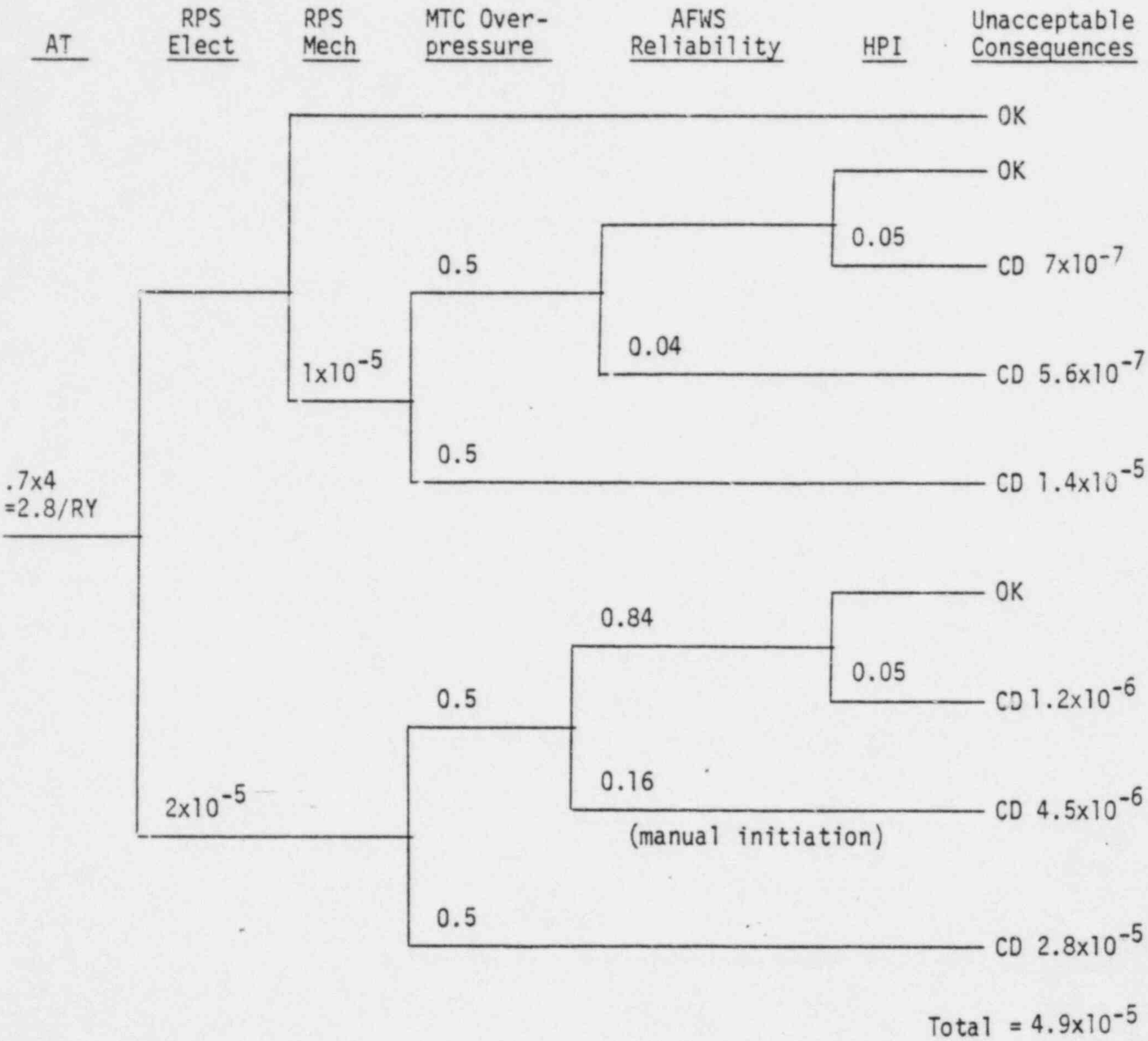


Figure 16

CE/B&W Base Case
Non-Turbine Trip Transients

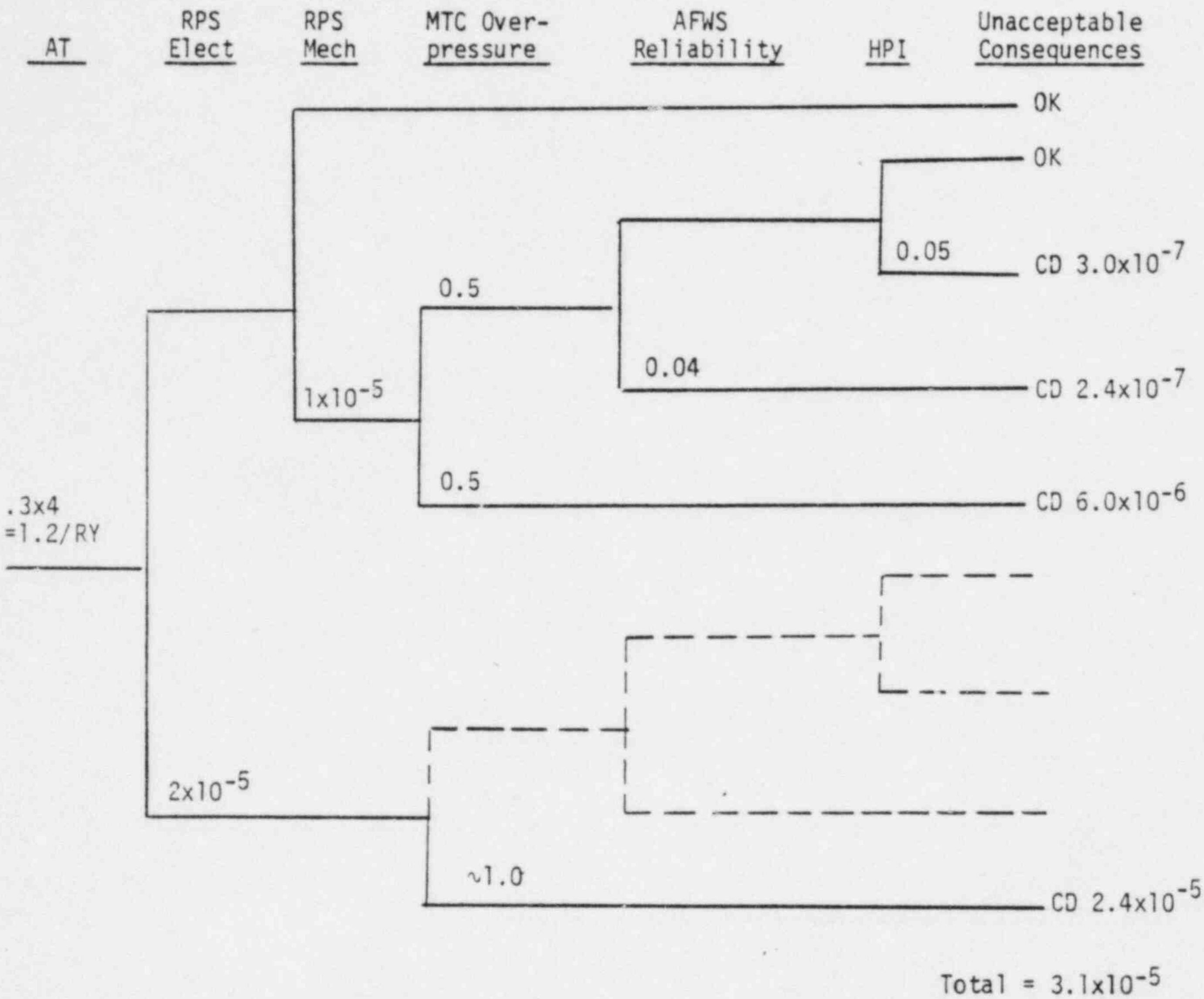
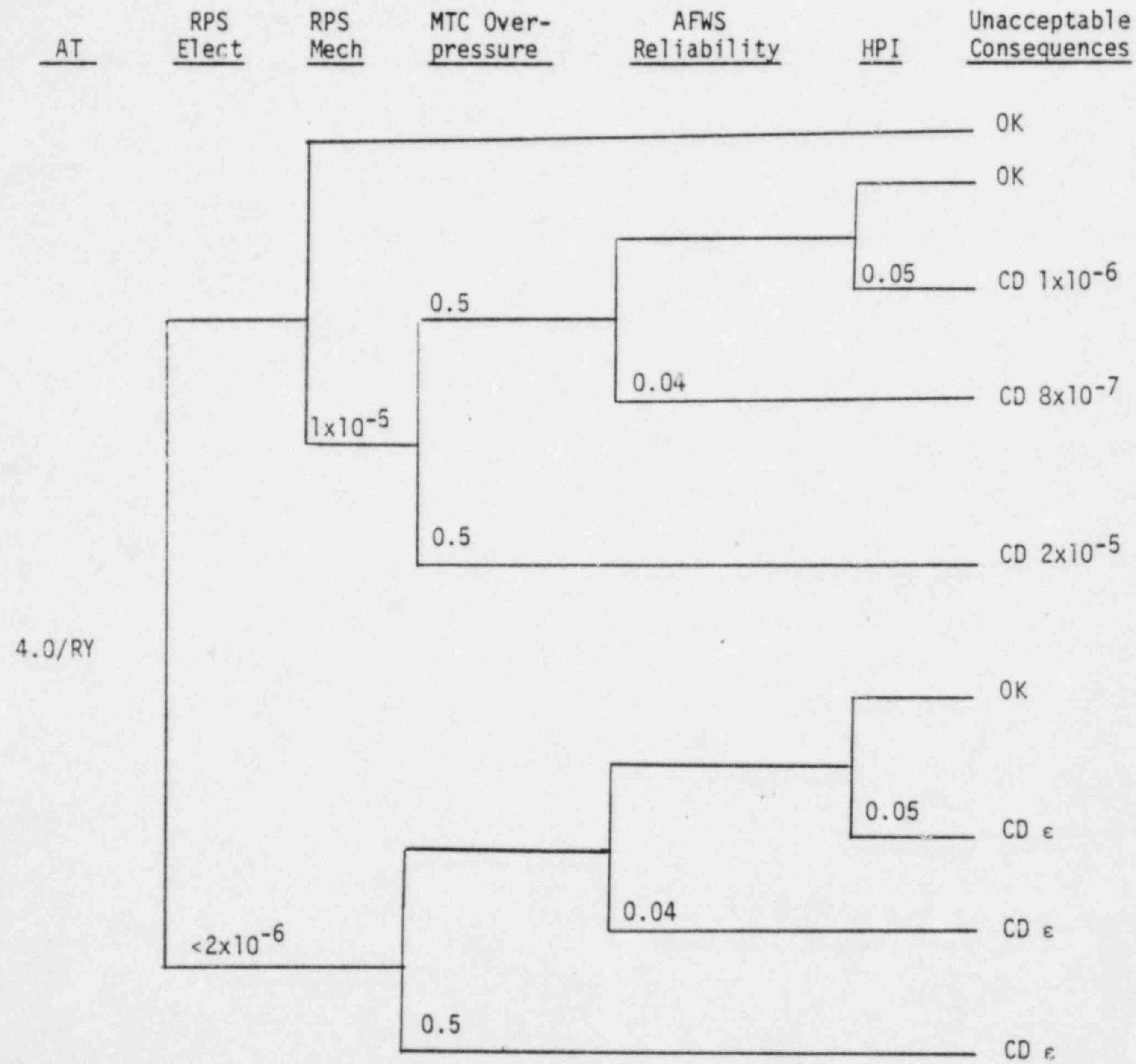


Figure 17

B&W/CE

Option 1

Diverse Scram System Installed
Diverse AFW and Turbine Trip Installed
(Utility Proposal)



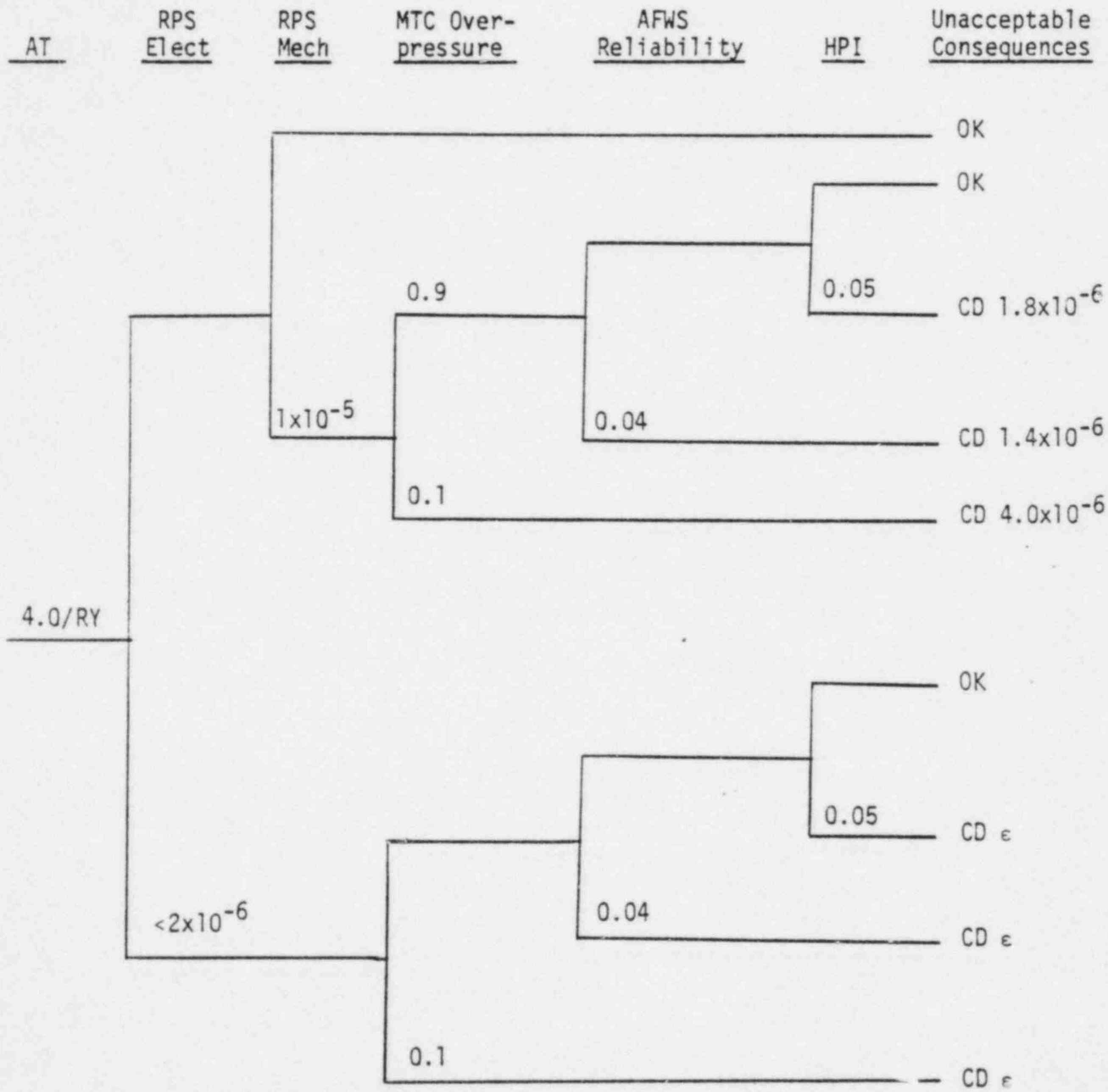
Total = 2.2×10^{-5}

Figure 18

B&W/CE

Option 2

Diverse Scram System Installed
Diverse AFW and Turbine Trip Installed
Extra Safety Valves Installed



Total = 7.2×10^{-6}

APPENDIX A

SCRAM RELIABILITY

Appendix A - Scram Reliability

Tables

Figures

Appendix A - Scram Reliability

A.1 - Analytical Approach

The Salem-1 common cause failure of both scram breakers on Febraury 25, 1983 prompted the ATWS Task Force to reevaluate the bases for the RTS unavailabilities used in the ATWS rule value-impact evaluations. Past reliability analyses did not generally quantify the common cause failure contributions of redundant components. Such quantification was and remains difficult because of the lack of good data on common cause failures. Past RTS reliability analyses concentrated on the independent component failure modes and for this reason simply did not foretell the importance of the ATWS precursors that have actually occurred. Thus, these past analyses should not be the sole source of reliability estimates. This appendix uses a different approach to the problem. It bases the RTS unavailabilities on worldwide experience to date. It is believed that this gives a reasonable estimate of RTS unavailability that includes the common cause contributions that are believed to dominate. The experienced based values are distributed across the four vendor designs based on a comparative reliability analysis that evaluates the major differences among the designs.

A.2 - Definition of Scram Success

For the purposes of this analysis a special definition of scram success has been adopted based on the concerns of overpressurization of the PWR primary loop or the overheating of a BWR suppression pool. PWR overpressurization can be prevented by a relatively few control rods successfully inserting whereas suppression pool overheating requires average power of the core to be controlled to low levels to allow timely recovery actions. Thus, scram success is defined as the insertion of one-half or more of the control rods into the core in a roughly checkerboard pattern. Under most situations this will be sufficient to shut down the reactor to hot standby conditions. This success criteria is compatible with the ATWS event trees used by the ATWS task force to analyze the reactor's tolerance to an ATWS event.

A.3 - Summary of Findings

Table A-1 reports the results of the analysis. The base case designs are the current designs without AMSAC or diverse scram and without automatic shunt trip coil actuation of the scram breakers for Westinghouse or B&W. The second set of numbers on the table report the adjustments to the base case designs given that the automatic shunt trip coil is added to Westinghouse and B&W. Adding the automatic shunt trip coil to B&W reduces the unavailability by only a small amount because the design is not highly dependent on breaker failure rate. Given that an automatic shunt trip coil is added to the Westinghouse design, as a short term requirement in response to the Salem Task Force findings, then the unavailabilities as determined by this appendix generally support the use of the generic $3x10^{-5}$/demand value used for the base case ATWS value-impact analyses. When a diverse trip system is added, the electrical contribution to RTS unavailability is reduced below 10^{-5}/demand; thus, the use of $1x10^{-5}$/demand for the diverse trip case is also generally supported by this appendix.

TABLE A-1

WORLDWIDE RTS EXPERIENCE
DISTRIBUTED ACROSS VENDORS

RTS UNAVAILABILITY ON DEMAND

	ELECTRICAL	MECHANICAL	TOTAL
BASE CASE			
GE	2E-5	3E-5	5E-5
W	2E-4	1E-5	2E-4
B&W	3E-5	1E-5	4E-5
CE	2E-5	1E-5	3E-5
SHUNT ON AUTO			
W	5E-5	1E-5	6E-5
B&W	3E-5	1E-5	4E-5
DIVERSE TRIP			
GE	*	3E-5	3E-5
W	*	1E-5	1E-5
B&W	*	1E-5	1E-5
CE	*	1E-5	1E-5

* = < 1E-5

A.4 - Experience Based, Non-Specific Scram Reliability

Commercial nuclear power plants have accumulated over 1100 reactor-years of operation through 1982. This includes both domestic and foreign experience. Excluded from this experience base are military, production, and USSR reactor units. From the early 1960's to the present, there have been four events that constitute potential failure to scram; these involve events at the Kahl, Monticello, Browns Ferry-3 and Salem-1 plants. This section describes these events, categorizes them according to whether they involve the electrical or mechanical portions of the RTS and calculates the experience based point estimates for RTS reliability.

A.4.1 - Kahl (Germany)

In 1963 a BWR of GE design experienced what could be termed a precursor to an ATWS. During a periodic surveillance test all or almost all of the scram relays failed. Only 2 weeks previously, the original relays had been replaced by a complete new set. The replacement relays were of foreign manufacture. The factory which produced them had only recently begun production. The relay failures were the result of inadequate heat curing of the protective coating on the contacts. During the 2 weeks the relays were in place, the environment they experienced during the operation of the plant softened the coatings and fused the contacts closed. This resulted in a situation where if a genuine scram had been required, it would not have occurred. The relays were part of the electrical portion of the RTS; thus, this event is classed as an electrical failure.

A.4.2 - Monticello

In 1971 this BWR was observed to have a number of relays which had failed in a manner similar to those which failed at the Kahl reactor. A number of differences, however, make this situation less qualified to be termed a failure. The number of relays which had failed was only a relatively small percentage of the total number of relays. The failure was the result of poor quality assurance during manufacture rather than being a design deficiency as was the case at Kahl. Most importantly, the failure was discovered during pre-operational testing before the plant went on line. This observed situation was examined by the NRC staff in NUREG-0460 and was discounted as a failure. For these reasons it has not counted as a failure in this analysis.

A.4.3 - Browns Ferry-3

On June 28, 1980 the Browns Ferry-3 BWR was to undergo a supposed routine shutdown, but of the 185 control rods, 76 (all on the east side of the core) failed to fully insert. The scram discharge volume (SDV), which receives water from the hydraulically powered control rod drives during a scram, reached a high-level bypass and was allowed to drain somewhat before a second manual scram was attempted. This drove 17 more rods home and advanced the other 59 part way. After another SDV high-level and draining, another scram was tried; 12 more were driven fully. The remaining 47, however, remained partially withdrawn until an automatic scram occurred; this scram was triggered by a SDV tank high water level signal when the scram reset switch was pointed to "normal." The last 47 rods then settled into place, about 15 minutes after the first attempt to scram. Core coolant flow, temperature, and pressure remained normal throughout. As the defect was part of the control rod mechanical system, this event is classed as a mechanical failure.

A.4.4 - Salem-1

On February 22, 1983 and again on February 25, 1983 this Westinghouse PWR experienced failures of its RTS scram breakers. Event #1 (February 22) occurred when the plant was at 20 percent power and in the process of transferring from offsite to onsite power. A reactor coolant and a main feedwater pump were lost resulting in a reactor trip signal from a low-low level in the steam generator. A manual trip was initiated nearly simultaneously and the failure of the scram breakers went undetected. On February 25 event #2 was initiated while the reactor was at 12 percent power and the generator synchronized with the grid. Again a low-low level in a steam generator initiated a reactor trip signal but this time the manual scram was not immediately performed. When the operators observed that plant parameters were not consistent with a scram, the reactor was manually shut down. The failure of the RTS to automatically scram the reactor was determined to be due to the failure of the scram breakers. The two scram breakers failed due to a common cause failure of improper lubrication and maintenance resulting in the binding of the undervoltage coils. This event was classed as a failure in the electrical portion of the RTS system.

A.4.5 - Experience Based RTS Unavailability

The point estimates of the RTS unavailability are summarized on Table A-2 using worldwide, U.S., and vendor-based experience. It is apparent that both time related and demand related failure mechanisms have contributed to RTS component breakdowns. For the purpose of this analysis, the number of failures have been divided by the estimated number of demands. This assumes demand related mechanisms are at work. Actually, in the case of the RTS system where the number of demands from testing are comparable to the number of demands from transients, the numerical values for unavailability using the demand related approach are comparable, but slightly higher, than if the time related approach was used. The 5 percent and 95 percent confidence bounds and the point values for the combined electrical and mechanical unavailabilities are calculated on work sheets found at the end of this appendix.

A.5 - Electrical Design-Base Case Comparison

The function of the electrical portion of the RTS is to generate an electrical signal that will activate the mechanical portion of the RTS to insert the control rods and shut the reactor down. The electrical portion includes a number of sensor channels, logic subsystems, and scram breakers (in case of a PWR) or scram solenoids (in case of a BWR) that cause the rods to insert. This section will discuss the important differences between the vendor designs and will establish a quantitative measure of the comparative reliability between the vendors based on failure experience, PRA perspectives,

TABLE A-2

Experience Based Estimates of RTS Unavailability*
(per demand)

	Electrical	Mechanical
Worldwide		
LWRs - 497 ry foreign through 12/82 631 ry U.S. through 3/83	8.7×10^{-5} two failures	4.3×10^{-5} one failure
United States		
LWRs - 631 ry	7.9×10^{-5} one failure	7.9×10^{-5} one failure
Vendors		
General Electric - 263 ry	-0-	1.9×10^{-4} one failure
Westinghouse - 249 ry.	2.0×10^{-4} one failure	-0-
Babcock & Wilcox - 59 ry	-0-	-0-
Combustion Engineering - 61 ry	-0-	-0-

*Point estimates based on number of failures divided by the estimated number of demands (12 test + 8 transients per reactor year).

and the degree of common cause sensitivity inherent in the base case designs. The base case designs exclude AMSAC and separate diverse trip systems for all vendors, and automatic shunt trip coil for Westinghouse and B&W. Section 3.0, "Reactor Trip System Issues," of NUREG-1000 should be referred to for further details of the RTS designs.

A.5.1 - Sensor Channels

Sensor channels for all vendor designs consist of several redundant channels for each reactor operating parameter important for the protection of the reactor. A reactor scram is initiated when trip signals from redundant sensor channels are determined by the logic subsystem to satisfy predetermined coincidence conditions. For instance, in a 2-out-of-3 system that has three channels for each reactor parameter, a trip will occur when two or more channels are activated. Though there are differences in the sensor channel design from vendor-to-vendor, all of the designs exhibit a comparable degree of diversity in the sensor portion by virtue of parameter selection and sensor design. The likelihood of an important common cause failure occurring in this part of the sensor system is judged to be low because of the variety of sensor types. Thus, if a required protective signal is not generated by the redundant channels of a given parameter, then another parameter is driven off-normal and produces the necessary signal. The trip portion of the sensor system consists of bistables that signal an out-of-tolerance condition. This portion of the system is vulnerable to bistable calibration errors and like component common cause failures. However, continuous monitoring of the sensor output, and the frequent testing of the trip values provide a good chance of discovery of such common cause problems. In addition, bistable failure on demand is reported to be in the order of 10^{-7}/demand which makes them a most reliable electronic component. Couple this low failure rate with the large number of sensed parameters and one must judge that the controlling common cause problems would probably not reside in the sensor subsystem, though human error such as miscalibration or valving closed sensor ports should not be ignored. In fact, the possibility of calibration errors was studied in the Reactor Safety Study (WASH-1400). This analysis selected the low pressure comparators and the overtemperature Δt comparators and estimated system failure based on miscalibration assuming the system was composed of only these two sensor types. System failure above 10^{-5}/demand was estimated. However, the results of the analysis were not included in the final RPS failure on demand value because of the number of other reactor trip parameters which were believed not to be closely coupled (in terms of human calibration error) with those included in the analysis.

From a vendor comparison viewpoint, the sensor portion of the RPS design appears to be comparable in terms of common cause susceptibility. Though differences exist in the level of redundancy and logic structure, these only influence the independent failure contribution which does not contribute significantly to the overall RPS unavailability. Therefore, for the purposes of this analysis, the sensor portion of the RTS will be ignored.

A.5.2 - Logic Subsystem

Though there is reasonable functional diversity in the sensor portion of the RTS, this is not the case in the logic subsystem. The transmitters, amplifiers, logic matrices, and relays that make up the logic subsystems do have redundancy to some degree, but generally lack diversity. The PRA's conducted to date generally have not quantified the contribution to unavailability caused by the possible common cause influences on the logic subsystems. The failure rates of these components are low and multiple failures are rare, although multiple failures caused by such influences as temperature degradation for certain logic components have been reported. Failures in these components are generally not announced at once and must await surveillance testing. In addition, comparator adjustments and calibrations can introduce human error.

The logic subsystem incorporates a number of logic channels, two in the case of Westinghouse and four in the case of the other vendors. A logic channel can be tripped by any one of six or more sensor channel inputs. NUREG-1000 provides a description of each of the vendor's designs and their variations. The important failure modes caused by the logic design will be discussed in general terms below.

A.5.2.1 Two Channel Logic System

The Westinghouse solid state protection system uses a two channel logic design. Such a design can be vulnerable to dependent faults which can result in a failure to scram. In addition, during maintenance or testing of the logic or breakers, a single channel must be depended on to trip the rods, thus maintenance and testing outage times are of importance and significantly contribute to RTS unavailability. Though the SSPS uses relays for its input trip signals and for the safeguards actuation outputs, it does not use relays to power the breaker undervoltage (UV) coils. Thus, relay faults can be neglected.

A.5.2.2 - Four Channel Logic System

The three other vendors use four channel logic. Though the arrangements of components differ from one vendor to another and also within a vendor group, the major types of faults are essentially the same. For instance, in a four channel system there are two combinations of double channel failures that can prevent trip. Thus, the contribution to unavailability of both the independent and dependent channel failures are twice as high as for a two channel system. However, a four channel system allows testing and maintenance of a single channel while it is in a tripped state without causing a reactor trip. If a trip signal is received during test or maintenance of a single channel, a trip of any one of the other three channels will cause reactor scram. Thus, there is essentially no test or maintenance outage contribution to unavailability. All of the four channel designs use relays to actuate the UV or shunt coils of the breakers (PWRs) or solenoids (BWRs) and thus relay faults should be considered.

A.5.3 - Scram Breaker (Scram Solenoid) Subsystems

The methods used to trip the rods vary greatly among the vendors. The PWR's use electromagnetic devices to hold the rods withdrawn. These reactors require a method that interrupts the electrical current to the electromagnets allowing the rods to fall by gravity into the reactor. The BWR's use a hydraulic system that upon a trip signal forces the rods into the reactor. The hydraulic system is activated by venting instrument air pressure via solenoid operated air pilot valves. Table A-3 summarizes the primary and secondary means for tripping the rods. The trip design of each vendor will be discussed in turn, after which a comparative reliability estimate will be presented for the entire electrical portion of each RTS design.

A.5.3.1 - General Electric (Refer to Figure A-1)

Each rod has two normally energized pilot valves, one controlled by relays actuated by trip logic channels A and C, the other by trip logic channels B and D.* When deenergized, these valves bleed away instrument air pressure moving hydraulic valves into their scram position. Both pilot valves must operate to result in rod insertion. Since each rod is individually controlled, there are about 360 of these normally energized pilot valves and 720 scram contactor relays in the typical BWR. To cause an ATWS serious enough to result in an unacceptable plant condition, multiple failures of a relatively large number of the pilot valves or relays must occur. The IREP generic experience data indicates a failure rate of 10^{-3}/demand for solenoid valves and 10^{-4}/demand for relays. Multiple failures of valves/relays are thus rare (there has been one instance where two rods failed to insert during operation because of failure of pilot valves). The independent failure contribution for multiple pilot valve failures is small. Common cause failures, such as a

*An alternate design uses a single pilot valve with two solenoids that requires both solenoids to deenergize before the valve changes state.

Table A-3

BREAKER SUBSYSTEM CONFIGURATIONS

	PRIMARY BREAKERS			BACKUP TO BREAKERS
	NUMBER	BREAKER FAILURE ON DEMAND	CUTSET FORM	
GE[a]	2 per rod (∿360)	1E-3[b]	Multiple Combinations	2 Slow Acting Pilot Valves
W	2	4E-3[c]	A·B	None
B&W	4[d]	4E-3[c]	2(A·B)[e]	Silicon Control Rectifiers
CE	8	1E-3[c]	4(A·B·C·D)	None

NOTES:

a. GE uses solenoid operated instrument air pilot valves.
b. Generic solenoid valve failure to operate value used in WASH-1400 and IREPs. Read 1E-3 as 1×10^{-3}.
c. Breaker values from Salem Task Force Report, NUREG-1000.
d. Four in the case of the Davis-Besse design, six for the Oconee design.
e. This cutset form applies only to the safety rod groups. For the Oconee design, the cutset form would be 2(A·B·C).

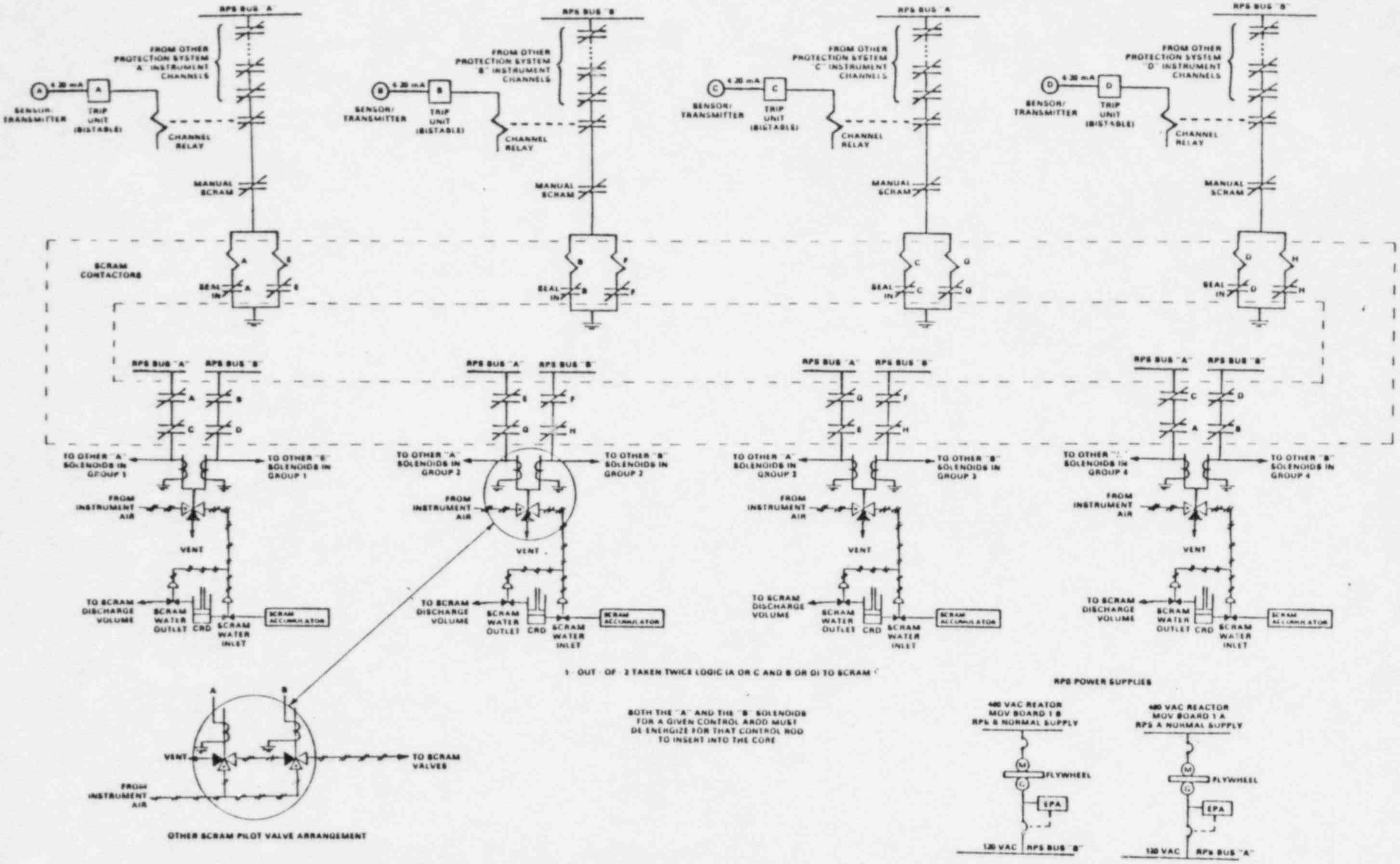


Figure A-1 **General Electric Reactor Trip System (Shown Normally Energized; De-energize to Scram)** contaminated instrument air supply can be hypothesized. In fact, during the construction phase of Grand Gulf, the pilot valve solenoids were exposed to overvoltage that failed a number of the normally energized pilot valves. Though this event occurred during construction and was discovered immediately, it does examplify a type of common cause failure that can defeat a system having very high redundancy.

The GE scram systems have a back-up to the individual pilot valves. Two normally deenergized back-up scram solenoid valves are positioned in the instrument air supply headers. When energized, these valves vent the instrument air from the supply headers, this method is slower acting but constitutes a somewhat diverse means of activating the hydraulic scram valves.

A.5.3.2 - Westinghouse (Refer to Figure A-2)

The Westinghouse PWRs use two scram breakers (DB-50s) in series to interrupt current to the electromagnets that hold the control rods out of the core. The successful operation of either breaker will result in rod insertion. Each breaker is tripped by a separate solid state logic subsystem. For breaker testing during operation, two bypass breakers, normally open, are included. Closing bypass breaker "A" allows the "A" scram breaker to be tested. The "A" bypass breaker is wired to the "B" logic subsystem and will open upon a "B" logic trip.

The Salem common cause RTS failure involved the simultaneous failure of both scram breakers. This event highlights several factors which make a one-out-of-two breaker configuration unreliable. Consider the following:

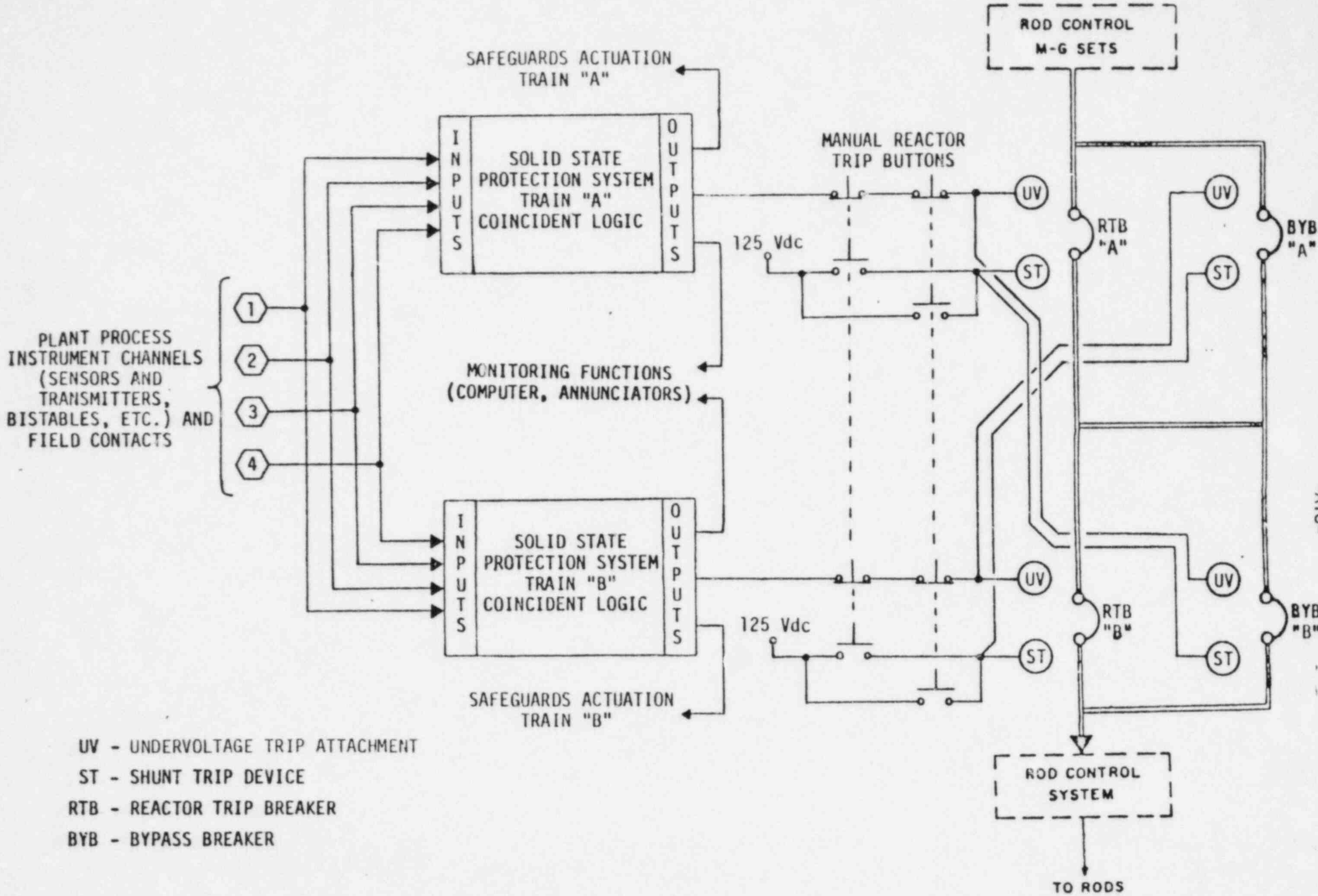


Figure A-2 **Westinghouse Reactor Trip System**

1. The two breaker system is highly sensitive to any degradation in breaker reliability. For instance, the current Westinghouse design uses the undervoltage coil to automatically open the breaker. This configuration has resulted in a $4x10^{-3}$ breaker failure on demand unavailability (see Salem report NUREG-1000). This relatively poor unavailability results in a $1.6x10^{-5}$ RPS unavailability solely from simultaneous, independent breaker failures. Add to this the contribution for partial breaker/logic subsystem failures, and test/maintenance outage contributions and this number is increased by a factor of three or more. Thus a straight reliability analysis without common cause considerations result in unavailabilities of $5x10^{-5}$/demand or higher.

2. The two breaker system has some redundancy but no diversity. Lacking diversity, it is particularly susceptible to common cause failure. Evidence has mounted that the DB-50 breaker suffers from design inadequacies, is sensitive to temperature and grit, requires careful maintenance and adjustments, responds randomly to testing (fails to open in one test but opens on the next or vice versa), and incipient failures cannot be easily detected. Thus, the scram breaker is proven to be a component with a high potential for having significant dependent failure modes which may not be easily detected.

The present Westinghouse breaker subsystem has no additional backup method for removing current from the electromagnets but relies entirely on the two scram breakers.

A.5.3.3 - Babcock and Wilcox (Refer to Figures A-3 and A-4)

The B&W plants use AK-2 breakers as opposed to the DB-50. However, they are used in a similar manner, i.e., a UV-coil automatically activates the breaker to open, they thus have the same failure rates as the DB-50s, based on NUREG-1000. The B&W plants use two distinct breaker subsystem designs. Herein one is called the Oconee design and the other the Davis Besse design.

The Oconee design uses parallel main and secondary buses to supply current to the control rod electromagnets. The safety rods (four groups making up about one-half of the rods) are controlled by four DC trip breakers (all four breakers must be opened to drop all the safety rods). The regulating rods (the other one-half of the rods) are controlled by four sets* of silicon controlled rectifiers (SCR), each set is required to appropriately operate to block current to its regulating rod group. AC breakers, one on the main bus and one on the secondary bus, can interrupt the current to all rods if both open as designed. There are no breaker failure combinations that result in more than half the rods failing to insert. Even if all breakers fail, the regulating rods will still insert as a result of the SCR operation. The cutset forms shown in Table A-3 refer only to failures involving the safety rod groups. The SCRs constitute a diverse means of assuring at least half of the rods insert. The most significant cutsets involving breakers would be a failure to open the main bus AC breaker in combination with the failure of the C logic channel or the failure to open of the secondary bus AC breaker in combination with the failure of the D logic channel. The failure of the logic channels in the above cutsets allow the gate voltage to remain on one half of the SCRs as

*Figures A-3 and A-4 have been simplified, where they show only a single SCR there is actually a set of six.

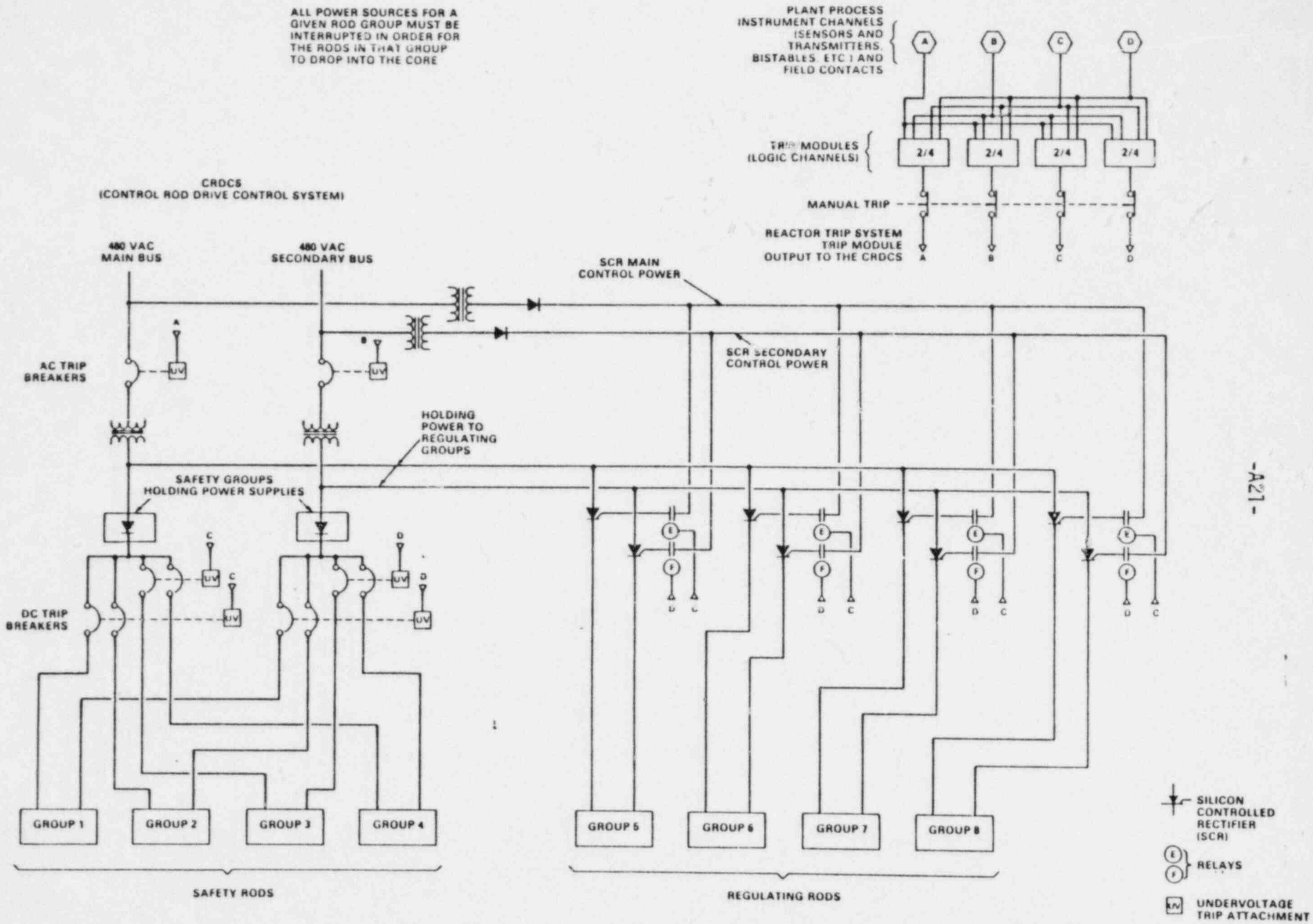


Figure A-3 Babcock & Wilcox Reactor Trip System (Oconee, TMI, CR-3, ANO-1, Rancho Seco)

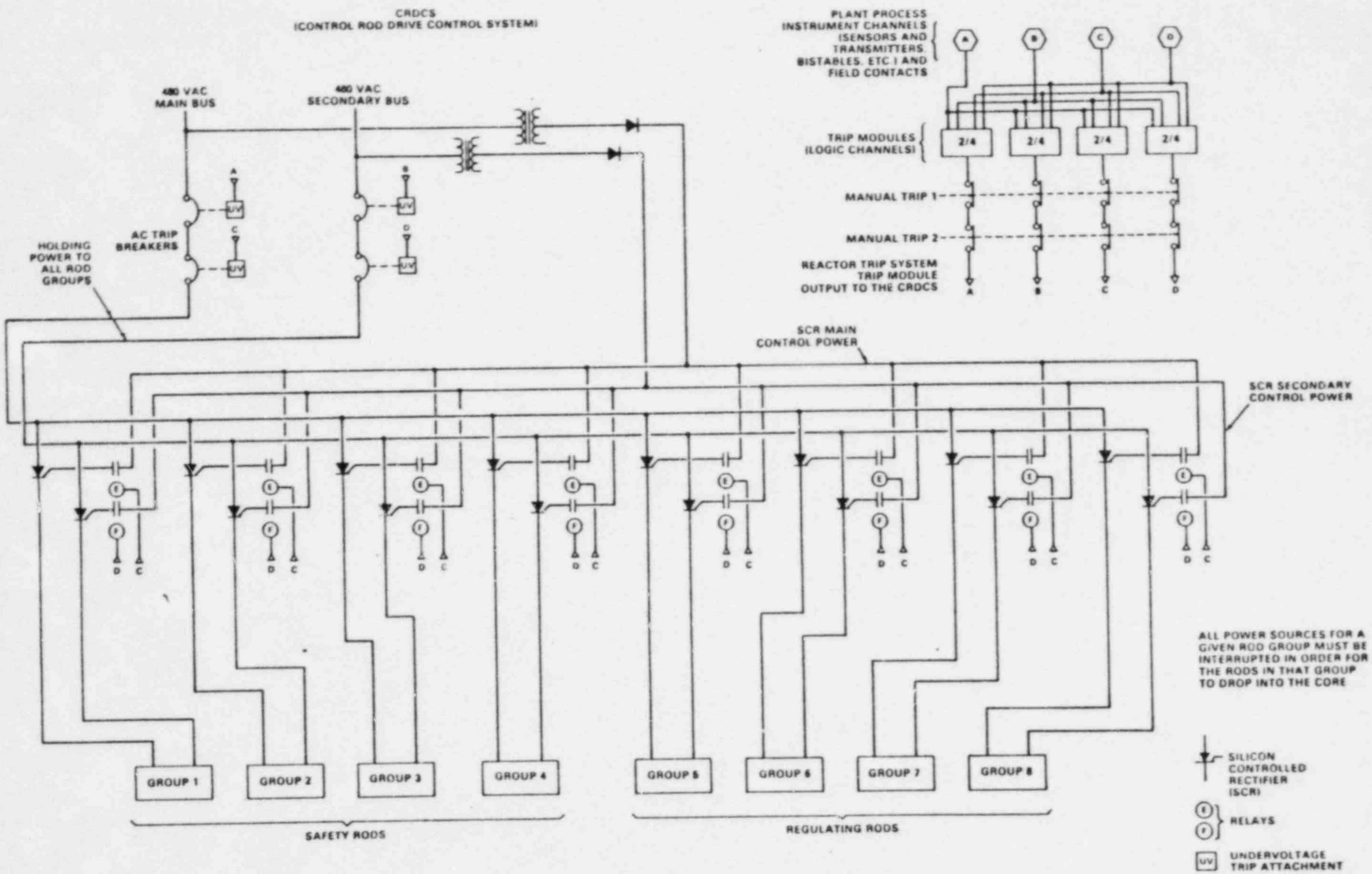


Figure A-4 Babcock & Wilcox Reactor Trip System (Davis-Besse) well as failing to open two DC breakers resulting in the continued supply of current to all rods. As a result of these double cutsets, the failure of the AC breaker are far more significant than failure of any of the DC breakers.

The Davis-Besse design uses a main bus and a secondary bus to supply, in two parallel paths, all of the four safety and four regulating rod groups. Two AC breakers are placed in series on both buses. Each rod group is fed by both the primary and secondary buses with a set of SCRs placed before each group. Though the configuration is somewhat different from the Oconee design, the significant double cutsets are the same for both designs.

A.5.3:4 - Combustion Engineering (Refer to Figures A-5 and A-6)

The CE plants use AK-2 breakers as in the case of B&W. However, in the CE design the shunt trip (ST) coil is actuated automatically upon a trip signal. This results in the automatic trip of the breaker having higher reliability than when the UV-coil is used alone. Figure A-5 shows the most common CE design while Figure A-6 shows an older design used at Fort Calhoun and Palisades. While there are differences between these designs, the major cutsets remain the same, and for the purposes of a top level comparison of vendor designs, it is proper to assume that reliability analyses of these designs will yield nearly the same results. Only the design in Figure A-5 will be discussed.

Referring to Figure A-5, note that each trip logic channel actuates a trip breaker control relay. This relay in turn controls the UV and ST coils of two AK-2 breakers. The electrical buses supplying current to the control rods

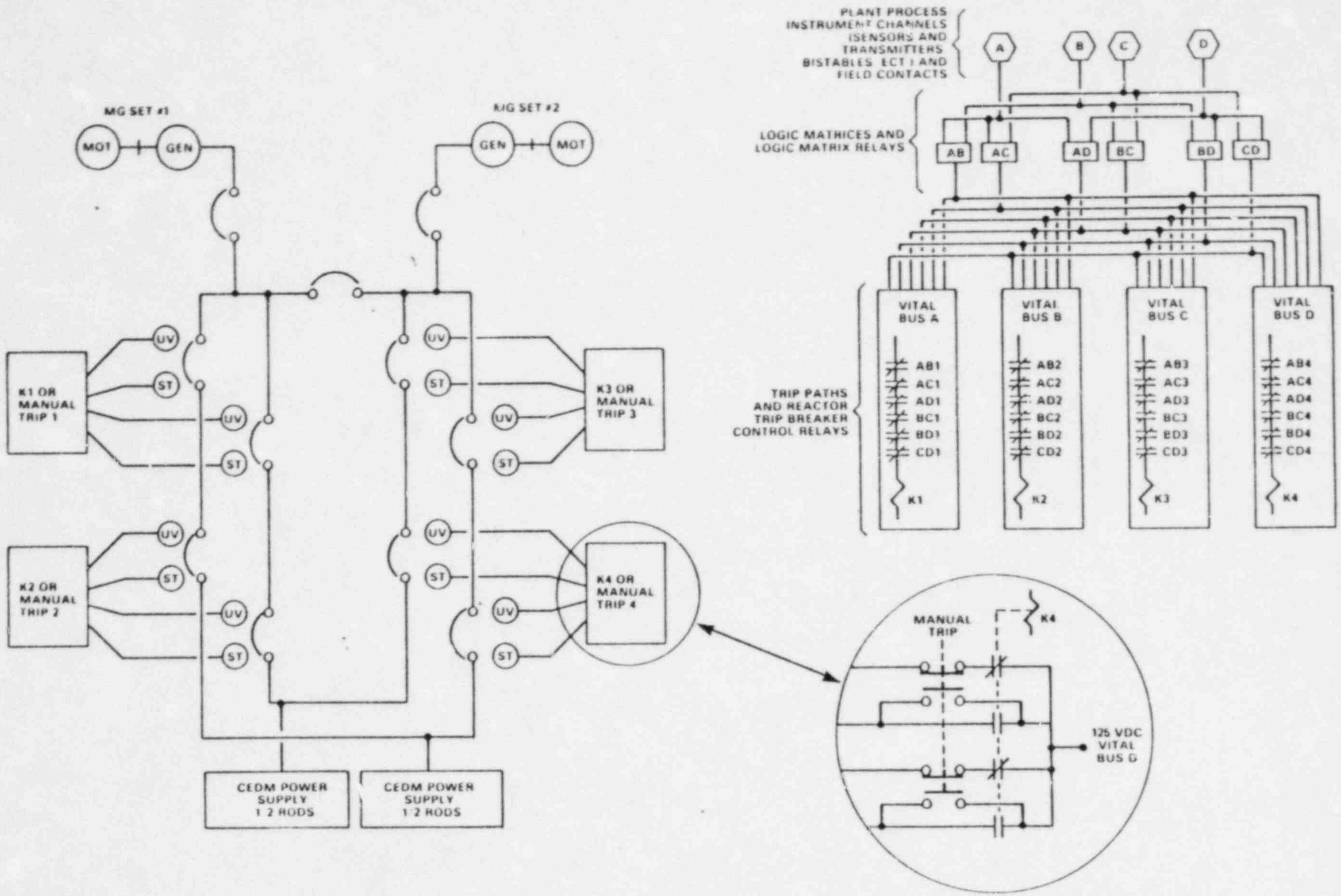


Figure A-5 **Combustion Engineering Reactor Trip System**

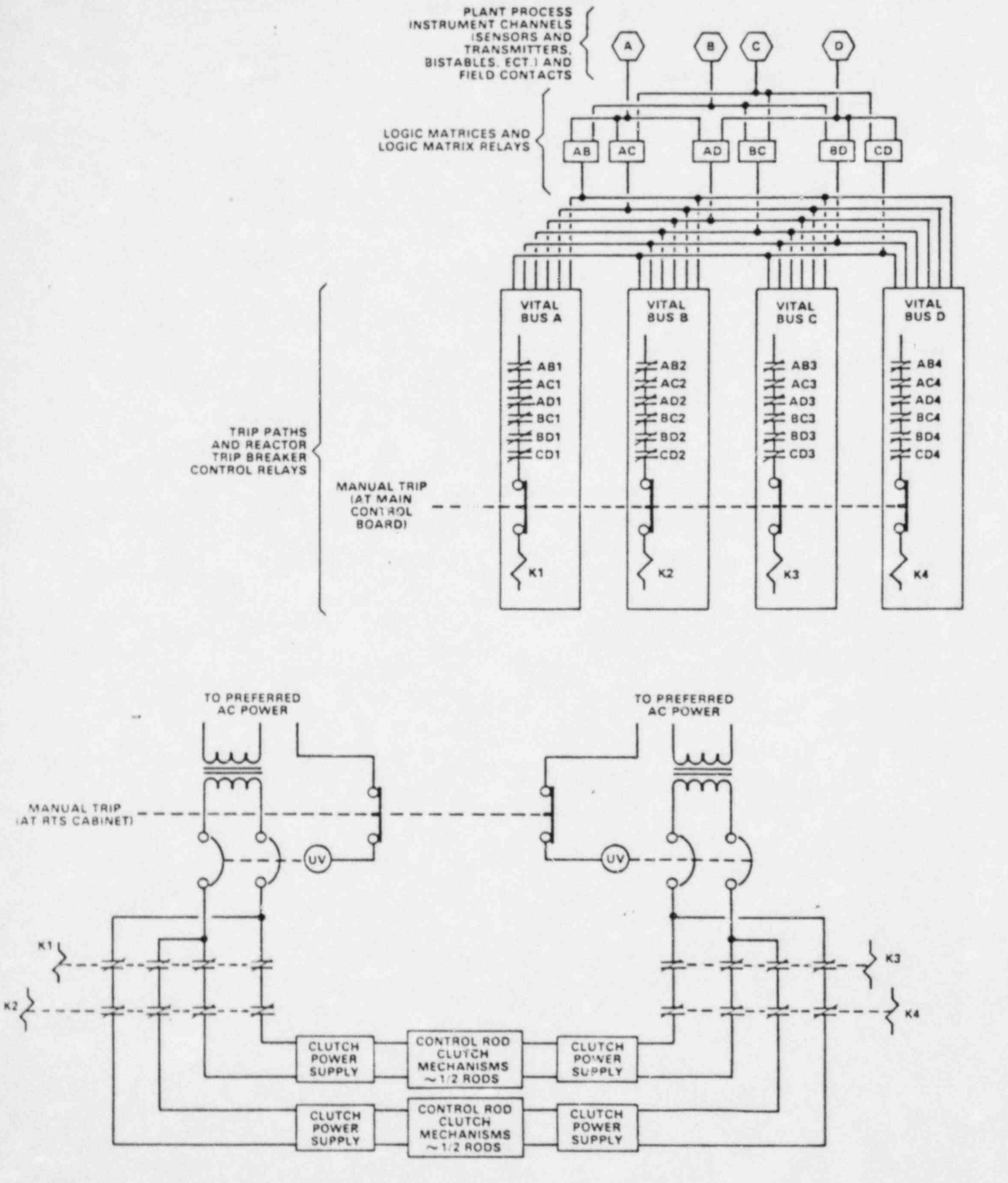


Figure A-6 Combustion Engineering Reactor Trip System (Fort Calhoun and Palisades) are set up in two paths. Each path supplies current to one half of the rods. Each rod group is supplied current by two sources via the MG sets #1 and #2. These two sources are interrupted by the AK-2 breakers as shown, i.e., two breakers in series in each path. The important cutsets are the failures of logic channels A and B or C and D. Relay failures K1 and K2 or K3 and K4 will give similar results. For the breakers to result in a full ATWS would require a failure of four selected breakers (four combinations are possible). Of course there are combinations of channel, relay, and breaker failures that would also yield a full ATWS.

A.5.4 - Comparative Analysis of Electrical Unavailability

A comparison of the electrical unavailabilities was conducted for the four major vendor designs. This comparison used generic failure rates of logic subsystems, relays, and breakers. It also accounted for maintenance and testing outages. The analysis is of a simplified nature stressing the influence of breaker reliability and treats common cause affects using beta factors. The absolute values of unavailability should not be considered highly accurate; nevertheless, they are judged to be usable for comparison among the designs. Table A-4 summarizes the failures on demand used in the analysis and Table A-5 summarizes the results of the analysis. The analyses show that the base case designs for B&W, CE, and GE have approximately the same unavailability where as Westinghouse, by virtue of the common cause failure potential for DB-50 breaker, has a higher unavailability. The comparative factors appearing on the table were used to distribute the worldwide experienced based RTS unavailabilities across the four vendors; this result is shown in Table A-1. The equation used to distribute this worldwide unavailability is as follows:

Table A-4

Failure Rate Values Used in Comparative Analysis
(These are rough generic values used to make
comparisons between the four vendor designs)

INDEPENDENT	PER DEMAND
Breaker	
Without Auto Shunt	4E-3
With Auto Shunt	1E-3
Logic Channel	1E-3
Relays	1E-4

DEPENDENT	2 FAILED	4 FAILED	BETA FACTORS 2nd	3rd	4th
Breaker					
Without Auto Shunt	2E-4	--	5%		
With Auto Shunt	2E-5	4E-7	2%	10%	20%
Logic Channel	1E-5	--	1%		
Relays	5E-6	--	5%		

MAINTENANCE

8 hr per year/8760 = 9E-4 outage probability per breaker or logic channel

TEST

1 hr per month/730 = 1.4E-3 outage probability per breaker

8 hr per 2 months/1460 = 5.5E-3 outage probability per logic channel

TABLE A-5

COMPARATIVE TOP LEVEL ESTIMATE OF RTS ELECTRICAL UNAVAILABILITY ON DEMAND [x10^{-5}]

	Westinghouse		Babcock & Wilcox		CE	GE
	w/o auto shunt	with auto shunt	w/o auto shunt	with auto shunt	with auto shunt	
Independent Cutsets						
Logic Channels	0.1	0.1	0.2	0.2	0.2	0.2
Breakers	1.6	0.1	ε	ε	ε	N/A
UV Control Relays	ε	ε	ε	ε	ε	ε
Logic/Breaker Combinations	0.8	0.2	0.8	0.2	ε	N/A
Other Combinations	ε	ε	ε	ε	ε	ε
Dependent Cutsets						
Logic Channels	1.0	1.0	2.0	2.0	2.0	2.0
Breakers	20.0	2.0	ε	ε	0.2	N/A
UV Control Relays	N/A	N/A	1.0	1.0	1.0	ε
Maintenance and Testing Outage			ε	ε	ε	ε
Maintenance	0.4	0.4				
Logic Testing	5.5	2.2				
Breaker Testing	0.3	0.3				
TOTAL [x10^{-5}]	30.0	6.3	4.0	3.4	3.4	2.2
Comparative factors used to distribute experienced based unavailability across base case designs	14.0	N/A	1.8	N/A	1.5	1.0

NOTES: (1) Success is defined as commanding one-half or more of the control rods to insert.

(2) ε denotes a negligable or zero value.

$$\overline{Q}_i = C_i \overline{Q}_{ww} (\Sigma Y/\Sigma CY)$$

where:

$\overline{Q}_i$ = averaged experienced based unavailability on demand for ith vendor

C_i = comparative factor for ith vendor

$\overline{Q}_{ww}$ = worldwide experienced based unavailability on demand

ΣY = sum of U.S. vendor reactor years

ΣCY = sum of product of comparative factors and reactor years over all vendors

A.6 - Mechanical Design-Base Case Comparison

The mechanical portion of the RTS includes the control rods and associated components that assist in the physical insertion of the rods. Past BWR risk analysis have shown that the possibility of multiple independent failures resulting in less than half the rods inserting can be ignored from a reliability standpoint. Thus, multiple failures of the magnetic clutches in PWRs or severe hydraulic seal failures in BWRs or such similar independent failure modes are unlikely to contribute significantly to overall RTS unavailability. Only subtle common cause failures, such as the scram discharge volume failure at Browns Ferry, are likely to be significant.

This incident is the only significant precursor involving the mechanical portion of the RTS. The BWRs appear to be somewhat more susceptible to common cause failure than the PWR as they require the scram discharge system as well as having a complex hydraulic design. Until a reevaluation is possible as a result of considerably more experience data or until an extensive common cause failure analysis is conducted, no detailed comparison of the vendors is possible. For the purposes of this analysis it was assumed that all PWRs are comparable while the BWR's common cause mechanical contribution was assumed to be 3 times that of the PWR's. The distributed worldwide experienced base unavailability for the mechancial portion of the RTS is found on Table A-1. These numbers were calculated using the equation given in Section A.5.4. To account for the substantial revision of the scram discharge system design as a result of the Browns Ferry event, the worldwide experienced based unavailability was reduced by a factor of two.

A.7 - Addition of Automatic Shunt Trip Coil

The current Westinghouse and B&W designs use only the undervoltage coil to trip the scram breakers. The high failure rates of these breakers have been associated with the marginal design of this coil and its associated linkage. The CE design, on the other hand, uses both the UV and ST coils to actuate the breaker; thus, it is a much more reliable method. NUREG-1000 reports a four fold increase in reliability with the addition of the ST coil. The actual beneficial effect may even be higher than this since the breaker failure data have been conservatively interpreted. The use of the ST coil also enhances the diversity of the breaker trip mechanism reducing the potential for breaker common cause failure. Table A-5 reports the estimated change in RTS electrical unavailability of the Westinghouse and B&W RTS with the addition of automatic shunt trip. The Westinghouse values are significantly affected because of its high sensitivity to a two breaker common cause failure whereas the B&W design is only moderately affected since a breaker failure must be combined with logic or SCR failures to result in an ATWS.

A.8 - Addition of a Diverse Trip System

The top level estimates of RTS unavailability used in this analysis are based on a simplified RTS model which treats the logic channels as supercomponents which have a moderate degree of common cause vulnerability. This results in dependent logic channel cutsets having unavailability contributions of 10^{-5}/demand or greater. This is not an unreasonable approach given the current RTS experience base and the lack of a comprehensive common cause analysis of the RTS system. Any attempt to accurately quantify the common cause influences, at this time, will no doubt result in controversy. However, the addition of a diverse trip system designed to scram the reactor under conditions that, given an ATWS, would result in unacceptable plant conditions, e.g., in the case of a PWR low feed flow or low steam generator level, would significantly reduce the possibility of common cause failure. If such a system is truly diverse such that it is independent from the RTS system, then there is solid reason to believe that the common cause susceptibility of the current RTS design would be eliminated. With installation of such a diverse scram system, the electrical contribution to RTS unavailability will be less than 10^{-5}/demand as indicated in Table A-1.

Work Sheets
Experience Based Estimates of RPS Unavailability

1. Total Worldwide LWR RPS Experience

   3 failures (Kahl, Browns Ferry, and Salem)

   497 Rx yrs foreign LWRs through 12/82

   631 Rx yrs U.S. LWRs through 3/83

   1,128 Rx yrs (total)

   20 demands/Rx yr (12 tests and 8 transients per year)

   22,560 demands

- Point estimate 1.3E-4 per demand scram failure (PFs)
- 95% upper confidence

$$PF_s = \frac{\chi^2_{.95}(2N+2)}{2D}$$

N = number of failures
D = number of demands

$$PF_s = \frac{15.5}{2(22,560)} = \underline{3.4E-4} \text{ per demand}$$

- 5% lower bound

$$PF_s = \frac{\chi^2_{.05}(2N)}{2D}$$

$$PF_s = \frac{1.64}{2(22,560)} = \underline{3.6E-5} \text{ per demand}$$

2. United States LWR RPS Experience

2 failures (Browns Ferry and Salem)

631 Rx yr

20 demands/Rx yr (12 test and 8 transients)

12,620 demands

- Point estimate = 1.6E-4 per demand
- 95% upper confidence

$$PF_s = \frac{X^2_{.95}(2N+2)}{2D}$$

$$PF_s = \frac{12.6}{2(12,620)} = \underline{5.0E-4} \text{ per demand}$$

- 5% lower confidence

$$PF_s = \frac{X^2_{.05}(2N)}{2D}$$

$$PF_s = \frac{.71}{2(12,620)} = \underline{2.8E-5} \text{ per demand}$$

3. GE RPS Experience

1 failure (Browns Ferry)

262.92 Rx yr

20 demands/Rx yr (12 test and 8 transients)

5,258 demands

- Point estimate = 1.9E-4 per demand
- 95% upper confidence

$$PF_s = \frac{\chi^2_{.95}(2N+2)}{2D}$$

$$PF_s = \frac{9.49}{2(5,258)} = \underline{9.0E-4}$$

- 5% lower bound

$$PF_s = \frac{\chi^2_{.05}(2N)}{2D}$$

$$PF_s = \frac{0.103}{2(5,258)} = \underline{9.8E-6}$$

4. Westinghouse RPS Experience

1 failure (Salem)

248.75 Rx yrs.

20 demands/Rx yr (12 tests and 8 transients per year)

4975 demands total

- Point estimate = 2E-4 per demand
- 95% upper confidence

$$PF_s = \frac{X^2_{.95}(2N+2)}{2D}$$

$$PF_s = \frac{9.49}{2(4,975)} = \underline{9.5E-4} \text{ per demand}$$

- 5% lower bound

$$PF_s = \frac{X^2_{.05}(2N)}{2D} = \frac{0.103}{2(4,975)} = \underline{1.0E-5} \text{ per demand}$$

5. <u>B&W RPS Experience</u>

-0- failures

58.67 Rx yrs

20 demands/Rx yr (12 tests and 8 transients per year)

1,173 demand

- Point estimate -0-
- 95% upper confidence

$$PF_s = \frac{X^2_{.95}(2N+2)}{2D}$$

$$PF_s = \frac{5.99}{2(1,173)} = \underline{2.6E-3} \text{ per demand}$$

- 5% lower bound

-0-

6. CE RPS Experience

-0- failures

61.08 Rx yrs

20 demands/yr (12 tests and 8 transients per year)

1,222 total demands

- Point estimate -0-
- 95% upper confidence

$$PF_s = \frac{X^2_{.95}(2N+2)}{2D}$$

$$PF_s = \frac{5.99}{2(1,222)} = \underline{2.5E-3} \text{ per demand}$$

- 5% lower bound

-0-

Draft Transmittal Letter to Congressional Committees

Dear Mr. Chairman:

Enclosed for your information are copies of a Notice of Final Rulemaking and a Notice of Proposed Rulemaking to be published in the Federal Register. These notices pertain to safety systems for power reactors.

The final amendment to the Commission's regulations in 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," will require licensees of light-water-cooled nuclear power reactors to make design improvements in systems for shutting down the reactors in the event of certain anticipated operational occurrences. These improvements are expected to increase the reliability of the shut down (scram) systems and thus reduce the probability of occurrence of adverse plant conditions which might lead to core melt and release of radioactivity to the environment.

Proposed amendments relating to this final rule were published for comment in November 1981. Forty comments were received. Thirty-two of the comments were from utilities and generally expressed the opinion that presently used reactor shut down systems are adequate and that improvements are not needed. The Commission does not share that opinion. Staff analyses, using probabilistic risk assessment techniques, indicate that the new requirements will improve reactor safety and be cost effective.

Enclosure "E"

The proposed amendments which were published in November 1981, included a requirement for an additional diverse scram system for pressurized water reactors (PWRs) designed by two of the three manufacturers of PWRs. The proposed amendments did not propose requiring such a system for PWRs designed by the third manufacturer, Westinghouse, because the low expected frequency of system failures and certain consequence mitigative features of Westinghouse PWRs were believed to provide adequate safety. Recent experience at a Westinghouse PWR, reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," has increased the staff's estimated failure rate of the Westinghouse scram system and caused the Commission to reconsider the need for a diverse scram system for Westinghouse PWRs.

The enclosed proposed amendment would require a diverse reactor scram system for Westinghouse PWRs as a means of improving scram system reliability. Current staff analyses indicate the requirement to be reasonably cost effective. Consideration of both prevention of a significant number of scram system failures and mitigation of consequences of failures is consistent with the Commission's regulatory safety philosophy of defense in depth.

The notice of proposed rulemaking allows 45 days for public comment after which time the Commission will further consider the need for a final rule.

Also enclosed are copies of a public announcement that will be released by the Commission in the next few days.

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosures:
1. Notice of Final Rulemaking
2. Notice of Proposed Rulemaking
3. Draft Public Announcement

DRAFT PUBLIC ANNOUNCEMENT

NRC REQUIRES ADDITIONAL ATWS PROTECTION:

PROPOSES FURTHER REQUIREMENTS

The Nuclear Regulatory Commission is amending its regulations to require additional safety improvements in the design and operation of light-water-cooled nuclear power reactors.

The new requirements will reduce the likelihood of failure of the reactor trip systems to scram the reactors following anticipated transients and will reduce the consequences should failures occur. An anticipated transient and concurrent failure of the reactor trip system could lead to melting of reactor fuel and the release of large amounts of radioactivity to the environment.

Anticipated transients are operating events, such as loss of offsite power to the reactor, expected to occur one or more times during the operating life of a reactor and scram is the rapid shut down of the reactor by the reactor trip system.

The new requirements are based on proposed rules published for public comment in November 1981 and are keyed to the type of reactor, such as pressurized water reactor or boiling water reactor, and its manufacturer.

At the same time, the Commission is proposing to amend one of the new requirements. This proposed amendment concerns an additional scram system requirements for pressurized water reactors manufactured by Westinghouse

Enclosure "F"

Electric Corporation. The NRC staff previously believed that the probability of ATWS events at the Westinghouse facilities was sufficiently low to preclude the need for additional protection. However, the occurrence of two ATWS events at Unit 1 of the Salem Nuclear Power Plant in New Jersey in February 1983 has increased the staff's estimate of the failure rate of Westinghouse reactor trip systems. An additional scram system requirement for pressurized water reactors manufactured by Babcock and Wilcox Company and by Combustion Engineering, Inc. already is in the new requirements.

The new requirements for pressurized water reactors are:

(1) Additional equipment, independent of the reactor trip system, to activate automatically the auxiliary (emergency) feedwater system and to initiate a shut down of the plant turbine under conditions indicative of an ATWS.

(2) For Combustion Engineering and Babcock and Wilcox plants an additional scram system (all of those components of the reactor trip system exclusive of control rods and control rod mechanisms) which also is independent of the reactor trip system.

For boiling water reactors, the new requirements are:

(1) An additional system to inject control rods into the reactor vessel.

(2) An increased capacity standby system to inject borated water into the reactor vessel.

(3) Additional equipment to shut down automatically the reactor coolant recirculating pumps under conditions indicative of an ATWS.

The new rule requires that each licensee submit by (date) a proposed schedule for meeting these requirements together with an explanation of the schedule. If the schedule calls for final implementation later than (date) for reactors already licensed for full power operation or the date of issuance of a license authorizing operation above five percent of full power for other reactors, the schedule must be justified.

At the same time, the Commission is urging, but not requiring, the establishment of a joint NRC-industry reliability assurance program to provide additional assurance that the desired high level of reliability of reactor trip systems will be achieved and maintained. If this voluntary program does not produce reasonable results, the Commission may then choose to require the development of such a program.

The Commission believes such a program should focus on an assessment of reactor trip system failures together with detailed analyses of the causes of such failures. In addition, it should provide for awareness of and evaluation of plant-specific and industry-wide operating experience with procedures for timely feedback of the information.

The final amendments to Part 50 of the Commission's regulations will become effective on (date). Written comments on the proposed amendments to Part 50 should be received by (date). They should be addressed to the Secretary of the Commission, Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch.

SECY PAPER DISTRIBUTION

AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

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JUL 5 1983

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

Enclosed for your consideration is a Commission paper which recommends issuance of a final amendment and a related proposed amendment to 10 CFR Part 50. The amendments would require design improvements in systems for shutting down a reactor in the event of anticipated transients, or for mitigating the effects of not shutting down the reactor. We believe the paper fairly presents the ATWS problem and recommends a reasonable course of action to the Commission.

The enclosed paper was developed with the assistance and guidance of the ATWS Task Force and Steering Group and the Generic Implications Task Force on Salem. The ATWS Task Force and Steering Group was established in July 1982, to resolve the ATWS rule. The Generic Implications Task Force on Salem was established immediately after the event on February 25, 1983. The Salem incidents significantly increased interest in the ATWS issue and caused a concerted effort to expedite action on the rulemaking.

The rule changes recommended in the enclosed paper are compatible with actions likely to be taken as a result of recommendations of the Generic Implications Task Force on Salem.

The ACRS's Subcommittee on ATWS has been briefed on this rulemaking, and a briefing of the ACRS Committee is scheduled for July 7, 1983. The rulemaking actions were discussed on three occasions with the CRGR, and we believe the enclosed paper is consistent with suggestions provided by the CRGR.

Concurring in recommendations of this paper are the Offices of Nuclear Reactor Regulation and Inspection and Enforcement, and the Division of Rules and Records. The Executive Legal Director has no legal objection. The Office of Public Affairs prepared the draft public announcement. Representatives from Regions I and II were on the ATWS Steering Group and copies of the paper have been furnished to all the Regional Administrators.

Original signed by:
ROBERT B. MINOGUE

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

EDO
WJDircks
7/ /83

Enclosure:
Commission Paper on ATWS Rule Changes

	NRR	ELD	DRR	PA
	6/27/83	6/23/83	6/27/83	6/23/83

OFFICE	RRB:RES	RRB:RES	RRB:RES	DRA:RES	RES	RES	IE
SURNAME	DPyatt	PBaranowsky	GBurdick	MErnst	DRoss	RMinogue	
DATE	6/27/83	6/ /83	6/27/83	6/15/83	7/1/83	7/5/83	6/ /83

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

Purpose: To obtain Commission approval for publication of a final rule and a related proposed rule in the Federal Register.

Category: This paper covers a major policy question.

Issue: What improvements, if any, should be required by regulation in the design of light-water-cooled nuclear power plants to reduce the risk of ATWS events to an acceptable level.

Discussion: An anticipated transient without scram (ATWS) is an expected operational occurrence (such as a loss of feed water, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor protection system to shut down the reactor. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to core melt and release of radioactivity to the environment. Anticipated transients occur on the average of ten times per year. The ATWS question involves the need, if any, to provide backup safety systems for shutting down the reactor or mitigating the effects of an ATWS, if there is an anticipated transient and a concurrent failure of the reactor protection system.

In 1980 the staff completed review and evaluation of information developed over the preceeding 10 years on ATWS events and the manner in which such events should be considered in the design and safety evaluation of nuclear power plants. This evaluation, reported in "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, suggested that the frequency of a severe ATWS event may be unacceptably high.

CONTACT:
D. W. Pyatt, RES
443-5960

In SECY 80-409, dated September 4, 1980, the staff recommended publication of a proposed rule to require improvements in the design of reactors to reduce the likelihood of ATWS events and to mitigate the consequences of such events. Following extensive consideration of SECY 80-409 by the Commission and of an alternative approach proposed by then Chairman Joseph Hendrie, two alternative proposed rules were published for public comment on November 24, 1981. In the November 24, 1981, Federal Register notice the Commission stated that it also was considering a third alternative proposed rule that was contained in a petition for rulemaking (PRM-50-29) received from a Utility Group on ATWS (which now represents 22 electric utilities) that had been published for comment on November 4, 1980. When the two alternative proposed rules were published, the comment period for the Utility Group's petition was reopened and the public was invited to comment on the three proposed rules.

While each of the three alternative proposed rules had the objective of reduction of risk from ATWS, each featured a different approach to achieving that objective. The staff-developed alternative emphasized individual reactor evaluation to identify needed improvements. The Hendrie alternative emphasized reliability assurance and would have also required certain hardware modifications. The Utility Group alternative prescribed specific changes keyed to the type of reactor and its manufacturer.

As noted in the June 28, 1982, status report to the Commission on this rulemaking (SECY 82-275), 39 public comments were received on the three alternative proposed rules. Thirty-one of the comments were from utilities. The most frequent comment was that no rule on ATWS was needed, but if one were to be adopted the preferred rule was the alternative proposed by the Utility Group.

The staff has reviewed all the comments and now recommends publication of a final rule and a related proposed rule. The final rule, set out in Enclosure "A", adopts the same approach that is used in the Utility Group's petition for rulemaking in that the prescribed changes are keyed to the reactor's type and manufacturer. With three exceptions the prescribed changes in the final rule were proposed in the Utility Group's petition. Those exceptions are the new requirements for the standby liquid control system (SLCS) used on boiling water reactors and the inclusion of reactor trip breakers for the diverse scram system on pressurized water reactors manufactured by Combustion Engineering and Babcock and Wilcox.

The final rule requires an increased capacity for the SLCS on all BWRs. The petitioner has proposed the increased capacity for SLCS only for BWRs that are yet to be constructed. The final rule also requires that SLCS initiation be automatic (in lieu of manual initiation by the operator) for new plants and for other plants that have already been designed to include this feature. Automatic initiation of SLCS was not proposed by the petitioner. The staff believes the SLCS is sufficiently important to reactor safety to require the increased capacity for both present and future reactors and that it would be cost effective to require automatic initiation of SLCS for future reactors. The staff believes that the inclusion of a diverse method of interrupting power to the diverse scram system is warranted based on the potential of common cause failure.

The proposed rule, set out in Enclosure "B", is a consequence of the recent ATWS events at the Salem Nuclear Power Plant. Salem uses pressurized water reactors (PWRs) manufactured by Westinghouse. The Salem events caused reconsideration of the need for diversity in scram systems in Westinghouse PWRs. The staff believes that the attached final rule's diversity requirement for CE and B&W PWRs should also apply to Westinghouse PWRs and should also include the reactor trip breakers.

The estimated cost and reduction in the probability of a core melt accident attributable to the requirements in the attached rules and of other requirements that were considered are discussed in Enclosure "D".

In evaluating comments on the three proposed rules and developing the attached final rule and related proposed rule, the staff used a combination of probabilistic risk assessment (PRA) techniques and engineering judgment. These PRA techniques were useful in comparing cost/benefit ratios for candidate regulatory changes and in developing a set of changes which should ensure an acceptable low risk from ATWS events. The results of the PRA are consistent with the staff's engineering judgment that the rules are needed to reduce the risk from ATWS events.

Alternatives: The staff is proposing that the Commission consider two alternative courses of action:

Alternative 1 - Withdraw the proposed rules that were published in the Federal Register on November 24, 1981, and maintain present regulations (thereby adopting the position that operating plants as designed are safe enough or, if changes in certain plants are needed, those changes can be required by order on a case-by-case basis).

Pro: Existing Reactor Protection Systems (RPS) were designed, manufactured, inspected, installed, operated, and tested to specified and approved codes, standards, guides, and procedures. In addition, this system incorporates independence, redundancy, and some diversity. These considerations suggest that the RPS design provides adequate protection and the costs of additional requirements should not be imposed on licensees. If certain licensees have reliability problems such as occurred at the Salem plant, those licensees can be given case-by-case consideration.

Con: While the RPS may provide adequate protection from multiple random independent failures, the design is susceptible to common mode failures. Common mode failures of reactor protection systems have occurred, e.g., at the Salem Plant in February 1983. The RPS is challenged on the average of 10 times per year and there is need for the high reliability that would be provided by the final and proposed rules. Cost/benefit analyses indicate the requirements in the rules would be cost effective. Remedy of the generic ATWS problem by rulemaking is administratively more efficient than licensee-by-licensee consideration.

Alternative 2 - Issue a final rule that prescribes generic safety improvements keyed to the reactor's type and manufacturer, and issue a proposed rule to include Westinghouse plants among PWRs that would be subject to the scram system diversity requirement.

Pro: Information gained over the past several years has led to a full understanding of all of the implications of the various prescriptive requirements. Therefore, the prescriptive rule approach avoids unnecessary extensive individual case analyses by licensees and the staff. Staff evaluations indicate that the prescribed changes would be cost effective in reducing the risk of ATWS events. The proposed rule for Westinghouse plants* would alleviate the potential for common cause failure of the Westinghouse trip system. The two rules could lead to completion of the rulemaking proceeding and to maintaining an acceptable level of risk from ATWS events without the expense to the NRC and the industry of plant-specific evaluations.

Con: Generic treatment of the plants causes difficulty in performing Regulatory analyses because of risk

*A proposed rule rather than a final rule is needed because the 1981 proposed rules did not explicitly include the scram diversity requirement for Westinghouse plants. Diversity was proposed for the scram systems for Combustion Engineering and Babcock & Wilcox PWRs.

important variations in design and operations among the plants. A prescriptive rule discourages consideration of alternative (and perhaps better) ways of achieving the desired safety for individual plants. Variations among the plants may result in differences from plant to plant in the effectiveness of the prescriptive design changes which will not be quantified.

Recommended Alternative - The staff recommends that Alternative 2 be implemented. The staff believes that the final rule set out in Enclosure "A" and the proposed rule set out in Enclosure "B", if made effective, would substantially reduce the ATWS risk in a cost effective manner and assure an acceptable level of risk from ATWS events.

Resource Requirements - Implementation of the rules would have only a minor impact on NRC resources. ATWS considerations are already being reviewed by the staff on a case-by-case basis for applicants, and since ATWS has never been specifically addressed in the Commission's regulations, it has become a contested issue by intervenor groups. The Regulatory treatment of ATWS provided in the attached rules should reduce the amount of staff time currently required to communicate with applicants and intervenors on this issue.

Initially, little effort would be required to implement the rules. NRR will need to agree with vendors on acceptable design modifications and implementation schedules will be worked out with licensees. The Regional Offices will be responsible for assuring that the modifications are properly installed.

Implementation of the rules will cost licensees an estimated \$3.5M to \$5.5M per plant. The estimated total cost to industry is on the order of \$525M. Enclosure "C" provides further information on the estimated costs.

Recommendations: That the Commission:

1. Approve publication in the Federal Register of the notices of:

    a. final amendment of 10 CFR 50 to add a new §50.62 regarding design improvements to reduce risk from ATWS events (Enclosure "A",) and

    b. proposed amendment of 10 CFR 50 which proposes change of the new §50.62 (c)(2) to include Westinghouse PWRs among the PWRs required to have a diverse scram system (Enclosure "B").

2. Note

a. The notice of proposed rulemaking in Enclosure "B" will be published in the Federal Register with a 45 day comment period.

b. No environmental impact statement, negative declaration, or environmental impact appraisal need be prepared in connection with the amendments because the actions taken or proposed will not significantly affect the quality of the human environment.

c. That pursuant to the Regulatory Flexibility Act of 1980 the notices contain statements that the Commission certifies that the final rule will not, and the proposed rule if promulgated will not, have a significant economic impact upon a substantial number of small entities. The rules would affect only nuclear power plants and companies that own these plants do not fall within the definition of "small entities." The Chief Counsel for Advocacy of the Small Business Administration will be informed by DRR of the certification regarding economic impact on small entities together with the reason for it.

d. That the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works, the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy Conservation and Power of the House Committee on Energy and Commerce, and the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations will be informed of the Commission's action by letter such as Enclosure "E".

e. The reporting requirements in connection with the implementation schedule required by the rule impose information collection requirements that are subject to the Paperwork Reduction Act. The requirements were approved by OMB.

f. That a Regulatory Analysis is attached as Enclosure "C".

g. That a public announcement will be issued (Enclosure "F".)

h. That copies of the Notices will be distributed by TIDC, ADM to affected licensees and other interested parties.

i. That the staff recommends the paper be placed in the PDR.

Scheduling: It is recommended that this paper be considered at an open session.

William J. Dircks
Executive Director for Operations

Enclosures:
"A" - Notice of Final Rulemaking
"B" - Notice of Proposed Rulemaking
"C" - Regulatory Analysis
"D" - Task Force Report
"EE - Draft Congressional Letter
"F" - Draft Public Announcement