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For Comment

Anticipated Transients Without Scram for Light Water Reactors

Resolution of Unresolved Safety Issue TAP A-9

Office of
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ABSTRACT

This is the fourth volume of the NRC staff's review of anticipated transients without scram (ATWS). It contains the proposed resolution of this unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants. A phased approach is used: near-term improvements in safety, both hardware and procedural, are required over the next 1 to 2 years to provide an expeditious safety increment. Later, more extensive requirements will be imposed on some plants, but the implementation of these major hardware changes may be stretched out to accommodate equipment acquisition and plant refueling schedules. This delay is intended to allow the changes to be accomplished with minimum disruption and downtime, thus with minimum expense consistent with the level of safety to be achieved.

Our present recommendation is more severe than the previous recommendations because the intended generic verification of the safety adequacy of the proposal was not achieved.

This report describes the proposed requirements and their phased implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information and the staff's evaluation of that information, technical details, and some related risk estimates associated with the revised requirements.

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USI - A-9

ANTICIPATED TRANSIENTS WITHOUT SCRAM
FOR LIGHT WATER REACTORS

1. INTRODUCTION AND SUMMARY

1.1 Introduction

This is the fourth volume of the NRC staff's review of anticipated transients without scram (ATWS). It contains the proposed resolution of this unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants. A phased approach is used with near-term improvements in safety, both hardware and procedural, being required over the next 1 to 2 years to provide an expeditious safety increment. Later, more extensive requirements will be imposed on some plants, but the implementation of these major hardware changes may be stretched out to accommodate equipment acquisition and plant refueling schedules. This delay is intended to allow the changes to be accomplished with minimum disruption and downtime, thus with minimum expense consistent with the level of safety to be achieved.

By comparison with our previous recommendation in Volume 3 of NUREG-0460, our present recommendation is more extensive for operating plants, for which the implementation of long-term major improvements were previously not to be required. The reason for this change is the failure of the program required of industry by the staff that was intended to achieve early generic verification of the adequacy of Alternative 3 modifications. This has not, in the opinion of the staff, been demonstrated over the lifetime of the approximately 100 plants originally proposed for Alternative 3. Serious problems remain with the information and analyses presented by the industry in support of the adequacy of Alternative 3; these problems are summarized in Section 1.3 and discussed in Appendix A.

In principle, further information gathered in a designated amount of time might demonstrate such adequacy. However, we believe that this would take too long and would expend an inordinate quantity of resources with too little chance of success. The present state of ATWS knowledge and information availability is the result of 11 years of work and discussion regarding ATWS requirements. We have therefore decided to establish requirements based on the existing knowledge as we interpret it.

Since we conclude that Alternative 3 is not adequate, we will require Alternative 4 (or a near approximation) on all plants. However, the rapid implementation of the major modifications that are part of Alternative 4 or many plants would be very expensive because of downtime on operating plants and delays on plants in the later stages of construction. Moreover, it would take several years to make some of the major modifications, even if cost were ignored.

Accordingly, we have instituted a phased program, requiring the prompt implementation of "Alternative 3A" (in the next 1 to 2 years) and "Alternative 4A" by 1984. This provides for (1) prompt implementation of significant safety improvements that are cost-effective to implement promptly, plus (2) major additional improvements that we judge to be needed for safety adequacy over the lifetime of many plants. Alternatives 3A and 4A, which are summarized in

Section 1.4 and given in detail in Chapter 2, are modifications of Alternatives 3 and 4 from Volume 3 of NUREG-0460, with the changes reflecting the additional information now available.

1.2 Background

In NUREG-0460, Volumes 1 and 2, we evaluated the information available to the staff at that time and concluded that the ATWS events presented an unacceptably high risk to the public during the service life of nuclear power plants. As more and more plants come on line, the risk to society increases further. Therefore, the staff concluded that some corrective measures were required to reduce the risk of severe consequences arising from possible ATWS events. It was further recommended that new systems (or modifications to existing systems) to mitigate the consequences of ATWS events be provided. The bases for these conclusions were the estimated frequency of severe ATWS events and the level of safety believed to be necessary. The required level of safety was specified in numerical terms.

In Volume 3 of NUREG-0460 (Ref. 1) that was published in December 1978, the staff reexamined the approach in earlier staff publications and concluded that a numerical safety objective is not satisfactory for use in nuclear regulatory decision-making at the present time, and recommended that the quantitative risk assessments be used as a supplement to engineering judgment. Based to a large extent on engineering judgment the staff recommended that the hardware modifications of Alternatives 2, 3, and 4 listed below be implemented. The staff's conclusion was stated to be subject to verification of the assumed level of protection provided by the plants as modified.

(1) Early Operating Reactors (Alternative 2) - Alternative 2 as proposed in Volume 3 would provide modifications aimed at reducing the likelihood of ATWS events. Because of the unique characteristics of these older designs, plant-unique analyses were to be performed to determine if any additional modifications would be required.

(2) Operating Plants and Plants with Construction Permits Issued Prior to 1/1/78 (Alternative 3) - Alternative 3 as proposed in Volume 3 would provide some degree of mitigation in addition to the preventive measures of Alternative 2. It required the same hardware features of Alternative 2 for PWRs, plus it required demonstration by analysis of the mitigation capability that is believed to exist already in PWRs. In the case of PWRs, Alternative 3 would require repiping and automation of the standby liquid poison injection function and demonstration by analysis of the resulting mitigation capability.

(3) New Plants and Plants with Construction Permits Issued on or After 1/1/78 (Alternative 4) - Alternative 4 as proposed in Volume 3 would rely on mitigation and did not include the additional measures of Alternatives 2 and 3 to improve the prevention of ATWS events. It would provide an implementation of the proposed licensing criteria of Volume 2 of NUREG-0460 that is acceptable to the staff.

In Volume 3 of NUREG-0460, the staff recommended that prior to the Commission's consideration of a proposed ATWS regulation, certain generic safety analyses should be performed. These analyses were to confirm that the proposed modifications for various classes of light water reactor (LWR) designs would accomplish the degree of ATWS prevention and mitigation described by the staff in Volume 3 of NUREG-0460.

Generic questions and information guidelines were provided to the industry in a letter dated February 15, 1979 from R. Mattson (Ref. 2) for Alternative 3 modifications (for plants receiving a construction permit prior to January 1, 1978) and Alternative 4 modifications (for plants receiving a construction permit after January 1, 1978). Originally, the answers to these 58 pages of questions and requests for information were scheduled for April 1979. A series of discussions resulted in separating the information needs for Alternative 3 into three categories, and rescheduling for May-October 1979. The Alternative 4 material was deferred.

The Three Mile Island accident in March 1979 forced deferral of all NRC work on ATWS, and most industry work was halted or delayed as well. As a result, the information submitted by the industry fell far short of the February 15, 1979, request and in most cases fell short of the Cateogry 1 scope.

1.3 Summary of Review of Industry Response

The nuclear power industry has always disagreed with the staff's evaluation of ATWS. Their general position can be summarized as follows:

(1) The industry believes that in pressure designs the probability of an ATWS event that might jeopardize public safety is already so low that no plant modifications or other protection measures are needed.

(2) If ATWS modifications are to be improved as a precautionary measure, the industry believes that they need only to be the preventive measures of Alternative 2 of Volume 3. They are not really needed, but they are also not very expensive.

(3) If additional mitigation measures beyond modifications in instrumentation and logic are to be required by the NRC in spite of the industry's showing that such are not needed, the requirements of Alternative 3 (of Volume 3) should be reasonably acceptable to all.

(4) The industry thus far has been unanimous in rejecting the need for any plant modifications beyond Alternative 3, maintaining that the analyses submitted demonstrate the adequacy of Alternative 3 and the low value of additional safety provided by the expensive Alternative 4.

The staff evaluation of the industry submittals provided in response to the February 1979 request are provided in Appendix A to this report. We conclude that the adequacy of the proposed Alternative 3 solution to ATWS as described in NUREG-0460, Volume 3, has not been demonstrated.

The areas judged to be inadequately addressed in the industry submittals are summarized in the following section.

1.3.1 PWR Designs

(1) Not all significant anticipated transients were analyzed. The stuck-open power-operated relief valve (PORV) anticipated transient has not been correctly analyzed.

(2) Long-term shutdown has not been adequately addressed. In particular, the impact of voids in the primary system after the initial pressure peak has passed, the timing of the reactor coolant pump trip, and the plants with low high-pressure safety injection (HPSI) shutoff head have not been addressed. The PWR transient codes used inthese analyses are unacceptable for situations where significant voids are calculated to be present in the primary.

(3) Combustion Engineering (CE) information reveals that some instrument capability will be lost due to high primary pressure; this is likely to be the case for the other PWRs also. Ability of the instruments and equipment needed for safe shutdown to withstand the pressure peak only partially addressed by CE and not addressed at all by Babcock & Wilcox (B&W) and Westinghouse (W).

(4) The impact of isolated PORVs on plant response to ATWS has not been adequately addressed.

(5) The calculated peak pressure for operating CE plants would exceed 4000 psi even with the vessel head lifting as calculated to relieve the primary pressure. Also, many components exceed service level "C" stress limit.

(6) No stress evaluation has been provided for balance-of-plant components.

(7) Overly optimistic assumptions were used in the B&W peak pressure calculation. The staff believes that some components would be calculated to exceed service level "C" limit if more appropriate assumptions are used.

(8) Many questions remain on radiological evaluations if the containment structure is not isolated soon after the initiation of an ATWS event.

(9) Design information on preventive and mitigative systems has been inadequately addressed.

(10) If HPSI is actuated early (automatically or manually) while the primary system pressure is above the HPSI design pressure, its operability and integrity are questionable.

(11) The effect of pressures substantially above the 3400-3500 psi range considered in Volume 3 is not well understood. In particular, the integrity and performance of safety and relief valves has not been assured; the TMI-related industry testing program is not expected to encompass this extreme pressure range.

1.3.? BWR Designs

(1) Most BWR designs, but BWR/4 in particular, are calculated by General Electric (GE) to produce severe power/flow oscillations several hundred seconds following a turbine trip ATWS event. Oscillations have been

observed in an operating BWR. The staff is concerned with the prediction capability of the codes, the impact on plant control systems, and the fuel integrity after the core undergoes these oscillations.

(2) The reports do not address the ability of equipment needed for safe shutdown to withstand the ATWS transients, including the oscillations.

(3) Insufficient information on the reactor coolant pressure boundary components integrity/operability is provided.

(4) ATWS containment loads are not shown to be bounded by design basis loads.

(5) Many questions remain on radiological evaluation if containment is not isolated early in the ATWS event.

(6) Design information on prevention and mitigation systems has been inadequately addressed.

The above concerns notwithstanding, the current information provides some degree of assurance that the consequences from ATWS events in BWRs are likely to be acceptable if the mitigating systems function as presumed and the oscillations do not cause an unforeseen damage.

The Westinghouse designs are still believed to be able to withstand ATWS events if the actuating circuitry of mitigating systems is modified.

B&W has not provided any new significant information, but we still believe it is unlikely that we would agree with the B&W analysis assumptions. Thus, the adequacy of Alternative 3 solution has not been demonstrated.

CE analyses (of operating plants, in particular) indicate that Alternative 3 is unlikely to be acceptable.

These factors are further discussed in Section 1.4 and Appendix A. While some of them are also of concern in evaluating Alternative 4A, the concerns are less severe for 4A because the ATWS transients are less severe.

1.4 Proposed Requirements

As summarized in the previous section and discussed in some detail in Appendix A, the staff has concluded that the industry has not provided adequate information to conclude that Alternative 3 provides the desired level of risk reduction from ATWS events. In the list below and in Table 1 we summarize the requirements, discussed further in Chapter 2, that we propose for licensees and applicants. Alternatives 2A, 3A, and 4A are defined in Chapter 2.

(1) Early operating plants (which total to 11 plants) must implement Alternative 2A by July 1, 1981.

(2) All other operating plants and other plants close to operation (staff to provide list at a later time) must implement Alternative 3A. Electrical items must be completed by July 1, 1981; piping changes by July 1, 1982.

TABLE 1. COMPARISON OF REQUIREMENTS

Vendor	Alt. 2 (Vol. 3)	Alt. 2A	Alt. 3 (Vol. 3)	Alt. 3A	Alt. 4 (Vol. 3)	Alt. 4A
B&W	BUSS[1] AMSAC[2]	BUSS[1] AMSAC[2] Analysis[3]	BUSS[1] AMSAC[2] Analysis[4]	BUSS[1] AMSAC[2] Cont. Isol.[5] Inst.[6]	AMSAC[2] Add safety valves Analysis[4]	BUSS[1] AMSAC[2] Safety Valves Analysis[7] OPT[8] Cont. Isol.[9] Inst.[6]
CE	SPS[1] AMSAC[2]	SPS[1] AMSAC[2] Analysis[3]	SPS[1] AMSAC[2] Analysis[4]	SPS[1] AMSAC[2] Cont. Isol.[5] Inst.[6]	AMSAC[2] Add safety valves Analysis[4]	SPS AMSAC Safety Valves OPT[8] Analysis[7] Cont. Isol.[9] Inst.[6]
W	AMSAC[2]	AMSAC[2] MSS[1] Analysis[3]	AMSAC[2]	AMSAC[2] MSS[1] Cont. Isol.[5] Inst.[6]	AMSAC[2] Analysis[4]	AMSAC[2] MSS[1] Analysis[7] Cont. Isol.[9] Inst.[6]
GE	ARI[1] SD[11] RPT[12] Logic[14]	ARI[1] SD[11] RPT[12] Logic[14] Analysis[3]	ARI[1] RPT[12] Logic[14] Auto 86 gpm SLCS[15] SD[11] Analysis[4]	ARI[1] RPT[12] Logic[14] Auto 86 SLCS[15] SD[11] Cont. Isol.[5]	RPT[10] Automatic, high capacity liquid poison injection Analysis[4]	ARI[1] RPT[13] Auto Hi-Cap Poison Logic[14] SD[11] Analysis[7] OPT[8] Cont. Isol.[5]

Footnotes to Table 1

[1]A system that is diverse and independent from RPS, meeting IEEE-279 and acting as backup to the electrical portion of the current scram system.

[2]ATWS mitigating system actuation circuitry satisfying criteria in Appendix C, Volume 3.

[3]Analysis of Alt. 2A plants to decide if mitigation capability exists or is necessary in overall safety context.

[4]Analysis remains to be performed and reviewed to confirm expected mitigation capability as described in Sections 2.2 and 2.3 of Volume 3.

[5]Provisions to close containment isolation valves quickly if fuel failure should occur.

[6]Providing instruments necessary for shutdown that can withstand the ATWS peak pressure.

[7]Analysis of Alt. 4A plants to verify adequacy of plant modifications.

[8]Optimization study for Alt. 4A plants where full implementation is not practicable (Alt. 3 1/2).

[9]A system satisfying the criteria in Appendix C, Volume 3, that isolates containment quickly in the event of ATWS fuel failure.

[10]Recirculation pump trip satisfying criteria in Appendix C, Volume 3.

[11]Modification of scram discharge volume.

[12]The approved Monticello design is an acceptable RPT design for all BWR 4 plants. The approved Zimmer design is an acceptable RPT design for all BWR 5 and 6 plants. There may be other acceptable designs which must be treated on a plant specific basis.

[13]As in footnote 10, except that RPT installed before July 1, 1981, may be in accordance with footnote 12.

[14]Changes in logic to reduce vessel isolation events and permit feedwater runback.

[15]Modified SLCS piping to assure delivery of 86 gpm of poison and automatic actuation circuitry satisfying parts A through H of Appendix C, Volume 3, with reliability equivalent to the mechanical portion of the SLCS.

(3) All plants other than early operating plants (Item 1, above) must implement Alternative 4A by January 1, 1984. This includes additional confirmatory information which must be provided as described in Chapter 2 and, for operating plants and those nearing completion, a study to optimize the performance and reliability of ATWS-related functions within constraints imposed by plant arrangement, diesel capacity, and completed seismic structures.

The analyses to be performed in completing Alternative 4A are required to verify that the modifications provide the intended level of safety. They are discussed in Appendix A for the four vendors' designs. As stated in NUREG-0460, Volume 1, and the 1975 status reports, the objective is to use evaluation models that realistically predict the course of the ATWS event sequences being analyzed, and conservatively predict the consequences. With certain exceptions, nominal values of system parameters and realistic assumptions concerning physical phenomena are to be used. The exceptions are (1) PWR moderator temperature coefficient less negative than will be experienced 99 percent of the time; (2) relief and safety valve discharge rate conservatively low; (3) conservative fuel damage models; (4) assumption of failures in mitigating systems; and (5) heat transfer rate in steam generators. In addition, the level "C" stress criterion and the suppression pool 200°F limit are conservative. For long-term cooldown, use of small-break LOCA codes involves additional conservatisms.

Use of realistic best-estimate analyses requires verification of the models. This can be accomplished by a combination of comparisons with other calculations and comparisons with test data. The status of verification in mid-1978 was given in the answer to Question 18 in Appendix B, Volume 3, of NUREG-0460.

Appendix B of this report gives our present views on additional testing for verification.

The models must be adequately verified to be acceptable for use in complying with the Alternative 4A requirements detailed in Appendix A. Where adequate verification is not obtained on a time scale consistent with the Alternative 4A implementation schedule, compensatory conservatisms may be required.

The bases for these requirements are the following:

(1) The need for measures to reduce the future likelihood of severe consequences arising from an ATWS event;

(2) A strong desire to achieve the required level of safety with the least cost to the public, since a large cost component is plant downtime or construction delay, timing and phasing of modifications to minimize document or delay;

(3) Given the time span necessary for major modifications and desirable for cost minimization, the need to make promptly such safety modifications as are practical in a short time;

(4) The failure to achieve early, generic verification of the adequacy of Alternative 3 proposed in Volume 3 for operating and near-term plants;

(5) The experience, at Three Mile Island and elsewhere, that plants do not always behave as they are calculated to behave, and the concomitant need felt to provide prevention and mitigation measures for credible postulated accident sequences.

1.5 Implementation

In Volume 3, the staff proposed an implementation consisting of (1) rulemaking to establish general requirements for all classes of plants, (2) guidance on implementation of general criteria in a Regulatory Guide to be issued after an effective rule is in place; (3) early, generic verification of the adequacy of Alternatives 3 and 4 hardware modifications to achieve the levels of safety that formed the bases of these alternatives.

In view of the staff's evaluation that early, generic verification has not been achieved, and the concomitant change in the proposed requirements, the plant has reconsidered the implementation options described in Chapter 4 of Volume 3. The staff has concluded that Alternatives 2A and 3A, which mandate prompt improvements in safety, would best be accomplished by Orders. For Alternative 4A, which is more controversial, requires additional verification, and has a longer time span, the staff recommends rulemaking.

2. MODIFIED ALTERNATIVES

2.1 General Considerations

In this chapter, we define the modification packages that constitute Alternatives 2A, 3A, and 4A. These are proposed for implementation on the different classes of plants as set forth in Sections 1.4 and 3.1.

The bases for establishing these sets of requirements are discussed in detail in the following sections of this chapter. In general, the reasoning includes the following considerations:

(1) The need for reduction in risk due to ATWS.

(2) The strong desire to accomplish the risk reduction in the least expensive, most cost-effective way consistent with the level of safety we believe to be needed.

(3) The lesson from the TMI-2 accident that low probability accident sequences may not proceed as predicted by analyses. This confirms the need to provide defense in depth by preventing the occurrence of accidents and also providing mitigation capability to permit a safe shutdown of the plant should a low probability accident sequence occur.

2.2 Alternative 2A

We continue to recommend treating the eleven early operating reactors on a case-by-case basis, as proposed in NUREG-0460, Volume 3, because of their unique designs. These plants are required to implement the Alternative 2 prevention system described in NUREG-0460, Volume 3 by July 1981, plus the modified scram system (MSS) on Westinghouse plants as described in Section 2.3.

In addition, the licensee of each Alternative 2A plant is required to submit, by July 1981, an analysis of the plant response to postulated ATWS sequences and mitigating systems alternatives, to decide whether mitigation capability exists or is necessary in context with the overall safety of these plants. Procedures and operator training must be modified to improve operator response to ATWS events.

The basis for these requirements is to provide appropriate improvement in ATWS prevention and mitigation consistent with the design of these early plants. An optimization of safety increment and cost-benefit is encouraged. Plant-specific analysis is needed because of the differences among the plants, and because these plants are not encompassed within the envelopes of the generic analyses that have been submitted.

2.3 Alternative 3A

This alternative is similar to Alternative 3 in NUREG-0460, Volume 3. However, the requirements have been somewhat modified on the basis of our review of the recent industry submittals.

This set of modifications is specified almost entirely in terms of required hardware. Prompt installation of these safety improvements is being imposed without any significant additional analysis. This is intended to have the effect of providing early, safety improvements. The industry reports claim that these modifications would achieve a large safety improvement. Although we cannot agree with their numbers (see Section 1.3 and Appendix A), we believe that the safety improvement would indeed be substantial.

In addition to hardware changes, emergency procedures must be developed and operators trained for recognition of ATWS events and for taking actions to control and mitigate such events with the modified equipment. These provisions, already required and under way at operating plants to improve operator response with existing equipment, must be revised to be consistent with the Alternative 3A modifications.

The Alternative 3A modifications are not intended by the staff to stand alone for the life of the plant. Rather, they are to be supplemented by the Alternative 4A modifications in a few years. Accordingly, speed of implementation of the specified hardware is emphasized; additional analysis and information requirements are minimized.

2.3.1. Hardware Description

2.3.1.1 Modifications to Reduce Susceptibility to Common-Mode Electrical Failures

Babcock and Wilcox

Plants must provide a diverse, four-channel backup scram system (BUSS) described by B&W. It must be diverse and independent from the reactor trip system and would actuate on high pressurizer pressure or level signals. The staff would require that such a system meet IEEE-279.

Combustion Engineering

Plants must provide a diverse, four-channel supplementary protection system (SPS) described by CE. It must be diverse and independent from the reactor trip system. The staff would require that such a system meet IEEE-279.

Westinghouse

Plants must provide a modified scram system (MSS) that is diverse and independent from RPS. The staff would require that such a system meet IEEE-279 criteria. This is a new item, the basis of which is given in Section 2.3.4 below.

General Electric

(1) Plants must provide an ATWS rod injection (ARI) system that has been described by GE. It would include separate sensors and logic and redundant scram air header exhaust valves. The staff would require that such a system meet IEEE-279.

(2) Plants must modify the hydraulic portion of the existing BWR design to reduce the common-mode failure potential of the single scram discharge volume in some designs. The modifications would provide independent and diverse level sensing for adequate scram reliability in the event of drain line failure.

(3) Plants must provide a recirculation pump trip (RPT) that is equivalent to the approved Monticello design or the approved modification of the Hatch design.

(4) Plants must change the control system logic described by GE to reduce the frequency of vessel isolation transients and permit feedwater runback.

2.3.1.2 Other Instrumentation Modifications

(1) For PWRs, until Alternative 4A modifications are implemented, the instruments exposed to primary system pressure must be capable of withstanding the Alternative 3A peak calculated pressures.

(2) ATWS instrumentation system relied upon by the operator for a safe shutdown of reactor should be designed so that a single failure does not result in loss of required information to the operator in accomplishing the necessary functions for an ATWS event.

(3) Instrumentation channels that are redundant must be electrically independent, energized from reliable power supplies, and physically separated in accordance with Regulatory Guide 1.75.

(4) Those ATWS parameters that provide transient or trend information necessary for the operator to perform his functions must be recorded.

2.3.1.3 Actuating Circuits

PWR and BWR designs must provide diverse actuation circuitry for mitigating systems (AMSAC). This diverse actuation circuitry may already be provided on some plants.

2.3.1.4 Additional Hardware Modifications for BWRs

(1) Plants must provide actuation circuitry for the SLCS that is automatic and diverse from the reactor trip system. A two-minute delay may be included to decrease the frequency of expensive false actuations.

(2) Plants must make necessary changes to the SLCS suction and delivery piping as follows:

(a) The basic requirement for Alternative 3A is to allow both installed SLCS pumps to provide full flow concurrently, yielding about 86 gpm in large plants. New injection points into the primary system are needed to accommodate the increased flow from the presently designed 43 gpm and to provide improved mixing in the reactor.

(b) Inasmuch as these plants will be required to implement Alternative 4A later, it would be desirable if the Alternative 3A piping could also serve for Alternative 4A. Conflicting requirements to accomplish this are the need for larger pipe for Alternative 4A flow rate compared with longer injection delay for Alternative 3A flow in a larger pipe sized for Alternative 4A flow. Some optimization would be appropriate.

To summarize, repiping to accommodate 86 gpm with an acceptably low injection delay is required. Accomplishing the repiping so it would not have to be revised for 3-400 gpm flow of Alternative 4A would be highly desirable.

(3) Plants must provide quenchers to replace ramsheads for all SRV discharge lines. A program to do this is already under way.

2.3.1.5 Containment Isolation

Plants must provide diverse actuation circuitry for early automatic isolation of containment. This diverse containment isolation may be already in place on some plants.

2.3.2 Criteria

Alternative 3A will encompass all the requirements and criteria stipulated for ATWS prevention and mitigating systems for Alternative 3 plants in NUREG-0460, Volume 3. In addition, as a result of the information reviewed thus far, more emphasis should be given for the following items in Alternative 3A modifications.

(1) As a part of ATWS equipment qualification, it should be assured by analysis or test that the integrity and functional capability of critical instrument sensors, sensing lines and isolation valves will be maintained for ATWS peak pressures.

(2) The AMSAC for CE, W and B&W plants must conform to the single-failure criterion in accordance with Section 2.2 of NUREG-0460, Volume 3.

(3) The SLCS automatic actuation circuitry must satisfy requirements A through H of Appendix C of NUREG-0460, Volume 3, and be shown to have an availability comparable to the remainder of SLCS.

(4) Credit for operator action can be taken no sooner than 10 minutes following the onset of an ATWS if the information displayed is adequate and the operator is trained for ATWS.

2.3.3 Required Information

The following information regarding Alternative 3A modifications is to be supplied to NRC by December 1, 1980. Details are discussed in Appendix A.

Licensees and applicants should design, procure, and install the Alternative 3A modifications independent of NRC review of this information. The design and installation of the modifications, and of the additional information required, will be reviewed by the NRC concurrent with the implementation of these modifications.

(1) BWR plants - Detailed design description and capability of the modified SLCS.

(2) PWR and BWR plants - Design information for the modifications to protection systems and AMSAC, including the system design description, design criteria and bases, functional logic diagrams, schematic wiring diagrams, electrical power supplies and physical arrangement drawings.

(3) PWR and BWR plants - Information confirming independence and diversity between modifications in protection system/AMSAC and normal scram system, conformance of RPS modifications to the requirements of IEEE-279 and the conformance of AMSAC to the criteria in Appendix C, Volume 3 of NUREG-0460, as applicable.

2.3.4 Basis for Changes from Volume 3 of NUREG-0460

The general basis for requiring Alternative 3A is set forth in Section 2.1 and in Volume 3. The rationale for determining the scope of Alternative 3A modifications is also set forth in Volume 3, except insofar as Alternative 3A described in Section 2.3 of this volume is different from Alternative 3 as described in Volume 3.

The principal differences betwe Alternatives 3 and 3A result from the information furnished by the industry over the last year and the staff's concerns thereon, as discussed elsewhere in this report. Three items merit additional discussion as set forth in the following sections.

2.3.4.1 Required ATWS Prevention Design Modifications in Westinghouse Scram System

A significant difference between Alternative 3 in Volume 3 and Alternative 3A in this report is a requirement to provide, for Westinghouse plants, the same sort of accident prevention safety improvement (supplementary reactor trip instruments and logic) as has been specified for other PWR designs.

The basic rationale for this new additional requirement is defense in depth and the lessons learned from the TMI accident and other experiences that actual accident sequences do not progress as foreseen in safety analyses. Thus, we propose to require an additional measure of ATWS prevention on W plants.

PROS	CONS
1. An effective way of reducing the risk from ATWS events is to reduce the likelihood of an ATWS event. The TMI-2 accident further strengthens the need to empnasize prevention of accidents even if features are incorporated to mitigate the consequences.	1. The consequences of an ATWS event in the W design is not as sev as in other NSSS designs. Further, this severity is reduced to acceptable level with adequate mitigating systems. Thus, there is no need on this basis to modify the existing scram system.

PROS	CONS
2. In the existing W scram system, there is no diversity in the scram breakers, thus a CMF in these (two) breakers would negate the successful completion of the scram functions. Problems have been experienced in these breakers with CMF potential.	2. The current scram breaker scheme is a simple system with two breakers in series. Tripping of either breaker results in reactor scram. It is not clear that modifications to this existing simple design will in fact substantially improve the reliability in achieving the reactor scram.
3. It is possible that the consequences of ATWS events under some conditions (e.g., different sequence, less favorable MTC) would be more severe than calculated.	
4. The cost of this modification is believed to be modest, less than $1,000,000 per plant.	

2.3.4.2 Containment Structure Isolation Requirement

We believe the benefits of early containment isolation are twofold: (1) increased public health and safety, and (2) reduced impact on analysis and review resources. The public reaction to the accident at Three Mile Island shows that the offsite radiological consequences should be reduced as much as possible and, in our opinion, early containment structure isolation would significantly reduce the offsite radiological dose values without a significant economic impact. Furthermore, detailed offsite consequence calculations will be necessary if containment is not isolated early. Therefore, we require all LWR plants to implement measures to assure that containment is isolated early and the isolation signals and actuation circuitry satisfy the criteria stated in Section 2.3.1.5. Some changes to containment isolation systems have already been implemented as a result of lessons learned from TMI-2.

BWR Plants

General Electric has provided radiological consequence analyses using rapid isolation of the Mark III containment as a design assumption. With rapid containment isolation, the GE calculations are stated to show that their plants can withstand 100 percent fuel cladding failures and still have consequences which are less than the guideline values of 10 CFR Part 100.

Even though we do not entirely agree with the submitted analyses (see Appendix A), the significance of rapid automatic isolation of the containment is clearly highlighted by the GE results.

PWR Plants

The staff has estimated that between 10 to 20 percent of the fuel cladding in the core may well experience cladding failure in a severe ATWS event as a result of pellet-clad interaction. Using this information, the staff estimated that, dependent upon the ATWS sequence, containment isolation within 30 to 60 seconds after fuel failure would assure that the radiological consequences would be less than the guideline values of 10 CFR Part 100. Scoping calculations in Volume 2, Appendix VI, clearly show the effect of significant containment leakage if fuel failure occurs.

The importance of rapid containment isolation in the event of fuel failure is clear from the time period and the dose rates for containment leakage. If rapid containment isolation is not provided, detailed offsite dose analysis and plant systems descriptions will be necessary in order to verify that the calculated consequences will not exceed the 10 CFR Part 100 guidelines.

2.4 Alternative 4A

This alternative is similar to Alternative 4 in NUREG-0460, Volume 3. The only major difference between Alternatives 4 and 4A is that the Alternative 4A would require modifications to reduce susceptibility to common-mode electrical failures in scram systems, whereas Alternative 4 did not require this modification.

Completion of Alternative 4A on each plant will resolve the ATWS concern for that plant. Accordingly, the Alternative 4A package consists of (1) hardware changes in accordance with Section 2.4.1 and (2) analysis that the criteria of Section 2.4.2 are satisfied. In addition to material already submitted, the additional information of Section 2.4.3, as applicable, shall be submitted.

2.4.1 Hardware Description

(1) All changes described under Alternative 3A will have to be made, except that the BWR SLCS shall be modified in accordance with item (b) below.

(a) In addition, PWRs shall be modified if required so that the peak calculated pressure (Volume 3, Section 2.3) shall not result in primary stress exceeding service level "C" limit.

(b) The BWR SLCS pumping capability shall be upgraded to approximately 300 to 400 gpm to provide for reactivity, inventory, and oscillation control, and protection against simple failures. The SLCS shall be modified as needed to conform to Appendix C of NUREG-0460, Volume 3.

(c) BWR RPT hardware installed before July 1, 1981, may conform to the approved modified Hatch or Monticello designs. RPT hardware installed after July 1, 1981 shall conform to Appendix C of Volume 3.

(2) For plants in operation and well along in construction, safety improvement above Alternative 3A is required. For each such plant not conforming to Alternative 4A as set forth in this report, a study of alternative modifications shall be submitted to the NRC, with the objective being to achieve a level of safety equivalent to Alternative 4A, in both performance and reliability, or as near to it as can be accomplished within the constraints of diesel capacity, completed seismically qualified structures, and the basic plant layout. (Some have called this "Alternative 3½".)

2.4.2 Criteria

(1) All actuation circuitry for mitigating systems must satisfy the requirements of Appendix C of NUREG-0460, Volume 3.

(2) Calculated primary stresses must be within service level "C" limit.

(3) Calculated severe oscillations in BWRs must be eliminated.

(4) Radiological, primary stress, and containment limits shall not be exceeded even if single failure is considered separately in each of the mitigating systems. These limites are described in NUREG-0460 Volume 2, Appendix IV.

2.4.2.1 Requirements for Mitigating System Reliability

Appendix C of Volume 3 sets forth the requirements for mitigating systesms; Paragraph I of that appendix specifies reliability criteria. Two options are provided: (1) IEEE-271 requirements for the system, or (2) calculated unavailability of the system to be no greater than approximately 10^{-3} per demand at the 50 percent confidence level. The problem is to define "the system" and thus provide clarification of our requirements.

The acceptance criteria for ATWS mitigating systems are given in Volume 1, Chapter 7. They are also listed here:

(1) Limit calculated radiological doses within 10 CFR 100 guidelines.

(2) Limit primary system stresses to maintain integrity and operability of equipment.

(3) Limit fuel distortion to maintain core-cooling capability.

(4) Maintain containment within design basis.

(5) Provide long-term shutdown and cooling capability.

In Appendix A, there are lists for each design of the 10 to 20 hardware systems that collectively are relied on to perform these functions. There is much overlap, with some functions involving several systems, and some systems involving several functions.

Most safety-related systems are composed of multiple trains of equipment, the redundancy providing the needed reliability. Conversely, there is a type of anti-redundancy in that successful performance of several functions is required to mitigate postulated ATWS events.

The matter is considered in a slightly different way in the answer to Question 1 in Appendix B of Volume 3. It is estimated there that almost five hardware systems must function at any time. If they are of approximately equal reliability, then the overall unavailability equals five times the unavailability of one system.

In addition, it is worth noting that the subdivision into "systems" and "functions" is arbitrary. Five "functions" are thus set equal to fifteen "systems" (see Westinghouse exercise, Appendix A, Section 1.2).

These considerations suggest that the collection of equipment required to mitigate ATWS should be specified as to reliability, rather than one of the rather arbitrarily defined "functions" or "systems."

This is consistent in some respects with the reliability requirements for emergency core cooling systems (ECCS). ECCS performance calculations are required to be made under the assumption that the most limiting single failure has occurred. Different single failures are limiting for different situations; for example, large, intermediate, and small breaks. ECCS calculations also require the assumption of loss of offsite power, which is not required for ATWS calculations.

In order to establish quantitatively the reliability required of the ATWS mitigating system, we start from Volume 3, which gives approximately 10^{-3} per demand for "the system". The answer to Question 1 in Volume 3, Appendix B, suggests that the 10^{-3} value is meant for "an individual system," defined here (and not in Volume 3) as one of several required for ATWS mitigation. This implies a judgment that ATWS mitigation as a whole should have an unavailability no greater than approximately 10^{-2} per demand. The staff therefore proposes this as an alternative interpretation of Volume 3, Appendix C, Paragraph I:

> The totality of the ATWS mitigating equipment relied on in any calculation to show compliance with ATWS criteria shall have an unavailability no greater than 10^{-2} per demand, calculated in accordance with Paragraph I, Option 2, Appendix C, Volume 3.

But reliability calculations are notoriously difficult and contentious, so Appendix C, Volume 3 provides in Option 1 an alternative form of reliability requirement, involving an extention of IEEE-279. For this discussion, it is the simple failure criterion that is relevant. By reasoning similar to that in the foregoing discussion, the staff arrived at the following recommended criterion:

> Where quantitative reliability assessment is not used, any calculation to show compliance with ATWS criteria shall include the assumption of the most limiting single failure in the ATWS mitigating equipment.

The staff expects that most ATWS calculations to show compliance with criteria will be performed generically, as such calculations have been so far. Exceptions foreseen include the following:

(1) Calculations to establish the required number of added relief or safety values in PWRs.

(2) Calculations to establish the required flow rate of the modified SLCS in BWRs.

2.4.3 Required Information

The information delineated in this section is required to close out the ATWS problem on each plant. A great deal of the analysis can be performed generically;

in fact, a great deal of generic analysis has already been submitted. It is not our intention to create a new design basis accident out of ATWS. It is for this reason that the specifications of Alternative 4A are hardware-oriented. However, enough information must be available to provide verification of the safety adequacy of the plants.

2.4.3.1 Analytical Methods

Analysis methods have been reviewed by the staff in the 1975 Status Reports and in Volume 2. Except as they must be modified by TMI-related and other recent concerns, the older staff evaluations are still in force. Appendix A of this volume contains additional staff evaluations. However, they pertain primarily to the analyses submitted in the effort to verify Alternative 3, so they should be considered only to the extent that they are applicable to Alternative 4A analyses.

The objective is to use evaluation models that realistically predict the course of the ATWS event sequences being analyzed. The appropriateness and realism of the calculations, however, must be supported by either experimental evidence or more elaborate calculations, which in turn have to be verified.

The short-term ATWS models, those used during the early portion of the transient for the prediction of the initial pressure rise, have already been reviewed by the staff (Reference 2). The review identified certain shortcomings in the various models and indicated a lack of integral verification. It is the staff's position that the shortcomings of the models must be corrected in a timely manner to facilitate implementation of Alternative 4A. This must be accomplished promptly to allow design, manufacture, and installation of plant modifications on the required schedule. Integral verification should follow. We recognize the difficulties associated with integral tests, and therefore recommend that implementation of Alternative 4A should proceed with the corrected models subject to verification on a schedule that will not delay implementation of these modifications. Should the verification require changes in all or part of an evaluation model and result in design changes, all plants will be required to show conformance with the Alternative 4A requirements without lengthening the implementation schedule.

The long-term effects of ATWS (that is, possible core damage and recovery from the transient conditions) have not yet been evaluated in a realistic manner. Licensees and applicants must propose evaluation models appropriate for the analysis of this phase of the transient. These models should account for void formation in the reactor coolant system. Verification of the long-term models will be required on a schedule similar to the short-term models.

If for any reason the verification of either the short-term or the long-term models is not available in time to justify the adequacy of the modifications on the required schedule, an assessment of the uncertainties of the calculations will be required. Based on this assessment, sufficient conservatism will be incorporated into the affected models to compensate for the uncertainties of the calculations. Individual plants will then have to meet the Alternative 4A requirements and schedule with the conservative calculations.

A summary status of these codes and the required information for PWR as well as the BWR designs are described in the following.

BWR Short-Term Behavior

Information submitted for the REDY code (Reference 26) has been reviewed by the staff. The staff required additional information relating to modeling concerns and required that the code be verified by comparisons to the ODYN code (Reference 7). To address our concerns, General Electric submitted NEDE-24222. This report did not address the modeling concerns, but it presented some comparisons between the REDY and ODYN codes. For the comparisons presented, the REDY code predicts similar results of peak pressure although neutron fluxes and rates of pressure increase are different in the beginning of the runs. In order to facilitate the implementation of Alternative 4A, the staff will require additional information on the ODYN comparisons presented in NEDE-24222, additional ODYN comparisons, and a response to the staff's previously identified modeling concerns (Volume 2, Appendix XVI).

BWR Long-Term Behavior

The REDY code has not been reviewed for acceptability in predicting the long-term behavior of transients. From the review of NEDE-24222, the staff concludes that the use of the REDY code for long-term ATWS applications should be justified, and the results of the REDY code should be verified using a small-break LOCA code meeting appropriate ATWS requirements. In Reference 28, the staff presented requirements for the small-break LOCA analyses, some of which may also be applicable for ATWS analyses. In addition, key models in the REDY code should be verified using appropriate separate effects test data described in Appendix B.

PWR Short-Term Behavior - No Significant Void in the Primary System

Reactor vendors predicted peak pressures in their ATWS analyses using best-estimate system transient codes (LOFTRAN, CADDS and CESEC-ATWS). The analytical models comprising these codes were described in References 3-5, 11-19, which were reviewed by the staff. The review for the short-term behavior (Volume 2, Appendices XIV, XV, and XVII) included the early portion of the transient until the peak pressures were predicted. The staff required additional justification of some models and requested certain changes in the Babcock & Wilcox and Combustion Engineering models. Furthermore, the staff required verification of the B&W, CE and Westinghouse system transient codes by appropriate startup tests. To address staff concerns, Westinghouse, Combustion Engineering and Babcock & Wilcox submitted References 8, 18, and 23. The submittals did not include verification of the transient codes. Comparison with system transient test results have been reported by Westinghouse Reference 34); however, these tests have not been challenging enough to verify codes used for overpressure transients. Codes used to calculate peak pressures for ATWS analyses should be verified against high-pressure integral tests as discussed in Appendix B.

PWR Long-Term Behavior - Significant Void Formation in the Primary System

If voids are predicted to be present in the primary system, the LOFTRAN, CADDs and CESEC-ATWS codes are no longer applicable. The calculations must be repeated using acceptable best-estimate small-break LOCA codes meeting appropriate ATWS requirements. Some of these requirements may be similar to those for the small-break LOCA analyses presented in References 22, 35, 36. These LOCA codes should have predictive capability for phase separation and natural circulation. Code verification should be performed as described in Appendix B.

2.4.3.2 Transient Analyses

PWR and BWR Plants

Supply the following information:

(1) Provide the transient analyses for the limiting transients (and justification that they are limiting) with respect to plant overpressure and thermal conditions for fuel integrity, including consideration of single failures in mitigating systems (see Section 2.4.2, Item 4).

(2) Provide all the important input parameters used in transient analyses and their justification.

(3) Demonstrate the capability for long-term shutdown including functionability of valves, pumps, and pipes and justification for any dependence on operator actions.

(4) Indicate the operator actions during the transient including the means to be used for diagnosing ATWS events.

PWR Plants

(1) Transient analyses should consider the formation of voids in the primary system during the transients involving loss of primary coolant inventory. Subsequent shutdown considerations must describe the impact of the reactor coolant pump operation and HPSI designs.

(2) Give the 99 percent MTC value using Appendix C of NUREG-0460, Volume 4, guidelines. Information supplied must include the values of MTC measured at power for confirmation.

(3) The PORV must be considered as follows:

  (a) The probability that it is blocked initially may be neglected if the limitations of Appendix D are imposed.

  (b) Spurious opening of the PORV is an anticipated transient but probably not a limiting one.

  (c) Failure of the PORV to open during the ATWS pressure peak must be assumed for loss-of-offsite-power ATWS, unless the plant is modified to put the valve and all associated instruments on a power source that will not be interupted; the diesels are too late for this function.

  (d) Failure of the PORV to open on demand for non-loss-of-offsite-power ATWS may be a limiting single failure.

(e) Failure of the PORV to reclose plus failure to close the block valve may be a limiting single failure. Manual block valve closure may be assumed at t = 10 minutes.

BWR Plants

(1) In the case of any calculated instabilities and oscillatory behavior during the course of any transient, demonstrate that coolable geometry is assure including PCI considerations; justify all fuel failure assumptions.

2.4.3.3 Component Stress Analyses

PWR and BWR Plants

Sufficient documentation must be submitted to show the basis on which the plant level "C" allowable ATWS pressure limit is established and to demonstrate the operability of components essential to safe shutdown after an ATWS event. The following should be provided:

(1) If generic analyses submitted as of March 1, 1980 by NSSS vendor and owners' groups are to be referenced for one or more components, the deficiencies as described in Appendix A of this report must be resolved.

(2) For components (NSSS and BOP supplied) where level "C" allowable pressure is established by simple factoring up of component design pressure by

multiplying it by ratio of ASME Code level "C" stress limit to the design stress limit, furnish list of components evaluated and the minimum allowable level "C" pressure thus determined. Most Alternative 4A components should be in this category. Details of these evaluations should be kept on file at the plant site, available for NRC review on a case-by-case basis.

(3) For NSSS and BOP-supplied components that must operate to assure safe plant shutdown (i.e., RHR system isolation valves, reactor coolant pumps, pressurizer heaters, etc.), information supplied shall be of the detail described in the February 15, 1979 letter from R. Mattson and Appendix A of NUREG-0460, Volume 4. Discuss the specific concerns noted for such equipment.

(4) For BWRs only, resolve the concerns expressed in Appendix A, Section 4.3, of this report regarding evaluation of component integrity and operability for combination of ATWS pressure load and safety relief valve (SRV) hydrodynamic load.

2.4.3.4 Containment Analyses

PWR Plants

No data apply for containment analyses of PWR plants.

BWR Plants

(1) The initial pool temperature used for best-estimate analysis shall be no lower than the annual average temperature experienced or predicted for the facility. A sensitivity calculation shall be performed using the technical specification limit for the initial pool temperature.

(2) The temperature difference between the bulk average pool temperature and the local temperature into which the bubbles are condensing (inflow temperature) shall be based on applicable in-plant tests, extrapolated to ATWS conditions using the methods provided in pressure suppression containment criteria under development under NRC Unresolved Safety Issue A-39.

(3) A local pool temperature limit of 200°F shall be applied to the best-estimate calculation (assumes quenchers are installed). The results of the sensitivity calculation shall be compared with applicable data and analysis to determine acceptability.

2.4.3.5 Fuel Behavior

PWR Plants

Using an approved PCI model (e.g., PROFIT), estimates must be provided of the number of fuel rods that are calculated to fail due to PCI during the "worst-case" power-increasing event. The "worst-case" (in terms of fuel failure probability) should be determined and may vary with the vendor and plant design. To make core-wide assessments of failure probability will require a census of initial power distributions, burnups, etc., and correlation of that information with the specific power history for a given transient. A bounding estimate of 10 percent may be used instead of the calculation.

BWR Plants

Based on the cladding temperature information submitting in the latest GE reports, 100 percent failed fuel may be a reasonable assumption for the present. GE claims that 100 percent of the fuel rods could fail without exceeding Part 100. This claim is evaluated later in Appendix A, Section 4.6. Such an assumption would preempt the need for a PCI or DNB analysis. If the ability to withstand 100 percent failed fuel is not verified, a fuel damage model that includes PCI must be used to give calculations of core damage fractions that are not optimistic.

In addition to the dose concern, there is also a concern about coolable fuel geometry. It appears that for some events temperatures may be high enough to cause cladding waisting and collapse at about the same time that centerline UO_2 melting and thermal expansion are causing severe pellet/cladding mechanical interaction. Depending on the amount of plastic deformation and the type of cladding breach, a variety of interactive effects are conceivable. However, with the implementation of Alternative 4A and the suppression of severe oscillations, we believe that most of the concern about maintenance of coolable geometry is eliminated.

The only remaining coolable geometry concern would then involve the high neutron flux (power) spike that occurs early in some of the BWR ATWS events. That behavior is quite similar to that of a control rod drop reactivity initiated accident (RIA), except that for ATWS the entire core is involved and the period of the reactivity insertion is different. Because of the similarities (in terms of fuel duty) between the RIA and ATWS events, however, RIA coolable geometry criteria may be used. A new (proposed) coolable geometry criterion is being discussed for RIA evaluation. This criterion would limit the local (not radially averaged) peak fuel enthalpy to 267 cal/g. Therefore, with the suppression of oscillations, maintenance of coolable geometry may be demonstrated by assuring that the local fuel enthalpy does not exceed 267 cal/g. If 267 cal/g is exceeded, it will be necessary to provide a mechanistic analysis to demonstrate maintenance of coolable geometry.

2.4.3.6 Radiological Evaluation

Westinghouse

(1) Justify terminating dose calculations at 30 hours and present the data as 0 to 8 hours, 8 to 24 hours, etc.

(2) Specify the assumed residual heat removal (RHR) operation time.

(3) Provide release information for plants of lower power and smaller containment volumes.

Combustion Engineering

(1) Provide and justify the steam generator leakage values versus time.

(2) Provide reactor coolant system (RCS) leakage and discharge (water and steam) to the containment as a function of time.

(3) Provide the information and analyses requested in the February 15, 1979 letter from R. Mattson.

Babcock and Wilcox

(1) Provide relative values of releases for t e 177-FA and 205-FA plant pathways to compare the doses from the two designs.

(2) Provide estimates of primary system leakage.

(3) Identify and use the most limiting X/Q values for B&W plants.

(4) Provide RCS leakage and discharge (water and steam) to the containment as a function of time.

(5) Provide the information and analyses requested in the February 15, 1979 letter from R. Mattson.

General Electric

(1) Provide dose analyses for turbine trip and MSIV closure ATWS events or justify why these events are not limiting.

(2) Provide releases to the environment prior to and following containment isolation. Include releases to the suppression pool and/or the main condenser as a function of time.

(3) Provide reactor coolant system leakage (e.g., through MSIVs) or justify its exclusion from the dose analyses.

(4) For BWR-4/5/6 plants, provide the information and analyses requested in the February 15, 1979 letter from R. Mattson.

2.4.3.7 Design Information

All vendors are required to furnish design information comparable to that required in a FSAR, thus what is usually available after the initial staff questions have been answered. They must include sufficient information to demonstrate compliance with the criteria of Section 2.4.2.

2.4.3.8 Optimization Study

For operating plants and those well along in construction (see Section 3.1.2) where full compliance with Alternative 4A is not proposed, furnish a study, including alternatives, showing the maximum ATWS capability that can practically be achieved. The objective is to consider various ways to maximize ATWS protection within the constraints imposed by basic plant arrangement, diesel capacity, and large seismically qualified structures. The staff wishes to encourage and reward innovation and ingenuity in optimization. Examples include the following:

BWR Plants

(1) Different chemical for liquid poison to increase negative reactivity inspection rate, e.g., enriched sodium pentaborate, gadolinium solution.

(2) Various combinations of SLCS pumps, with switching and valving alternatives to maximize the effectiveness of existing diesel generators.

(3) Where diesel capacity imposes a limit on liquid poison injection flow, provide some pumping capability that can operate only on offsite power (which is predicted to be available for most ATWS sequences).

(4) Provide high-pressure SLCS accumulators to give enhanced initial flow rate.

PWR Plants

(1) Various possible locations besides the pressurizer for additional safety or relief valves; higher pressure settings to decrease probability of

their spurious opening. Additional power-operated relief capacity may also help combat extended loss-of-feedwater events in plants with limited HPSI shutoff head.

(2) Tradeoffs among core design, boron concentration during operation, solid burnable poison, and relief capacity. Figure 1 shows a proposal from an industry source that may have merit.

2.4.4 Basis for Alternative 4A

The rationale for requiring Alternative 4A on virtually all plants is summarized in Chapter 1. Basically, it is the need for ATWS safety improvement and the failure to obtain verification that Alternative 3 is adequate over the lifetime of many plants.

The technical basis for Alternative 4A is Alternative 4 in Volume 3, modified in the light of the information developed since Volume 3 was written. In addition, the cost-benefit ideas from Volume 3 have been applied in a modified form to the Alternative 4A requirements, for operating plants and those nearing completion, to take into account the constraints imposed by basic plant arrangement, diesel-generator capacity and seismically qualified structures.

Detailed discussion of value-impact considerations are given in Volume 3 and in Chapter 4 of this report. In this section, we give a simplified discussion of the major considerations.

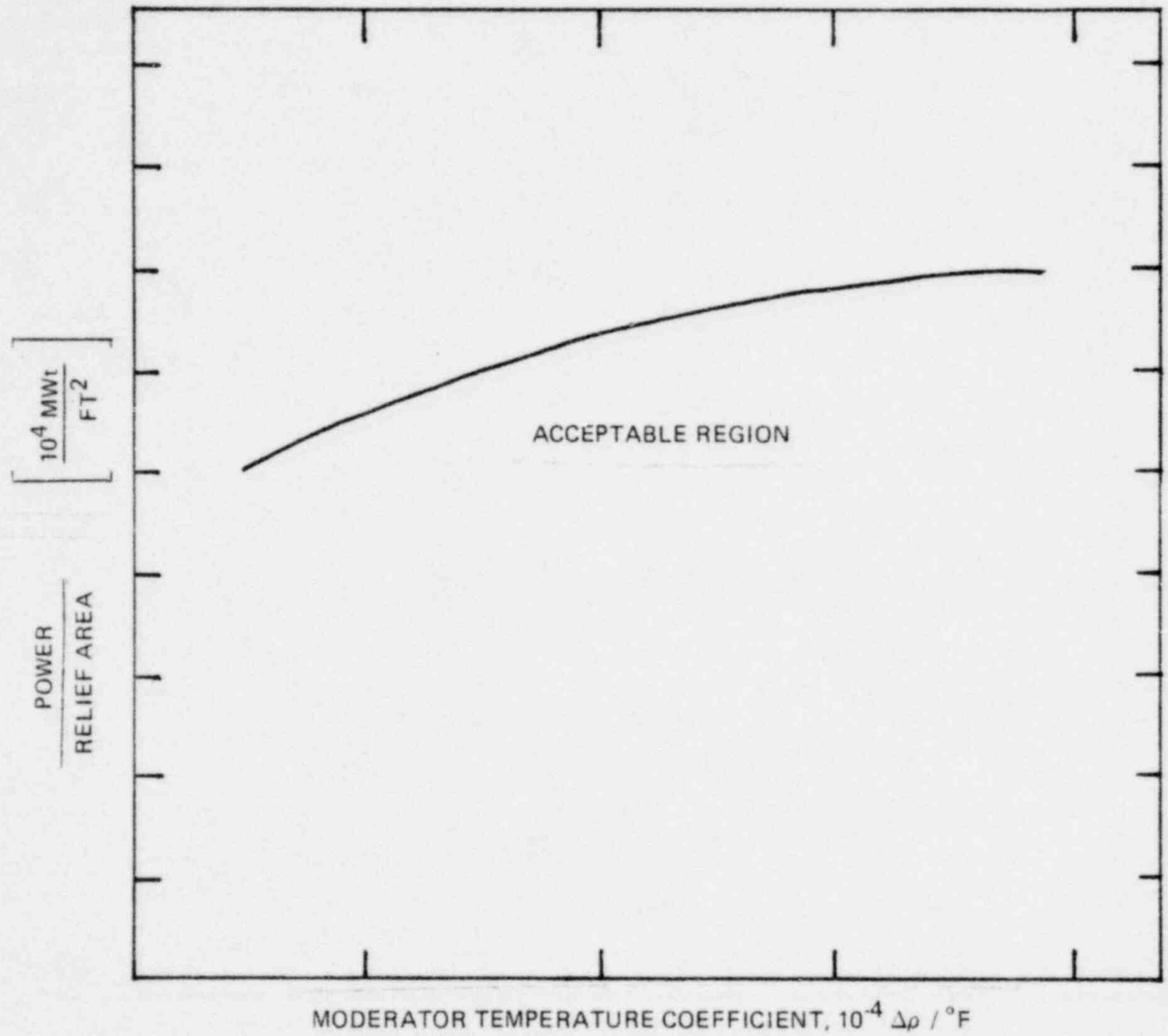

Figure 1. ATWS mitigation parameter diagram.

Listed below are the principal pros and cons of requiring Alternative 4A, compared to requiring only Alternative 3A.

PROS	CONS
A. General	
1. Reduce likelihood of reactor damage, radioactivity release, or core melt; see Appendix E and Volume 3.	1. Director and indirect cost of additional modifications.
2. Minimize additional analysis and information since Alternative 4A is much more likely to be verifiably adequate; thus reduce uncertainty in requirements.	2. Occupational exposure from modifications in operating reactors.
3. Phased implementation to minimize downtime.	
4. Decreased dependence on operator action.	
5. The proposed mechanism (non-DBA) for implementing the necessary plant modifications achieves the desired safety and yet minimizes manpower resource impact.	

	PROS	CONS
B. PWR		
	1. Reduce primary peak pressure to value assur ng integrity and coolability.	1. Increase probability of small-break LOCA if extra relief or safety valves are installed.
	2. Eliminate need to count on and analyze reactor vessel head lifting.	
	3. Reduce likelihood of steam generator tube leakage or rupture from ATWS.	
	4. Reduce likelihood of failure of instruments exposed to primary fuid.	
C. BWR		
	1. Shut down reactor faster with reduced energy, inventory requirement.	1. Increase probability of inadvertent boron injection-cleanup cost (of course this is a con for Alternative 3A also).

PROS	CONS
2. Eliminate or greatly reduce severity of oscillations.	
3. Reduce containment temperatures and loads.	

3. IMPLEMENTATION

The proposed implementation for ATWS requirements is given in Section 1.4. The rationale behind these requirements, and the schedules for their implementation, is given in Chapter 1 and a value-impact discussion is given in Chapter 4.

3.1 Classes of Plants

As set forth in Section 1.4, ATWS requirements are aggregated into two groups of plants.

3.1.1 Early Plants (Alternative 2A)

Perform plant-specific ATWS analysis, and propose plant modifications based on optimization study by July 1981.

These plants are the following:

Dresden 1	Haddam Neck
ankee-Rowe	Lacrosse
*Indian Point 1	Oyster Creek 1
*Humboldt Bay	Nine Mile Point 1
Big Rock Point	Ginna
San Onofre 1	

3.1.2 All Other Plants

The basic requirement for all plants (except the eleven early plants listed in the preceding) is Alternative 4A. However, since this can take a long time and since doing it as quickly as possible could be very expensive, the staff is recommending both interim improvements and equipment optimizing for operating reactors and those plants that are well along in construction.

No list or criteria are used to segregate these plants into categories. Of course, generic solutions are encouraged for classes of plants to decrease the resources required for industry design and NRC review.

(1) All plants shall implement Alternative 4A, as described in Section 2.4, or as modified by Paragraph C by January 1984.

*These two plants are shut down indefinitely.

(2) All plants in operation or coming into operation that do not confirm to Paragraph A shall implement Alternative 3A by July 1, 1981 (electrical) and July 1, 1982 (piping) or by the date of the operating license, whichever is later.

(3) Each plant for which conformance to Alternative 4A is deemed not practicable, because of constraints improved by basic plant layout, diesel capacity, or completed seismically qualified structures, shall submit by December 31, 1980, the optimization study set forth in Section 2.4.1, including alternatives for achieving a level of safety equivalent to Alternative 4A. This alternative (sometimes called "Alternative 3½") is intended for operating plants and those well along in construction.

Duplicate plants at the same site may be modified identically, even if the second unit is not as far along in construction as to fall within this provision, if the first unit qualifies.

Duplicate and replicate plants at different sites, and plants referencing standard designs, shall be judged the same as non-standard plants; that is, improvement above Alternative 3A is required on a schedule that is consistent with avoiding inordinate costs and is within the constraints of hardware already completed.

3.2 Form of Licensing Requirements

Volume 1, 2, and 3 of NUREG-0460 as well as this volume have specified the need to reduce risk from ATWS events. Volumes 3 and 4 have considered the potential impact and benefits that are believed to be achieved for a variety of different alternative plant modifications. In Volume 3, the staff stated that they intended to propose to the Commission modified regulations to include a requirement for protection from ATWS events.

The full rulemaking procedure may take a long time, but the results are binding on NRC and the industry and thus would result in minimum additional delays prior to implementation of the necessary plant modifications. However, for reasons discussed in Chapter 1 of this report, we have not yet been able to propose an ATWS rule to the Commission.

As discussed in the preceding section, we have developed a two-step process to achieve the necessary protection from ATWS events. The first step involves requiring early implementation of hardware modifications described in Chapter 2 as Alternatives 2A and 3A; the second, later upgrading to Alternative 4A to achieve the level of protection deemed to be necessary over the lifetime operation of LWRs. Early implementation of Alternatives 2A and 3A would permit an opportunity for considering all points of view (possibly through a rulemaking procedure) prior to implementation, if required, of Alternative 4A.

Implementation of ATWS requirements could be required using any one of the following three alternatives:

(1) Promulgate interim effective rule.

(2) Establish staff position and require compliance (e.g., through orders on a case-by-case basis).

(3) Issue regulatory guide for comment and interim guidance.

The staff intends to propose using orders to implement Alternatives 2A and 3A for the early phase of ATWS resolution.

Implementation of Alternative 4A, for which more time is available and more controversy exists, should be in accordance with a regulation adopted following rulemaking proceeding.

4. VALUES AND IMPACTS

Table 2, Summary of Values and Impacts, in this section updates Table 9, Appendix XII, Volume 2, NUREG-0460. This modified table reflects current information available. Appendix XII, Section 1.4.8, "Cost of Replacement Power," has also been updated. The updated analysis assumes complete loss of one nuclear unit and unavailability of the second unit for 3 years. This differs from the original analysis because the first analysis assumed both units would be permanently removed from service following an ATWS-caused core melt. The replacement power cost for the present analysis is based on an average plant capacity factor of 65 percent, whereas the original analysis assumed a 60 percent capacity factor. The years 1985 and 1990 were the estimated dates of installation for plants under construction and future plants.

All costs for this updated analysis are in 1980 dollars. For this analysis, a real discount rate (the rate after adjusting for inflation) is 3 percent. The original analysis used a discount rate of approximately 5 percent (that is,

TABLE 2. SUMMARY OF VALUE AND IMPACT ALTERNATIVE ATWS REQUIREMENTS, 1980 DOLLARS IN MILLIONS PER PLANT LIFETIME

	Operating Plants				Plants Under Construction (Construction Permit Issued)				Future Plants (Construction Permit Not Issued)	
Plant Designer	Alternative 3A		Alternative 4A		Alternative 3A		Alternative 4A		Alternative 4A	
	Impacts	Values	Impacts	Values	Impacts	Values	Impacts	Values	Impacts	Values
Babcock & Wilcox	2.8	2.6 - 7.4	4.4	6.0 - 17.0	2.0	2.6 - 6.8	2.9	6.2 - 12.8	2.5	5.4 - 11.7
General Electric	2.6	2.6 - 7.4	4.4	6.0 - 17.0	1.8	2.6 - 6.8	2.9	6.2 - 12.8	2.5	5.4 - 11.7
Westinghouse	1.7	3.8 - 8.6	2.5	3.8 - 8.6	1.2	4.0 - 7.2	1.8	6.2 - 12.8	2.0	5.4 - 11.7
*General Electric	3.5	31.8 - 54.8	13.0	42.7 - 87.0	3.2	48.8 - 80.5	10.8	51.4 - 84.7	6.7	44.0 - 64.6

*No cost was included for cleanup and downtime resulting from inadvertent actuation of poison injection system (estimated $200,000 to $8,000,000 per plant lifetime).

(1. ÷ 1.05) - 1 = 0.0476). This has the effect of making the current cost estimates higher. The capital cost of a nuclear plant in the original study was $600 per kW in 1978 dollars; this analysis uses $880 in 1980 dollars. Finally, the replacement power costs and all other values in Tables 2.2, 2.4, 2.6, and 2.8 of Appendix XII, Volume 2, were adjusted by the change in probabilities for each alternative used in this analysis. The remainder of Appendix XII remains unchanged.

4.1 Values

The values of implementing the staff ATWS requirements are the improvement in safety from the equipment and operating changes, the increased availability of power production facilities that might otherwise be disabled or shut down due to an ATWS event, and any other benefits that might be derived by averting risk. The principal values were the reduction in man-rem exposure valued at $1000 per man-rem, reduction in onsite and offsite property damage, averted cost of the need to replace a disabled plant, averted cost of shutting down other light water reactors in response to an ATWS event, and unfavorable impacts of increased oil imports for replacement power. Other values which were recognized, but not incorporated into the analysis, were averted risk of adverse effects on domestic nuclear reactor sales, and averted sociological impacts and disruptions.

4.2 Impacts

The impacts consist of direct and indirect costs. The direct costs are the cost of the equipment and its installation. The indirect costs are estimated to be equal to the direct cost for this analysis. However, Volume 3 of NUREG-0460 states that for particular plants and utilities these costs could exceed the direct cost by several factors. The indirect cost includes items such as licensing costs, operating and maintenance, if any, unidentified analysis, financing and escalation, taxes and insurance, contingency funds, and radiation exposure. A review of cost estimates from the Washington Public Power Supply System, January 2, 1979 letter to R. Mattson, shows the indirect cost to be considerably less than the direct cost. Downtime costs were not included because of the period of time allowed for the installation of the major equipment.

For B&W and CE, the major difference between Alternatives 3A and 4A is the addition of safety valves. (Of course, other less expensive solutions may be proposed by the industry.) Plants under construction will not be forced to install Alternative 4A immediately; however, Alternative 4A could be installed with little impact on construction completion or operating license for plants in the early stage of construction. This would avoid any radiation exposure, and testing would be accomplished along with startup testing. This is believed to lower the cost considerably; however, this option will be left to the utilities. The staff has not included the cost of replacement instruments, if required to withstand high pressures, because the staff does not have information on the additional sensors needed and the cost associated with these sensors.

For Westinghouse reactors, there are essentially no differences in the hardware modifications required under Alternatives 3A and 4A.

For GE BWRs, the major difference between Alternatives 3A and 4A is the requirement of higher flow capability of SLCS for the Alternative 4A plants. As discussed in Section 3.1.2 for operating plants and those well along in construction, an optimization study is to be performed to determine the additional changes to SLCS beyond those required under Alternative 3A that could be implemented, taking into consideration the constraints of plant layout, diesel capacity, or seismically qualified structures. As in the case of PWRs, downtime costs were not included because of the long period of time allowed for installation of this equipment.

4.3 Occupational Exposure

Occupational exposures would occur during any installation of modifications in operating plants that might be required. Some exposure would occur in all plants during operation and maintenance. The average occupational exposure for replacement of steam generators at Turkey Point was estimated to be less than 250 to 500 man-rem per steam generator. In consideration of the relative difficulty of steam generator replacement, the staff expects exposure resulting from installation of ATWS modifications to the primary system to be less than (probably much less than) 250 man-rem per operating plant. If one assumes a value of $1000 per man-rem, the monetized impact would be less than (probably much less than) $250,000.

In the July 13, 1978 presentation at the ATWS Working Group Meeting, Stone & Webster estimated what appear to be occupational exposures for operation and maintenance from implementing NUREG-0460 ATWS mitigation. The estimates were $30,000 and $320,000 for PWRs and BWRs, respectively. These figures would support our earlier judgment that such impacts would probably be negligible.

4.4 Cost of Inadvertent Operation

Inadvertent actuation of the ATWS mitigation system for PWRs would not appear to have high probability or severe consequences. Therefore, the following discussion is concerned with BWRs. The staff does not have data for frequency of inadvertent trips of a BWR automatic boron injection system since no system is now installed. However, EPRI NP-801, "Frequencies of Anticipated Transients," provides the following related experience:

(1) Inadvertent startup of HPCI* in BWRs - zero events in 49 reactor-years.

(2) Inadvertent safety injection signals resulting in scram in PWRs - three events in 118 reactor-years.

The staff expects the logic for the automatic boron injection system to minimize its inadvertent operation. The staff has conservatively used the above data to assume an inadvertent actuation frequency of 2.5×10^{-2} per reactor-year for the automatic boron system.

Experience with boron cleanup is limited to one instance at Dresden, which required a cleanup time of 54 hours. For specific plants, cleanup time would be influenced by waste storage tank capacity and evaporator capacity. Therefore, some plant-to-plant variation would be expected in cleanup time. GE estimates

*Data are only provided in the range of 25 percent of 110 percent power.

the cleanup time as 3 days to 2 months. Based on these considerations, our estimate of cost in millions of dollars per reactor lifetime (Table 3) would be the following:

TABLE 3. COST OF INADVERTENT OPERATION, MILLION DOLLARS PER REACTOR LIFETIME

Cleanup Time	Cost ($ million)
54 hours	0.2 - 0.3
2 weeks	1.5 - 1.8
2 months	6.5 - 7.8

5. CONCLUSIONS

The staff has reached the following conclusions:

(1) The staff has seen no information to change its previous conclusion that the present likelihood of severe consequences arising from an ATWS event is acceptably small, but that the future likelihood of severe ATWS consequences could become unacceptably large and measures should be taken to diminish such consequences.

(2) These measures should include both ATWS prevention and ATWS mitigation.

(3) The desired level of safety should be achieved at as low a cost as possible.

(4) Early generic verification of the acceptability of Alternative 3, proposed in Volume 3 for all operating plants (except the 11 oldest plants) and many plants under construction, has not been achieved. The staff believes that further efforts along this line would be a diversion of scarce resources.

(5) Based on current information, the staff estimates (see Appendix E) that implementing Alternative 3A would decrease the ATWS risk by a factor of 20 for BWRs, a factor of 2 or more for CE and B&W plants, and a factor of 2 or 80 for W plants, the latter figure being for plants where the scram and mitigating systems are not now diverse.

(6) Based on current information, the staff estimates (Appendix E) that going from Alternative 3A to Alternative 4A would decrease the ATWS risk by an

additional factor of 10 for BWRs and a factor of 25 for B&W and CE plants; for W plants, there is no hardware or risk difference between Alternatives 3A and 4A.

These risk estimates are approximate point estimates that are based primarily on hardware reliability and plant response analysis. Additional factors such as unforeseen human errors and undiscovered common failure modes would tend to limit the risk improvement achievable through hardware changes.

(7) The additional safety improvements of Alternative 4A, which the staff believes should be provided, would be very expensive in downtime if implemented quickly on operating plants, or if its implementation required major plant rearrangement. Accordingly, the staff has provided for implementation over several years, and for "Alternative 3½" to optimize ATWS protection in plants where major changes would be required for Alternative 4A.

(8) In view of the delay involved in implementing Alternative 4A (or 3½) as the staff proposes for improved cost-effectiveness, the staff also proposes prompt implementation of Alternative 3A for interim safety improvement.

(9) Additional analyses and verification will be required for Alternative 4A.

(10) The staff continues to recommend Alternative 2A for the 11 oldest plants.

(11) Alternatives 2A, 3A, and 4A are modifications of Alternatives 2, 3, and 4 in the light of information developed in the past year. The differences are shown in Table 1, Chapter 1, and are discussed in Chapter 2.

(12) The ultimate ATWS requirements, Alternative 4A, should be imposed by regulation after a rulemaking proceeding.

(13) Near-term improvements, Alternative 3A, should be imposed by orders. A regulation, although significantly slower, could also be used.

APPENDIX A

REVIEW OF RECENT INDUSTRY SUBMITTALS

INTRODUCTION

In Volume 3 of NUREG-0460 (Ref. 1), the Nuclear Regulatory Commission's (NRC's) staff report on anticipated transients without scram (ATWS), it was recommended that prior to the Commission's consideration of a proposed ATWS regulation, certain generic safety analyses should be performed. These analyses were to confirm that the proposed modifications for various classes of light water reactor (LWR) designs accomplish the degree of ATWS prevention and mitigation described by the staff in Volume 3 of NUREG-0460. If the generic analysis approach was successful, the rule to be proposed for Commission action would not treat ATWS as a design basis accident and would not require a new safety analysis of ATWS on each licensing case. There could be specific exceptions in the future where an analysis for a particular design would be desirable or necessary because the present generic analyses do not envelop that specific design or some future, unanticipated mode of normal operation.

Generic questions and guidelines were provided to the industry in letters dated 2/15/79 from R. Mattson (Ref. 2) for two kinds of plant modifications recommended in Volume 3 of NUREG-0460. These are the Alternative 3 modifications for plants receiving a construction permit prior to January 1, 1978, and the Alternative 4 modifications for plants receiving a construction permit after January 1, 1978. The plants which began operation prior to Dresden 2 will be treated according to Alternative 2 of Volume 3 and will be examined on a case-by-case basis.

This appendix provides the staff evaluation of the industry responses to the February 15, 1979 set of questions (Ref. 2).

1. WESTINGHOUSE DESIGNS

1.1 Analytical Methods

Status

The analytical models used in the analyses have been presented in References 3 through 6. The evaluation of these models by the staff has been presented in Reference 7 when these models were accepted for the ATWS analyses subject to completion of the review of the TRANFLO code (Ref. 6) and verification of the LOFTRAN code (Ref. 3) using plant experimental data to be obtained from various startup tests. The LOFTRAN code is a system transient code used to calculate various system parameters such as flows, temperatures, and pressures. The TRANFLO code is used to calculate the steam generator water level in the secondary side in order to input the heat transfer coefficient to the LOFTRAN code. The staff has not completed the review of the TRANFLO code. However, the staff review has progressed to the point of determination that there is a reasonable assurance that the code will be acceptable to the staff after completion of the review. Conclusions derived from our completed review will be applicable to the analyses.

The LOFTRAN code had been compared against some plant data even before Reference 7 was published. However, these tests are relatively mild and the verification of the code is not complete.

Impact of TMI-2 Event

The TMI-2 accident demonstrates the importance of extending analyses to show that stable plant conditions are obtained. Therefore, analyses demonstrating this capability should be performed. Information submitted to the staff (Ref. 3) indicates that the models in the LOFTRAN code predict the system behavior adequately when the primary loop fluid flow is in single phase. It is our position that the system transient code, LOFTRAN, is an inappropriate code to predict system behavior after the formation of the voids in the primary loop.

The analysis performed using the LOFTRAN code indicates that approximately 3000 ft^3 of fluid has been discharged through the valves in a four-loop Westinghouse 3423 MWt plant between 60 and 250 seconds following a loss of load ATWS event. The total volume of the reactor coolant system is reported to be 12,520 ft^3. Hence, there is approximately 25 percent vapor fraction in the primary loop at 250 seconds, assuming that all of the valves function properly and there is no leakage through the seals of the primary loop. Our judgment is that two-phase flow conditions will exist with phase separation in the primary loop. The main coolant pumps are predicted to trip due to cavitation in approximately 150 to 200 seconds. Thereafter, the core is cooled via natural circulation. HPI is initiated at about 10 minutes depending on the system pressure and the HPSI design. The phase separation phenomenon would redistribute the voids in the core and primary loop. Void redistribution in the core would change the reactivity feedback and would consequently cause power variation until boron injection. In order to predict these phenomena, an acceptable small break LOCA code should be used.

Verification of the Codes

Verification of the ATWS codes will be performed as described in Appendix B.

1.2 Transient Analysis Evaluation

NUREG-0460, Volume 3 (Ref. 1), specifies that in order to meet the requirements of Alternative 3 all Westinghouse plants should be provided with the alternate mitigating system actuation circuitry (AMSAC) which should be separate and diverse from the regular reactor protection system (RPS).

In the system analysis part of the submittal (Ref. 8), Westinghouse performed transient analyses for five most limiting transients in an attempt to demonstrate that with AMSAC all Westinghouse plants could withstand ATWS events without exceeding the safety limits.

The analyses were performed using the following Westinghouse codes:

- LOFTRAN (system code)
- FACTRAN (thermal fuel code)
- THINC-III (thermal-hydraulic code for DNBR calculation)
- TRANFLO (general purpose code for calculating steam generator performance)

These codes were used to evaluate system performance during the following transients, assuming failure of the normal scram function:

(1) Loss of External Electrical Load and/or Turbine Generator

(2) Complete Loss of Normal Feedwater

(3) Loss of Offsite Power

(4) Accidental Depressurization of the Reactor Coolant System

(5) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

These transients were considered by Westinghouse to be limiting even though no explicit basis was given for making this decision. The transients analyzed could be divided in two categories: one containing the transients in which failure of the normal scram function could be mitigated by the system without intervention of the AMSAC (this group includes the ATWS events that do not lead to a loss of heat sink and do not need short term mitigating actions) and second containing the transients in which the failure of the normal scram function require AMSAC operation for mitigating ATWS. In these transients, AMSAC actuates the auxiliary feedwater system and/or trips the main turbine. Transients 4 and 5 belong to category 1 events and transients 1, 2 and 3 to category 2 events.

The evaluation analyses were performed for a standard W four loop plant with four different steam generators (5 Series, 44 Series, Model D and Model F). The input to the analysis represented a composite of conservative parameters so that many plants could be bracketed by the reference analysis. In addition, sensitivity studies for the most sensitive parameters were performed.

The systems relied on in those analyses are as follow:

(1) ATWS Mitigating System Actuation Circuitry

(2) Reactor Primary Coolant System including Main Coolant Pumps

(3) Pressurizer Pressure Control (sprays and heaters)

(4) Pressurizer Power Operated Relief Valves (two valves)

(5) Pressurizer Safety Valves (two valves in 2 loop plants and three valves in 3 and 4 loop plants)

(6) Main Feedwater Supply System

(7) Condensate Supply System

(8) Auxiliary Feedwater System (two trains)

(9) Atmospheric Dump Valves

(10) Trubine Bypass Valves

(11) Steam Generator Safety Valves

(12) Main Steam Isolation Valves

(13) Turbine Electrohydraulic Control and Mechanical Hydraulic Emergency Trip Systems

(14) High Pressure Safety Injection System (two trains)

(15) Residual Heat Removal System

In the system transient analyses, two acceptability criteria were used: (1) system pressure, and (2) DNBR attained during the transient. It was shown in Table A.1 that for all the transients analyzed with the assumptions conforming with Alternative 3, both these criteria were met.

TABLE A.1 ACCEPTABILITY CRITERIA

Transient	Peak RCS Pressure, psia	DNBR
Loss of load	2979	> Initial value
Loss of normal feedwater	2857	> Initial value
Loss of offsite power	2611	1.30
Accidental depressurization	< Initial value	1.45
Uncontrolled rod withrawal	2428	

In addition, Westinghouse has indicated that no additional hardware change would be required to meet the criteria of Alternative 4. Additional analysis with 99 percent MTC would, however, be required.

Staff Evaluation

The staff has reviewed the Westinghouse submittal (Ref. 8) and found several areas which were deficient. These areas are detailed below:

(1) Westinghouse provided analyses for five transients representing limiting Condition II events. Explicit justification was not given why these particular events were chosen.

(2) In four out of five transients analyzed, credit is taken for operation of the PORVs. Westinghouse also performed a sensitivity study assuming one PORV inoperative. However, no analysis was provided assuming both PORVs

isolated. This is a credible situation which actually exists in some operating plants.

(3) For each transient analyzed, Westinghouse did not identify the systems required to bring the plant to shutdown condition and maintain the plant at that condition.

(4) Westinghouse did not justify the range of the parameters used in the sensitivity studies to cover system characteristics expected for each class of plants analyzed. The sensitivity studies did not include all the parameters required by the February 15, 1979 letter (Ref. 2). Westinghouse also did not provide sensitivity studies with multiple parameters changed simultaneously.

(5) Westinghouse did not provide all the initial/boundary conditions and parameters required by the February 15, 1979 letter (Ref. 2). Westinghouse did not consider the effect of different HPSI design on ATWS. This could have an important impact on the course of some transients such as a stuck-open PORV.

(6) In evaluating the long-term shutdown after an ATWS, Westinghouse did not consider the potential effect of the voids generated in the primary system during the stuck-open pressurizer safety/relief valve event or as a consequence of another ATWS event). These voids may prevent natural circulation of the primary coolant. Also, the role of reactor coolant pumps as well as HPSI for long-term shutdown has not been addressed.

(7) Westinghouse did not consider the effect of high pressures achieved during the postulated ATWS events on the plant control systems and instrumentation.

(8) The operator actions taken 10 minutes after the beginning of the postulated ATWS events and the bases for these actions were not provided.

In addition to the analyses based on the Alternative 3 assumptions, Westinghouse has performed some analyses based on the Alternative 4 requirements. Two transients were analyzed using 99% MTC and in the sensitivity studies failure of one PORV to open was considered. However, single failure in other systems, including the failure of one auxiliary feedwater train, was not included in the analyses.

1.3 Electrical Evaluation

1.3.1 Summary of Information Submitted

Westinghouse has provided a discussion of AMSAC design bases (Ref. 8). The mitigating systems relied upon by Westinghouse in the ATWS analyses, are Auxiliary Feedwater System and Turbine Trip. As these systems are not in Westinghouse scope, no specific design information is presented for these mitigating systems.

1.3.2 Technical Concerns

(1) In the discussion of design bases for AMSAC (Ref. 8), Westinghouse has stated that "AMSAC does not need to be designed to the single failure

criterion." This interpretation is in conflict with AMSAC criteria stipulated in Appendix C of NUREG-0460, Volume 3 (Ref. 1).

(2) Westinghouse design bases permit sharing of sensor signals between RPS and AMSAC. Specific design information needs to be reviewed for any potential violation of independence requirements between RPS and AMSAC.

1.3.3 Information Needed

(1) Specific design information for the Modified Scram System (MSS)/AMSAC including system design description, functional logic diagrams, schematic wiring diagrams, electrical power supplies and physical arrangement drawings.

(2) Demonstration of independence and diversity between MSS/AMSAC and normal scram system.

(3) Conformance discussion of how MSS satisfies the requirements of IEEE-279.

(4) Qualification requirements for ATWS equipment utilized in short-term mitigation and long-term shutdown functions; specifically assure that the integrity and functional capability of critical instrument sensors, sensing lines and isolation valves will be sustained for ATWS peak pressure conditions.

1.4 Component Stress Evaluation

This staff has reviewed the information supplied by W in both the June 1979 and December 1979 submittals (Refs. 8, 9). In these submittals, W has reported that Reactor Coolant Pressure Boundary Components in the W scope of supply can be exposed to an ATWS pressure of 3200 psi without exceeding the ASME Code Level C Service Limit. In general based upon the information provided, we are in agreement with the W conclusions.

For a few W-supplied components, however, as specifically noted below, the staff has concluded that more documentation is required to evaluate the conservatism of the analyses performed by W to arrive at the reported 3200 psi pressure.

For some major pressure vessel components (i.e., reactor vessels, steam generators, and pressurizers), W has stated that it has reviewed all of the designs used in Alternate 3 plants and has performed analyses to determine the pressure at which the Level C stress criteria are reached. In the December 1979 submittal W has stated that it is reporting from its review and analysis, the allowable pressure for the "limiting" highest stress areas of these components, those with lowest allowable pressure at Code Level C Service Limit. As a conservatism, W has performed the evaluation for these vessels for an assumed component average temperature of 700°F. This is approximately 50°F higher than the maximum average vessel temperature that would occur at the time when the system pressure is at a maximum for the highest pressure ATWS event. For materials such as those used in a W Alternate 3 plant reactor coolant system component, the Code allowable stress limits decrease with increasing temperature. Determining a maximum allowable pressure assuming a conservatively high material temperature provides additional assurance that the limiting pressures reported by W are conservative.

It is possible to calculate Level C allowable pressures using the following simple, but conservative, method. The ratio of the Level C limit to the design limit can be determined. This ratio, always larger than one, when multiplied by the component design pressure provides a conservative value for the Level C allowable pressure.

Applying this conservative method and utilizing the information supplied by W with regard to material types, highest stressed areas, etc., for the "limiting" vessels we have, with the exception of the 4-loop reactor vessel closure bolts, calculated that these vessels can be exposed to an ATWS pressure of 3000 psi without exceeding the Level C Service limit. The analytical results reported by W indicate that the most "limiting" component for Level C allowable pressure is one of the reactor vessel designs used in W 4-loop plants (i.e., nozzle safe ends and closure bolts). Each applicant or licensee must confirm the applicability of the Level C allowable pressure as reported by W for his particular plant and the specific basis for that confirmation (i.e., safe end wall thickness, etc.). Based upon the evaluation made by W and our understanding of the normal industry design and manufacturing practice for RCPB components, we believe that no Alternate 3 W plant licensee or applicant will have any difficulty providing confirmation for the 3200 psi allowable pressure for reactor vessels, steam generators, and pressurizers.

In the December 1979 submittal (Ref. 8), W has reported an allowable Level C pressure of 3474 psi for Control Rod Drive Mechanisms on the basis of the results of an "elastic stress analysis." As for the vessel nozzle safe ends, licensees and applicants should confirm this and provide sufficient details of the analysis to specifically clarify the basis for the reported pressure.

With regard to Reactor Coolant System Piping in the W Scope of Supply for Alternate 3 plants, W has reported in both the June and December 1979 submittals (Refs. 8, 9) that the ASME Code Level C Service Limit allowable pressure is in excess of 3700 psi.

Based on the staff's understanding of the materials used for W supplied RCPB piping, the staff agrees that the Level C acceptable pressure for W supplied RCPB piping is considerably above the previously mentioned allowable pressure of 3200 psi reported for some other RCPB components. W has reported a Level C allowable pressure somewhat higher than 3200 psi for this piping based upon the ASME Level C criterion which permits such piping to reach pressures of 1.5 times the design pressure under certain loading conditions. Although the staff has not specifically attempted to confirm the allowable piping pressure reported by W (that is, above 3700 psi), we are in agreement that somewhere above 3200 psi is acceptable without exceeding Level C for the loading condition resulting from an ATWS event.

With regard to RCPB valves within the W scope of scope of supply for Alternate 3 plants, W has reported in the December, 1979 submittal (Ref. 8) that all ASME Section III valves used in such plants have been hydrostatically tested at 100°F, as required by the Code, at 5625 psig. Taking into account the fact that the valves are subjected to temperatures of around 550°F during an ATWS event and recognizing the change in Code-allowable material properties from 100°F to 550°F, W has determined an equivalent allowable pressure for the valves of 4725 psi, considerably in excess of 3200 psi. Based on the fact that test results are usually preferable to analytical results, when available, for component qualification and that the change in Code-allowable material properties is based on considerable testing, we have concluded that the structural capability of W-supplied RCPB valves has been adequately verified for at least the 3200 psi generically discussed in the June and December 1979 submittals (Refs. 8, 9) for the Alternate 3 W NSSS components.

With regard to the W-supplied Reactor Coolant Pumps utilized in W Alternate 3 Plants, W has reported in the December 1979 submittal (Ref. 8) that, based on a finite element analysis which has been performed, the RC pump used for Alternate 3 plants can withstand a pressure of 3231 psig from an ATWS event without exceeding the Level C Service Limit. Based on discussions with W regarding the type of analysis performed, the staff has concluded that the "allowable" pressure reported by W is acceptable.

However, as noted below, confirmatory documentation is required to confirm these discussions of analyses which have been performed.

The structural integrity of the pressurizer safety valves and power-operated relief valves has been confirmed, at least for the static maximum pressure conditions associated with an ATWS event, in much the same way as described above for other W-supplied valves. The structural integrity and operability of these valves under the dynamic fluid relieving conditions associated with an ATWS event is further addressed below.

The operability and functional capability of a few RCPB components is essential to obtaining ultimate safe shutdown after the ATWS event. In the December 1979 submittal (Ref. 8), W addressed the operability of W-supplied ASME Section III isolation valves which must be operable so that such systems as RHR and HPCI can function when RCP pressures have returned to system design pressure or lower after the exposure to the peak ATWS system pressure.

For such valves, W has stated that qualification tests are performed to pressurize the valves with disk in place at over 3700 psi at ambient (100°F) temperature. Taking into account the change in valve material properties with temperature, this is equivalent to hydrostatic testing the valve at about 3150 psi at 550°F, the temperature at which the valve would be exposed to the maximum ATWS pressure. W has stated that the disks have routinely been exercised after the hydrostatic testing. We are in agreement and find acceptable the reported testing as justification of operability for such valves up to pressures of 3150 psi at 550°F.

For some ATWS events, it may be necessary to be able to run the reactor coolant pumps to assure safe shutdown. In the December 1979 submittal (Ref. 8), W has described the results of finite element analyses used to check such things as pump clearances to evaluate the effect of the ATWS pressure and any resulting deformations on pump operability. W has concluded that after exposure to a pressure of over 3200 psi, sufficient clearances are maintained so that operability would not be impaired. As reported below, we require additional information on the details of this pump analysis to be able to evaluate the W findings.

It may be necessary after an ATWS event to utilize the pressurizer heaters to assist in the safe shutdown of the plant. In the December 1979 submittal (Ref. 8), W has specifically stated that the heater tubing integrity and the integrity of the pressurizer bottom head to heater welds have been evaluated and have been found to meet the Code Level C Service Limit. In the submittal, W has not specifically addressed the operability of the heaters. Based on independent discussions that we have had with suppliers of these heaters, we believe that they probably would be functional at some pressure above 3200 psi. However, we require that applicants and licensees provide confirmation of this.

Based on our review of the June and December 1979 W submittals (Refs. 8, 9) intended to address structural integrity and/or operability of plant Reactor Coolant System Components, we have reached the above conclusions, qualified in some cases with expressed need for confirmatory documentation. In addition, there are some components for which considerably more documentation is required to verify component structural and/or functional adequacy. Listed below are those areas where additional documentation is needed to confirm conclusions reported in the June or December 1979 submittals (Refs. 8, 9) whether or not previously noted above.

(1) Each applicant or licensee shall confirm the applicability of the W reported Level C allowable 3200 psi pressure for his plant. The basis for the allowable reactor vessel safe end and closure bolt allowable pressures shall be specifically described in detail.

(2) Each applicant, licensee, or W shall provide detailed description of the Finite element analysis performed to determine Reactor Coolant pump

structural integrity and operability. Additionally, a more detailed description of the elastic analysis performed for determining the Level C allowable pressure for the Control Rod Drive Mechanisms should be provided (Ref. 2).

(3) The information presented regarding the operability of pressurizer safety valves and power operated relief valves and the structural and functional integrity of safety and relief valve discharge piping is inadequate. Each applicant and licensee shall establish the operability of these valves for ATWS conditions and similarly the integrity and functional capability of the associated discharge piping. It is expected that these areas cannot be adequately addressed until sometime after the completion of the EPRI Safety and Relief Valve and Piping Test Program about to be undertaken as discussed in NUREG-0660 (Ref. 10).

(4) Instrumentation - Structural Integrity of Primary Reactor Coolant Pressure Boundary Instrumentation in the W scope of supply necessary for Safe Shutdown has only been addressed in a cursory manner and functional capability of such instrumentation has not been addressed at all. Much more detailed information is required to complete our evaluation.

(5) BOP-Supplied Equipment - Neither the June nor December 1979 submittals (Refs. 8, 9) address structural integrity and/or operability of any equipment outside the W scope of supply. This information must be supplied and must also address the effect of ATWS on steam generator tube plugging criteria consideration.

(6) The December 1979 submittal (Ref. 8) addresses the structural integrity and operability aspects of W Supplied Reactor Coolant Pressure Boundary valves that were manufactured to ASME Section III requirements. There are undoubtedly some plants that were constructed prior to inclusion of valve design requirements in ASME Section III. Each licensee or applicant must either confirm for its plant(s) the applicability of the Section III valve information as described by W in that submittal or provide assurance of structural integrity and operability for non-ASME Section III W-supplied valves.

(7) Each licensee and applicant must provide confirmation of pressurizer heater operability after exposure to a minimum ATWS pressure of 3200 psi.

1.5 Containment Evaluation

The new information does not lead us to change our previous conclusion that the ATWS containment pressure and temperature are within the containment design specification.

1.6 Fuel Behavior

In Appendix XVII of NUREG-0460 (Ref. 7) and in other communications with W, the staff noted that whereas W contended that no fuel failures would occur during an ATWS in a W plant (because DNBR remained above the 95/95 value), an assessment of the likehood and consequences of PCI failure was needed. W has not provided a quantitative estimate of the probability for PCI failure for

ATWS events that involve a power increase. Such assessments may be made with the PROFIT PCI model.

To provide core-wide assessments of PCI failure probability, however, a census needs to be made of initial rod power distribution and burnup. PROFIT provides a failure probability on a rod basis, that information needs to be correlated with the specific power history for a given transient. Regardless of whether W or NRC were to perform the calculations (NRC would have acquire and collate additional data), there would be a further delay, which would be inconsistent with the current ATWS resolution schedule. Therefore, as an interim measure, we propose that 10 percent of the rods should be assumed failed due to PCI resulting from power-increasing PWR ATWS.

1.7 Radiological Consequence Evaluation for Westinghouse Plants

The recent Westinghouse ATWS report (Ref. 8) contains generic radiological consequence calculations for both Alternative 3 and Alternative 4 plants.

1.7.1 Deficiencies of Submitted Reports

The following is a list of items which are inadequate in the W report.

(1) Westinghouse provides dose calculations for a period of 30 hours after the postulated ATWS event. Westinghouse does not provide any explanation to support terminating the dose calculations at 30 hours. This data should

be presented in a manner consistent with past staff practice (0 to 8 hours, 8 to 24 hours, etc.).

(2) The assumed operation time (duration) for the RHR system should be provided.

(3) RCS release information for plants of lower power and smaller containment volumes should be proviced to show that the releases from the reference plant chosen, do present the worst case dose conditions.

1.7.2 Information Requirements Alternative 3A Plants

Given the consequence information in the W reports for Alternative 3 plants, the staff currently believes that the radiological consequences can be maintained less than the guideline values of 10 CFR Part 100 if containment isolation is achieved quickly (see Section 2.3.4.2 of the main text of the report). If the isolation capability is assured, no more information is required on Alternative 3A plants.

1.7.3 Information Requirements for Alternative 4A Plants

Given the information in the W report on Alternative 4 plants, the staff currently estimates that the radiological consequences can be maintained less than the guideline values of 10 CFR Part 100 if containment isolation is achieved quickly. Therefore, as discussed in Section 2.3.4.2 of the main text of this report, the staff will require rapid containment isolation as a part of the Alternative 4A solution and no more information need be required for Alternative 4A W designs.

2. COMBUSTION ENGINEERING DESIGNS

2.1 Analytical Methods

Status

The analytical models used in the analyses have been presented in References 11 through 16. The evaluation of these models by the staff has been presented in Volume 2 of NUREG-0460 (Ref. 7) in which the staff required minor modeling changes and verification of codes using experimental data to be obtained from appropriate startup tests. The present calculations are performed by Combustion Engineering complying with the requirements of modeling changes. No verification against plant data has been submitted by CE.

Impact of TMI-2 Events

In the meantime, the TMI-2 event has occurred and one of the important lessons learned was the importance of the capability of a plant for long-term safe shutdown. Therefore, analyses demonstrating this capability should be performed. It is our position that the system transient code, CESEC-ATWS, is an inappropriate code to predict system behavior after the formation of the voids in the primary loop. Audit calculations for the TMI-2 event performed by Brookhaven National Laboratory indicate that the CESEC-ATWS code would predict the void fraction in the primary loop very inaccurately and would not predict the phase separation. Therefore, the use of this code is unacceptable to predict long-term behavior when voids are formed in the primary loop.

The analysis performed using the CESEC-ATWS code indicates that approximately 140,000 lb of fluid has been discharged through the valves in a nominal 2560 MWt plant during the first 400 seconds following a complete loss of feedwater ATWS event. The reactor coolant initial inventory is reported to be 471,370 pounds. Thus, appropriately 30 percent of the inventory has been lost through the valves assuming that all of them function properly and that there is no leakage through the seals of the primary loop. Our judgment is that two-phase flow conditions with phase separation in the primary loop will exist. The main coolant pumps are predicted to trip due to cavitation in approximately 120 seconds. Thereafter the core is cooled via natural circulation. HPI is initiated manually at about 600 seconds depending on the system pressure and the HPI design. The phase separation phenomenon would redistribute the voids in the core and primary loop. Void redistribution in the core would change the reactivity feedback and would consequently cause power variation during the long-term shutdown. In order to predict these phenomena, an acceptable small-break LOCA code should be used. This small-break LOCA code should meet the requirements as appropriate to ATWS events.

Verification of Codes

Verification of the ATWS codes will be performed as described in Appendix B.

2.2 Transient Analysis Evaluation

The CE submittals have concerned only Alternative 3 analyses. The majority of the analyses are in CENPD-158, dated May 1976 (Ref. 17), updated by a recent but much abbreviated report, CENPD-263-P, dated November 1979 (Ref. 18).

The older repor. described analyses of a considerable number of individual transients for two classes of CE plants. The new report presents only the loss-of-feedwater ATWS for the same two classes of plant (2560 MWt and 3800 MWt) plus the 3410 MWt class. It should be noted that actual plants vary from 1565 MWt to 3817 MWt, so the three plant classes are rather "broad" classes. The peak calculated pressures for the three classes of plants (256 MWt, 3410 MWt, 3210 MWt) were reported to be 4220, 4290, 3800 psia. The systems relied on are listed below.

Systems Involved in ATWS Mitigation

- Auxiliary Feedwater System
- Pressurizer Safety Valves and PORV
- Reactor Coolant Pumps
- Pressurizer Pressure Control (Sprays and Heaters)
- Safety Injection System, HPSI Mode
- Chemical and Volume Control System
- Condensate and Feedwater System
- Turbine Bypass Valves
- Main Steam Isolation Valves
- Steam Generator Safety Valves
- Atmospheric Dump Valves
- Shutdown Ceiling System

The 1979 report deals exclusively with the complete loss of feedwater (CLOF) transient because the CLOF led to the highest peak primary system pressure in the older report. However, the peak pressures calculated in the new report are significantly lower than those calculated in the old report. The reason for this change is that CE now takes credit for leakage of primary coolant past the vessel flange O-ring seal as a means of pressure relief.

CE has calculated the behavior of the vessel head, flange, hold-down studs, and O-rings under high pressure conditions. CE calculated that, within a rather narrow pressure interval, the vessel head seal components will lift open. This would open a small gap around the entire circumference of the vessel, providing a significant amount of supplementary relieving capacity. CE has calculated the size of this gap on a function of pressure, modified by some hysteresis.

This extra relieving capacity significantly reduces the peak pressure. CE also predicts that the vessel head will re-seat itself when the pressure drops again.

The moderator temperature coefficient is now calculated to be unfavorably high in CE plants (see Appendix C). Calculated peak pressures for these plants would be much higher even than the 4000-plus psi of Reference 18 without the head-lifting relieving capacity.

2.2.1 Evaluation

The introduction of vessel head O-ring seal leakage is the major new factor in CE ATWS analysis. A major difficulty is that study of this effect must depend entirely on calculation--there is no practical way of performing tests. Given this, the review of the calculations, and the search for any possible phenomena not modeled in the calculations but capable of interfering with the desired outcome, would have to be thorough indeed. More briefly, although we concede that this lifting of the vessel head is indeed probable, we find it more difficult to believe that the phenomenon proceeds as smoothly as if it were part of the vessel design.

The description of these calculations occupies less than two pages in the report, and is primarily qualitative in nature. We would need a much more complete description of these calculations to accept Alternative 3 in these plants, including a consideration of the stability of the vessel head when raised in this manner, and an error analysis carried through to confidence limits on the relief flow rate. An audit calculation would also be desirable.

In addition, we would need more detail on how tightly the vessel head re-seats itself after passing a jet of coolant driven by more than 4,000 psi. The report gives no detail on this.

Although the vessel head o-ring seal leakage credit is the major new feature in the CENPD-263-P report (Ref. 18), our review, plus our recent experiences with operating plant accidents, have highlighted some other potential problems:

(1) The use of the CLOF as the bounding ATWS may well be valid, but needs more justification. Given the range of plant configurations and the existence of other transients (e.g., zero power CEA withdrawal) which are close to the CLOF in peak pressure, we cannot accept the CLOF as always limiting based on what information we have.

(2) The sensitivity studies in the new report are limited to studies of the effect of pressurizer total relief area and of the moderator temperature

coefficient of reactivity. This is not sufficient to demonstrate the applicability of the analyses to all the plants in each class.

(3) More information on two-phase fluid discharge from the safety valves (and PORVs) is required. This information is expected to be available from the results of the upcoming EPRI text program.

(4) The question of operator action has not been addressed in detail. The operators must be trained in ATWS mitigation and provided with written procedures which will enable them to unambiguously diagnose an ATWS, refrain from interfering with any mitigating system, and perform any supplementary investigating and monitoring actions that may be appropriate.

(5) Experience has clearly shown that the means of bringing the plant to cold stable long-term post-ATWS shutdown must be well planned ahead of time. In the case of these CE plants, the transients lead to voids in the primary system and tripped main coolant pumps. Analyses must be performed which demonstrate that the core can and will be adequately cooled, safely and reliably over an extended period of time, under these condition In this regard, the impact of lower shutoff head of some HPSI designs also needs to be carefully evaluated.

2.2.3 Electrical Considerations

2.2.3.1 Summary of Information Submitted

The CE report (Ref. 18) includes a brief description of the systems used for ATWS mitigation. CE has also presented a general discussion of the diversity principles considered in the design of the NSSS.

CE has identified a list of instruments and equipment used in ATWS analyses and summarized the environment to which the equipment was designed or qualified to perform their functions. In Section 3.7, CE has provided the results of their review of the integrity of instrumentation transmitters, which form part of Reactor Coolant Pressure Boundary (RCPB) at ATWS pressures. These results indicate that majority of these transmitters are likely to lose their functional capabilities after ATWS pressure surges. It is also important to note that CE's review did not include the assessment of instrumentation sensor lines and isolation valves, which are in applicant's scope.

2.2.3.2 Technical Concerns

Functionability of critical instrumentation systems (sensors, sensing lines and associated isolation valves) during and subsequent to ATWS pressure conditions.

2.2.3.3 Information Needed

(1) Specific design information for the Supplementary Protection System (SPS) and AMSAC including the system design description, design criteria and

bases, functional logic diagrams, schematic wiring diagrams, electrical power supplies and physical arrangement drawings.

(2) Demonstration of independence and diversity between SPS/AMSAC and normal scram system.

(3) Conformance discussion as to how SPS meets the requirements of IEEE Standard 279 and the AMSAC meets the criteria in Appendix C, Volume 3 of NUREG-0460 (Ref. 1).

2.3 Component Stress Evaluation

We have reviewed Combustion Engineering Topical Report CENPD-263-P (Ref. 18) for information relative to mechanical component structural integrity and operability. The report indicates that for the "worst case" ATWS transient (i.e., Loss of Feedwater resulting in highest reactor coolant system pressure), the ATWS pressure, depending on the plant type, ranges from 3800-4300 psi.

The report describes the results of analyses that CE has performed for some Alternative 3 plant reactor coolant system components within the CE scope of supply. CE has provided a general description of the numbers and types of components which were evaluated for the ATWS pressure loading environment. It appears from the description that some types of components were evaluated for each of a selected list of Alternate 3 plants and others chosen on the basis of being representative of one of three classes of Alternate 3 plants (i.e., 2560, 3410, or 3800 MWt). It is not clear to the staff from the descriptions provided in the report exactly which components have been evaluated and which have not. Additionally, CE has described a finite element analysis performed to determine the response of the reactor vessel closure to the ATWS pressure and has taken credit for leakage through the calculated flange opening for determining their peak ATWS pressures.

For the components that have been evaluated, the report provides a brief general description of each of several methods used to evaluate component stress and in a few cases deformation level. The report states that methods used included scaling of originally performed design stress analysis results and elastic or inelastic finite element analysis for some components.

With a few exceptions, it is not clear from the report exactly which method of analysis was used to determine the stress or deformation level for a particular component. The report notes that of the components evaluated, seven were found to exceed ASME Code level C by varying amounts, and thus have at least portions of the component pressure boundary exposed to stresses above the material yield strength. For a couple of these highly stressed components (i.e., reactor coolant pumps and pressurizer heater elements), component operability may be required for safe shutdown of the plant. With the exception of a 12-inch surge line piping elbow there is little or no description of how these components were analyzed, whether deformations were evaluated for effect on operability, etc.

At 4300 psi, some of these components would probably experience a significant amount of permanent deformation, the effects of which should be evaluated using inelastic analysis. The inelastic analysis method recognizes the nonlinearities in the relationship of stress to strain in the material and computes the component structural behavior under the known loading environment considering the strain hardening characteristics of the actual component material(s), its permanent deformations, and stress redistribution occurring in the component.

CE has stated that they performed finite element inelastic analyses for some components, which are exposed to stresses above the material yield strength and also for the analyses performed to evaluate the amount of reactor vessel flange o-ring leakage. Our specific comments regarding the vessel flange analyses are provided separately below. However, regarding finite element inelastic analyses in general, the staff has the following remarks.

The accuracy of the results are quite dependent on the characteristics of the model that is utilized for the analysis and a correct utilizetion of what is usually a very complex computer code which has the capability to take into account the changing material properties referred to above. This type of analysis and the computer codes used to perform finite element analyses have not been in widespread use for very many years, i.e., the technology is relatively new.

Because of the many complexities and cost involved in performing these analyses, the method is only used in high stress/strain critical applications. It is used where it is less expensive to perform a high cost analysis to demonstrate equipment adequacy than to replace the equipment. We require that such analyses be described and reviewed in a fair amount of depth. For some applications, it is also preferable to have an independent analysis or a test performed where possible to confirm the results.

The report is not responsive to the February 15 (Ref. 2) questions in that it does not contain sufficient detailed information to enable us to evaluate the adequacy of the analyses that were performed.

A few components were specifically addressed in the report and for these the staff has the following comments.

The report states that a finite element analysis was performed for one active 16-inch shutdown cooling isolation valve. It is stated that the analysis indicated that no plasticity occurred in the valve body and no adverse effect on operability would be expected. The only "design" information provided about the valve analyzed, other than its nominal size and type, i.e., 16-inch gate, is that the valve body is thicker than the pipe to which it is attached. The report then concludes that similar results, i.e., no adverse effect on operability, would be expected to occur for other active valves of all sizes and types provided that the valve body is thicker than the pipe to which it is attached.

In our view, CE has provided essentially no information about the details of the analysis performed for the 16-inch valve. Also, the staff does not believe that the extrapolation of results to active valves of other sizes and types solely on the basis of valve body thickness can be supported technically, even if CE were to provide more detailed information on the one valve analysis. The information requested in the February 15, 1979 letter to the vendors for each type and size of active valve, possibly supplemented with test data, if available, is needed to verify operability.

In summary, CE has apparently performed evaluations for some reactor coolant system components within the CE scope of supply. However, as a result of the extremely brief summary format utilized in the report for describing the analyses and the "results" there is not sufficient information available for us to meaningfully evaluate what has been done. Specific additional documentation requirements are listed below.

A few comments should be made regarding the description in the report of the analyses, that were performed to determine the response of the reactor vessel closure to the ATWS pressure surge so that credit could be taken for the pressure mitigation effect of fluid leakage past the vessel gaskets. In addition to reviewing the information in the report, the staff also had discussions with CE to obtain further clarification as to what was done. Parts of the analysis are considered proprietary by CE.

Consequently, only a few concerns are noted here:

(1) Finite element inelastic analysis methods were used. The preceding discussion gives information requirements when this method of analysis is used.

(2) It is the staff's understanding that two "representative" vessel closures were analyzed to "represent" at least three different vessel designs. Insufficient information is available to evaluate the validity of this approach.

(3) It is not clear to the staff that tolerance on the amount of preload applied to the reactor vessel bolts has been adequately taken into account. This is a significant parameter which can greatly affect the system pressure at which closure leakage would begin, and of course could thus greatly affect the maximum system ATWS pressure.

(4) It does not appear that the analysis has taken into account possible deformations and movements of the closure head dome itself. The control rod drive penetration housings are installed into the vessel head dome with relatively small partial penetration welds which by ASME code rules are not to be exposed to any bending moment type loading. Assurance must be provided that at these high stress levels, closure head movement or deformation will not impose a severe enough moment loading on one or more of the control rod drive mechanism penetrations such that combined with the high pressure housing to dome weld failure would result.

(5) There is also a concern raised by the staff and an NRC consultant that with head lifting and leakage past both closure gaskets, that the head may "cycle", i.e., continue to alternately lift and reseat somewhat analagous to "chattering" of a safety valve. It would result in large dynamic loads being imposed on the closure bolts and also fluid leakage would be much lower than assumed resulting in higher than calculated maximum system pressures. Assurance would have to be provided that this would not occur.

A summary listing of areas where adequate information has not been provided follows:

(1) No information has been provided for BOP supplied components. Additionally it is not clear which CE supplied components have been evaluated versus those that have not. This should also address the effect of ATWS on steam generator tube plugging criteria considerations.

(2) It is not clear in all cases exactly where finite element elastic or inelastic analysis has been applied. The information requested in the February 15, 1979 letter to vendors (Ref. 2) is considered the minimum necessary to perform a review. For finite element inelastic analysis, information needs were further discussed above.

(3) Vessel Closure Analyses - This analysis is the foundation on which the entire Alternate 3 evaluation for ATWS is based. As such, it must be thoroughly understood and demonstrated to have been performed in a conservative, not a realistic manner. As a minimum, the concerns noted in the text above must be satisfied, including the documentation mentioned for finite element analyses in the February 15, 1979 letter (Ref. 2). It is also fairly obvious that for such a critical application an independent confirmatory analysis would be invaluable input to reaching a final decision.

(4) Structural Integrity and Operability of Active Valves - Technical justification must be provided for each size and design, not arbitrary extrapolations from a single analysis.

(5) Letdown Heat Exchanger Piping - The report indicates that this piping may fail. Documentation has not been provided indicating that the dynamic effects of this failure have been evaluated and shown to have no effect on safe shutdown.

(6) Pressurizer Safety Valves, Relief Valves (PORVs), and Associated Discharge Piping - The information provided on safety valve structural integrity is lacking in detail as per comments made above for components where finite element analysis is used. Information on Safety Valve operability is qualitative and probably somewhat speculative. Assurance of operability in the 3000-psi range will probably have to wait for results of EPRI Safety and Relief Valve Test Program as discussed in NUREG-0660 (Ref. 10). No tests in the 4000-psi range are planned. CE has not provided any information on structural integrity or operability of PORV's.

Additionally, no information has been provided on the structural integrity and functional capability of safety and relief valve discharge piping.

(7) Non-Active Valves - The brief evaluation description provided appears to indicate that for these valves on extrapolation of capability was also made based on the one 16-inch active valve that was analyzed. Such approach may be technically feasible where operability is not a concern. However, the information in the report is too brief to provide the required technical justification for the validity of the extrapolation.

(8) Instrumentation - Information presented is fairly detailed and indicates many "typical" instruments probably would need upgrading for Alternative 3

plants. It would appear that this equipment capability ultimately would have to be addressed on a plant specific basis by each Applicant or Licensee.

2.4 Containment Evaluation

The recent industry submittals did not lend us to change our previous conclusion that ATWS containment pressure and temperature are within the containment design specifications.

2.5 Fuel Behavior

As noted in Appendix XV of NUREG-0460 (Ref. 7), the most-limiting ATWS DNB (under-cooling) event was described (Ref. 17) as a partial loss of fluid flow, for which minimum DNBR of 0.97 was said to occur at 102 seconds. In response to staff questioning, CE later performed another analysis using a different code (TORV-CW-1) which has a different DNB correlation from that used in CESEC. The TORC-CE-1 analysis indicated that the minimum DNBR was above the 95/95 value. That result is reaffirmed in CENPD-263-P (Ref. 18), and it is again concluded that "no fuel is expected to experience DNB and subsequently fail." It is further stated that no cladding collapse is predicted.

Although the staff indicated in NUREG-0460 (Ref. 7) that CE should provide an assessment of the likelihood for PCI failures resulting from ATWS events (such as rod withdrawal) that would involve reactivity insertions, PCI has not been addressed in any CE ATWS analysis submitted to date. Such assessments may be made with the PROFIT PCI model so that the radiological dose consequences of those events can be estimated. To provide core-wide assessments of PCI failure probability, however, a census needs to be made of initial rod power distribution and burnup, since PROFIT provides a failure probability on a rod basis. That information needs to be correlated with the specific power history for a given transient. Regardless of whether CE or NRC were to perform the calculations (NRC would have to acquire and collate more data), there would be a further delay, which would be inconsistent with the current ATWS resolution schedule. Therefore, as an interim measure, we propose that 10 percent of the rods should be assumed failed due to PCI resulting from power-increasing PWR ATWS.

2.6 Radiological Consequence Evaluation for CE Plants

The recent Combustion Engineering (CE) ATWS reports (Refs. 17, 18) contain a generic radiological consequence calculation for Alternative 3 plants only.

2.6.1 Deficiencies of Submitted Reports

The following is a list of items that need to be resolved before the CE reports can be considered complete.

(1) The section on steam generator tube leakage needs additional discussion and more data. The CE report does not provide a table or figure for SG leakage vs time nor does it provide sufficient backup data and information to determine whether the CE approach and numerical values used in this area are acceptable.

(2) The recent report (Ref. 18) does not address the potential effects of increased containment pressure as a result of the O-ring venting on the calculated containment leak rate. This topic should be discussed.

(3) The CE reports do not discuss the effects of "typical" RHR leakage or the potential for damage to seals on ECCS equipment as a result of the ATWS system pressures and how these two pathways could contribute to the ATWS doses for Alternative 3 plants.

(4) No information or analyses of Alternative 4 CE plants as requested by the February 15, 1979 letter (Ref. 2) was provided.

2.6.2 Information Requirements for Alternative 3A Plants

Given the information (Refs. 17, 18), the staff currently believes that the radiological consequences can be maintained less than the guideline values of 10 CFR Part 100 if containment isolation is achieved rapidly (see Section 2.3.4.2 of the main text of this report). If the isolation capability is provided, no more information is required on Alternative 3A plants.

2.6.3 Information Requirements for Alternative 4A Plants

As discussed earlier in Section 2.3.4.2 of the main text, the staff has estimated that rapid isolation of the containment as part of an Alternative 4A solution will provide some protection that the radiological consequences following an ATWS should not exceed the guideline values of 10 CFR Part 100. However, since no previous analyses have been provided for the Alternative 4 CE designs, the staff believes that before a definitive statement on radiologi   consequences can be made, the staff needs the information identified in Section 2.6.1, item 4, of this appendix.

3. BABCOCK & WILCOX DESIGNS

3.1 Analytical Methods

3.1.1 Status

The analytical models used in the analyses have been presented in Reference 19. The analyses have been presented in Reference 20. The evaluation of these models by the staff has been presented in Reference 7 where the staff required minor modeling limitations and verification of the codes using appropriate startup tests. B&W performed reanalysis of ATWS events using guidelines of Alternative 3 (Ref. 21). This resulted in significantly changed input parameters in feedwater coastdown, auxiliary feedwater actuation and physics parameters as tabulated in Reference 21. The results of the most recent calculations (Refs. 23, 25) are not as severe as in Reference 20. This is further discussed in Section 3.2 of this appendix.

3.1.2 Impact of TMI-2 Event

In the meantime, the TMI-2 event has occurred and one of the important lessons learned was the importance of the capability of a plant for a long-term safe shutdown. Therefore, analyses demonstrating this capability should be performed.

It is the staff position that the system transient code, CADDS, is an inappropriate code to predict system behavior after the formation of the voids in the primary loop. B&W realizes that this code should not be used for the conditions in which voids form in the primary and yet has not conducted analyses to predict the long-term behavior of the plant for an ATWS event. B&W refers to Reference 22 that was produced to demonstrate the shutdown capability of the plants in a small break LOCA. However, under ATWS conditions, because of absence of scram, the power is higher than that in a small-break LOCA and in addition, because of a possible phase separation and redistribution of the voids the reactor power can be higher that that predicted using homogeneous models. Under these conditions, there is also concern of interruption of natural circulation.

Referring to a small-break LOCA analysis is not sufficient to conclude that safe shutdown can be achieved in an ATWS event. B&W should demonstrate that the plants can be shut down after an ATWS event using an appropriate acceptable small-break LOCA code modeling physical phenomena accurately. The small-break LOCA code should meet the requirements appropriate to ATWS.

3.1.3 Verification of Codes

Verification of ATWS codes will be performed as described in Appendix B.

3.2 Transient Analysis Evaluation

In their recent submittal (Ref. 23), B&W attempted to demonstrate that the presently existing plants possess sufficient protection and mitigation capabilities required by Alternative 3. In the system analysis part of the submittal, B&W discussed the individual transients specified by the February 15, 1979 letter (Ref. 2). Three of these transients were specifically analyzed using the input conditions listed in the submittal. For other transients, B&W quoted the results of the previously performed analyses which were sometimes based assumptions not in complete conformance with the requirements of the February 15, 1979 letter. Finally, evaluation of some of the transients was reported without support of any definite analysis or references.

All the analyses referenced in the submittal were performed using the following B&W codes:

CADDS - System code with point kinetic representation of the core

LYNXT - Thermal-hydraulic code for DNBR calculation consisting of modified COBRA IV code

CONTEMPT - Containment code

These codes were used to evaluate system performance during the transients assuming failure of the scram function. The following three transients were analyzed with the assumptions based on requirement of the February 15, 1979 letter (Ref. 2):

(1) Complete loss of normal feedwater

(2) Loss of offsite power

(3) Loss of primary flow

The transients for which evaluations were based on the results of the previously performed analyses are listed below:

(1) Rod withdrawal - Reference 20

(2) Primary system depressurization (stuck-open pressurizer safety valve) - Reference 24

(3) Boron dilution - Reference 20

Finally, the transients for which no analyses were provided are shown below:

(1) Load increase

(2) Excessive cooldown

(3) Inactive primary loop startup

(4) Loss of electrical load

In the system transient analyses, B&W used three acceptability criteria: (1) primary system pressure, (2) containment peak pressure, and (3) doses for radiation released from the containment. It was shown by B&W that for the three most-limiting transients all criteria were met (Table A.1).

TABLE A.1 B&W LIMITING TRANSIENTS

Acceptance Criterion	Parameter of Interest	Limiting Event	Results	
RCS pressure boundary stress limit	Peak RCS pressure	LOFW	177-FA 3464 psig	205-FA 3762 psig
10 CFR Part 100 dose	2-hour/30-day dose (thyroid)	LOOP	2/48 rem[1]	
Containment integrity	Peak containment pressure	PORV[2]	11 psig	

[1]Comparable to 177-FA plant results.
[2]Comparable to 205-FA plant results.

B&W has included sensitivity studies in the submittal. These studies were based on the previously performed evaluations (Refs. 20, 24) or on some unspecified sources of information.

The following systems were relied on in these evaluations:

(1) Backup Scram System

(2) Alternate Mitigating System Actuation Circuitry

(3) Reactor Primary Coolant System including Main Coolant Pumps

(4) Pressurizer Pressure Control (Sprays and Heaters)

(5) Pressurizer Power-Operated Relief Valve (One Valve)

(6) Pressurizer Safety Valves (Two Valves)

(7) Main Feedwater Supply System

(8) Condensate Supply System

(9) Auxiliary Feedwater System

(10) Steam Generator Code Safety Valves

(11) Turbine Bypass Valves

(12) Main Steam Isolation Valves

(13) High-Pressure Injection System (Two Trains)

(14) Core Flooding System

(15) Reactor Building Fan Coolers

(16) Reactor Building Spray System (No audit taken in the analysis)

(17) Turbine Electrohydraulic Control System

(18) Integrated Control System

3.2.1 Staff Evaluation

The staff has reviewed the B&W submittal and found several areas where either more information is needed or the conclusions reached by B&W were not sufficiently justified. These areas are detailed below:

(1) In determining limiting ..lues of the moderator temperature coefficient (MTC), B&W used the data steady state reactor operations. As discussed in Appendix C of this report, if the MTCs were based on including the plants performing load follow maneuvers and shutdowns the corresponding MTCs would be less negative by about 1 pcm. This would increase the calculated peak pressures (CE and W evaluations already include this correction).

(2) The pressurizer PORV throat area assumed in the analysis is larger than the throat areas in some plants (e.g., Oconee). This is an overly optimistic assumption for these plants.

(3) In the sensitivity studies, B&W studied the effect of PORV failure to open during loss of feedwater transients and demonstrated that it would have only a relatively small effect on peak system pressure (8 percent). Since unavailability of PORV is a credible situation which may occur in operating plants due to a closed block valve, B&W should evaluate PORV failure for other transients. For the loss of offsite power, PORV failure to open must be assumed unless hardware modifications are implemented to assure its opening under LOOP conditions.

(4) For each transient analyzed, B&W did not identify the systems required to bring the plant to shutdown condition and maintain the plant at that condition (requirement of the February 15, 1979 letter). This should include evaluation of the role of reactor coolant pumps and the HPSI characteristics.

(5) B&W did not demonstrate that the feedwater coast down time of less than 10 seconds used in the analyses of Reference 23 is justified or that it applies to the plants constructed by different AEs. B&W has not justified the early time at which AFWS is assumed in the analyses to be available.

(6) In evaluating the long-term shutdown, B&W did not discuss the potential effects of the voids generated in the primary coolant system during its depressurization through the stuck-open pressurizer safety/relief valve. These voids may prevent natural circulation after the main coolant pumps become inoperative. B&W did not discuss the impact of the time at which these pumps are tripped and the shutoff head of the HPSI pumps.

(7) B&W has not evaluated the effect of high pressures achieved during the postulated ATWS events on the plant control systems and instrumentation sensors, sensing lines, and block valves.

(8) B&W should specify the status of the plant 10 minutes after the beginning of the postulated ATWS events. It should also describe the operator actions after that time with the appropriate justifications.

In this submittal B&W did not address the Alternative 4 requirements.

3.2.2 Electrical Considerations

The B&W report (Ref. 23) does not address electrical, instrumentation and control information relating to ATWS requested in the NRC's letter of February 15, 1979.

3.2.3 Information Needed

(1) Specific design information for the backup scram system (BUSS) and AMSAC including the system design descriptions, design criteria and bases, functional logic diagrams, schematic wiring diagrams, electrical power supplies and physical arrangement drawings.

(2) Demonstration of independence and diversity between BUSS/AMSAC and normal scram system.

(3) Conformance discussion as to how BUSS meets the requirements of IEEE Standard 279 and AMSAC meets the criteria in Appendix C, Volume 3 of NUREG-0460 (Ref. 1).

(4) Qualification requirements for ATWS equipment utilized in short-term mitigation and long-term shutdown functions; specifically assure that the integrity and functional capability of critical instrument sensors, sensing lines/isolation valves will be sustained for ATWS peak pressure conditions.

(5) Operator actions and the period within which such actions are assumed to be completed.

3.3 Component Stress Evaluation

The staff has reviewed the information relative to mechanical component structural integrity and operability supplied by B&W in the January 1980 draft report entitled "Analysis of B&W NSSS Response to ATWS Events" (Ref. 23).

The B&W Evaluation of Alternate 3 components, performed only for components within the B&W scope of supply, is not complete. The objective of the evaluation apparently is to demonstrate component integrity and operability for ATWS pressures of up to 4000 psi. From reviewing the referenced draft report, we conclude that there is a large amount of analysis still to be completed. By letter dated January 22, 1980 (Ref. 25), B&W has advised that a schedule for submittal of this on-going work is being developed.

Although the Alternate 3 component evaluations are not complete, the results of "final" evaluations for a few components and the results of partial evaluations for some other components are discussed in the draft report.

For major RCPB components such as reactor vessels, steam generators, pressurizers and Reactor Coolant Primary Piping, B&W reports that they have utilized a simple but conservative calculational procedure to determine the minimum that the components could be exposed to without exceeding the ASME Code Level C Stress Criterion. The component design pressure, 2500 psi for all the components, is multiplied by the ratio of the Level C primary membrane stress limit to the design primary membrane stress limit. For the materials of construction for B&W Alternate 3 RCPB components, the ratio of the Level C limit to the design limit is 1.2 for austenitic stainless steel components and 1.5 for ferritic materials. The procedure thus yields minimum allowable Level C pressures of 3000 psi for the austenitic components and 3750 psi for the ferritic comoponents. We are in agreement with B&W that conservative Level C allowable pressures would be determined in this way.

As B&W correctly points out, there can be other conservatisms that may have been included in the component's original design analysis. In some cases, performing a reanalyses of the component using techniques (such as finite element analysis) would provide a more accurate understanding of the component's true stress levels. By performing such additional in-depth reviews and reanalyses, it would be possible to show in many cases that a particular component can be exposed to much higher pressures without exceeding, the Level C stress limit, than would be indicated by using the simple stress limit ratio method discussed above.

In this draft report (Ref. 23), B&W indicates that by more in depth review of existing design analyses it has been able to determine that a "majority" of components can be exposed to pressures of up to 4000 psi without exceeding the Level C limit. The report does not contain any detailed information about the reviews that were performed and how conclusions were reached from the existing analyses. For example, it is not obvious that many of these components could be exposed to pressures as much as 1000 psi above the minimum Level C value, calculated by the stresses above Level C. Assuming that the B&W conclusions about component pressure capability is correct, the results would seem to indicate that many, if not most, B&W scope of supply RCPB components are designed with margins of safety considerably higher than required by the ASME Code.

B&W further notes that for those portions of the major components, previously referred to, in which review of existing design analyses did not indicate sufficient capability to withstand 4000 psi, without exceeding Code Level C, additional analyses were being performed to attempt to verify such capability.

Because B&W's discussion of "major components" included the reactor vessel and the pressurizer, but was not sufficiently detailed in nature to address specific areas of these components, there are two areas of concern regarding these vessels.

In 1977, B&W submitted component integrity information in an ATWS topical report (Ref. 20). In that report, B&W reported that analyses performed at that time to evaluate the pressure retention capability of B&W NSSS components indicated the pressurizer heater tubes would fail (pressure boundary collapse) at 3230 psi. It was also reported that reactor coolant leakage through the reactor vessel closure gaskets would begin at 3250 psi for 177-FA plant reactor vessels and 3300 psi for 205-FA plant vessels.

As was clearly shown during the recent accident at the Three Mile Island 2 reactor, the operability of the pressurizer heaters may be needed to assure safe shutdown of the plant. Unless new information to the contrary is available, it would appear that the pressurizer heaters presently utilized in Alternate 3 plants cannot be expected to function after exposure to the ATWS pressure that is being calculated for these plants.

As noted above, B&W had also reported reactor vessel closure leakage at pressures of 3250 to 3300 psi. We believe that at these pressures, leakage would literally be quite small (i.e., possibly drops per minute) and involve no gross loss of reactor coolant and complete lifting off of the closure head as has been reported by CE at somewhat higher pressures. However, B&W has indicated as an objective to demonstrate component structural integrity at a pressure of 4000 psi. Certainly at 4000 psi, possible head lifting, deformations, and gross closure leakage become items of serious concern.

The draft report also provides some information on other B&W supplied Alternate 3 components other than those referred to thus far as "major components."

Some very preliminary information is provided relative to analyses that are not complete for the reactor coolant pumps. A qualitative description is provided as to how each of the different pumps utilized in B&W Alternate 3 plants operates and generally at what physical location on the pump the highest stressed location has been determined to be. A table is provided that indicates how much of the pump "area", a term that is not defined, is exposed to stresses higher than Level C for both 3500 and 4000 psi. We have evaluated the brief description of the basic evaluation procedure that B&W uses to determine pump acceptability.

The only critical comment we have is that based on the information provided it is not clear that the high stressed areas and their associated deformations are being evaluated for their effect on pump operability; i.e., it may be necessary to run those pumps after the ATWS Pressure surge, to assure safe shutdown. From the report, it appears that other concerns of basic structural integrity, such as "leakage rates" (presumably seal integrity), are being evaluated.

The draft report also includes some information apparently in "final" form with regard to RCPB valve pressure retention capability for those valves within the B&W scope of supply. It describes a cataloging effort whereby technical information for all of the valves supplied by B&W has been tabulated in a series of tables included in the report. The information provided includes the manufacturer of each valve, the Code required hydrostatic test pressure for the valve, and a listing of the Code allowable stress values (design values) as given for each of the material types used in the valves, as specified in the 1965, 1968, 1971, and 1974 ASME Code editions for both the hydrostatic test temperature of 100°F and the ATWS temperature 670°F. Apparently, the Code editions used for this evaluation encompass design time period applicable to valves for Alternate 3 plants. This should be confirmed.

With the exception of the safety and relief valves, the discussion of the data indicate that of all the B&W RCPB valves, the lowest Code hydrostatic test pressure used to qualify any one valve was 5400 psi which was applied at a temperature of 100°F.

B&W has determined the ratio of the Code allowable stress value at 100°F to that at 670°F for all the tabulated stress values as noted above. Of all the ratios thus determined, the most conservative ratio (i.e., largest numerically) indicating greatest decrease in material strength from 100°F to 670°F, 1.248 in this case, was used to conservatively calculate an "equivalent hydrostatic pressure" for the ATWS temperature of 670°F. The "equivalent" 670°F pressure thus calculated using the lowest hydrostatic pressure of 5400 and the above referenced material property reduction ratio (i.e., dividing 5400 by 1.248) yields a pressure slightly over 4300 psi.

Although when using this pressure there is no way to evaluate the exact state of stress within the valve at 4300 psi against a specific criterion such as the Level C stress limit, the staff is in agreement with B&W as to the basic structural integrity up to 4300 psi provided the valve is hydrotested in the same configuration that would be exposed to the ATWS pressure. It involves extrapolation of known valve pressure containing capability data at one temperature (100°F) to that at the ATWS temperature of 670°F. The staff believes that this procedure can be used to provide adequate assurance at 670°F of the pressure retention capability without failure, and that the hydrotest already has demonstrated this capability at a higher pressure at 100°F, as long as the Code recognized reduction in material strength with increasing temperature is adequately or conservatively taken into account. B&W has used the most conservative (i.e., greatest) reduction in strength with temperature of any of the applicable materials, provided in any one of the editions of the Code applicable to the time period during which valves for Alternate 3 plants were designed.

As noted above, the staff found one significant prerequisite for utilizing hydrotest data to provide assurance of pressure retention capability under ATWS conditions: (1) The configuration that would be exposed to the hydrotest pressure must be the same configuration that would be exposed to the ATWS pressure. (2) For most of the valves tabulated by B&W and for which B&W desires to use hydrotest data to demonstrate pressure retention capability, again excluding safety and relief valves, the standard industry practice would be to expose the valve body to the hydrotest pressure with the disc at least partially, if not all the way open. Such a test is valid for expolation to ATWS temperatures only for the valve body. Thus, for valves that would be open during the ATWS pressure surge, extrapolation of such test data could be used to assure that the valve body would not be expected to pressure retention capability up to an ATWS pressure extrapolated per the procedure described above.

Typically after the valve bodies have been tested as described in the preceding paragraph, the valves are tested with the disc in place at some lower pressure. In the draft report (Ref. 23), B&W does not discuss the position of the valve disc for the hydrotest, except for the safety and relief valves.

In the conclusion of the valve section of the draft report, B&W stated that the hydrotest pressure extrapolation procedure provides assurance not only of valve body integrity under ATWS pressures but also of disc integrity. These conclusions about disc integrity are not acceptable in view of what has been stated about the standard industry procedures for testing such valves.

The staff finds unacceptable the information provided relative to safety and relief valve pressure retention capability. The draft report (Ref. 23) states that safety and relief valves of the type used in Alternate 3 plants have been exposed to very high hydrotest pressures, with some as high as 9000 psi. B&W then wishes to apply a similar extrapolation procedure to these valves to demonstrate basic pressure retention capability at ATWS pressures. However, for these valves B&W has attached to the draft report copies of portions of valve manufacturers' hydrotest procedures which apparently were used in hydrotesting the Alternate 3 plant safety and relief valves. The procedures indicate that only portions of each of these valve types are exposed to such high pressures. The highest pressure used to test an <u>assembled</u> valve is 3750 psi. Such a test, performed at 100°F, cannot be extrapolated to an equivalent pressure at the

ATWS temperature which would be high enough to provide assurance of valve structural integrity. In addition, both safety and relief valves would be exposed to large dynamic ATWS loads associated with the discharge of subcooled liquid and two phase flow. For this reason extrapolation of hydrotest data even if it were available, for high pressures, would not provide assurance of safety and relief valve integrity.

A summary of specific areas where information has not been provided for Alternate 3 plant components follows:

(1) Integrity and operability of BOP-supplied RCPB components and the effect of ATWS on steam generator tube plugging criteria considerations.

(2) Additional analyses for major components, reactor coolant pumps, and valve operability analyses.

(3) Safety and Relief Valves and Associated Discharge Piping - Information in the draft report regarding structural integrity of the valves is unacceptable. No information was provided on safety and relief valve operability or structural integrity and functional capability of discharge piping. All of this information may be dependent on results of EPRI valve test program as discussed in NUREG-0660 (Ref. 10). However, no tests in the 4000-psi range are foreseen.

(4) Instrumentation - No information provided on pressure capability of primary system instrumentation.

(5) Pressurizer heater operability and reactor vessel closure leakage.

3.4 Containment Evaluation

The recent industry submittal did not lead the staff to change its previous conclusion that ATWS containment pressure and temperature are within the containment design.

3.5 Fuel Behavior

In its January 1980 draft submittal (Ref. 23), B&W provides the results of further analyses, which concentrate on the 177-FA core on the grounds that the earlier results had demonstrated comparable fuel duty for 205-FA plants. The following points are noteworthy:

(1) Two events, two-pump coastdown (2-PCD) and loss of offsite power (LOOP), were further examined to determine DNBR response, percent of core in DNB, and potential for cladding collapse. PCI was examined for the rod withdrawal event only.

(2) For the 2-PCD, the transient minimum DNBR was 1.9 using a fifteen-node LYNXT model. Because the 95/95 DNBR criterion was not violated for this event, no fuel failure was expected.

(3) For LOOP, a minimum DNBR of 0.82 was obtained, corresponding to ~16 percent of the core in DNB for the 177-FA case versus ~10 percent in DNB for the 205-FA plant, based upon reference design peaking distribution, which

reportedly bounds all plants for all times in core life. However, based upon "various existing" designs, a "more realistic assessment" of the percent of the core in DNB was given by B&W as "0 to 8 percent." This is particularly notable because B&W's most recent dose calculations provided by letter dated January 22, 1980 (Ref. 25) are based on 8 percent fuel failure. The 30-day thyroid dose (48 rem) for the LOOP (assuming 8 percent failure) is the most limiting.

(4) For the LOOP event, maximum cladding temperature and differential pressure were reported to be 1400°F and 2200 psi, respectively, for the hottest rod. For those conditions, calculations performed with the B&W CROV code indicate that creep collapse will occur and the stress induced in the cladding will exceed the yield stress during the LOOP event. Based on existing licensing criteria all the collapsed rods would be considered failed for dose calculations, but failure of the rods would also be inferred by application of the DNB failure criterion. B&W contends, however, that although the fuel is in DNB and has collapsed, the temperatures remain low enough to prevent release of fission products. It is unlikely that the staff would agree with this B&W approach without substantial additional information.

(5) For the rod withdrawal ATWS, B&W concludes that no PCI failure is expected. That conclusion appears to be based primarily on the premise that stress corrosion cracking requires a "significant" hold time at power, exceeding ~5 minutes, whereas the power is predicted to decline after ~50 seconds in this event. The staff believes that the hold-time argument lacks

conclusive support, and so the staff cannot accept B&W's contention at this time. The staff is attempting to obtain a more quantitative assessment of the significance of hold time as part of this year's technical assistance program at Battelle Pacific Northwest Laboratories.

PCI assessments may be made with the PROFIT PCI model so that the radiological dose consequences of those events can be estimated. To provide core-wide assessments of PCI failure probability, however, a census needs to be made of initial rod power distribution and burnup, since PROFIT provides a failure probability on a rod basis. That information needs to be correlated with the specific power history for a given transient. Regardless of whether B&W or NRC were to perform the calculations (NRC would have to acquire and collate more data), there would be a further delay, which might be inconsistent with the current ATWS resolution schedule. Therefore, as an interim measure, we propose that 10 percent of the rods should be assumed failed due to PCI resulting from power-increasing PWR ATWS.

3.6 Radiological Consequence Evaluation for B&W Plants

The recent B&W report (Ref. 23) and earlier report (Ref. 20) contain a generic radiological consequence calculation for Alternative 3 plants only.

3.6.1 Deficiencies of Submitted Reports

The following is a list of items which were incomplete in the B&W reports:

(1) The B&W report (Ref. 23) is written for both the 177-FA and 205-FA plants yet the radiological calculations are performed for only the 177-FA plants. To only say that the doses are comparable in a footnote on page 1-2 is inadequate. Additional information presenting relative values of releases for the 177-FA and 205-FA plants pathways should be provided to show that the doses from the two designs will be comparable.

(2) Realistic estimates of normal primary system leakages should be provided.

(3) The X/Q values should be taken as the most limiting X/Q is for B&W plants and identified in the current report.

(4) The gap activities used by B&W do not reflect the guidance provided in the February 15, 1979 letter (Ref. 2). While the activity of some of the noble gases is higher than the staff assumption of 2 percent in the gaps, each iodine isotope is less than the staff recommended values. It is suggested that B&W revise their calculations to incorporate the staff values or to provide a comparison of the doses using the staff values versus the B&W gap activity levels.

(5) Information and analysis of Alternative 4 plants as requested by the February 15, 1979 letter (Ref. 2) needs to be provided.

3.6.2 Information Requirements for Alternative 3A Plants

Given the information of the B&W reports to date, the staff currently believes that the radiological consequences can be maintained less than the guideline values of 10 CFR 50 Part 100 if containment isolation is achieved rapidly (see Section 2.3.4.2 of the main text of this report). If the isolation capability is provided, no more information is required on Alternative 3A plants.

3.6.3 Information Requirements for Alternative 4A Plants

As discussed earlier in Section 2.3.4.2 of the main text, the staff has estimated that rapid isolation of the containment as part of an Alternative 4A solution will provide some protection that the radiological consequences following an ATWS should not exceed the guideline values of 10 CFR Part 100. However, since no previous analyses have been provided for the Alternative 4 B&W designs, the staff believes that before a definitive statement on radiological consequences can be made, the staff needs the information identified in Section 3.6.1, item 5, of this appendix.

4. GENERAL ELECTRIC DESIGNS

4.1 Analytical Methods

4.1.1 Status

The analytical models used in the analyses have been presented in Reference 26. The evaluation of the models for the ATWS analyses by the staff has been presented in Reference 7, where additional information requirements were specified. The same requirements were reiterated in Reference 27. General Electric responded in NEDE-24222 but did not satisfy all of the staff's concerns.

4.1.2 Impact of TMI-2 Event

In the meantime, the TMI-2 event occurred and one of the important lessons learned was the importance of the capability of a plant for a long-term safe shutdown. Analyses showing the long-term behavior of the plant are provided in NEDE-24222 (Ref. 28). There are new concerns as presented in the following sections.

4.1.3 Impact of Peach Bottom Tests

In April 1977, General Electric, EPRI, and Philadelphia Electric conducted turbine trip tests at the Peach Bottom plant. The tests showed that the computer code, REDY, was predicting the peak neutron fluxes nonconservatively by a factor of two or three. Both pre- and post-test predictions of the neutron flux were nonconservative. When these predictions were used to calculate ΔCPR, ΔCPR values were also nonconservative by a factor of two to ten. However, predictions of peak system pressures were slightly conservative.

In order to calculate the results of this type of overpressurization transient, General Electric developed the ODYN code. The ODYN code predicts the Peach Bottom test data accurately and has been approved by the staff (Ref. 27). The ODYN code contains the modeling of the pressure pulses in a steam line. It has a detailed one-dimensional neutronics model that replaces the void sweep model in the REDY code. The staff has required the verification of the REDY results using the ODYN code in Reference 7.

4.1.4 Evaluation of Short-Term Behavior of the Plant

General Electric conducted ODYN and REDY analyses for turbine trip (with and without bypass) and MSIV closure transients. The results indicate that the predictions of peak system pressure and suppression pool temperature and pressure are similar. Differences in the rate of pressure increase or neutron flux spike which happen early in the transient do not appreciably change peak pressure and suppression pool pressure and temperature. The prediction of these quantities apparently do not depend on the initial flux surge. We will require additional information on the ODYN comparisons presented in NEDE-24222, additional ODYN comparisons and a response to the previously identified modeling concerns.

4.1.5 Evaluation of the Long-Term Behavior of the Plant

The analysis shows that limit cycle oscillations are predicted by the code. The staff has not reviewed the REDY code for the long-term behavior of the plant. However, there are some concerns related to limit-cycle oscillation and other aspects of the long-term behavior of the plant.

4.1.6 Capability of the Code to Predict Limit Cycle Oscillations

According to General Electric, these limit-cycle oscillations occur because of the instability of the system. An important modeling consideration in prediction of instability threshold is the void sweep model used in the REDY code. This model should be verified against plant test data in the region of interest and/or against detailed computer code predictions which can calculate void-power feedback using the frst principles in a three dimensional core. The safety significance of the oscillations is discussed in Section 4.5 of this appendix.

4.1.7 Capability of the Code to Predict Inventory in Core and Downcomer

During the transient, as calculated by the REDY code, the level of water in the downcomer drops about 12 feet below the nominal steady state operating value reducing the thermal head substantially. The core inlet flow may not be zero but it is reduced substantially. Limit-cycle oscillations are predicted to occur for some transients at the same time. It is important to verify the calculations of core inlet flow and inventories in the core and downcomer under these conditions, whether limit-cycle oscillations occur or not, using separate effects tests and/or another detailed verified computer code.

4.1.8 Capability of the Code to Predict the Effect of HPCI

HPCI is injected into the feedwater line when the water level in the downcomer drops to Level 2. There is an approximate 20-second delay until water enters the reactor vessel. According to the plots, the water level is approximately 3 feet below the nominal when the HPCI has started and continues to drop to about 12 feet while the HPCI is starting up. When the water level drops below approximately 3 feet, the feedline starts to uncover and the subcooled HPCI water mixes with the steam or two-phase fluid. This condition generally causes a sudden pressure drop particularly under thermal equilibrium conditions. The sudden pressure drop in the feedline would cause a similar pressure drop in the core causing a sudden increase in voids and possible expulsion of water from the bundles in the core. Although voids would increase the void reactivity feedback and tend to reduce the neutron power, excessive voids beyond boiling transition change the heat transfer characteristics by wetting and rewetting the surface. It is possible that some of the voids will be swept by liquid coming from the inlet plenum and some of the voids may be collapsed as the liquid drops back into the core from the upper plenum. It is conceivable that large flow and pressure oscillations in the core due to the HPCI may occur. This possible phenomenon should be analyzed using an acceptable and detailed LOCA code to verify the REDY results. This ATWS event may be more challenging to the fuel than a small-break LOCA since the power during the ATWS event is on the order of 20 to 30 percent.

4.1.9 Fuel Failures During Limit Cycle Oscillations

The REDY code calculated that there are limit-cycle oscillations on an average core basis. Translation of the core average cycles into those cycles occurring in individual channel boxes is very difficult. Amplitude and phase differences occur because of radial power and flow distributions between the individual channel boxes. The sum is the average core oscillation. Multi-channel calculation is needed to understand this phenomenon, to determine the worse channel or channels and estimate the amount of fuel failure over the whole core. The present REDY code does not have this capability.

The individual channels experience power, heat flux, pressure, flow and temperature oscillations. The heat transfer characteristics in a rod bundle and overall flow and vapor distribution inside the bundle under the conditions of interest would have to be verified to demonstrate acceptability of Alternative 3. Wetting and rewetting phenomenon may occur and the amplitude of temperature oscillations is a strong function of this phenomenon. The problem is similar to the reflood problem in a LOCA except that in ATWS the power is higher and the inlet flow is oscillating. However, the staff has decided to require Alternative 4A, and thus to eliminate or greatly reduce the oscillations. Therefore, code verification is required only for calculated behavior of Alternative 4A plants.

4.2 Transient Analysis Evaluation

The General Electric submittals (Refs. 29, 30) have thus far addressed only Alternative 3 plants. The analyses presented cover the BWR/3, BWR/4, BWR/5, and BWR/6 product lines. No Alternative 4 analyses have as yet been submitted for any product line. Table A.2 summarizes the results.

4.2.1 Description

The transient analyses were done using the REDY code to simulate plant behavior. The reports present dome pressure, water level, net reactivity, neutron and heat flux, core inlet flow, feedwater flow, relief valve flow, HPCI flow, and total vessel steam flow (Refs. 29, 30). Thus, it is possible to derive a good understanding of the course of each transient and how the various systems affect it.

Analyses of all anticipated operational occurrences, assuming failure of the scram, are presented. From these analyses, GE concludes that the MSIV closure transient is limiting. Primary emphasis is item given to MSIV closure, plus the inadvertant opening of a relief valve (leads to relatively high suppression pool temperatures) and the turbine trip with bypass (typical of the more common transients).

These simulations are then used to determine maximum vessel pressure, minimum water level, maximum containment pressure, and maximum suppression pool temperature. In addition, the percentage of fuel in DNB is calculated for use in radiological evaluations. Results of the evaluations are presented in Table A.2.

Mitigation of ATWS involves the systems listed in Table A.3. As can be seen from this table, some new systems will be introduced, and some current systems will be modified.

(1) The alternative rod injection (ARI) systems will increase the reliability of the existing scram system. This system will use diverse logic to open new valves on the scram air discharge header on either high vessel pressure or low vessel water level. The scram discharge volume level sensor diversity is also required.

(2) A recirculation pump trip (RPT) on high pressure or low-low level will introduce negative reactivity, and thus lower reactor power, in the initial portion of the transient.

(3) A feedwater runback feature on high pressure (not low level) will help the RPT reduce power during pressurization transients by reducing the core inlet subcooling.

(4) The standby liquid control system (SLCS) will be repiped to discharge borated water from the jet pump instrumentation lines in BWR/3 and BWR/4, and from the HPCS sparger in BWR/5 and BWR/6. This will allow both SLCS pumps to run simultaneously, thus doubling the injection rate to 86 gpm. The SLCS will actuate on high pressure or low level, after a 2-minute time delay, provided all control rods are not fully inserted and core power (as measured by neutron flux) is not zero. Although the RPT (aided by feedwater runback) will limit initial power surges during a transient, the augmented SLCS is necessary to limit the energy deposited to the suppression pool, prevent eventual uncovering of the core, and ultimately bring the reactor to cold shutdown.

(5) The water level setpoint for MSIV closure will be lowered from Level 2 to Level 1. This change decreases the likelihood that nonisolation transients would trigger MSIV closure, which is a more severe ATWS event.

From the results of these calculations, GE concludes that the subject plants can safely withstand any ATWS. Our concerns are listed below:

(1) The three transients considered by GE to be worst case (MSIV closure, IORV, turbine trip with bypass), and on which the sensitivity studies are based, are just barely limiting and in some cases are not limiting. For

example, for BWR/4, the pressure regulator failure maximum demand ties with MSIV closure for peak pressure and exceeds it in percent of fuel in DNB (Ref. 29). Thus, the use of sensitivity studies to make the results generic may not always be valid.

(2) The BWR/4 turbine trip with bypass produces severe power oscillations; moreover, other transients and product lines appear to be close to such an instability. Although GE does conclude that this chugging does not lead to loss of coolable geometry, overpressurization, or any other unacceptable consequence, we are not yet convinced that this phenomenon can be analyzed in a simple manner. We believe a better understanding of the oscillation phenomenon and its causes would be necessary to acceptance of Alternative 3; upgrading to Alternative 3A may well eliminate the problem.

(3) The question of operator action has not yet been addressed. There must be an unambiguous means for the operator to diagnose the onset of an ATWS, assurance that he will not interfere with the ATWS mitigating systems, and an investigation into what supplementary actions he would undertake. Therefore, implementation of Alternative 3 would have to include detailed operating procedures to guide the operator and training in these procedures for the operator.

(4) The discussion of how the plant will be brought to cold long-term shutdown is inadequate. When the primary system is pressurized, normal leakage of primary coolant through pumps seals, valve seals, etc., is great enough

to indicate a danger of recriticality via dilution of dissolved boron after about 5 days by our (conservative) estimate. There must be assurance that the system can be depressurized or otherwise held indefinitely in a stable shutdown state.

(5) None of the reports discuss the ability of equipment to function after an ATWS, other than to state that our requirements will be met, and to evaluate the effect of the pressure surge. All mitigating systems must be shown to conform to Volume 3, Appendix C.

4.2.2 Generic Aspects

The analyses in this report (Refs. 29, 30) are best-estimate calculations (with some conservatisms). The sensitivity studies are intended to demonstrate the applicability of the results to all of the plants. This approach is valid when the various individual plants differ only in the values of the parameters on which the sensitivity studies are based. Unfortunately, there are qualitative as well as quantitative differences between the various plants. This is especially true for the BWR/3 product line, which, for example, includes some plants with isolation condensors instead of a RCIC system (which injects a significant surge of cold water when first activated), and some with unpiped safety valves discharging steam directly into the drywell (which complicates equipment qualification and suppression pool response).

4.2.3 Summary and Conclusions

Although not complete as outlined above, the GE ATWS reports (Refs. 29, 30) present a very comprehensive investigation of ATWS in the context of Alternative 3. Even though we do not entirely agree with GE's conclusions, and have raised explicit questions as described above, it is clear that implementation of Alternative 3 as described in the GE reports would result in a significant improvement in plant safety from the standpoint of ATWS.

4.2.4 Electrical Considerations

4.2.4.1 Summary of Information Submitted

GE has provided single line diagrams on the conceptual functional controls for the ATWS prevention and mitigating systems. GE has stated (Ref. 29) that the primary diversity provided by ARI is in the use of an "energized-to-trip" circuit versus a "deenergized-to-trip" circuit in the output devices of a current scram system. Also, the relays in the ARI will be from a different manufacturer than the scram contractors used in the current scram system. In addition, GE has depicted in Table 5.1-1 (Ref. 29) the functional diversity that currently exists in the sensors of the scram system. With regard to criteria conformance, GE has stated in Section 2.4.1 (Ref. 29) that the RPT and the logic for SLCS will be designed to meet the criteria in NUREG-0460 Volume 3 (Ref. 1), Appendix C, Items A-H and ARI will meet the requirements of IEEE Standard 279-1971.

TABLE A.2

SUMMARY OF ATWS TRANSIENTS BWR/3
S/RV Capacity 68.6% NRB Steam Flow Safety
Valves Capacity 0%

Transient	Peak Neutron Flux (% NBR)	Peak Heat Flux (% NBR)	Peak Steamline Pressure (psig)	Peak Vessel Bottom Pressure (psig)	Minimum Water Level Reached (ft)	Peak Pool Temperature (°F)	Peak Containment Pressure (psig)
MSIV	670.4 (4 sec)	142 (4.9 sec)	1354 (10.7 sec)	1370 (11 sec)	34.2 (320 sec)	179 (50 min)	10.2 (60 min)
TTWB 15% Bypass	762.4 (0.83)	133.4 (2.1)	1231 (5.9)	1253 (6)	34.7 (330)	172 (60 min)	8.9 (70 min)
IORV					(35.0) (1770 sec)	189 (73 min)	12.6 (73 min)
TTNB 0% Bypass	1109.9 (0.75	136.9 (1.9)	1329 (7.7)	1346 (7.7)	34.3 (325)	179 (53 min)	10.2 (60 min)
FW Controller Failure, Max	798.7 (18)	137.4 (22)	1225 (23)	1247 (23)	34.5 (350)	174 (60 min)	9.3 (73 min)
Pres Reg Failure (Open)	466.3 (24)	120.6 (25)	1330 (32)	1346 (32)	34.5 (130)	179 (50 min)	10.3 (65 min)
Loss of FW Flow	561.2 (32)	103.8 (34)	1240 (38)	1256 (38)	30.4 (75)	174 (53 min)	9.1 (63 min)
Loss of Feed- water Heater	111.47 (188)	110.7 (193)	1020 (188)	1065 (187)		N/A*	N/A*
Recirc Flow Failure (Increasing)	515.5 (2.4)	97.9 (5.3)	1011 (5.6)	1038 (5.6)		N/A*	N/A*

TABLE A.2 (continued)

SUMMARY OF ATWS TRANSIENTS BWR/3
S/RV Capacity 68.6% NRB Steam Flow Safety
Valves Capacity 0%

Tran ient	Peak Neutron Flux (% NBR)	Peak Heat Flux (% NBR)	Peak Steamline Pressure (psig)	Peak Vessel Bottom Pressure (psig)	Minimum Water Level Reached (ft)	Peak Pool Temperature (°F)	Peak Containment Pressure (psig)
Loss of Condenser Vacuum	762.4 (0.83)	133.4 (2.1)	1231 (5.9)	1253 (5.8)	34.1 (330)	178 (50 min)	10.0 (60 min)
Loss of Normal AC Power	148 (5.1)	100.1 (6.0)	1130 (6.3)	1146 (6.0)	33.8 (269)	185 (37 min)	11.5 (37 min)

*No S/RV's are actuated, therefore no impact on containment pool temperature and pressure.

Level 2 setpoint at 35.7 ft above vessel zero
Top of Active Fuel at 29.3 ft above vessel zero
Relief Valve capacity 68.8% NBR, Safety Valve Capacity 0% NBR.
Times in brackets are times when peaks occur, in seconds.

TABLE A.2 (continued)

SUMMARY OF ATWS RESULTS - BWR/4
ARI FAILURE, 2 PUMP SLCS, 2-MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum Pressure Vessel Bottom (psig)	Minimum Water Level (ft below separator skirt)	Maximum Suppression Pool Temp. (°F)	Maximum Containment Pressure (psig)
MSIV Closure	527	143	1280	11.59	186	10.3
Turbine Trip with Bypass	392	133	1193	11.77	102	0.6
Inadvertent Opening of a S/R Valve	100	100	1044	-	183	9.7
Loss of Condenser Vacuum	403	133	1195	10.98	188	10.7
Loss of a Feedwater Heater	113	112	1046	-	90	No change
Feedwater Controller Failure-Max Demand	511	137	1195	8.90	99	0.5
Pressure Regulator Failure-Max Steam Demand	585	139	1280	11.14	189	11.0
Loss of Normal Feedwater Flow	100	100	1044	9.35	90	No change

TABLE A.2 (continued)

SUMMARY OF ATWS RESULTS - BWR/4
ARI FAILURE, 2 PUMP SLCS, 2-MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum Pressure Vessel Bottom (psig)	Minimum Water Level (ft below separator skirt)	Maximum Suppression Pool Temp. (°F)	Maximum Containment Pressure (psig)
Loss of Normal AC Power	258	100	1172	10.55	182	9.5
Recirculation Flow Controlled Failure-Max Demand	530	92	1020	-	90	No Change
Turbine Trip with Bypass Failure	655	138	1267	10.92	191	11.4

TABLE A.2 (continued)

SUMMARY OF ATWS RESULTS - BWR/5
ARI FAILURE, 2 PUMP SLCS, 2-MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (%NBR)	Maximum Average Fuel Heat Flux (%NBR)	Maximum Pressure Vessel Bottom (psig)	Maximum Steamline Pressure (psig)	Maximum Suppression Pool Temp (°F)	Maximum Containment Pressure (psig)
MSIV Closure	614	150	1247	1193	179	9.1
Turbine Trip with Bypass	426	134	1192	1126	104	0.7
IORV	---	---	---	---	187	10.6
Loss of Condenser Vacuum	433	134	1193	1126	176	8.3
Loss of Feedwater Heater	114	113	1071	987	90	No change
Feedwater Contol-ler-Failure -Max Demand	450	140	1196	1127	108	1.0
Pressure Regulator Failure - Open	399	151	1238	1180	175	8.2
Loss of Normal Feedwater Flow	100	100	1056	988	90	No change
Loss of Normal AC Power	468	109	1205	1191	170	7.3
Recirculation Flow Failure - Open	382	92	1007	969	90	No change
Turbine Trip Without Bypass	643	143	1230	1171	178	9.1

TABLE A.2 (continued)

SUMMARY OF ATWS RESULTS - BWR/6
ARI FAILURE, 2 PUMP SLCS, 2-MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (%NBR)	Maximum Average Fuel Heat Flux (%NBR)	Maximum Pressure Vessel Bottom (psig)	Maximum Suppression Pool Temp (°F)	Maximum Containment Pressure (psig)
MSIV Closure	745	147	1299	167	6.9
Turbine Trip with Bypass	358	135	1225	100	0.5
Inadvertent Opening of a S/R Valve	-	-	-	170	7.3
Loss of Condenser Vacuum	367	133	1235	163	6.3
Pressure Regulator Failure - Zero Steam Demand	404	143	1283	167	6.9
Loss of a Feedwater Heater	115	114	1071	90	No change
Feedwater Controller Failure-Max Demand	396	126	1214	95	0.3
Pressure Regulator Failure-Max Steam Demand	509	153	1296	167	6.9

TABLE A.2 (continued)

SUMMARY OF ATWS RESULTS - BWR/6 (Continued)
ARI FAILURE, 2 PUMP SLCS, 2-MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (%NBR)	Maximum Average Fuel Heat Flux (%NBR)	Maximum Pressure Vessel Bottom (psig)	Maximum Suppression Pool Temp (°F)	Maximum Containment Pressure (psig)
Loss of Feedwater	100	100	1061	90	No Change
Loss of Normal AC Power	546	101	1218	150	4.5
Recirculation Flow Controlled Failure-Max Demand	247	88	1013	90	No Change
Turbine Trip with Bypass Failure	733	144	1285	168	7.0

TABLE A.3. SYSTEMS INVOLVED IN ATWS MITIGATION

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a. Recirculation Pump Trip (RPT)*
b. Safety/Relief Valves (S/RV)
c. Control Rod Drives
d. Rod Insertion (ARI) Valves*
e. High Pressure Coolant Injection (HPCI); High Pressure Core Spray (HPCS); Reactor Core Isolation Cooling (RCIC)
f. Repiped Standby Liquid Control System (SLCS)*
g. Suppression Pool and Containment
h. Residual Heat Removal System (RHR)
i. Main Condenser (not mandatory for ATWS mitigation)
j. Main Steam Isolation Valves (MSIV)
k. Feedwater System (runback function)*
l. Turbne Pressure Control and Bypass System (not mandatory for ATWS Mitigation)
m. Condensate Storage Tank (CST)
n. Reactor Water Cleanup System (isolation function)
o. Standby Gas Treatment System
p. Diesel Generator
q. ATWS Logic*
r. Instrumentation (Primary functions)

    (i) Reactor Power - LPRM/APRM

    (ii) Control Rod Position Indication

    (iii) Dome Pressure

    (IV) Vessel Water Level

    (v) Suppression Pool Temperature

4.2.4.2 Technical Concerns

(1) Demonstration of component diversity between the current scram system and ARI/RPT systems especially in the relays used in the logic matrix.

(2) Exclusion of recirculation pump motor generator set field breaker/ recirculation pump motor breaker from environmental and seismic qualification, for plants where RPT is installed after July 1, 1981.

4.2.4.3 Information Needed

(1) Specific design information for the ARI and ATWS mitigating systems actuation circuits (AMSAC), including the system design description, design criteria and bases, functional logic diagrams, schematic wiring diagrams, electrical power supplies, and physical arrangment wirings.

(2) Component diversity between the current scram system and ARI/RPT systems in their logic matrix relays (as distinguished from scram contractors, for which diversity has been described).

(3) Conformance discussion as to how ARI meets the requirements of IEEE Standard 279 and RPT and logic for SLCS meet the criteria in Appendix C, Volume 3 of NUREG-0460 (Ref. 1).

(4) Logic changes to reduce vessel isolation and feedwater runback and improved level instrumentation for scram discharge volume.

(5) Conceptual design utilizes flux level permissive in SLCS actuation. Results of consideration of the potential for a common mode failure in the neutron flux signal, thus disabling the scram function and inhibiting the SLCS actuation.

(6) Assurance that ATWS mitigating systems can function in the ATWS environment. Specifically assure that the integrity and functional capability of critical instrument sensors, sensing lines/isolation valves will be sustained for ATWS peak pressure conditions.

4.3 Component Stress Evaluation

4.3.1 BWR-4/5/6 Plants

We have reviewed the information provided by General Electric relative to mechanical component structural integrity and operability in topical reports NEDE-24222 Volume 1 dated May 1979 (Ref. 31) and NEDE-24222 Volume 2 dated December 1979 (Ref. 29).

As discussed in these reports, the largest loads are imposed on Reactor Coolant System Components for the MSIV Closure Event. This event results in the highest Reactor Coolant System Pressures.

In addition, and not well described in these reports, some components in the reactor coolant pressure boundary and some components associated with safety systems that take suction or discharge from the suppression pool are also exposed to large vibratory loads resulting from the discharge of safety/relief valves to the pool. The high ATWS pressure loading occurs simultaneously with the safety/relief valve vibratory loads. Thus, the structural integrity and operability of these mechanical components must be evaluated for this combined loading environment.

In Appendix A.3 of Volume 2 of the report (Ref. 29), GE has provided a fairly detailed tabulation of the maximum calculated reactor coolant system pressures for many locations within the reactor coolant system for the "worst case" MSIV closure event.

For GE-supplied components at these various locations, a tabulation of ASME Code Level C allowable pressures was provided. In general, the Level C pressures were determined in a very conservative manner. The component design pressure was simply multiplied by the ratio of the applicable Level C stress criterion to the component design stress criterion.

In some cases, the "allowable pressures" were based on hydrostatic tests performed on the component in accordance with the requirements of the ASME Code or the similar requirements of some other standard as may be applicable for a specific component.

Based on the tabulated results, it would appear that for most components the maximum calculated system pressure resulting from the MSIV Closure ATWS event is no higher than, at the most, slightly above 10 percent over the component design pressure, thus considerably below the reported Level C allowable pressure. If the system static pressure resulting from the ATWS event were the only load to which the component is subjected, it could easily be concluded that the stresses in reactor coolant system components, supplied by GE, were quite low under ATWS conditions, i.e., as noted above considerably below ASME Level C, in fact in most cases close to Code Level B.

However, the reported results, both in Appendix A.3 of the GE report (Ref. 29) and also to some extent as qualitatively discussed in Sections 4 and 5 of Volume 2 of the report (Ref. 29), are not adequate because they do not report or discuss component stress levels or operability capability for the combined loading case of the ATWS pressure load combined with the SRV vibratory load.

In discussions, GE has been attempting to provide justification that the existing "design basis" non-ATWS SRV loads, which involve lower system pressures and fewer SRV valve actuations and must be calculated using a "more conservative" methodology than is required for calculating ATWS SRV loads somehow envelope any SRV loads associated with ATWS. To date, GE has not provided technical justification for this position. Until further information is provided by GE or licensees and applicants regarding the combined effects of ATWS pressure and SRV vibratory loads, we cannot complete an evaluation of BWR-4/5/6 component integrity and operability.

For the BWR-4/5/6 plants documentation is needed in the following areas:

(1) Information regarding component structural integrity/operability for the combined loading case of the ATWS pressure plus associated SRV vibratory loads.

(2) Information regarding structural integrity and operability of BOP-supplied components affected by ATWS loads. For BOP-supplied piping exposed to SRV vibratory loads, the evaluation of the effect of these loads on piping functional capability should be evaluated.

(3) Although believed to be an oversight in the drafting of NEDE-24222, no specific confirmation of the Level C "allowable" pressure has been provided for any of the BWR-4/5/6 reactor vessels.

4.3.2 BWR-3 Plants

We have reviewed General Electric Topical Report NEDE-24223 (Ref. 30) for information relative to component structural integrity and operability. Under the "worst-case" ATWS event, the MSIV Closure, Reactor Coolant System Components are exposed to high system pressures. Additionally, Reactor Coolant System Components and some components associated with safety systems that take suction or discharge from the suppression are exposed to large vibratory loads resulting from the discharge of safety/relief valves to the pool.

The only information provided in NEDE-24223, is that Reactor Pressure Vessel emergency (we presume Code Level C) limit is 1500 psi. No information has been provided regarding any other GE- or BOP-supplied components for either pressure capability alone or the effect of the combined effects of the ATWS and the SRV vibratory loads.

4.4 Containment Evaluation

There are two areas of concerns related to the containment integrity during ATWS events, namely, hydrodynamic loads and suppression pool temperature limit during the discharge of the safety/relief valve.

The evaluation of these concerns is based on the information provided in the NEDE-24222 (Ref. 29) and NEDE-24223 (Ref. 30) reports. The following sections present the results of our evaluation.

4.4.1 SRV-Related Hydrodynamic Loads

In response to the primary system pressure transient, the safety-relief valves will be actuated to provide overpressure protection by discharging the primary system energy and mass through the SRV discharge line and discharge device (quencher) into the suppression pool. Upon SRV actuation, the air column within the partially submerged SRV discharge line is compressed by the high-pressure steam and, in turn, accelerates the water column into the suppression pool. Following water clearing, the compressed air is also accelerated into the suppression pool forming high-pressure bubbles. These bubbles execute a number of oscillatory expansions and contractions before rising to the suppression pool surface. During this transient, oscillatory loads created by the bubbles are imposed on structures, piping and other equipment.

Extensive experimental and analytical efforts have been undertaken to investigate the SRV loads. Results of these investigations indicate that primary system pressure, suppression pool temperature, and suppression chamber pressure are among the most important parameters influencing the SRV loads. For ATWS events, these three parameters are all substantially higher than that during normal plant operational transients. Consequently, the SRV loads are expected to be more severe both with respect to bubble frequency and pressure amplitude.

GE has analyzed the SRV loads for ATWS events and the results of the analysis are stated by GE to indicate that the ATWS SRV loads are bounded by the loads specification for structures, piping, and equipment. However, the methodology used to predict the loads was not provided for the staff review. In particular, the SRV loads for plants using Mark II and Mark III containments were stated to be predicted by an updated analytical model based on the Caorso test data. This model has not been submitted for staff review. Therefore, we are unable to complete our review of the adequacy of the predicted ATWS SRV loads due to the lack of relevent information. We will require that GE 'dress the following:

(1) Provide a detailed description of the methodology used to predict the ATWS SRV loads.

(2) Provide a detailed discussion and justification for the calculated ATWS SRV loads and show how the design specification for SRV loads wil bound the ATWS loads. It should be noted that the comparison should consider the entire forcing function both for design and ATWS conditions.

4.4.2 Suppression Pool Temperature Limit

Following the air clearing phase and associated hydrodynamic loads as described in Section 1.1 of this report, pure steam is injected into the pool where the steam is condensed. Extended steam injection will heat up the pool. At some threshold level, the steam condensation becomes unstable. As a result, severe vibratory loads could be induced. Current practice to deal with this phenomeon is to limit the allowable pool temperature. Based on the data available to the staff, we have established a local temperature limit of 200°F and presented the results of our evaluation in the 1975 status report on ATWS (Ref. 32).

On the basis of their interpretation of the test data, GE finds that the quencher device has demonstrated the capability for smooth steam condensation up to saturated temperature. However, the staff has maintained that the tests were conducted up to only the nominal boiling point (212°F), not the local saturated temperature (223°F). Although the phenomenon of unstable steam condensation did not occur during the tests, extention of the allowable pool temperature to the local saturated temperature without additional data base is not justified.

In response to this position, GE gathered all the supporting data and presented this information in NEDE-24222 (Ref. 29) and NEDE-24223 (Ref. 30). The staff has reviewed the report and has reached the following conclusions:

(1) The new data base was provided on the basis of the tests conducted using a small-scale model of a cross-quencher. Results of the tests also confirm that the quencher device provides smooth steam condensation up to pool temperatures approaching the nominal boiling point. This new data base, however, did not reveal any evidence to support the argument that the quencher device could perform satisfactorily under local saturated temperature conditions.

(2) The report presents the previously submitted data base, which the staff used to establish the 200°F temperature limit, in a different way. Instead of using the measured pool temperature, the degree of subcooling (difference between saturated temperature and measured pool temperature) is used as the reference. Based on this new interpretation, GE concludes that the quencher device will perform adequately up to a local pool temperature corresponding to 7°F subcooling. However, the information to support this conclusion was not provided. Although the staff believes that the use of subcooling has a good technical rationale for the interpretation of the steam condens n phenomenon, the staff also finds that additional information is needed to demonstrate the adequacy of the newly defined pool temperature limit proposed by GE.

Based on the evaluation discussed above, the staff concludes that the current pool local tempeature limit of 200°F should be maintained until the additional information is provided to justify its modification.

With respect to the difference of local and bulk temperature, GE (Ref. 29) presents a simple analytical model to establish a limiting ΔT of 12.0°F for Mark I and 3.3°F for Mark II and III. These calculational results were scaled down from Monticello and Caorso test results and the ratio of calculated ATWS steam flux to the tested steam flux.

The staff discussed with GE this key assumption of linear dependence on steam flux. The staff's major concern regarding the adequacy of the asumption relates to the behavior of the suppression pool thermal mixing in response to the change in steam flux. The staff believes that this simplified assumption cannot be substantiated without experimental evidence or a sophisticated analytical model. At the present time, plants with Mark I containment will be required to perform in-plant tests to determine the appropriate ΔT. For Mark II and III containments, in-plant tests will be required to determine the ΔT or use the ΔT to be defined as a result of our evaluation of the Caorso tests. The Caorso test report relating to this issue is currently scheduled for submittal to the staff by the second quarter of 1980. The results of the staff's evaluation will be produced as a part of the Task Action Plan A-39 effort.

4.5 <u>Fuel Behavior</u>

Significant new information has been supplied in three GE submittals (Refs. 29, 30, 33) that address early verification Alternative 3. The major points are as follows:

(1) The BWR-3 ATWS analyses indicate enormous increases in peak neutron flux for several cases (e.g., >1100 percent NBR for turbine trip without bypass). No estimates have been provided of the number of rods in boiling transition or of cladding or fuel temperatures for any BWR-3 ATWS events, nor have any estimates been provided of the number of rods failed by PCI.

(2) Several events are shown to have significant, periodic oscillations in neutron flux following an initial, large neutron flux spike. The staff has never before encountered this type of accident behavior prediction, and so it has never been specifically considered in previous PCI evaluations. However, one way to partially address this type of phenomenon is

to calculate a probability of failure (POF) for each major spike in neutron flux (with the PROFIT PCI model for example) and then to sum up the probabilities. That approach is consistent with the PCI mechanistic hypothesis (proposed by C. Vitanza at Halden) that small, part-wall cracks propagate through the cladding in a step-wise fashion with each ΔP (change in power).

(3) Whereas previous GE submittals had indicated that the BWR/6 MSIV closure ATWS produced the greatest number of rods in transition boiling (17 percent), the latest information we have received indicates that the pressure-regulator-failure ATWS will yield more than twice that number.

(4) Regardless of the number of failed rods, whether from overheating or PCI effects, GE contends that Part 100 radiological dose guidelines would not be exceeded even with 100 percent failure. Our problems with this analysis are given in Section 4.6 that follows.

(5) Plots of cladding and peak rod volume average temperature (in Reference 29) time indicate that the cladding may reach collapse temperatures (~1500 to 1700°F) at about the same time (~10 seconds) that the UO_2 approaches or undergoes centerline melting. Thus the UO_2 volumetric explosion, coupled with cladding waisting and collapse, could effect severe pellet/cladding interactions and unusual dimensional changes. Depending on the magnitude of plastic deformation and type of cladding breach, a variety of thermal and mechanical effects are conceivable. The oscillations in neutron flux and fluid flow that then occur later in the event could produce some unique mechanical loading effects. None of this has been modeled.

(6) The combined effects of high nuetron flux spikes, resulting in high cladding and boil temperatures, followed by oscillation in flux, fluid flow, etc., raise questions not only about fuel future, but also about the potential for loss of coolable (rod-like) geometry. The 2200°F, 17 percent oxidation LOCA limits that GE proposes as evidence of coolable geometry are not applicable here because those limits address cladding oxidation and embrittlement effects only. They do not address the potential effects of oscillating mechanical loads on wasted and collapsed cladding that might be "locked onto" the fuel pellets as a result of a BWR ATWS involving a high flux spike, nor do they consider center-melted oxide.

GE would have to provide further analysis of the rod-dimensional changes and the potential effects of the flux/flow oscillations on fuel rod mechanical loads and integrity to justify acceptability of oscillatory behavior.

4.6 Radiological Consequence Evaluation for BWR-4/5/6's

The GE report (Ref. 29) contains a generic radiological consequence calculation for Alternative 3 plants only.

4.6.1 Deficiencies of Submitted Reports

The following is a list of items that are incomplete in the GE report.

(1) The previous GE report (Ref. 31) provided the dose analyses for three specific event sequences; namely, the turbine trip with bypass (TTWB),

the inadvertent opening of a safety relief valve (IORV) and the main steam line isolation valve closure (MSIV). What is most notable about the December 1979 report (Ref. 29) is that only one of the three initiating events (Ref. 31) appears in the much expanded list of initiating events found in Section 4.5.2 of Reference 29. Missing is the evaluation of the TTWB or MSIV initiating events. GE did not provide any justification or explanation for excluding the evaluation of the two events previously analyzed in the May 1979 report (Ref. 31).

Because the May 1979 GE report (Ref. 31) shows that the largest calculated offsite consequences occur from one of the two missing initiating events, the staff could not find the recent GE report (Ref. 29) acceptable for Alternative 3 until either the dose analyses of the TTWB and MSIV events are included in the report or adequate justification is provided by GE in the report explaining the reason the dose evaluations of these two initiating events need not be considered.

(2) Item 5 of Table 4.5.1 of Reference 29 needs considerable expansion. The information which should be supplied by this item is critical to determining the accuracy of the calculations for the offsite dose consequences. Not only should the releases to the suppression pool or main condenser be provided as a function of time but also the releases to the environment both prior and post containment isolation. This is a major deficiency in information necessary for an accurate evaluation of GE ATWS calculations.

(3) GE still does not provide information about reactor coolant system leakage, such as through the MSIVs, or an explanation of why this potential release

pathway should not be analyzed as part of the ATWS dose analyses. This information or explanation must be provided.

(4) Information and analyses of Alternative 4 plants as requested by the February 15, 1979 letter (Ref. 2) need to be provided.

4.6.2 Information Requirements for Alternative 3 Plants

Given the information in NEDE-24222 (Refs. 29, 31), the staff believes that if the GE-assumed isolation capability is provided in the plants, the radiological consequences for Alternative 3A plants should be less than the guideline valves of 10 CFR Part 100, and therefore no more information is required for BWR-4/5/6 Alternative 3A plants.

4.6.3 Information Requirements for Alternative 4A Plants

As discussed in Section 2.3.4.2 of the main report, the staff will require rapid containment isolation as a part of the Alternative 4A solution. The staff also believes that the deficiencies of Section 4.6.1, Items 3 and 4, of this appendix need resolution before a final statement on the potential ATWS radiological consequences of Alternative 4A plants can be made.

4.7 <u>Radiological Consequence Evaluation for BWR-3 Plants</u>

The GE report (Ref. 30) contains a generic radiological consequence calculation for Alternative 3 and BWR-3 designs only.

4.7.1 Deficiencies of Submitted Report

The following items are incomplete in NEDE-24223 (Ref. 30).

(1) In Section 4.5.2 of NEDE-24223 (Ref. 30), GE provides some information about steam flow to the condenser and suppression pool. This information must be expanded to include the release of activity to the environment both prior to and past isolation.

(2) GE still does not provide information about reactor coolant system leakage at normal rates or through the MSIVs or an explanation of why this potential release pathway should not be analyzed as part of the ATWS dose analyses. This information or explanation must be provided.

4.7.2 Information Requirements for BWR-3 Alternative 3A Plants

As mentioned in Section 2.3.4.2 of the main report, rapid isolation of the containment will be required as a part of any Alternative 3A solution. Provided the BWR-3 plants can meet the GE assumed isolation capability, the radiological consequences should be less than the guideline values of 10 CFR Part 100 and no further information need be required for BWR-3 on Alternative 3A.

APPENDIX B

VERIFICATION OF THE CODES

The system transient or LOCA codes which are used in ATWS analyses must be verified against (1) appropriate test data, and/or (2) other codes which have been verified against test data. The appropriate test data can possibly be obtained from appropriately modified facilities of LOFT, Semiscale and other facilities built for separate effects tests. The role of the utilities and vendors in the development of the necessary test program and the possible use of NRC facilities needs to be assessed. In general, the purposes of tests are (1) to verify the predictive capability of the codes; i.e., to determine if the models in the code behave as they should and if the trends are correct and (2) to assess the uncertainties associated with the calculations.

Two types of tests are of interest: integral plant tests and separate effects tests. We are considering a variety of potential tests which could satisfy the need to assure adequate evaluation of the Alternative 4A plant modifications. We intend to review any available data to determine which of the following tests should be considered. Some of these tests are also relevant to the ongoing NRC and industry small LOCA programs.

1. INTEGRAL PLANT TESTS

1.1 Maximum Pressure Tests (PWR)

The purpose of this test is to verify code predictions of maximum pressure and temperatures as the subcooled water is being discharged through relief and/or safety valves while the whole system is still solid; i.e., while there is no void in the primary system. Tests in LOFT and/or Semiscale should be considered.

1.2 Void Tests (PWR)

The purposes of these tests are to verify code predictions of (1) phase separation (where and when it occurs), (2) natural circulation (whether or not it is interrupted and amount of natural circulation flow present with different flow patterns, phase separation and heat transfer conditions), and (3) core uncovering (if or when it occurs and how it is related to items (1) and (2) above). The LOFT small break test series is expected to provide some information on these points. However, additional ATWS tests may be required.

2. SEPARATE EFFECTS TESTS

2.1 Steam Generator Performance Test (PWR)

The purposes of these tests are to verify code calculations of (1) secondary side level changes, (2) tube uncovering, (3) natural circulation, and

(4) stability of two-phase flow in the steam generator tubes during natural circulation. The tests should cover a wide range of feedwater and auxiliary feedwater flow rates.

2.2 Valve Discharge Tests (PWR)

The purpose of these tests is to verify the calculated valve discharge rates under subcooled and two-phase conditions. The EPRI tests are expected to provide some information, but the pressure range of the ATWS-related tests has not been established. When established, this range should cover the condition that the valves are expected to operate with Alternative 4A modifications.

2.3 Inventory Verification Tests (BWR)

The purpose of these tests is to verify the code predictions of water inventory in the core and downcomer of a boiling water reactor when the water level in the downcomer drops substantially. Tests in plants or TLTA should be considered.

2.4 HPCI Tests (BWR)

The purpose of these tests is to verify the code predictions of pressure changes as the subcooled high-pressure injection water moves with steam or two-phase primary fluid. Tests in TLTA should be considered.

APPENDIX C

TREATMENT OF MODERATOR TEMPERATURE COEFFICIENT IN PWRs

To describe the models used by three PWR vendors to estimate the 99 and 95 percent probability moderator temperature coefficient (MTC) and make some comparisons among these models and results, as presently known to the staff, it is necessary to discuss briefly some general MTC characteristics. These characteristics are described in tutorial detail in NUREG-0460, Volume 2, Appendix VIII. Only a simplified summary will be considered here.

An ATWS transient analysis uses a variable MTC that depends on transient state conditions but is based on the MTC assumed to exist at the initial condition of full power and other state parameters nominal. (This is usually referred to as the ATWS MTC.)

This MTC is a function of core lattice design, fuel enrichment, burnup, power, temperature and control states, and especially the moderator boron content. Elements controlling the boron content such as enrichment (determining cycle length) and burnable poison may vary from cycle to cycle, and those such as burnup, xenon content, and control state during the cycle. For a given cycle, if the reactor is taken immediately to full power and left there for the entire cycle without shutdown or control changes other than boron changes to compensate for xenon and burnable poison and fuel burnup, the MTC would rapidly become more negative over the first few (about 3) days from xenon buildup (the MTC change is from boron removal) and then (generally) more slowly over the remainder of the cycle from burnup. The MTC change from xenon is about 5 pcm

(pcm will be used to represent $10^{-5}\Delta k/^{\circ}F$ here) and for burnup about 20 pcm. The burnup change is usually monotonic and nearly linear with burnup, although sufficient burnable poison may flatten or even temporarily reverse the effect early in the cycle (e.g., CE reactors).

For a given cycle (and for the reactor life if all cycles are the same) if operated in this steady state mode, the 99 percent probability MTC would be about equal to the BOC, equilibrium xenon MTC (which we will call BOL-MTC), since it takes "3 days out of a 300-day cycle" to get to this value. The BOC-MTC value depends on the design parameters of the core. The 95 percent MTC value for this steady state mode is about 1 pcm less than BOC-MTC for a typical equilibrium cycle burnup.

For the W and CE design models, the first and subsequent cycle BOC-MTC are stated to be nearly the same (CE strong first-cycle burnable poison causes large early part of cycle MTC increase) and thus the steady state mode 99 percent values would be about equal to the equilibrium cycle BOC-MTC. For B&W the first cycle design values are stated to be considerably less negative than the subsequent cycle values. This modifies the probability distribution slightly but the 99 percent steady state mode value remains near the equilibrium cycle BOC-MTC.

For W design models, the core design parameters are such that the BOC-MTC is (stated to be) about -8 or 9 pcm for all reactors. For CE, it is generally about -2 to -4 pcm for the older reactors and about -6 pcm for the newer designs. For B&W, it is apparently about -10 or 11 pcm for the 177 reactors and about -12 or 13 pcm for the 205 reactors. The staff review to confirm these values and provide core parameter variation reasons for the difference has been minimal since the MTC normally used in transient and accident analyses other than ATWS are considerably more conservative than these values. However, the staff has reviewed startup physics test results for many cycles of some of these reactors (some classes are not yet operating) and these values are generally compatible with the results of these tests. It may be noted, however, that the BOC-MTC depends on cycle parameters such as enrichment (cycle length) and burnable poison amount which have occasionally changed in the past and could change in the future without a specific design commitment to maintain a given value.

Non-steady-state operation such as load follow or shutdown and restart introduces a potential different and variable MTC state and thus probability distribution, primarily from xenon-induced boron changes (although control rod insertion also plays a role). The differences depend on the assumed events and modes of control. There is not available a detailed written report from the vendors on the analysis of such modifications. The staff, however, did have some detailed discussions with them several years ago and in particular with CE on the details of their analysis. The CE analysis considered a conservative mix of load following shutdowns and increased cycle lengths (the strongest influence is from extended shutdowns), and did not take credit for burnup at part power. (It should be noted that the severity of the consequences of a typical PWR limiting ATWS analysis would generally be expected to decrease with decreased initial power even when considering the possibility layer MTC which might accompany such a decrease.) The study indicated that (for CE) the base steady-state mode MTC at a given probability should be increased (less negative) by 1 to 2 pcm, making 95 percent MTC about equal to the BOC-MTC and the 99 percent MTC about 1 to 2 pcm less negative. The same increase should probably also apply to the W case and likely to B&W. (It might be noted that the 90 percent MTC would be expected to be about 1 pcm more negative than the 95 percent value.)

Thus the CE 95 percent MTC should be about -6 pcm for the newer reactors and about -2 pcm for the older ones, and the 99 percent MTC about -5 and -1 pcm. These 95 percent values (-6, -2) have been used in the recent CE analyses and no 99 percent values have been submitted. (Past CE analyses seen by the staff have used only -6.) The recent W 95 and 99 percent values are stated (details are not presented) to be based on non-steady-state analyses similar to those of CE. (Note that this differs from the W ATWS report of 1974 in which time at part power was used as a basis.) The W 95 and 99 percent values are -8 and -7 pcm, which would be expected from their steady-state values and the CE analysis. Older W analyses performed in 1974 also used these values. The recent B&W report uses 95 percent values of 10.5 pcm for 177 reactors and 12 pcm for 205 reactors (the 1977 analyses used 12.7 for 205 at 99 percent), which, unlike the above, are based on the steady state probabilities and taking credit for part power burnup, but does use (conservatively) an 18-month cycle. A transient analysis increase would presumably make these values about 1 pcm less negative. A reason stated for non use of power variation modifications is that these are the results of "recent" control rod use which demonstrates that a pervasive mechanical fault is not present.

APPENDIX D

POWER-OPERATED RELIEF VALVES IN PWRs

The reactor coolant system is required by ASME Boiler and Pressure Vessel Code to be protected from transient overpressure conditions. This protection is accomplished by several means, including reactor trip, operation of code required pressurizer safety valve, and operation of the power-operated relief valves (PORV).

All PWR designs (except CESSAR-80) are (or will be) equipped with one to three power-operated relief valves. Typically, B&W designs have one PORV whereas W and CE designs incorporate two PORVs. The valves were designed to prevent lifting of the pressurizer code safety valves and for B&W and W designs to allow the reactors to stay on line for load rejection transients. As a result of the accident at TMI-2 in March 1979, the staff has required (or is considering requiring) that (1) the PORV position be indicated in the control room, (2) the PORV be capable of being opened/closed without offsite AC power, and (3) isolation valve (upstream of PORV) be closed when the reactor coolant system pressure is below some prespecified value to prevent pressure blowdown through a stuck-open PORV.

Although anticipated transient (Condition II events - ANSI-N.18.2) analyses do not rely on PORV discharge capability to stay within the overpressure limit, credit for PORV discharge capability is taken in the ATWS overpressure calculations. The analyses make the following assumptions for Alternative 3A and Alternative 4A plants.

(1) Alternative 3A Plants - Assume all PORVs are available to limit the peak primary pressure.

(2) Alternative 4A Plants - Assume that the peak pressure limit is not exceeded even if a single failure disables one PORV (i.e., one PORV is unavailable).

Experience with operating plants has shown that PORVs may be isolated (sometimes for long periods of operation) due to problems of leakage through valves. Since those valves were not considered as part of the plants' safety systems, there are no technical specifications which limit the operation or isolation of these valves. In this appendix, we address the potential impact of isolated PORVs and recommend a procedure for providing alternative means of limiting the peak calculated pressure to within acceptable limits.

Calculations with and without PORV relief (and some judgmental estimates) have given the sensitivities tabulated in the following Table D.1. The values in the third colume (ΔP) give the pressure increase for unavailability of the PORV(s) to open in an ATWS sequence that otherwise would result in a peak pressure of $\sim$3200 psi, the range in Alternative 4A.

As a result, we recommend the following requirement on PORV outage for Alternative 3A and 4A plants: Plants shall not operate at power levels in excess of 75 percent with PORVs isolated for more than five days per year unless the impact on peak calculated pressure due to the isolated valves is compensated by other factors such as power level, MTC, etc., being at more favorable values at the time than the values used in the ATWS analysis.

TABLE D.1 CALCULATED SENSITIVITIES

Designs	Number of Unavailable Valves (PORVs)	Estimated ΔP (3200 psi range)	Reference
Combustion Engineering (non-CESSAR)	One	+250 psi	CENPD-158, Rev. 1
	Two	+500 psi	CENPD-158, Rev. 1
Babcock & Wilcox	One	+200 psi	NUREG-0460, Vol. 2
Westinghouse	One	+160 psi	Dec. 79, W Report
	Two	+320 psi	Judgment

Assumptions Used to Develop Allowable PORV Outage

(1) The likelihood of exceeding calculated peak pressure as a result of isolated PORV(s) shall not be greater than 0.01.

(2) If the plant is operating at 75 percent of its nominal power level, the peak pressure would be within the peak calculated at 100 percent power with the PORVs isolated.

(3) The plants operate at power level in excess of 75 percent during approximately 80 percent of time.

APPENDIX E

EVENT TREES AND PROBABILITY ESTIMATES

The staff has developed simplified event trees to determine the scenarios that could result in unacceptable consequences and assigned probability values to various branches of the event trees based on available information and judgment. Although we acknowledge that the probability estimates may have large uncertainties, the event tree technique yields valuable information on relative importance of alternative sequences. Point estimates are used.

Figures E.1 and E.2 are simplified event trees for BWRs and PWRs, respectively. The event trees consider a number of ATWS scenarios subject to equipment availability. The major systems relied on to limit the consequences are identified. At each branch point, the upper branch denotes successful operation of that system (or function), whereas the lower branch represents failure of that equipment to perform its function.

The sequences range from ATWS (that is, an ATWS without any failures in the mitigating systems) to ATWS followed by failures in mitigatiing systems (e.g., ATWS-U sequence in Figure E.1 means an ATWS followed by failure to inject high pressure makeup flow).

The consequences of concern are characterized as core melt (PWR, BWR) high pool temperature for BWRs (potential loss of containment integrity), excessive pressures in PWRs (potential impact on ability to safely shut down the plants). Sequences with estimated frequency significantly below 10^{-6} are not considered.

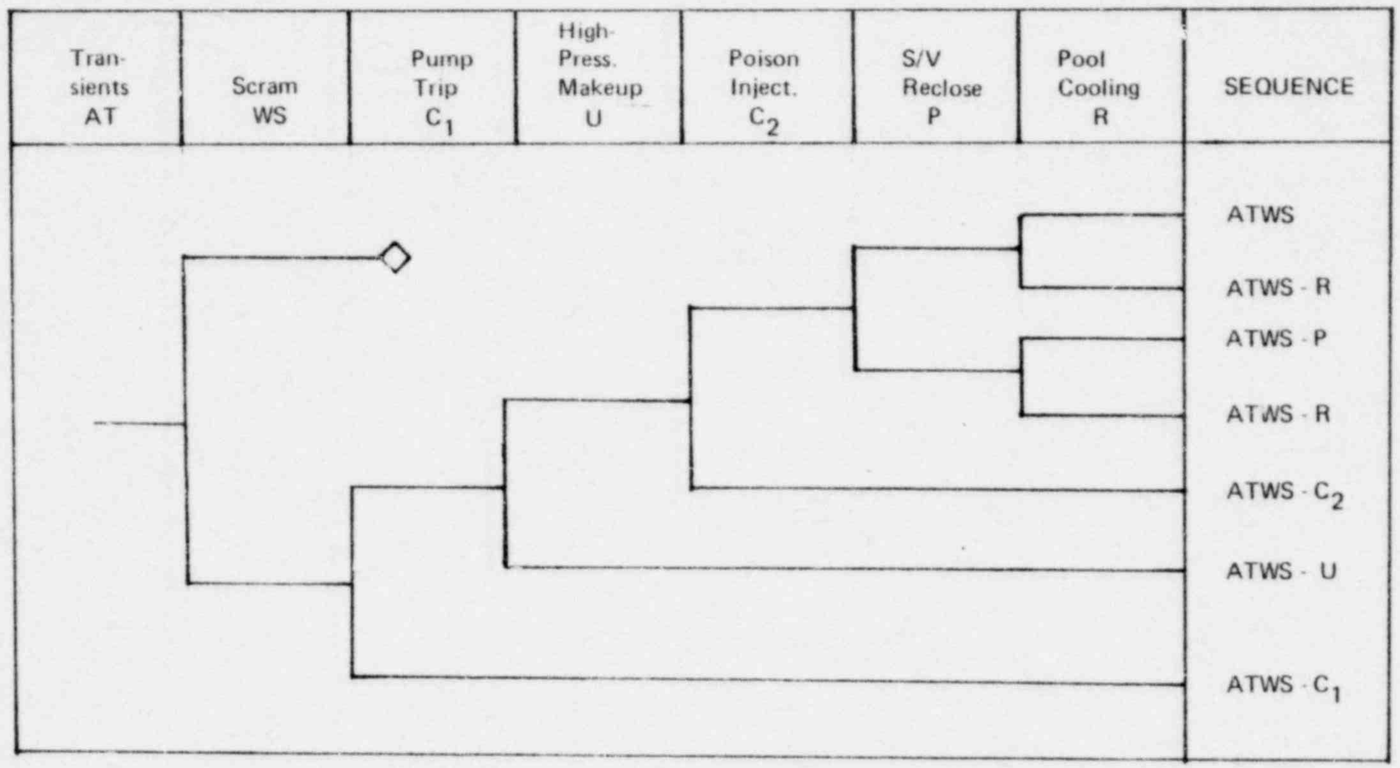

Figure E.1. Simplified BWR event tree.

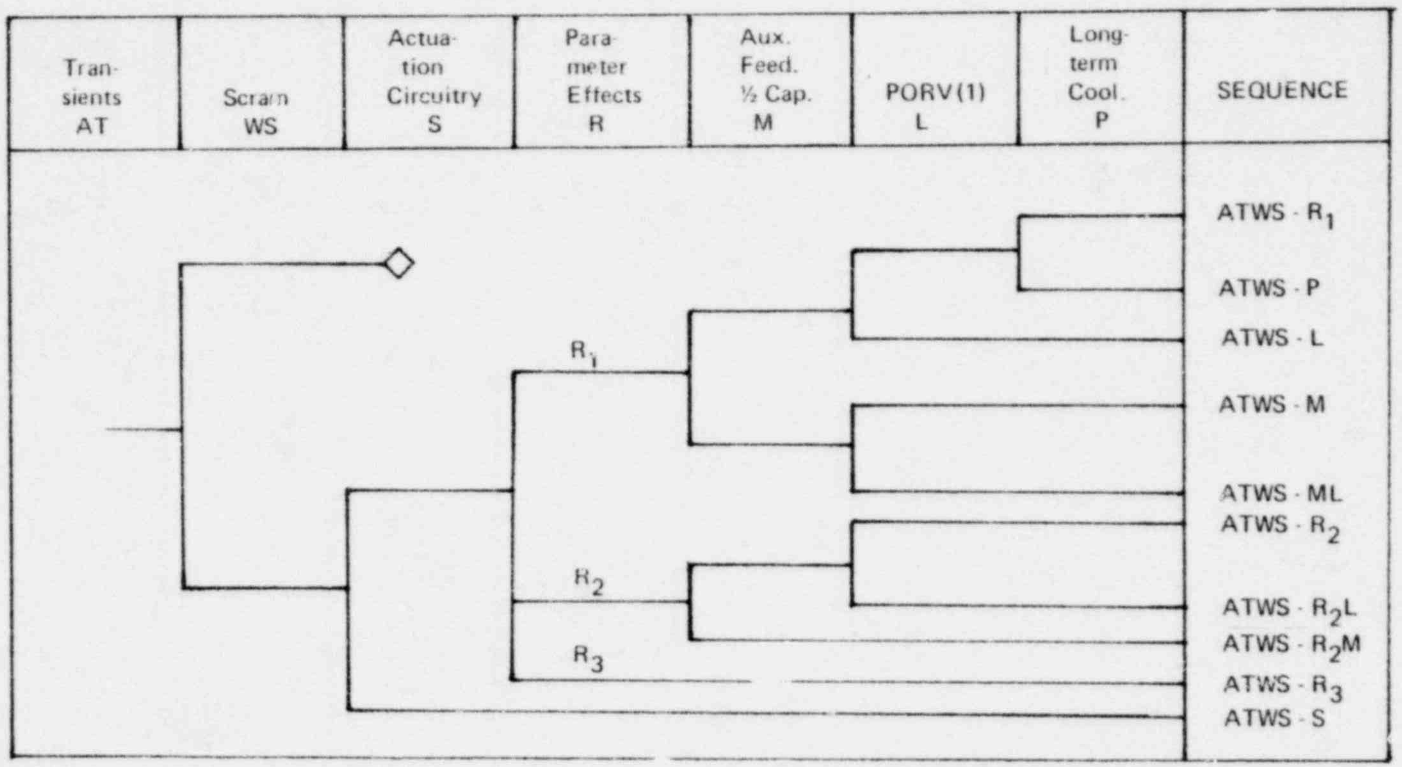

Figure E.2. Simplified PWR event tree.

BWR Plants

Table E.1 includes the impact on frequency and consequences due to the three alternative modifications considered in this report.

The summary of improvements from these alternatives are provided in Table E.2. We believe that implementation of Alternative 3A or 4A would reduce the probability of severe consequences from ATWS by approximately a factor of 10 or 100, respectively.

PWR Plants

Table E.3 includes the impact on frequency and consequence due to Alternative 2A, 3A or 4A. For PWR designs, the proposed hardware modifications for Alternatives 2A and 3A are essentially identical, and thus the resultant improvements are essentially identical. The summary of results is provided in Table E.4. These results show that implementation of Alternative 3A would reduce the probability of severe consequences to below $\sim 4 \times 10^{-5}$/RY for B&W and CE designs, and $\sim 10^{-6}$/RY for W designs (see Appendix G for the reasons for lower risk from W plants).

TABLE E.1 ATWS FREQUENCY AND CONSEQUENCE IN BWRs

Sequence	Alt. 2A Solution[1]		Alt. 3A Solution[1]		Alternative 4A Solution[1]	
	Frequency	Consequence	Frequency	Consequence	Frequency	Consequence
ATWS	$\sim 9\times10^{-5}$ (2)	Likely core melt(3)	$\sim 5\times10^{-5}$ (4)	Likely OK (5)	$\sim 5\times10^{-5}$	Acceptable
ATWS-C_1 (6)	$<10^{-6}$ (7)	Core melt	$<10^{-6}$	Core melt	$<10^{-6}$	Core melt
ATWS - U	------(8)		$\sim 5\times10^{-6}$ (9)	Likely core melt(10)	$\sim 5\times10^{-6}$	Acceptable(11)
ATWS-C_2	------(8)		$\sim 3\times10^{-6}$ (12)	Likely core melt	$<10^{-6}$ (13)	Likely core melt
ATWS - P	------(8)		$\sim 5\times10^{-6}$	High containment temperature	$\sim 5\times10^{-6}$	Acceptable
ATWS - R	------(8)		$\sim 5\times10^{-6}$ (14)	High containment temperature	$\sim 5\times10^{-6}$	Acceptable

FOOTNOTES FOR TABLE E.1

[1]See Chapter of this report.

[2]Current scram systems are highly reliable and thus difficult to significantly improve. This analysis assumes, conservatively, we believe, that modifications in the electrical portions would reduce unreliability for scram system from 3×10^{-5} to 1.5×10^{-5}.

[3]Inability to shut down reactor; containment failure; makeup water depleted; core melt may occur.

[4]Changes in setpoints and feedwater runback (in conjunction with at least 86 GPM boron system) may reduce isolation events to 4/RY (estimate).

[5]Core wide oscillations, severity plant dependent, safety impact unknown, pool temperature may exceed limit.

[6]ATWS followed by failure of recirculation pumps to trip.

[7]Failure of one to two safety/relief valves to reclose (estimate derived from the RSS) following an ATWS.

[8]These sequences are included in the core-melt sequence above.

[9]Failure to provide high-pressure makeup coolant (10^{-1} from RSS). HPCS (BWR 5/6 plants) may be more reliable (factor of 2 to 5).

[10]Inability to keep core covered.

[11]Alternate 4A increased SLCS capability (flow rate) is expected to keep the core covered in the event HPCI(s) fails.

[12]Assumed unreliability of delivering 86 or more GPM poison in 2 minutes (not operator (manual) actuated). If conservative values for the initial pool temperature and the ΔT between local and bulk temperature are used, the calculated peak local pool temperature may be as high as 240°F with 86 GPM System.

[13]The unreliability of the Alternative 4A SLCS design is required to be much lower than 10^{-2}.

[14]Operator action required (10 to 15 minutes) to begin pool cooling. Further delay of several minutes could result in significant increase in containment temperature. Probability of failure of this action is assumed to be 0.1 (similar value is suggested in the RSS).

TABLE E.2 FREQUENCY OF SEVERE CONSEQUENCES IN BWR

Plants	Frequency
Current	$\sim 2 \times 10^{-4}$/RY*
Alt. 2A	$\sim 9 \times 10^{-5}$/RY
Alt. 3A	$\sim 1 \times 10^{-5}$/RY
Alt. 4A	$\sim 1 \times 10^{-6}$/RY

*Volume 3, NUREG-0460

Table E.3 ATWS FREQUENCY AND CONSEQUENCES IN PWRs

Sequence	NSSS Designer	Present Design(1) Frequency	Present Design(1) Consequences	Alt. 2 and 3A Designs(2) Frequency	Alt. 2 and 3A Designs(2) Consequences	Alternative 4A Designs Frequency	Alternative 4A Designs Consequences
ATWS - R_2(3)	W	10^{-6} - 8×10^{-5}(4)	Core melt	$\sim4\times10^{-5}$	Acceptable Excessive Press.(3)	$\sim4\times10^{-5}$	Acceptable
	CE	8×10^{-5}	Core melt	$\sim4\times10^{-5}$	Acceptable Excessive Press.(3)	$\sim4\times10^{-5}$	Acceptable
	B&W	8×10^{-5}	Core melt	$\sim4\times10^{-5}$	Acceptable Excessive Press.(3)	$\sim4\times10^{-5}$	Acceptable
ATWS - P	All PWRs	-	-(5)	$<10^{-6}$(6)	Core melt(7)	$<10^{-6}$	Core melt
ATWS - M(8)	W	-	-(5)	$\sim3\times10^{-6}$(9)	Acceptable(10)	3×10^{-6}	Acceptable
	B&W, CE	-	-(5)	$\sim3\times10^{-6}$(9)	High pressure(10)	3×10^{-6}	Acceptable
ATWS - L(11)	W	-	-(5)	$<10^{-6}$	Acceptable	$<10^{-6}$	Acceptable
	B&W, CE	-	-(5)	$<10^{-6}$	High pressure(10)	$<10^{-6}$	Acceptable
ATWS - R_3(12)	W	-	-(5)	$<10^{-6}$(13)	Acceptable	$<10^{-6}$	Acceptable
	B&W, CE	-	-(13)	$<10^{-6}$	High pressure	$<10^{-6}$	Acceptable
ATWS - S(14)	All PWRs	-	-(13)	$<10^{-6}$(15)	Core melt likely	$<10^{-6}$	Core melt likely

FOOTNOTES FOR TABLE E.3

(1) Some plants may already have mitigation capability - AMSAC with unknown reliability.

(2) Alternative 3 and 3A changes to PWR designs are essentially the same except 3A includes assurance of instrument capability. For W this alternative also includes modifications in scram system.

(3) The response of a PWR is highly dependent on the initial values of parameters. For example:

For B&W and CE designs an initial MTC (moderator temperature coefficient) value experienced by the reactor ~ 30 percent (for CE this is likely to be 50 percent) of the time would be expected to exceed service level "C" limit on pressure (3200-3400 PSI). However, from our statistical studies (App. VII, Vol. 2, NUREG-0460) we learned that other parameters could take on more severe values than those assumed (nominal) in these calculations and also result in high primary pressures. Therefore, 50 percent of the time (R_j - 0.5) the parameter values are assumed to be such as to not result in pressures in excess of service level "C" limit.

For W designs, service level "C" limit is not expected to be exceeded unless the MTC value (initial) is more severe than that experienced by the reactor 99.5 percent of time the reactor is at specific power level.

(4) Current scram systems are highly reliable and thus difficult to significantly improve. This analysis assumes, conservatively, we believe, that modifications in the electrical portions would reduce unreliability for scram system from 3×10^{-5} to 1.5×10^{-5}.

(5) These sequences are included in the core melt sequences above.

(6) P = assumed unavailability of long-term heat removal systems. $<10^{-2}$

(7) ATWS followed by failure to remove long-term heat removal expected to result in core melt.

(8) ATWS followed by failure of 1/2 capacity of auxiliary feedwater flow.

(9) Unavailability of 1/2 aux feed flow $\sim 4 \times 10^{-2}$ (RSS estimate).

(10) Excessive Pressure (>4000 psi) for B&W and CE designs. Acceptable pressure for W designs.

(11) ATWS followed by failure of a PORV to open.

(12) ATWS occurs when the system parameter values (e.g. MTC) are at extremely unfavorable conditions.

(13) R_3 = 0.01 - See Appendix VII of NUREG-0460, Vol. 2.

(14) ATWS followed by failure of actuation circuitry (AMSAC) for auxiliary feedwater system and/or other mitigating systems.

(15) The actuating circuitry unreliability is judged to be less than 10^{-2}.

TABLE E.4

FREQUENCY OF SEVERE CONSEQUENCES IN PWRs

Plants	W	B&W, CE
Current	10^{-6} 8×10^{-5}	$\sim 8 \times 10^{-5}$
Alt. 2A, 3A	$\sim 1 \times 10^{-6}$	$\sim 4 \times 10^{-5}$
Alt. 4A	$\sim 1 \times 10^{-6}$	$\sim 1 \times 10^{-6}$

*See footnote (1) in Table E.3.

APPENDIX F

EARTHQUAKE CONSIDERATIONS

This appendix treats the question of whether, and how, seismic and ATWS loads must be combined in evaluating the integrity and operability of RCPB components. The combination of seismic and ATWS events is credible, because earthquakes can induce transients; severe earthquakes are expected to induce transients. If a semismically-induced transient became an ATWS, then the primary system would be subjected to the ATWS peak pressure as well as the seismic forces. Definitive prediction of the relative timing of the earthquake and the peak pressure seems not to be possible. They are therefore conservatively assumed to be concurrent.

The event sequence to be evaluated is given in the following:

(1) Seismic event

(2) Transient

(3) Failure to scram

A spectrum of earthquakes is considered. Return intervals of 10, 100 and 1000 years are used to represent earthquakes of different frequencies.

These representative earthquakes were chosen on the basis of recurrence frequency. They are therefore not easily compared with the design-basis earthquakes used for safety analysis: (1) safe-shutdown earthquakes (SSE) and (2) operating-basis earthquake (OBE). The problem is the inability to determine the recurrence frequency of these earthquakes, whose magnitudes are established in nonprobabilistic ways. Therefore, the staff chose for this analysis to use representative earthquakes that were defined probabilistically.

The earthquake magnitude of the 10-, 100-, or 1000-year earthquake is different at each site. Moreover, these magnitudes are not known; the sites have not been evaluated in this way. However, the 10- and 100-year earthquakes are smaller than the OBE as it is currently being established as 1/2 the acceleration of the SSE. The 1000-year earthquake is smaller than, or at most comparable to, the SSE.

The probability of failure to scram is assumed to be independent of earthquakes of the magnitudes considered here. In about 500 reactor-years, 50 ten-year earthquakes have been experienced, of which on the order of 5 were hundred-year earthquakes. None have been observed to inhibit scram (or mitigating system operability). Most earthquakes of this frequency have in fact gone unnoticed in every respect.

Thus for the 10-year and 100-year earthquakes, the independence of scram failure is easily accepted. For the 1000-year earthquake, there is little or no experience, so one has to rely on the required qualification of the scram system for the SSE. Analyses or tests are used to demonstrate such qualification. The staff sees no reason to question the independence of scram failure and these earthquakes.

The remaining factor is the probabiliy of an earthquake initiating a plant transient in the class of potentially severe ATWS events. We have no analysis of this probability, but can estimate it from experience. As far as we know, none of the earthquakes so far experienced in U.S. power plants has caused a plant transient. For the 50 ten-year earthquakes, that gives 1/50 or less as the probability of such an earthquake initiating a transient. For the 100-year earthquakes, we obtain similarly an estimate of 1/5 as the probability of transient initiation. For a 1000-year earthquake, we have no data so we conservatively take a probability of unity that a transient would ensue.

The results are shown in Table F.1. Probabilities are used; the error bands on these numbers can be large,so we used them only for aids to engineering judgment. Point values are given. The conclusion to be drawn is that the calculated frequencies of seismic-induced ATWS events are very small and can be neglected.

The critical assumptions are (1) independence of scram failure from these earthquakes; and (2) validity of estimates used for the probability of earthquake-induced transients. These would be worth checking on the few U.S. plants located at sites with high seismicity. Except for these sites, we conclude that it is not necessary to consider seismic and ATWS loads in combination when evaluating primary-system and interfacing components against level C limits.

For sites of high seismicity, further study is indicated. Areas to explore include (1) a more careful evaluation of earthquake probabilities as a function of earthquake magnitude; (2) whether the loads due to a 10- or 100-year earthquake would be significant compared to ATWS loads; (3) other margins of seismic resistance; and (4) experience of electrical grid outage and other transients in similar plants in earthquakes.

TABLE F.1 SEISMIC - ATWS ANALYSIS

Earthquake Return Period, Years	10	100	1000
Frequency of earthquake (EQ) per reactor year	1×10^{-1}	1×10^{-2}	1×10^{-3}
Prob. of transient as a result of earthquake	1/50	1/5	1
Frequency of EQ and transient per reactor year	2×10^{-3}	2×10^{-3}	1×10^{-3}
Prob. of scram failure, given EQ and transient	3×10^{-5}	3×10^{-5}	3×10^{-5}
Frequency of earthquake-induced ATWS per reactor year (RY)	6×10^{-8}	6×10^{-8}	3×10^{-8}

APPENDIX G

COMPARISON OF PEAK PRESSURES FOR PWR DESIGNS

This appendix discusses reasons for the differences in the calculated peak pressure in the various PWR designs. In this regard, we have identified those parameters which we believe have a significant impact on the calculated pressure. Staff comments regarding the adequacy of chosen values are provided in Appendix A in this report.

Important Parameters

Reactor Power Level - As the initial power level is increased, the calculated peak pressure also increases. The peak pressure sensitivity per percent change in power ranges for a few psi to 40 - 50 psi, depending on the design.

Reactor Coolant System Volume - Following the imbalance in the heat generation rate and the heat removal capability, the rate of primary system pressure rise is a function of the power level and the volume of the coolant. The larger the power to volume ratio, the greater would be the calculated peak pressure. We do not have specific sensitivity values for this parameter.

Total Relief Capacity of Safety/Relief Valves - As the capability to transfer energy to the steam generators decreases, the primary system fluid discharge increases through the safety/relief valves. The rate of pressure increase depends strongly on the amount of primary coolant that can be discharged through these valves. Thus the greater the discharge capability of these valves, the lower the peak calculated pressure would be.

Steam Generator Inventory - The rate of imbalance between the energy produced and the energy removed via the steam generators depends on the rate at which the steam generator tubes are uncovered. If the rate of imbalance is slow, the power level can be reduced by the negative reactivity feedback and hence reduce the pressure rise of the primary coolant. Thus the greater the steam generator inventory, the lower the calculated pressure would be.

Moderator Temperature Coefficient (MTC) - During the period of increasing pressure the moderator temperature rises, which produces negative reactivity feedback. For more negative values of the MTC, the reduction in power produced would also be larger. Thus, if the MTC is more negative initially, the calculated peak pressure would be lower (due to reduced power generation).

In summary, the calculated peak pressure would be lower if:

(1) The initial power level is reduced,
(2) RCS volume is increased,
(3) The safety/relief valve capacity is increased,
(4) The steam generator inventory is increased, and
(5) The MTC is more negative.

In the case of B&W plants, because of lower steam generator inventory, the coastdown time for main feedwater and the actuation time for auxiliary feedwater are also important.

Results

Table G.1 summarizes the plant type parameter data and includes the calculated peak pressures assuming all systems are functional. The most significant differences are in the relief capacity, steam generator inventory and the MTC.

W designed plants have features which make the consequences of ATWS less severe than those for B&W and CE designs. As far as the B&W and CE designed plants are concerned, the initial MTC value in the CE plants is so much less negative than in the B&W plants that all other differences are swamped and the calculated pressure in this class of CE designs is much higher than that in the B&W designs.

Table G.1 ATWS - Comparison of Plant Parameters

Parameter	Westinghouse			Combustion Engineering			Babcock and Wilcox	
	2-loop	3-loop	4-loop	2560 MWt	3410 MWt	3800 MWt	177-FA	205-FA
Core Power, MWt	1650	2785	3423	2710	3410	3817	2772	3800
RCS Volume ft^3/MWt	3.78	3.43	3.66	4.10	3.46	3.64	4.11	3.37
Total relief capacity lbm/hr/MWt	648	522	491	330	330	529	249	329
SG liquid inventory lbm/MWt	123	114	119	95	96	86	33	23
MTC, pcm per °F	-8	-8	-8	-2.0	-6.3	-6.8	-10.5	-12.0
Peak RCS pressure during LOFW accident, psia	2753	2783	2848	4220*	4290*	3800*	3464	3762

* The calculated peak pressures would be significantly greater than these values if credit were not taken for leakage through vessel flange O-rings.

APPENDIX H

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