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March 6, 1979

MASTER

Mr. Andrew C. Millunzi
Office, Assistant Secretary for
  Energy Technology
U.S. Department of Energy
Washington, D.C. 20545

Re: Grant #EN-77-C-03-1465 - Breeder Safety Program

- Task 1 - Provide Technical Review and General Support Service
- Task 2 - Examine Issues in Formulating Risk Assessment Criteria
- Task 3 - Assess the Implications of NUREG-0460 (LWR-ATWS Study) on the Breeder Safety Program

Dear Andy:

This letter reviews our progress on the subject Tasks. We are currently awaiting your direction to begin Task 1. Our Task 2 effort is progressing well. A draft Report will be forthcoming within a few months.

A draft Report--written under Task 3--is attached for your comments. At your request, we would be happy to send a copy to Atomics-International, General Electric, and Westinghouse, for their review. Should you concur, we would like to publish the enclosed as a Rand Report--a publically available document--and possibly as a journal article. May we have your comments no later than March 30, 1979?

We are looking forward to continuing our work with you.

Very truly yours,

Ken Solomon

Kenneth A. Solomon

KAS:heb
Enclosure as noted

cc: Mr. Jerry Griffith
  ✓Mr. Eddie Simms

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author(s) William E. Kastenberg and Kenneth A. Solomon

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PREFACE

In this report, we assess--in a preliminary fashion--the potential impact that NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vols. I, II and III may have on the Liquid Metal Fast Breeder Program.

This report, supported under Department of Energy Grant Number En-77-C-03-1465, is one of a series of forthcoming studies, assessing levels of acceptable risk applied to energy systems--most specifically--the nuclear fuel cycle. The present report is intended for a technical audience, and as such does not address the social issues of risk acceptability. These issues are to be discussed in others of the forthcoming studies.

William E. Kastenberg is Professor of Engineering and Applied Sciences at the University of California, Los Angeles, and a consultant to the Rand Corporation.

CONTENTS

SUMMARY

In this report we assess--in a preliminary fashion--the potential impact of the proposed LWR-ATWS acceptance criteria contained in NUREG-0460 on the LMFBR safety program, with reference to the Clinch River Breeder Reactor Plant. In particular, this report addresses the following issues:

- To what extent and in what manner can NUREG-0460 impact the LMFBR safety program?
- What are the implications for LMFBR safety design if the Nureg-0460 proposed criteria are accepted?
- What policy decisions or changes (if any) need to be implemented if the NUREG-0460 proposed criteria are adopted?

Our approach is to review the current safety considerations for LWRs, the proposed LWR-ATWS acceptance criteria, and the recent preliminary licensing review of the Clinch River Breeder Reactor Plant (CRBRP). We then compare and contrast NRC's preliminary safety criteria for CRBRP and the proposed LWR-ATWS criteria. From this approach we derive our preliminary conclusions and recommendations.

Our preliminary conclusions include:

- The proposed ATWS acceptance criteria for LWRs (as specified in NUREG-0460) are in principle, applicable to LMFBRs.
- For LMFBRs, the major criterion will be the assurance that the plant protection system (shutdown or scram) has a sufficiently high reliability (low failure rate) so that Core Disruptive Accidents (as currently defined) will lie outside the design basis. For early plants, however, mitigating systems may also be required.
- Alternative accident scenarios for LMFBRs, which are initiated from the shutdown state or may lead to potential Core Disruptive accidents even following scram, need to be examined in greater detail.

- The proposed LWR-ATWS criteria do not appear to present any new or unforeseen design and/or safety questions for LMFBRs. They do, however, specify design goals for mitigating systems which may insure conformance with NRC policy.

Our preliminary recommendations include:

- Continued research and development aimed at insuring a sufficiently high reliability for the plant protection system, on intrinsic plant features to minimize scram demand and on alternate approaches to mitigating systems design. Studies should be conducted to determine the contribution to risk of accidents which initiate from the shutdown state.
- DOE should expend additional effort comparing LWR and LMFBR safety criteria. Fundamental to this effort DOE should assess the desirability and the extent to which LWR and LMFBR safety criteria can be more more uniform.
- DOE should assess the degree to which resolution of LMFBR safety criteria can be effectively "piggy backed" onto the LWR safety program.

This latter recommendation is significant in view of the possibility of a reduced LMFBR safety program and the conflicting need to keep as many viable energy options open as cost effectively as possible.

Since our preliminary conclusions and recommendations are based on a comparison of commercial LWR criteria with a prototype LMFBR, some specific details contained within this report may be less applicable to later, commercial LMFBR designs. The general conclusions should remain valid, however, for any LMFBR in order to be licensable in the U.S.

I. INTRODUCTION

In the design of Light Water Reactors (LWRs), protection against anticipated transients (e.g., loss of normal electric power and control rod withdrawal) is provided by a highly reliable scram or shutdown system. At issue is the concern that should this scram or shutdown system become inoperable, the transient could then lead to a core melt down situation. This concern has led the Nuclear Regulatory Commission (NRC) to propose in NUREG-0460 [1], new requirements (or acceptance criteria) for the consideration of Anticipated Transients Without Scram (ATWS) events and the manner in which they could be considered in the design and safety evaluation of LWRs.

In this report we assess--in a preliminary fasion--the potential impact of the proposed LWR-ATWS acceptance criteria [1] on the Liquid Metal Fast Breeder Reactor (LMFBR) safety program, as represented by the Clinch River Breeder Reactor Plant.

In particular, this report addresses the following issues:

- To what extent and in what manner can NUREG-0460 impact the LMFBR safety program?
- What are the implications for LMFBR safety design if the NUREG-0460 proposed criteria are accepted?
- What policy decisions or changes (if any) need to be implemented if the NUREG-0460 proposed criteria are adopted?

These issues are addressed in the following manner. In Section II we provide background information on LWR and LMFBR safety. We review the various classifications of accidents, their potential frequencies of occurence, and their impact on design and risk. In Section III, we review the proposed ATWS acceptance criteria for LWRs. In Section IV, we review the recent preliminary licensing experience for the Clinch River Breeder Reactor Plant (CRBRP) and discuss NRC's preliminary safety criteria for it. We then compare and detail the LWR-ATWS criteria with those proposed for CRBRP, so that the applicability of the proposed LWR-ATWS criteria to the LMFBR safety program can be determined. In Section V we provide a preliminary assessment of how NUREG-0460 may impact LMFBR design. Our conclusions and recommendations follow in Section VI. This section also contains several policy implications with key effects on any future LMFBR program. These include:

- In light of potential reductions in the LMFBR safety program, and the need to keep all energy options open, DOE should consider the possible benefits of "piggy-backing" the LMFBR program onto the LWR safety program.
- The possibility of making LWR and LMFBR safety criteria more uniform should be assessed, both on the basis of economic benefit, and potential increases in overall safety.
- The LMFBR safety program should especially consider those aspects unique to LMFBRs, such as accident energetics, which might pose special problems in meeting safety criteria.

II. BACKGROUND INFORMATION

LIGHT WATER REACTORS (LWRs)

The safety goal for Light Water Reactors (LWRs) is concerned with insuring that radiological doses to the public are within "acceptable limits" following a postulated reactor accident. These acceptable limits are defined in 10 CFR 100 [2]. For the purpose of evaluation, accidents are usually characterized into three categories [3]:

A. Events of moderate frequency (anticipated operational occurrences) leading to no abnormal radioactive releases from the facility;
B. Events of small probability with the potential for small radioactive release from the facility; and
C. Highly unlikely accidents (potentially severe accidents of extremely low probability, postulated to establish performance requirements of engineered safety features and site acceptability).

Table I, taken from WASH-1250 [3] contains some examples of these events and postulated accidents which are used in preparation of both the preliminary and final safety analysis reports (SARS), required by NRC. It is important to note that in the 1978 NRC publication, NUREG-0438 [4], the approximate frequencies assigned to these categories are:

Category A	Several times/year
Category B	1/10 to 1/100 per year
Category C	1/1000 to 1/10,000 per year

For these accidents, it must be shown that the radiological consequences of the accident are within the guidelines set forth in

TABLE I

EXAMPLES OF POSTULATED REACTOR FACILITY ACCIDENTS BY CATEGORY [3]

A. Moderated Frequency Events (no abnormal radioactive release from the facility)

1. Withdrawal of control rod at maximum speed due to malfunction or error
2. Failure of one safety rod to scram
3. Partial loss of normal forced reactor coolant flow
4. Unintentional startup of an inactive reactor coolant loop
5. Loss of external electrical load and/or turbine trip
6. Loss of off-site electrical power
7. Excessive load increase
8. Loss of normal feedwater flow
9. Inadvertent depressurization of the primary coolant system

B. Infrequent Accidents of Small Probability (abnormal radioactive release possible, but not expected) *

1. Small leaks and breaks in pipes (or minor leaks in large primary or secondary system pipes)
2. Inadvertent loading of a fuel assembly into an improper position
3. Complete loss of normal forced reactor coolant flow
4. Complete loss of all A-C power (station blackout)
5. Major leakage in radioactive waste decay tank

C. Highly Unlikely Accidents (postulated for evaluating site acceptability) **

1. Major rupture of pipes containing reactor coolant up to and including double-ended rupture of largest pipe in the primary coolant system (loss of coolant accident)
2. Major secondary or steam system pipe rupture up to and including double-ended rupture of a main steam pipe
3. Control rod ejection
4. Severe fuel handling accident
5. Tornadoes, flooding and earthquakes

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*May exceed the guidelines of 10 CFR Part 20.

**May not exceed the guidelines of 10 CFR Part 100.

10 CFR 100 [2]. Furthermore, Category C accidents are termed Design Basis Accidents because they are used to set performance requirements for engineered safety features of the plant (e.g., the emergency core cooling system).

Table II illustrates an alternative classification of accidents for LWRs which is used in the preparation of environmental impact reports. We note that Classes 1 through 8 are basically the same as Categories A through C (which appear in the SAR). Of particular interest is the Class 9 accident which lies outside the design basis. Should a Class 9 accident occur, it would result in radiological consequences greater than those specified in 10 CFR 100.

It is important to note that the precedent used in setting the design basis accident is: "an overall safety objective of 10^{-6} per year and an objective for individual events of 10^{-7} per year" [1].

The Reactor Safety Study, WASH-1400 [5], attempts to estimate the probability and consequences of all accidents (including Class 9) that could occur at LWRs. In general, the results indicate that the probability of the dominant contributor to risk (core melt) in the plants studied is about 5×10^{-5} per reactor year, with only about two percent of the core melt events resulting in early fatalities [1]. Table III, obtained from the draft version indicates the range of frequencies considered. (It should be noted that PWR-8, PWR-9, BWR-5, and BWR-6, do not result in core melt.)*

An important conclusion of the Reactor Safety Study is that transient events are small contributors to the overall probability of a core melt in the Pressurized Water Reactor (PWR) studied, while they dominate the Boiling Water Reactor (BWR) risk. The NRC staff has reviewed the extent to which Anticipated Transients Without Scram (ATWS)**

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*PWR = Pressurized Water Reactor, BWR = Boiling Water Reactor. The integer corresponds to a particular release category described in WASH-1400.

**An Anticipated Transient Without Scram is defined loosely as the failure of the plant protection system (shutdown or control system) following the initiation of a Category A event. As defined above, Category A events have a frequency of several times per year.

TABLE II

REACTOR FACILITY

CLASSIFICATION OF POSTULATED ACCIDENTS AND OCCURANCES [3]

No. Of Class	DESCRIPTION	EXAMPLE(S)
1	Trivial Incidents	Small spills Small leaks inside containment
2	Misc. small releases outside Containment	Spills Leaks and pipe breaks
3	Radwaste System Failures	Equipment Failure Serious malfunction or human error
4	Events that Release Radioactivity into the Primary System	Fuel defects during normal operation Transients outside expected range of variables
5	Events that Release Radioactivity into Secondary System	Class 4 and Heat Exchanger Leak
6	Refueling Accidents Inside Containment	Drop fuel element Drop heavy object onto fuel Mechancial malfunction or loss of cooling in transfer tube
7	Accidents to Spent Fuel Outside Containment	Drop fuel element Drop heavy object onto fuel Drop shielding cask - loss of cooling to cask, transportation incident on site
8	Accident Initiation Events considered in Design-Basis evaluation in the Safety Analysis Report	Reactivity Transient Rupture of Primary Piping Flow of Decrease - Steamline Break
9	Hypothetical Sequences of Failures More Severe than Class 8 (but having much lower probability of occurrence)	Successive Failures of Multiple Barriers normally provided and maintained

Table III

PROBABILITIES OF INDIVIDUAL RELEASE CATEGORIES [5]

PWR Release Category	PWR Accident Probability per Year	BWR Release Category	BWR Accident Probability per Year
PWR 1	7×10^{-7}	BWR 1	9×10^{-7}
PWR 2	5×10^{-6}	BWR 2	2×10^{-6}
PWR 3	5×10^{-6}	BWR 3	1×10^{-5}
PWR 4	5×10^{-7}	BWR 4	3×10^{-5}
PWR 5	1×10^{-6}	BWR 5*	1×10^{-5}
PWR 6	1×10^{-5}	BWR 6*	1×10^{-4}
PWR 7	6×10^{-5}		
PWR 8*	4×10^{-5}		
PWR 9*	4×10^{-4}		

*Assumed not to result in core melt.

contribute to the overall probability of core melt [1]. Based on this review, the Staff in Volumes I and II of NUREG-0460, concluded that ATWS events would be "significant" contributors to the overall probability of core melt in future reactors. Some of the factors prompting this conclusion are:

a) The PWR studied in WASH-1400 is not representative of all PWRs, (not necessarily of future designs);
b) The inclusion in the analysis of non-core melt sequences such as steam-generator tube failure;
c) The lack of the recirculation pump trip in some BWRs; and
d) The inclusion of analyses not considered in WASH-1400 such as unavailability of various mitigating systems.

Based on these considerations, Volumes I and II of NUREG-0460 contain proposed acceptance criteria for existing and proposed LWRs.

Concurrent with the NRC staff review of ATWS events, the NRC commissioned a review of the Reactor Safety Study, the results of which appeared in NUREG/CR-0400 [6] several months after publication of NUREG-0460. Of particular interest to this study are the following findings of the Review Group:

1. WASH-1400 was largely successful in at least three ways: in making the study of reactor safety more rational, in establishing the topology of many accident sequences, and in delineating procedures through which quantitative estimates of the risk can be derived for those sequences for which a data base exists.
2. We are unable to determine whether the absolute probabilities of accident sequences in WASH-1400 are high or low, but we believe that the error bounds on those estimates are, in general, greatly understated. This is true in part because of an inability to quantify common cause failure, an inadequate data base in many cases, and in part due to some questionable methodological and statistical procedures.

Among the relevant recommendations, the Review Group stated:

1. Reevaluate NRC's inspection and quality assurance system, and licensing criteria to determine the extent to which they incorporate those things that have been learned from WASH-1400 and other relevant literature.
2. In general, avoid the use of the probabilistic risk analyses methodology for the determination of absolute risk probabilities for subsystems unless an adequate data base exists and it is possible to quantify the uncertainties.
3. Fault tree/event tree analysis should be among the principal means used to deal with generic safety issues, to formulate new regulatory requirements, to assess and revalidate existing regulatory requirements and to evaluate new designs.

It is interesting to note, that in a small subsection on ATWS events, the Review Group quotes some of the results of NUREG-0460, Volumes I and II, claiming an improvement of the ATWS results in WASH-1400 but caution extrapolation to a full nuclear industry.

Following release and publication of the Review Group's report, the NRC staff published Volume III of NUREG-0460 which reevaluates their position on ATWS acceptance criteria. In Section III, the originally proposed ATWS acceptance criteria and the reevaluated criteria are reviewed.

It should also be noted that following publication of these three reports (NUREG-0460, Vols. I, II and III and NUREG/CR-0400) the NRC Commissioners issued a statement concerning the Reactor Safety Study [7]. Of particular interest here is the following statement included in their cover letter:

> The quantitative estimates of event probabilities in the RSS* should not be used as the principal basis for

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*Reactor Safety Study

any regulatory decision. However, these estimates may be used for relative comparisons of alternative designs or requirements provided that explicit considerations are given to the criticisms of those estimates as set forth in the Report of the Risk Assessment Review Group.

LIQUID METAL FAST BREEDER REACTORS (LMFBR)

The major emphasis on safety research for LMFBRs has been on Core Disruption Accidents (CDAs), because of their potential for large energy releases. A CDA is defined loosely as an accident involving loss of core coolable geometry.* The potential for large energy release stems from the fact that LMFBR cores are not designed to be in their most critical configuration, and various material motions and/or composition changes can lead to large reactivity additions.

The two accident scenarios receiving the most attention are the transient overpower accident (TOP) and the loss of flow (LOF), both without scram. The TOP is an accident in which a postulated reactivity insertion causes the power to increase but heat removal capability is assumed to remain nominal. The LOF is an accident in which the heat removal capability is postulated to decrease while the power generation is assumed to remain nominal. In both cases, the reactor is unprotected, i.e., the plant protection system (PPS) or scram system is assumed to fail.

Recently, there has been interest in core disruptive accidents initiating from the shut-down state. Two examples are:

- o the loss of heat sink with scram and
- o the loss of cold leg piping with scram.

The most recent licensing experience for an LMFBR that may resemble future commercial plant experience is the review of the PSAR

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*A distinction is sometimes made between pressure driven disassembly and a slow progression of melting. In this report CDA will cover both.

for the Clinch River Breeder Plant (CRBRP). Although the formal review has been suspended by the NRC, certain preliminary, yet relevant decisions were made by both the applicant and the regulatory agency.

The initial PSAR contained two designs for CRBRP; the reference design in which CDAs were not considered as part of the design basis and the parallel design (or fallback design) in which systems to accomodate or mitigate against CDAs were included. Subsequent to the initial docketing of the PSAR, an ammendment was submitted withdrawing the parallel design, but, "in keeping with past practice of first-of-a-kind plants, the project planned to incorporate features designed on the basis of accomodating a range of events including those having an exceedingly low probability of occurrence" [8]. Included were plans to, "incorporate features designed to mitigate consequences of accidents from loss of in-place coolable geometry" [8].

Before suspending formal review of the PSAR for CRBRP, the NRC staff issued preliminary comments and guidance with respect to the PSAR for CRBRP [9]. Although the views and positions of the NRC staff were intended to be specifically for CRBRP and not intended to establish precedents for future LMFBR reviews, they are important because they parallel the proposed criteria for LWR-ATWS acceptance.*

In Section IV, we review the consideration of ATWS events in the safety analyses of CRBRP and the proposed NRC licensing criteria.

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*It should be noted that prior to the suspension of the licensing review for CRBRP, the applicant was in the process of appealing some of these preliminary decisions reached by NRC.

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III. PROPOSED ATWS CRITERIA FOR LWRs

The objective of designing for safety is to protect against "Anticipated Events" and "Unlikely Events" and to design engineered safety features to prevent or mitigate against "Highly Unlikely Events." These events, and their assumed frequency of occurrence were discussed in Section II. The significance of ATWS is that some of the transients considered under "Anticipated Events"--if not controlled by scramming the rods or by the actions of other systems--can result in core melt. Surveys of transients at operating reactors show that their frequency vary with the age of the plant. For established plants, the rate was nearly 20 per year early in life and decreased to 6 per year later in life. The NRC staff believes that the rate of occurrence of anticipated transients is approximately 10 per reactor year, with approximately 6 per reactor year resulting in significant consequences had scram not occurred and need be considered in ATWS evaluations.

The estimation of scram reliability is difficult because the systems are highly reliable and very few failures have occurred. Table IV reproduced from NUREG-0460 [1], shows the variability of different estimates.

Table IV

SCRAM FAILURE PROBABILITY PER DEMAND[1]
(Assuming Monthly Testing)

Confidence Level	EPRI Part I	WASH-1270	Current Staff Estimate
50%	3.0×10^{-6}	6.9×10^{-5}	1.1×10^{-4}
90%	1.3×10^{-5}	1.6×10^{-4}	3.0×10^{-4}

Based upon available data, the Staff concludes that the probability of undetected scram system failure lies between 10^{-5} to 10^{-4} per demand and arbitrarily used a value of 3×10^{-5} per demand in its assessment. From these considerations the Staff proposes that the frequency of an anticipated transient without scram resulting in significant consequences is approximately 2×10^{-4} per reactor year. The Staff concluded that, "If the frequency of ATWS events resulting in core melt were reduced to approximately 10^{-6} per reactor year, ATWS would be a small fraction of the overall risk from nuclear power even if further improvements were to reduce the probability of other sequences" [1].

Reduction of ATWS Risk

Based upon the discussion presented above, the NRC staff has concluded that some corrective measures to reduce the probability or consequences of ATWS events are required. Three general means of achieving this objective were proposed in NUREG-0460, Vol. I:

a) Reduction in the number of transients. To meet the proposed 10^{-6} per year objective would require a reduction of 100 in the frequency of transients. Such a large reduction appears to be impractical.
b) Improve scram reliability. The Staff considers this means difficult to achieve because it may not be readily demonstrated and/or may not be attainable. However, this may be accomplished by a second shut down system.
c) Mitigation of ATWS consequences. Provide mitigating systems to limit pressure rise, to provide make-up water and core cooling, and to limit leakage of radioactive material.

Proposed Requirements for LWRs (NUREG-0460, Vols. I and II)

The method first proposed by the NRC staff in NUREG-0460, Vols. I and II, is the use of deterministic calculations and probabilistic criteria to specify ATWS licensing requirements. For each plant a selected set of ATWS events would be analyzed using specific methods in order to determine whether certain performance and engineering acceptance criteria can be met.

In particular, the following acceptance criteria were proposed:

a) Radiological consequences; the calculated radiological doses from postulated ATWS events shall be within the guidelines of 10 CFR 100.
b) Primary system integrity; places limits on coolant system pressure and temperature.
c) Fuel integrity; no significant core distortion.
d) Containment integrity; calculated containment pressure, temperature and other variables shall not exceed design value.
e) Long term shut-down and cooling capability; plant shall be shown capable of returning to a safe cold shut down condition indefinately.
f) Mitigating systems design; includes automatic initiation, high availability, independent, separate and diverse, qualified, monitored and periodically tested.
g) Reactor protection system design; can be used as an acceptable means of reducing the probability of ATWS events if it meets the criteria posed under (f) above for mitigating systems design.

The deterministic calculations are to include the transients listed in Table V. Given an ATWS frequency of 2×10^{-4} per reactor year, and the proposed goal of not exceeding 10 CFR 100 guidelines at 10^{-6} per reactor year, suggests that no more than 1 in 200 of these transients should result in calculated consequences exceeding 10 CFR 100 guidelines.

TABLE V

ATWS INITIATING EVENTS [1]

Pressurized Water Reactors (PWR)	Boiling Water Reactors (BWR)
Rod Withdrawal	Primary Coolant Flow Decrease
Boron Dilution	Reactor Water Temperature Decrease
Loss of Primary Flow	Reactor Coolant Flow Increase
Inactive Primary Loop Startup	Reactor Water Inventory Decrease
Loss of Electric Load	Primary Pressure Increase
Loss of Normal Feedwater	Rod Withdrawal
Loss of Normal Electric Power	Loss of Normal Electrical Power
Load Increase	Stuck Open Safety/Relief Valve
Primary System Depressurization	60° Step Loss in Feedwater Heating
Excessive Cooldown	

Proposed Requirements for LWRs (NUREG-0460, Vol. III)

As mentioned in Section II, the Staff issued a third volume on LWR-ATWS considerations based upon new safety and cost information, new insights on the general subject of quantitative risk assessment (the Review Group report) and from internal NRC review (ACRS and RRRC).* In addition, nuclear utilities, architect engineering firms and reactor manufacturers responded to the Staff recommendations in Volumes I and II, maintaining that the proposed requirements for mitigating systems are unnecessarily conservative and that the costs would be significant, particularly if applied to plants under construction or operation.

In Volumes I and II, the Staff proposed a numerical safety objective (the proposed goal of not exceeding 10 CFR 100 guidelines at 10^{-6}

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*ACRS = Advisory Committee on Reactor Safeguards
RRRC = Regulatory Requirements Review Group.

per reactor year, suggesting that no more than 1 in 200 transients should result in consequences exceeding the ATWS Acceptance Criteria). In Volume III, the Staff states:

> We now believe that a numerical safety objective is not satisfactory for use in nuclear regulatory decision making in the manner suggested in the first two volumes of NUREG-0460. We continue to believe, however, that quantitative risk evaluations have provided a valuable input to our understanding of ATWS. We now believe that the resolution of ATWS concern should rest on engineering evaluation and judgment of the appropriateness of alternative plant modifications, rather than rest on quantitative risk analysis.

Based on these considerations, the Staff discusses four alternative plant modifications, the different degrees of assurance of safety they are judged to provide, and addresses the question of how ATWS should be resolved for new plants and plants under construction or in operation.

The four alternative plant modifications are:

Alternative 1. <u>No Plant Modifications</u>. This represents the long-standing position of the nuclear industry.

Alternative 2. <u>Modification to Reduce Susceptibility to Common Mode Electrical Failures</u>. This is aimed at improving the prevention of ATWS events (i.e., decrease their probability).

Alternative 3. <u>Modifications to Reduce Susceptibility to Common Mode Electrical Failures and to Provide Mitigation of Most ATWS Events</u>. This would provide mitigation measures in addition to the prevention measures of Alternative #2.

Alternative 4. <u>Modifications to Provide Mitigation of ATWS Events</u>. This relies on mitigation and does not contain the additional measures of improving prevention. Alternative #4 would provide an implementation of the proposed licensing criteria of Vol. II of NUREG-0460 that is acceptable to the Staff.

The Staff then discusses the different degrees of assurance of safety within the context of each vendor's design. In particular, Alternative #3 is less stringent than the conclusions and recommendations of Vol. I, insofar as mitigation is concerned. Alternative #4 corresponds to the recommendations of Volumes I and II for new plants, <u>except that the requirement to make ATWS a design basis event has been removed</u>.

While most of the discussion in Vol. III involving implementation is focused upon existing plants and plants under construction, a recommendation for new plants is also presented. The Staff proposal, is to establish, through rule making, general NRC requirements on plant modifications contained in Alternative #4 for new plants. This proposal follows closely Volumes I and II with the exception noted above. The rule would eventually become a Regulatory Guide.

IV. ATWS EVENTS IN LMFBRs

In this section, we review and compare the transient events considered in the safety evaluation of the Clinch River Breeder Reactor Plant (CRBRP). We then compare the proposed criteria for ATWS events in LWRs to NRC's proposed criteria for CRBRP. From this comparison, we then determine the impact of the proposed LWR - ATWS criterion on LMFBRs.

TRANSIENT EVENTS IN CRBRP

The safety approach for the reference design of CRBRP was developed upon three levels of design and was intended to be consistent with NRC's approach for licensing LWRs as reviewed in Section II:

- The first level focuses on the reliability of operation, accident prevention, and intrinsic design features;
- The second level focuses on protection against "Anticipated Faults" and "Unlikely Faults"; while
- The third level focuses on the inclusion of "Extremely Unlikely Faults" in the design basis.

As stated in Section II, consideration was also given to events beyond the design basis because CRBRP was intended to be a "first-of-a-kind" plant. In particular, the Loss-of-Flow (LOF) and Transient Overpower (TOP), both without scram, were analyzed in detail. These sequences lead to a Core Disruptive Accident (CDA) and a potentially large energy release. In addition, preliminary analysis of the loss of cold leg pipe integrity was presented in the PSAR.

Table VI, taken from the CRBRP-PSAR [8], contains examples of the events described above. Several important points come to light from an inspection of Table VI:

Table VI

LMFBR TRANSIENT EVENTS [8]

REACTIVITY INSERTION DESIGN EVENTS	UNDERCOOLING DESIGN EVENTS
Anticipated	**Anticipated**
Control assembly withdrawal at startup	Loss of off-site electrical power
Control assembly withdrawal at power	Spurious primary pump trip
Seismic reactivity insertion (core, radial blanket and control rod) - OBE	Spurious intermediate pump trip
Small reactivity insertions	Inadvertent closure of one evaporator or superheater module isolation valve
Inadvertent drop of single control rod at full power	Turbine trip
	Loss of normal feedwater
	Inadvertent actuation of the sodium/water reaction system
Unlikely	**Unlikely**
Loss of hydraulic holddown	Single primary pump seizure
Core radial movement	Single intermediate loop pump seizure
Mal-operation of reactor plant controlls	Small water-to-sodium leaks in steam generator tubes
	Failure of the steam bypass system
Extremely Unlikely	**Extremely Unlikely**
Cold sodium insertion	Steam or feed-line pipe break
Gas bubble through core	Loss of normal shutdown cooling system
Seismic reactivity insertion (core, radial blanket and control rod) - SSE	Large sodium/water reaction
Control assembly withdrawal at startup-max. mech. speed	Primary heat transport system pipe leak
Control assembly withdrawal at power - max. mech. speed	Intermediate heat transport system pipe leak

- o The anticipated initiating events for CRBRP transients are very similar to the anticipated initiating events for LWRs (Table V). Rod withdrawal, loss of electric power and coolant flow perturbations are examples of transients common to all reactors.
- o The two major CDAs considered--the LOF and TOP without scram--are basically Anticipated Transients Without Scram (ATWS). They are postulated events in which the reactor is unprotected.
- o The CDAs for LMFBRs are similar to Class 9 accidents in that they will be required to be beyond the design basis. The loss of heat sink and loss of cold leg piping (both with scram)for LMFBRs are similar to the loss of coolant accident (LOCA) for LWRs. The TOP and LOF sequences without scram for LMFRBs are similar to ATWS events for LWRs, with the exception of the potential for large energy releases in LMFBRs.

Of interest are the relative contributions of these events to the overall risk. As stated in Section II, the Reactor Safety Study concludes that ATWS events are small contributors to PWRs (the LOCA was the dominant contributor) and a major contributor to the BWR risk. For CRBRP, the applicant attempted to assess the risk by using WASH-1400 methodology [10]. Without commenting on the validity of the results, the applicant found that although the probability of a highly energetic CDA is contributed to about equally by the loss of heat sink and the LOF sequences, the major contributors to overall risk were:

- o LOF with failure to scram;
- o partial or total loss of AC power; and
- o earthquakes.

<u>Preliminary Criteria for CRBRP</u>

In a letter dated May 6, 1976 the NRC staff offered preliminary comments and criteria regarding the licensing review for CRBRP. Of interest here are comments concerning Design Safety Approach and Core Disruptive Accidents, some of which are overlapping. Although the views and positions stated in the letter were intended specifically for CRBRP, and not intended to establish precedent for future LMFBR license reviews, they are useful in assessing NRC's recent ATWS proposals for LWRs.

Under Design Safety Criteria, NRC proposes that in addition to assuring that the level of safety achieved for the CRBRP be comparable to that for LWRs (i.e., that the consequences of accidents within the design basis envelope are within the guidelines of 10 CFR 100), they propose the safety objective (Criteria #1):

> ...there be no greater than one chance in one million per year for potential consequences greater than 10 CFR 100 dose guidelines for an individual plant...

To meet this objective, 5 design features are specified as listed in Table VII (Items 2 through 5 and 7 under Design Criteria).

For Core Disruptive Accidents, NRC states:

> ...the probability of core melt and CDAs can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum.

Because of the uniqueness of the plant, however, NRC further stipulates that additional measures be taken to limit consequences and reduce residual risks from accidents having a lower probability than design basis accidents. To meet this objective, NRC proposes (Criteria #2): "...containment integrity be provided for at least 24 hours following a postulated core disruptive accident."

The evaluation of this requirement should be based upon:

- o A core mechanical work energy release of 1200 MW-sec (fuel vapor expansion to 1 atmosphere);
- o A sodium relase of 1000 pounds from the head; and
- o Vaporization of 10% of the core fuel inventory and direct release of this fraction from the head.

Table VII

THE EQUIVALENCY OF LWR-ATWS AND CRBRP PROPOSED SAFETY DESIGN AND CDA CRITERIA

ATWS CRITERIA FOR LWRs (NUREG-0460, Vol. I)	CRBRP CRITERIA Design Criteria	CDA Criteria
1. ATWS dose $\leq$ 10 CFR 100	1. Doses which exceed 10 CFR 100 must have frequency $< 10^{-6}$/yr.	
2. Primary system integrity (ASME Code, Level C-service limit stress).	2. Heat transport integrity must be high.	2. Withstand mechanical work energy of 1200 MW-sec.
3. Fuel Integrity (no significant core distortion).	3. Means to detect, cope and protect against subassembly faults.	3. DNA
4. Containment Integrity (not to exceed design value).	4. Protection from sodium releases be provided.	4. Provide containment integrity for 24 hrs following CDA.
5. Long term shutdown/cooling (subcritical state w/o rods and maintained in cold shutdown).	5. Two independent, diverse and functionally redundant decay heat removal systems must be required.	5. 24 hour integrity based on 1200 MW sec, 1000 lbs sodium, and 10% fuel vaporization.
6. Mitigating Systems Design	6 DNA	6. a. Head hold down and missile barrier; b. Sodium and fuel vapor deflector; c. Prevent hydrogen producing reactions.
7. Reactor Protection System must meet criteria of mitigating systems design.	7. Provide two indepent, diverse and functionally redundant shut down systems.	7. DNA

Note: DNA = Does not apply.

Three additional design features to provide containment system protection are specified in Table VII (6a, 6b, and 6c).

COMPARISON OF LWR-ATWS AND CRBRP CRITERIA

In Table VII, we compare the proposed ATWS criteria in Volumes I and II of NUREG-0460 for LWRs with the proposed criteria for CRBRP on a criterion by criterion basis. Criterion 1 stipulates that because ATWS events for LWRs will have a frequency greater than 10^{-6} per year they should become part of the design basis and must have resultant doses less than 10 CFR 100 limits. For LMFBRs, Criterion 1 stipulates that for CDAs to be considered beyond the design basis, the frequency of doses greater than 10 CFR 100 limits must be less than 10^{-6} per year. We believe that the implication here is that if a frequency of less than 10^{-6} per year is not achievable (or demonstrable) for LMFBRs, *CDAs would then become part of the design basis* if this proposal is implemented.

In Volume III of NUREG-0460, it is proposed to remove the requirement making ATWS a design basis event; i.e., to implement the remaining criteria without case-by-case accident analyses or dose calculations, We do not believe NRC would extend this proposal to LMFBRs if CDA initiators had frequencies greater than 10^{-6} per year, because of the potential energetics and limited operating data for LMFBRs. Indeed, engineering judgment, would dictate the implementation of mitigating systems for CDAs.

We suggest that the proposed criteria 2 through 5 for LWRs have a corresponding counterpart in the CRBRP criteria list as well. Although they are not identical, the criteria concerning primary system integrity, fuel integrity, containment integrity and long term shutdown/coolability, we believe, are analogous. The CRBRP criteria for CDAs appear to be more restrictive because they are based on a specified time, energy release and material composition (mass and thermodynamic state).

Criterion 6 for LWRs specifies Mitigating Systems Design while that for CRBRP actually exemplifies systems which are intended to mitigate the effects of a CDA. However, it is not clear to us whether or not these specific systems meet the design criteria.

Criterion 7 applies to the reactor scram or shut down system which we believe is at the heart of the issue. For LWRs the proposed criterion is intended to give specifications for an independent system. These specifications are basically those of a mitigating system. For CRBRP, both the applicant and the NRC specify two independent, diverse and functionally redundant shutdown systems. We believe that the key question is whether this can satisfactorily be achieved and/or demonstrated.

We can summarize our findings as follows. The CDAs considered for LMFBRs are similar to ATWS events considered in LWRs but with the potential for core energetics. As in LWRs, CDAs could be considered beyond the design basis if it can be shown that doses which exceed 10 CFR 100 limits occur with a frequency less than 10^{-6} per year. In contrast to LWRs, it is planned to achieve (and/or demonstrate) two independent, diverse and functionally redundant shutdown systems. Because CRBRP is a "first-of-a-kind" plant, criteria for fuel integrity, primary system integrity, containment integrity, long term shutdown/cooling and mitigating systems design have been proposed. These correspond in principle to the proposed LWR-ATWS criteria. Uncertain at this time is whether or not LWR-ATWS events with frequencies greater than 10^{-6} per year will be considered in the design basis. Volumes I and II of NUREG-0460 propose that they be considered in the design basis, while Volume III proposes that they not be considered. The implication for LMFBRs is not clear.

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V. DESIGN IMPACT

In Chapter IV, we have shown that the proposed ATWS acceptance criteria for LWRs (NUREG-0460) are quite similar to the preliminary Safety Design and CDA criteria proposed for the Clinch River Breeder Reactor. This similarity is not surprising because the initiators of CDAs considered for LMFBRs were shown to be similar to those considered in ATWS. From Table VII, it would appear that the proposed ATWS criteria have general applicability to LMFBRs as well.* Furthermore, it appears that NUREG-0460 (Vols. I and II) reinforces the notion that CDAs are outside the Design Basis if doses exceeding the 10 CFR 100 guidelines occur with a frequency of $< 10^{-6}$/year. Of particular interest in this report, is the potential impact on future designs for LMFBRs and policy implications regarding these criteria.

The major impact on design arises when considering the guidelines set forth in 10 CFR 100. One must either show that the accident sequences (CDAs) are outside the design basis envelope (less than 10^{-6} per year or 10^{-7} per year individual event) or mitigating systems must be included so that consequences fall below the 10 CFR 100 guidelines. For ATWS, a key design impact is in providing a Plant Protection System (PPS) or shutdown system with a failure rate of less than 10^{-6} per year.

For CRBRP, the PSAR states:

> ...the shutdown system shall have a probability of failure of less than <u>one chance</u> in a million per year (failure rate less than 10^{-6}/year) when required to prevent loss of cooled geometry [8].

It is proposed by the applicant to accomplish this with two redundant, independent, fast acting systems; each system containing:

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*It is interesting to note that in the case of CRBRP, the applicant agreed in principle to the safety design criteria [11], but disagreed with the proposed CDA criteria [12], in particular the 1200 MW-sec decision and the 24 hour containment integrity requirement.

1) diverse sensors, 2) diverse logic and 3) diverse circuitry, and each actuating separate, diverse sets of neutron absorber rods. Although the CRBRP Risk Assessment report is based upon achieving this value, there is lettle evidence to support the claim that it has been achieved and/or demonstrated.*

The NRC staff considers that this approach is acceptable in meeting ATWS requirements. However, if sufficiently high reliability (or low failure rates) cannot be achieved and demonstrated to the satisfaction of NRC, then the other criteria or approaches become overriding in design. The acceptance criteria for primary systems integrity, fuel integrity, containment integrity, long term cooling and shutdown, and mitigating systems design may become dominant design considerations.

For LMFBRs, the primary factor and the major difference compared to LWRs in meeting system integrity is the magnitude of the work/energy release. For mitigating system desing, the magnitude of the work/energy release as well as the energy partitioning (amount of fuel vaporized, sodium released, etc.) will be required. Alternative containment concepts (vented vs. nonvented) alternative engineered safeguards and alternative accident scenarios, will need to be developed so that proposed criteria can be met in an optimal way.

An alternate approach may be to demonstrate that the demand on the scram system can be reduced sufficiently in LMFBRs (as compared to LWRs) to justify the exclusion of DCAs from the design basis. The demand on the scram system could conceivably be reduced because of the low pressure primary system, and the large inventory of sodium in the primary and secondary loops; the so called "third-level thermal design margins" [8]. Because of the high heat capacity of the system, and the effects of natural circulation, it might be possible for corrective or manual action by the operator, negating the need for a second rapid, automatic scram. The time scales may be long enough for some events to limit core disruption, while the operator acts.

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*It is useful to note that some more recent designs for commercial-sized LMFBRs have included second, or even third shutdown systems using inherent actuation devices. These would bypass the need for any electronics or logic circuits and might, as a back-up for more conventional systems, increase the reliability of the shutdown system. Tests of some prototype systems have already begun.

VI. CONCLUSIONS, RECOMMENDATIONS AND POLICY IMPLICATIONS

In this report, we have reviewed briefly the Nuclear Regulatory Commission's (NRC) recent considerations of Anticipated Transients Without Scram (ATWS) for Light Water Reactors (NUREG-0460 [1]) and their proposed acceptance criteria for existing and new plants. In addition, we have reviewed NRC's preliminary Design Safety Criteria and Core Disruptive Accident (CDA) criteria for the Clinch River Breeder Reactor. From these, we have discussed some of the potential implications of ATWS on Liquid Metal, Fast Breeder Reactor (LMFBR) safety.

As a result of our investigation, we conclude the following:

- Many of the major CDAs considered in the safety analysis for LMFBRs (Transient Overpower Accident and Loss of Flow Accident) are basically ATWS type events but with the potential for a large work/energy release. Therefore, the proposed ATWS acceptance criteria for LWRs (as specified in NUREG-0460, Vols. I and II) are *in principle*, applicable to LMFBRS. Indeed, the preliminary NRC criteria proposed for CRBRP correspond to the LWR-ATWS criteria, but are based on core energetics.
- For LMFBRs the major criterion will be the assurance that the plant protection system (shutdown or scram) has a sufficiently high reliability (low failure rate) as we defined in Sections II through IV. Therefore, if a sufficiently reliable scram system cannot be achieved and/or demonstrated, then design criteria based on *CDA work/energy releases* will be required for mitigating systems design.
- Alternative accident scenarios for LMFBRs, such as loss of heat sink, loss of decay heat removal, and loss of coolant pipe integrity, which are initiated from the shutdown state, are similar to the loss of coolant accident (LOCA) in LWRs.

Although preliminary analysis indicates that the CDAs are the major risk contributor in LMFBRs little attention has been placed on the contribution of these LOCA type events.

- o The proposed ATWS criteria do not appear to present any new or unforeseen design and/or safety questions for LMFBRs. They do, however, specify design goals, especially for mitigating systems, which may insure conformance with NRC policy.
- o Although Vol. III of NUREG-0460 recommends removal of ATWS as a Design Basis Accident even though it may not be demonstrated that they have probability less than 10^{-6} per year, we do not believe this would hold for LMFBRs, because of CDA energetics and limited operating experience.

As a result of these conclusions, the following steps are recommended:

- o Research and development should be continued with the aim of insuring that sufficiently high reliability be achieved for the plant protection system (shutdown or scram).
- o Advantage should be taken, in design, of the intrinsic features of the LMFBRs, so that the demand on the plant protection system be minimized. In this context, CDA initiators should be examined to determine whether or not there is sufficient time for operator action (manual or otherwise) to reduce the demand. Operating experience for exisiting LMFBRs (Phénix, PFK, etc.) should also be examined to the extent possible, to gather data on scram demand.
- o For the first generation of LMFBRs it might be required to include systems designed to mitigate against the consequences of energetic CDAs. In this context, alternate approaches to engineered safeguards and mitigating systems should be examined. These include, alternate containment design, post accident heat removal and dedicated shutdown heat removal systems. These systems should also be measured against the proposed acceptance criteria for LWR-ATWS mitigating design.

- o Studies should be conducted to determine the contribution to risk of alternative accident scenarios such as loss of heat sink, loss of decay heat removal and primary pipe rupture. In this context, the appropriateness of mitigating systems design for CDAs should be established for these alternate scenarios.
- o Continued effort should be expended in the examination of designs which minimize CDA energetics.

What are the policy recommendations eminating from our brief but revealing study? We identify three recommendations which could effect the LMFBR safety program.

1) We recommend that DOE expend additional effort comparing LWR and LMFBR safety criteria. To what extent are these criteria parallel and overlapping? To what extent can LMFBRs be designed so that energetics are limited in CDAs? This recommendation is especially significant in view of the fact that in our brief--non all-inclusive-study--we could not identify any prior effort addressing this crucial comparison.
2) Fundamental to this additional effort, DOE must assess the desirability and extent to which LWR and LMFBR Safety Criteria can be made more uniform. What would be the cost (dollar, safety, and institutional) of doing so; what would be the cost of not doing so?
3) To the extent to which these criteria can be made more uniform, DOE needs to assess the degree to which the resolution of LMFBR safety concerns and the establishment of LMFBR safety criteria can more effectively be "piggy backed" onto the LWR Safety Program. This last, dramatic recommendation is significant in view of two--potentially conflicting--constraints. First, the possibility of a reduced LMFBR safety program over the next few years is very real. And second, the vital need to keep as many viable energy options open as cost effectively can be done is also very real.

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