

A RAND NOTE

SOME IMPLICATIONS OF THE THREE MILE ISLAND
ACCIDENT FOR LMFBR SAFETY AND LICENSING:
THE DESIGN BASIS ISSUE

Kenneth A. Solomon

August 1980

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PREFACE

This Note reviews the concept of *design basis accidents*--an artificial boundary separating those accidents which are considered in the licensing process from those which currently are not--and discusses how the concept enters into light water reactor (LWR) and liquid metal fast breeder reactor (LMFBR) design. It also re-examines the impact of the accident at Three Mile Island on the design basis concept, and how it might affect licensing for LWRs and LMFBRs.

This document, supported under U.S. Department of Energy contract AS03-78ET-37300, is one in a series of studies assessing nuclear safety issues. It should be of interest to those who design nuclear safety systems, who regulate nuclear safety, and who finance safety research.

Already published in the nuclear safety series is Rand Note N-1188-DOE, *Anticipated Transients Without SCRAM for Light Water Reactors: Implications for Liquid Metal Fast Breeder Reactors*, by W. E. Kastenberg and K. A. Solomon, July 1979.

SUMMARY

The 1979 accident at the Three Mile Island-Unit 2 (TMI-2) nuclear generating plant prompted numerous studies identifying relevant safety issues and recommending both short term and long term fixes to improve the general safety of the light water reactor (LWR). One of the major safety issues which has evolved from the Action Plan of the Nuclear Regulatory Commission's (NRC) Task Force investigations concerns the use of the design basis accident (DBA) in the licensing process. The DBA is an artificial "boundary" separating those "credible" accidents which are considered in the licensing process from those "incredible" ones which are not.

While these recommendations focus on current light water reactors, they will also affect design and operation of the liquid metal fast breeder reactor (LMFBR). This Note assesses the impact of the TMI-2 accident on the LMFBR. Specifically, it:

- o Reviews the DBA concept and its use in the licensing process.
- o Assesses the impact of the TMI-2 accident on the DBA concept in general and the LMFBR licensing process in particular.
- o Considers how the concept of risk can be used in setting the DBA criteria.
- o Discusses key implications of the TMI-2 accident on other issues in addition to the DBA concept.

There will be many significant similarities in the licensing approach for LWRs and LMFBRs; specifically, the range of accident initiating events considered, and their frequency of occurrence will be similar. I suggest that many of the changes required by the NRC Action Plan will apply directly or indirectly to the LMFBR.

Further, the TMI-2 accident may have a significant impact not only on the design of the future LWR Safety Program--but equally important--on the design of the future LMFBR Safety Program. Currently, the safety emphasis in both programs is on reducing the magnitude of very

high consequence, very low probability accidents. However, one of the major lessons of TMI is that more attention needs to be focused on higher probability (or even anticipated) events which could potentially propagate to large consequences, either through design failures or operator errors. Future considerations for both LWR and LMFBR risk reduction will include emphasis on these more likely events, in addition to "beyond the current design basis events." Recommendations include:

1. DOE should become actively involved with the NRC, which is actively pursuing the possibility of developing quantitative risk criteria for LWRs.
2. The several key factors addressed by the NRC Staff Action Plan must be resolved, including siting, degraded core accidents, reliability engineering, and safety vs. non-safety systems.
3. More detailed analysis of a wider range of complex transients is likely to become a future licensing requirement. This will affect operator training, emergency preparedness, and other activities.
4. A design basis for the LMFBR should be adopted based on quantifiable risk criteria analogous to, but not necessarily the same as, those for LWRs.

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GLOSSARY OF TERMS

Applicant	The Applicant is usually the reactor plant owner and is the designated license holder under NRC rules.
CDA	<i>Core disruptive accident</i> -refers to a reactor accident where at least part of the core integrity and core coolable geometry are lost.
CRBR	<i>Clinch River Breeder Reactor</i>
DBA	<i>Design Basis Accident</i> -an artificial boundary separating those accidents which are considered in the licensing process from those which are not.
EPA	<i>Environmental Protection Agency</i>
FFTF	<i>Fast Flux Test Facility</i> -a testing facility modelled after an LMFBR core. It is constructed on Federal land.
LOCA	<i>Loss of Coolant Accident</i> -refers to a reactor accident where the ability to cool the core is lost due to a break in the primary loop.
LOF	<i>Loss of Flow</i> -refers to a reactor accident where the ability to cool the core is either impeded or fully lost due to reduced or fully inoperable pumping power.
LWR	<i>Light Water Reactor</i>
LMFBR	<i>Liquid Metal Cooled Fast Breeder Reactor</i>
REM	<i>Roentgen Equivalent Man or Rad Equivalent Man</i> -refers to a measure of radiation in units of energy per unit mass.
RISK (of an event)	The probability of an event (often an accident) times its consequence.
SAR (also PSAR and FSAR)	<i>Safety Analysis Report (or Preliminary or Final Safety Analysis Report).</i>
SCRAM	The act (or ability) of (to) shut down the reactor and remove decay heat.
SER	<i>Safety Evaluation Report</i>
Staff	Refers to the NRC staff.

I. INTRODUCTION

The assurance of safe nuclear energy receives increasing public attention. Most recently, this attention was focused on the March 1979 accident at the Three Mile Island-Unit 2 (TMI-2) nuclear generating plant.

What is the impact of the TMI-2 accident on the safety, licensing, and development of nuclear energy? Although concerned primarily with the current generation of light water reactors (LWRs), ultimately this question must include the liquid metal fast breeder reactor (LMFBR) as well. In response to this longer term focus, Kastenberg and Solomon [1], in an earlier piece, assessed the potential impact of the proposed LWR-Anticipated Transient Without SCRAM (ATWS)* criteria [2] on the LMFBR safety programs as represented by the Clinch River Breeder Reactor plant (CRBRP). In the present Note, I consider another question: How does the TMI-2 accident affect the design basis accident criteria for LMFBRs?

BACKGROUND

The basic regulations governing nuclear power plant licensing in the United States are described in Title 10, Code of Federal Regulations-Energy [3]. As new issues arise, the Nuclear Regulatory Commission (NRC) can establish rules as a basis for regulation, which become part of the code and which provide policy and technical guidance for licensing. The NRC staff can issue Regulatory Guides describing methods acceptable for implementing specific parts of the Commission's regulations. In addition to these documents, the Standard Review Plan [4] sets forth internal review procedures followed by the NRC staff in evaluating documents and other information submitted for licensing review.

*An anticipated transient without scram is defined as the failure of the plant protection system (shutdown or scram system) following the initiation of a Category A event. Category A events have a frequency of several times per year.

Although there are several steps in the licensing process, two of the more important ones are (1) the Applicant's* submittal of the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR) and (2) the NRC's submittal of the Environmental Impact Statement (EIR) and the Safety Evaluation Report (SER).

The intent of the PSAR is to demonstrate that the design and construction will comply with regulations. Successful review of the PSAR usually leads to a construction license. The PSAR, along with supporting documentation, a set of technical specifications, a preoperational test program and a qualifications program for operating personnel is required before issuance of an operating license which will allow the Applicant to load full power. The technical specifications denote the manner in which the plant will operate.

Concurrent with the safety review, the NRC is required by the National Environmental Protection Act (NEPA) to consider environmental issues associated with a given license application review. The evaluation of NEPA requirements is usually contained in an Environmental Impact Report (EIR) which is published by the NRC staff along with a Safety Evaluation Report (SER). In these two reports--the EIR and the SER--the major safety and environmental issues relating to a particular license application are addressed.

BASIS FOR CURRENT STUDY

As a result of the recent accident at the Three Mile Island-Unit 2 (TMI-2) nuclear generating plant, several groups have made recommendations on how nuclear power plants should be licensed, sited, regulated, and operated. The groups have included the President's Commission [6], the Nuclear Regulatory Commission's Special Inquiry Group [7], the NRC staff's Short Term Lessons Learned Task Force [8], and the NRC staff's

*The Applicant is usually the plant owner and is the designated license holder under NRC rules. The Applicant is usually assisted in the preparation of the PSAR and FSAR by the Nuclear Steam Supply System (NSSS) vendor who supplies the reactor, the architect-engineer (A-E) who designs the power plant, and other consultants. A review of their functions can be found in Reference [5].

Long Term Lessons Learned Task Force [9]. One of the major safety issues emerging from the NRC Task Force investigations concerns the use of design basis accidents in the licensing process. As will be discussed in Section II, the design basis is an artificial "boundary" which specifies which accidents shall be considered and which shall not in the review of license applications and hence safety for a given plant. This issue arises because some of the events occurring at TMI-2 exceeded the design basis upon which the plant was licensed.

The recent safety review for the proposed Clinch River Breeder Reactor (CRBR), intended to be the nation's first fully licensed LMFBR, was patterned after the safety review for LWRs. As discussed in Section IV, a major issue in the safety review for CRBR was the definition of the design basis and the inclusion of events beyond the design basis in the review process. Hence, any new consideration of the design basis for LWR licensing and safety will naturally have implications for LMFBRs.

SCOPE OF PRESENT WORK

This Note begins by reviewing the concept of the design basis, design basis accidents, and risks (Section II). I then show how the current design basis issue arises from the accident at Three Mile Island and review the potential changes in regulatory requirements that may result (Section III). The safety and licensing issues for LMFBRs are also reviewed within the context of this experience and designs for future generation (i.e., commercial-sized) LMFBRs (Section IV). I show how risk considerations can be employed to resolve these issues (Section V) and how safety criteria for LMFBRs may be determined. Since design basis accidents are not the only issues affected by the Three Mile Island accident, Section VI presents a short discussion of some of the other key considerations arising from it. Finally, I draw some conclusions and offer recommendations for further work (Section VII). In the Appendix, I expand the discussion of moderate, small, and unlikely events as applied to both LWRs and LMFBRs.

II. THE DESIGN BASIS CONCEPT

In this section I review the concept of the design basis and the use of the design basis accident (DBA) in the regulatory review of light water reactors (LWRs). In addition, I consider how the concept of risk enters into the licensing process through the design basis concept.

DESIGN BASIS ACCIDENTS

During the licensing review of an LWR, the Applicant is required to analyze the course of certain accidents and their potential consequences. These accidents generally fall into three loosely defined categories:

- a) likely events,
- b) unlikely events, and
- c) extremely unlikely events.

Likely accidents are those which are expected to occur often enough (up to several times per year) that the intrinsic design of the plant can cope with them. An example is a turbine trip, for which the plant protection system and residual heat removal system are activated, bringing the plant to a safe shutdown. *Unlikely and extremely unlikely events* are expected to occur with sufficiently small probability such that the inherent features of the plant may not cope with them. However, because of their potential risk to the health and safety of the public, engineered safety features are installed to mitigate against their effects. For example, an emergency core cooling system (ECCS) is provided in LWRs to protect against the consequences of a loss-of-coolant accident (LOCA). Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), paragraphs 50.46 and Appendix K, specify the requirements for the design of such an ECCS. A LOCA, such as a double ended guillotine pipe rupture, is considered a design basis accident, and paragraphs 50.46 along with Appendix K specify the design basis upon which the applicant designs the ECCS.

Of interest also are (1) accidents that lie beyond the design basis and (2) the concept of "defense in depth." For example, in the Reactor Safety Study (WASH 1400), accidents for which the ECCS is *inoperable* were analyzed in terms of public risk. Because of the high reliability and redundancy *assumed* for the ECCS, however, such accidents are considered to lie beyond the design basis; thus they *are not* in the licensing review. To provide conservatism into the design of LWRs, the concept of "defense in depth" has been employed. Multiple barriers (or mitigating devices) are used in the event of fission product release from the fuel. Such barriers include fuel clad, the reactor vessel, and the reactor containment with mitigating systems such as the containment spray system, hydrogen recombiners, and pressure relief valves. Each barrier and/or system is designed to some design basis.

During the recent accident at the Three Mile Island--Unit 2 Generating Plant, the ECCS operation was terminated by operator error several times during the initial phases. This led to an uncovering of the core with subsequent fission product and hydrogen release. This degradation of ECCS performance, by human or other (mechanical or electrical) means, was outside the design basis. Because of the defense in depth concept, however, the reactor containment was successful in performing its function, i.e., mitigating the consequences of ECCS partial failure. The apparent burning of hydrogen in the containment produced a pressure of 28 psig for about 10 minutes, well within the 58 psig design basis. Fission product release was due, in part, to the failure to isolate containment when the sump pumps transferred primary coolant water to the auxiliary building.

The Nuclear Regulatory Commission's decision to exclude certain accidents from the design basis is predicated, in part, on their anticipated low probability of occurrence. Hence, a judgmental approach, in terms of risk, has been utilized by the NRC in setting the requirements for obtaining a license for a nuclear power plant.

THE CONCEPT OF RISK AND THE DESIGN BASIS

Qualitatively, the concept of risk is used to describe the negative impacts of technology on society. These negative impacts may arise due to normal (routine) operation of the technology (e.g., health hazards resulting from the use of fossil fuels) or during off-normal (accident) conditions (e.g., release of radioactive gas during a nuclear reactor accident). The quantitative determination of accidental risk requires a knowledge of both the frequency (probability per unit time) and the consequences of undesirable events.

Because severe nuclear reactor accidents may occur with very small frequencies, few, if any, actuarial data exist. Rather, quantitative measures of risk are based upon models and calculation, and there exist large uncertainties in risk estimates. However, the methodologies employed in risk assessment are particularly useful in evaluating the contribution to risk of potential accidents.

In setting the design basis, for example, some aspects of risk consideration have been employed. With respect to risk, the design basis can be interpreted as excluding accidents with frequencies based on [2]:

"an overall safety objective of 10^{-6} per year and an objective for individual events of 10^{-7} per year."

It is of interest to compare this frequency with frequencies given in the 1978 NRC publication NUREG-0438 [10] for different classifications of accidents, and the frequency of core melts given in WASH-1400, the Reactor Safety Study [11]. In NUREG-0438 frequencies are given as:

- | | |
|---------------------------------|-----------------------------|
| A. Events of moderate frequency | Several times/year |
| B. Events of small probability | 1/10 to 1/100 per year |
| C. Highly unlikely accidents | 1/1000 to 1/10,000 per year |

yielding a potential limit for the design basis of 10^{-4} /yr.

In WASH-1400, the frequency of the dominant risk contributor (core melt) in the plants studied was estimated to be 5×10^{-5} per reactor year (with only about 2% of these core melt events resulting in early fatalities). Core melt has usually been considered to be beyond the design basis in the regulatory process. Hence, one might interpret the design basis for an LWR, in terms of frequency, as lying between 10^{-4} and 10^{-6} per year, with 10^{-6} from the Standard Review Plan [4].

In terms of consequence, the design basis is set in Title 10, Part 100 of the Code of Federal Regulations (10 CFR 100),* which specifies the allowable dose limits to which the design basis must conform (i.e., sets the design on engineered safety features intended to mitigate against design basis accidents). From the above discussion we see that accidents with frequencies less than 10^{-6} per year (10^{-7} per source [4]) are excluded, regardless of their consequences. Accidents that occur with frequencies greater than 10^{-6} per year must have mitigating system (ESFs) that yield consequences below the 10 CFR 100 limits.

An expanded discussion of these frequencies and example consequences, as well as a further description of design basis accidents, are given in the Appendix.

* At the present 10 CFR 100 specifies that a person at the fence post will not receive a whole body dose greater than 25 rem and a thyroid dose greater than 300 rem as a result of a design basis accident [11].

III. THE DESIGN BASIS ISSUE RAISED BY TMI

As a result of the accident at the Three Mile Island-Unit 2 (TMI-2), a number of issues relating to LWR safety have been raised. Of interest here are those issues relating to design basis accidents. The NRC staff, in both its initial study [8] and its final report [9] on TMI-2 lessons learned, discuss the potential inclusions of "beyond the design basis" accidents. Discussion of the consideration of beyond design basis accidents stems primarily from the generation of hydrogen during those periods when the TMI-2 core was uncovered.*

Hydrogen can be produced during an LWR-LOCA by metal water reactions, radiolysis, and corrosion. At TMI-2, approximately 40% of the clad material reacted with water to produce hydrogen in the vessel and subsequently the containment building [13]. The design basis for combustible gas control is given in 10 CFR 50.44, and Regulatory Guide 1.7 - Revision 2 describes methods acceptable for implementation. These methods are based on an assumed clad/water reaction which is between 1 and 5%, depending upon the accidents considered. In the case of TMI-2, the design basis was clearly exceeded.

In the short-term lesson learned report [8], the NRC staff recommendations regarding post-accident hydrogen control systems for LWR containment were:

1. Provide penetrations for external recombiners or post-accident external purge system,
2. Require inerting BWR containments, and
3. Provide capability for install hydrogen recombiners at each LWR.

* When the water level in the reactor was below the top of the fuel elements.

In the final report on lessons learned [9], the NRC staff recommends that "the [NRC] Commission issue within three months a notice of intent to conduct rule making to solicit comments on the issues and facts relating to the consideration of design features to mitigate accidents that would result in (a) core-melt and (b) severe core damage, but not substantial melting." Provisions to cope with hydrogen generation and the consideration of the use of vent-filtered containment would be part of the rule-making process.

The rationale for the rule-making can be paraphrased from the Staff report as follows [9]. Accidents that result in substantial melting of the core are the most significant--in terms of public risk--of the events not included in the design basis. Even though they should have a lower frequency (less than 10^{-6} /yr) than design basis accidents, their consequences make them risk significant. From the accident sequence point of view, it is the potential failure of containment which yields this high risk. Hence, prevention of containment failure, such as the case with the vent-filtered containment, should reduce significantly the consequences of core melt.

The recommendations of the Staff represent a potential shift in the licensing process toward the inclusion of events beyond the design basis. While it is not clear that rules will be adopted that require the consideration of core melt (or degraded cores) in the licensing process, the recommendations do have a significant effect on future licensing reviews of LMFBRs. In particular, potential high consequence, low frequency events could become part of the design basis. Alternatively, mitigation features may be required, without these events being made design basis accidents.

IV. LICENSING AND SAFETY OF LMFBRs

The recent experience with licensing an LMFBR has been limited to the Fast Flux Test Facility (FFTF) in Washington state and the Clinch River Breeder Reactor (CRBR) in Tennessee. FFTF is unique because it is a test facility constructed on federal land and as such is not required to undergo any formal licensing procedure. However, the NRC staff has reviewed both the Preliminary and Final Safety Analysis Reports (PSAR and FSAR) and has issued their Safety Evaluation Reports (SER) in each case. Although the recommendations of the Staff are not binding, they have for the most part been considered by the Applicant (in this case DOE).

More recently, the licensing experience with CRBR may be indicative of future commercial plant experience. Following submittal of the PSAR in April 1975, certain preliminary but relevant decisions were made by both the Applicant and the Staff. Because of the uncertainty in its completion, the licensing review for CRBR has been suspended. In the following subsection, a description of the current status of LMFBR licensing and safety is described with special emphasis on the difference between LMFBR and LWR safety considerations.

LMFBR SAFETY QUESTIONS

The major emphasis in the CRBR review, and indeed a key concern for any large LMFBR, was the ability of the plant to withstand the consequences of a core disruptive accident (CDA), which involves loss of core coolable geometry and has a potential for large energy releases. The importance of this issue in LMFBR safety considerations stems from two key differences between LMFBR and LWR concepts--the core configurations, and the reactivity effects of the coolant-moderator media. In the case of LWRs, this fluid is water. In the event of a failure to adequately cool the core, either resulting from a loss of water or leading to such a loss due to boiling, the reactivity effect of the loss of water in the core tends to shut down the reactor. The primary concern is then one of a gradual melting of the

core, slumping to the bottom of the vessel, and either an energy release due to a fuel-water thermal interaction in the vessel bottom, or a melt through of the vessel. If the containment fails, radiation could be released to the environment.

LMFBRs, however, have two inherent features with the potential for producing a much more energetic release. First, a loss of sodium from the core tends to *increase* the reactivity of the reactor core.* This places more importance on rapid insertion of control rods to reduce the total reactivity of the reactor core plus sodium void reactivity well below criticality. Second, because the core of an LMFBR contains several critical masses of fuel and is not in its "most critical" configuration we can hypothesize that if damaged, it might at some point reach a new configuration and attain criticality, even after shutdown, if lack of adequate core cooling led to core melting or significant fuel motion. Thus, although a distinction is sometimes made between a rapid pressure-driven disassembly and a slow progression to melting, various material motions (e.g., fuel motion or compaction) can lead to reactivity additions in either case. These reactivity additions, in turn, could lead to energy release in the event of failure of the plant protection system.

The two accident scenarios receiving the most attention for CRBR are the transient overpower accident (TOP) and loss of flow accident (LOF), both with failure to scram (failure of the plant protection system to shutdown the reactor). In either case the reactor is assumed to be operating at steady state (near design power) at the time of accident initiation. The failure to scram causes a power/flow mismatch[†] with ultimate material motions (e.g., fuel melting) and composition changes (e.g., coolant voiding).

Before the licensing process was suspended, regulators placed a greater emphasis on accidents initiating from the shutdown state, for example,

* This is known as the positive sodium void coefficient.

† In the TOP the power rises at constant coolant flow; in the LOF the coolant flow decreases at constant power.

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

In these scenarios, the reactor is shut down, but the ability to remove decay heat is impaired. Although the time scale is now expanded to hours (the TOP and LOF events range from milliseconds to seconds), lack of cooling will ultimately result in fuel melting. Although it is not clear how energetic such events will be, they clearly belong to the class of core disruptive events considered.

It is important to note that the initial PSAR contained two designs to CRBR: the reference design in which CDAs were not considered part of the design basis, and the parallel design (or fall-back) design in which systems to accommodate or mitigate CDAs were included. Subsequent to the initial docketing of the PSAR, an amendment 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 accommodating a range of events including those having an exceedingly low probability of occurrence" [14]. Included were plans to "incorporate features designed to mitigate consequences of accidents from loss of in-place (core) coolable geometry" [14].

Before suspending formal review of the PSAR for CRBR, the NRC staff issued preliminary comments and guidance with respect to CRBR [14]. Although the views and positions of the NRC staff were intended to be specifically for CRBR and not intended to establish precedents for future LMFBR reviews, they are important because LWR criteria are moving in their direction.

Before continuing, two additional key differences between LWR and LMFBR safety must be pointed out, both of which result from the use of sodium, rather than water, as the coolant-moderator in LMFBRs. First, sodium is well known to be highly reactive with air and water; the air in the containment and the water from the steam generators

can lead to sodium fires in the event of a sodium leak. This could, in turn, lead to generation of sodium aerosols and to overpressure problems of major concern since the sodium aerosols would be highly radioactive due to activation of the primary coolant sodium in an LMFBR.

A second difference is important in light of the concern over H_2 generation resulting from the TMI accident. Such a large production of H_2 , which resulted from the H_2O -cladding reaction in TMI, will not occur in the core of an LMFBR because there is no water in the core. On the other hand, there is the possibility of significant sodium-concrete interactions within the containment itself should the reactor vessel or sodium piping and components be breached and sodium released into the containment building. Two problems that could result include H_2 generation and production of radioactive sodium aerosols released into the containment. Although the specific recommendations for post-accident hydrogen control systems for the LWR primary containments may not be directly applicable to LMFBRs, the question of how to minimize overpressure within the secondary containment still arises.

PRELIMINARY LMFBR DESIGN BASES

In their preliminary comments on the PSAR, the Staff proposed that in addition to assuring that the level of safety achieved for CRBR 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 [15]:

"...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, five design features are specified [15] which include two independent, diverse, and functionally redundant decay heat removal and shutdown systems.

It was further stated that [15]:

"...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:

"containment integrity be provided for at least 24 hours following a postulated core disruptive accident."

To meet this objective, the following design basis was postulated by the NRC:

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

The 24 hour containment integrity requirement was later withdrawn because it was based in part on considerations contained in WASH-1400, which were dropped following the Lewis Review Report [16].

RISK CONSIDERATIONS

Concurrent with the submission of the PSAR, the applicant estimated the risk [17] of CRBR, and compared it to the risk estimated for LWRs as given in WASH-1400. The validity of the study is not considered here, but the results show that the major risk contribution stems from accidents which can be considered beyond the design basis. This result parallels that obtained in WASH-1400, namely that accidents involving core melt in LWRs are the major risk contributors.

It is of interest to note that in both cases the design basis

in terms of exceeding the 10 CFR 100 guidelines is 10^{-6} per year. Consequences greater than the 10 CFR 100 guidelines can occur, regardless of their magnitude, providing they occur with sufficiently small frequency. Hence two systems such as the LWR and LMFBR may have quite different risk estimates, yet they can both satisfy the design basis "acceptance" criterion. However, as noted above, the NRC requires both that the probability of a CDA in an LMFBR should be reduced to a low enough level to eliminate it as a design basis accident, and that measures be taken to mitigate the effects of such a potential CDA. Thus even events beyond the design basis may have to be considered, at least in general terms, in order to license future LMFBRs.

V. RISK CONSIDERATIONS FOR SETTING FUTURE
LMFBR AND LWR DESIGN BASES

As discussed in Section II, the quantitative determination of accidental risk requires a knowledge of both the frequency and the consequence of undesirable events. For example, suppose a given undesirable event occurred with a frequency of once in 10 years, and it resulted in 100 deaths. One measure of risk^{*} (expected value) is their product: 100 deaths times 1/(10 years), yielding 10 deaths/yr. The total risk, in terms of expected value, is then the sum over all possible events.

Risk-reduction, in terms of design, may involve the use of redundant and reliable systems (e.g., multiple coolant loops) designed to minimize the incidence of potentially serious accidents; highly reliable safety systems (e.g., the plant protection system and shutdown system) intended to prevent the progression of initiating events into major accidents; and further engineered safety systems to mitigate the consequences of those events which may occur (e.g., the containment spray system). Consideration of additional safety systems or features for existing plants, as well as integrated designs for new plants, can and should be based on their potential for risk-reduction versus their cost. Such tradeoffs are generally referred to as "risk-benefit" analysis or "value-impact" analysis.

As a result of the accident at the Three Mile Island-Unit 2 (TMI-2) plant, the Nuclear Regulatory Commission (NRC) is considering means to cope with hydrogen generation resulting from an accident. This leads to the consideration of events beyond the design basis because such amounts of hydrogen are generated in degraded cores (cores having had inadequate cooling for periods of times that leave them only partially disrupted). In addition to hydrogen, a degraded core

^{*} Other approaches include probability density distribution functions frequency versus consequence and complementary cumulative functions of frequency versus consequence.

will also be accompanied by fission product release to containment. Hence, the use of vent-filtered containment will be considered for the case of the molten, but not partially disrupted, core [8,9].

It is apparent, then, that any decision made by the Commission that requires additional safety features raises substantial issues concerning an appropriate design basis. Furthermore, one of the major lessons of TMI is that more attention needs to be focused on dealing properly with higher probability (or even anticipated) events which can potentially propagate to large consequences, either through design failures or operator error. Future considerations for both LWR and LMFBR risk reduction would include emphasis on these more likely events, in addition to "beyond the current design basis events."

The NRC is really concerned with three classes of accidents, all of which--not merely the design basis accidents--must be carefully considered: (1) moderate-to-high probability, relatively low consequence events; (2) design basis accidents; and (3) accidents beyond the design basis. As we have seen, the current emphasis is to design for a DBA, under the assumption that lower consequence events will be accommodated if DBAs are^{*} and that the probability of higher consequence accidents is low enough not to warrant their explicit consideration.

At what level of detail must these categories of accidents be analyzed? One previously known fact that TMI has illustrated more graphically is that consequences approaching those of a DBA are not limited to initiating events such as a large pipe leak; the key to avoiding such events appears to be in a recognition of the types of events and combination of events which could ultimately lead to serious consequences--usually involving loss of core-coolable geometry.

* Actually, there is a whole series of events of varying consequences and probabilities (e.g., Table A.1 and A.2 of the Appendix), many of which are specially included in the design.

A major LMFBR effort is being carried out to assess:

1. those types of major loss-of-flow or transient overpower accidents which lead to the onset of boiling in the core; *
2. the sequence of events following the initiation of sodium boiling;
3. possible mechanisms to reduce the likelihood of sodium boiling, or to mitigate its effects; and
4. the effects of, and possible designs to reduce the impact of, core meltdown or disassembly accidents.

This effort includes both experimental studies (e.g., the TREAT reactor) and computer simulations (e.g., the SAS and MELT codes [18]). However, what should be done to help assure that a more likely, supposedly benign, initiating event does not lead to such an accident?

The methods used in the Reactor Safety Study [11] (RSS) are being used in the LMFBR industry to assess the potential impacts of all types of initiators. That is, all conceivable initiating events (e.g., loss of offsite power, loss of one or more pumps, steam generator failure, etc.), including those where scram has occurred, are first modeled by event trees, which consider the probabilities and consequences of all subsequent system and operator actions. As was done in the RSS, some sort of probabilistic cutoff needs to be considered at this point, lest the total number of accident sequences becomes unmanageable. This is one of the key issues which must be addressed--the selecting of design basis safety criteria for LMFBRs.

* Since the specific conditions required to produce loss of core-coolable geometry or a core disassembly accident in an LMFBR are subject to much debate, the onset of sodium boiling at the top of the core is often used as a signal that some type of core damage could ensue.

I suggest that the basis for such criteria lies in the concept, expressed earlier for CRBR, that the level of safety^{*} to be achieved for an LMFBR be comparable to that attached for LWRs [15]. An example of how such criteria might be quantified [19] would be to first determine an overall design basis criterion (e.g., a probability of 10^{-6} or 10^{-7} per year of exceeding 10 CFR 100).^{**} Next, the probability that loss of core-coolable geometry leads to exceeding 10 CFR 100 would be determined for a specific concept. Finally, the maximum allowable probability of an accident leading to loss of core-coolable geometry would be derived from the previous two criteria. Note that, since the consequences of an energetic CDA in an LMFBR can be more severe than an LWR meltdown, it may be necessary to (1) require lower probabilities on accident initiators for LMFBRs, (2) require additional safety systems to lower the probability that any initiating event could lead to loss of core-coolable geometry, or (3) provide increased containment capability.

Thus it might be demonstrated that an adequate level of safety for the LMFBR requires analysis of even more types of accidents, both high probability-low consequence and low-probability-high consequence. Furthermore, differences between LWR and LMFBR safety will involve the specific differences associated with water and sodium systems, for example:

- o H_2 generation in LWR core and containment vs. the sodium fire problem and sodium concrete interaction in LMFBRs.
- o The sodium void effect in LMFBRs, and
- o The difference in heat transfer characteristics; for example, LMFBRs may be more "forgiving" for short-term transients due to the large heat capacity in the sodium,

^{*}I distinguish risk from safety in the following way: Risk and safety are complementary terms, the safer something is, the less risky it is.

^{**}Note that this has not been achieved by LWRs.

and near-atmospheric pressure operation far below the boiling point of the coolant.

In spite of these differences, there appear to be enough similarities in these systems so that many general safety criteria can be employed in either case. Although details of the physical phenomena may be greatly different and the trade-off between accident probabilities and consequences will be different, there are many areas of similarities. Specifically, shutdown systems and control room logic will require similar considerations; in addition, the type and nature of initiating events are quite similar [17]. And, finally, the future design basis for both LMFBR and LWR safety review can and should be based on similar risk considerations.

VI. OTHER CONSIDERATIONS RAISED BY
THE THREE MILE ISLAND ACCIDENT

Although the purpose of this Note has been to consider the design basis accident issue, other issues of at least equal significance to the safety and licensing of LMFBRs have arisen from the TMI-2 accident. Although they will not be discussed in detail here, their importance justifies commenting on several of them.

First, the level of operator training at Three Mile Island was considered "inadequate and contributed significantly to the seriousness of the accident," although "the TMI training program conformed to the NRC standard for training" [6]. Because of the potential for large consequence accidents with LMFBRs, the adequacy of training of reactor operators is just as important in both cases. Furthermore, many of the specific recommendations which may eventually be implemented for LWR operator training will be directly applicable for LMFBR operators. For example, the President's Commission [6] cited the facts that "simulator training did not include...multiple-failure accidents," and that "there was no formally defined training program for [auxiliary operators]." LMFBR designs can, however, provide some margin for operator error in many situations, since the high heat capacity of concepts with large primary sodium inventories can allow several hours of "grace period" before emergency core cooling becomes necessary.

In light of the importance of operator actions, the President's Commission also recommended that "Equipment should be reviewed from the point of view of providing information to operators to help them prevent accidents and to cope with accidents when they occur." This includes monitoring, warning and diagnostic equipment for both normal and abnormal situations, and should utilize computer technology where relevant. (See, for example, Reference 20.) Other equipment-related recommendations include the need to design systems and maintenance procedures to mitigate accident consequences. Specifically, systems

for isolation, filtering, venting, hydrogen recombining, and measuring coolant levels are key items mentioned.

The Commission also pointed out that increased awareness should be given to accident recovery, clean-up and waste disposal, and the health hazards arising from these operations. This will be even more important for LMFBR licensing and safety due to the extremely reactive nature of the coolant, the activation of the primary coolant, and contamination of the core, piping, and all loop components with solidified sodium even after their removal from the system.

Other health and safety measures were recommended which have no specific bearing on LMFBR safety. However, LMFBR safety should consider these measures (e.g., the biological effects of low-level radiation) *as they specifically relate to the levels and types of isotopes problematic to LMFBRs.*

Some reliability and risk assessment considerations can also be pointed out which supplement those corresponding to the Design Basis Accident issues. First, there is a need to analyze multiple failure accidents, especially considering human failures. Second, the President's Commission emphasized the need to continue "in-depth studies...on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown" [6]. Finally, as a supplement to Design Basis Accident analysis, the Commission reiterated the necessity of considering those accident initiators of lower consequence (but greater likelihood) than DBA's, again "with particular attention to human failures."

In addition to these factors, the TMI accident has brought up such fundamental issues as reactor siting, emergency preparedness, consideration of degraded cores, the current differentiation between safety and non-safety systems, and the entire licensing and review process itself. An in-depth discussion of such issues is not appropriate here; such questions should be given more detailed treatment separately. However, I will briefly review some of these points.

First, with regard to emergency preparedness planning and siting policy, the former has "had a low priority in the NRC and the AEC before it" [6]. The only NRC requirement has been to develop such

emergency plans for the low population zone (LPZ)--an area around each plant with low enough population density that such plans were expected to be feasible. In the case of TMI, this zone was a 2-mile radius around the plant. In this specific instance, Pennsylvania law was stricter and required planning within a 5-mile zone. However, during the accident itself, the specific circumstances led the NRC to believe that the consequences could exceed even the 5-mile radius; thus, evacuation plans extending out to 20 miles "were hurriedly developed" [6]. The implications for LMFBR siting and emergency preparedness are clear. Due to the potentially severe consequences of major LMFBR accidents, and in addition to engineered systems to mitigate the effects of such accidents, the following [6] should be strongly considered:

1. New siting policies, including remote siting where possible, must be considered for LMFBRs to be licensable.
2. Emergency plans must be carefully drawn up encompassing a wider range of accidents, with separate plans drawn up for each of a wide range of events. These plans should cover several zones of population density, extending well past current LPZs, and with different levels of action to be taken in each zone.
3. Plans should extend out to include areas receiving even lower levels of radiation than currently considered.
4. Means of *counteracting or protecting against* radiation hazards should be considered.
5. The public should be better informed about such plans, their purpose, and the hazards involved, well in advance.

I have already alluded to some problems resulting from degradation of core integrity. TMI-2 was not the first nuclear reactor to sustain core damage.* In fact, one LMFBR, the Enrico Fermi Fast Breeder

* Estimates for TMI-2 include cladding failures in 90% of the fuel rods, and fuel temperatures in the range of 4000-5200°F, possibly including fuel melting [6].

Reactor, had previously sustained a partial meltdown within two fuel elements in 1966, due to a coolant inlet blockage [21]. Although the damage was confined to the fuel elements directly affected, and the reactor was safely shut down, this incident, along with the TMI accident, indicates that core damage is not as remote a possibility as was once believed. It is now doubly important not only to more carefully examine the potential impacts of degraded core accidents upon the health and safety of the general population, but also to consider systems, both in-core and ex-vessel, to contain and/or mitigate the effects of a damaged core. Such systems would include auxiliary core-cooling systems, ex-vessel cooling systems, in-vessel and ex-vessel "core catchers," containment isolation and venting systems, etc.

Next we can point out a potential problem concerning the traditional separation of nuclear power plant systems into safety-grade and non-safety-grade classes. In the past, those systems considered specifically as safety systems were subject to significantly more rigorous criteria and qualification than "non-safety" systems. It may have taken the accident at TMI to indicate the importance of many of these "non-safety" systems in performing distinctly safety-related functions during an accident situation, or in initiating or complicating an accident situation. As recent events now suggest, it may be time to consider either 1) softening this rigid distinction and reclassifying many non-safety systems as safety-related, or 2) providing additional dedicated safety systems as a back-up to non-safety systems.

Finally, although this Note is not directly concerned with changes that may occur in the licensing or review process, I have indicated some areas where specific changes are likely to occur. As a result of TMI, changes in the NRC's operation and in the overall licensing process are occurring and will continue to occur over the next few years. There should be ample time and incentive for those involved in LMFBR safety and licensing to learn from the near term LWR licensing experience and apply it to the case of the LMFBR.

VII. FINDINGS AND RECOMMENDATIONS

This Note has reviewed the concepts of the design basis and design basis accidents and their use in the licensing process for nuclear power plants. I have also reviewed the issues relating to these concepts raised by the accident at Three Mile Island-Unit 2 and how they may affect the LMFBR option. Lastly, I considered the concept of risk and how it can be used in setting design basis.

At present, the design basis for nuclear power plant licensing is the assurance that accidents which occur with a frequency of approximately 10^{-6} /yr or greater, must have a dose no greater than 25 REM whole body (300 REM thyroid) at the site boundary. This design basis has been used to establish criteria for the design of engineered safety features incorporated into Light Water Reactors. Accidents whose frequency is less than 10^{-6} /yr are excluded from consideration regardless of their consequences. As a result, accidents postulated to occur in LWRs, and leading to core melt (e.g. LOCA) are excluded from the design basis. This exclusion is usually justified on the basis of low probability.

Similarly, during the preliminary licensing review for CRBR, and as I postulate for future LMFBRs as well, the goal was to insure a level of safety "comparable to the LWR" by using a similar design basis. One objective was to prove that core disruptive accidents were of sufficiently low probability that they were excluded from the design basis. If this cannot be shown for commercial-sized LMFBRs, then other criteria, as I discussed in Section V, should be instituted. Accidents falling within the design basis were required to conform to the dose limits set for LWRs.

Because CRBR was to be the first of a kind, additional safety features were to be incorporated in an effort to reduce residual risk. As a result of the TMI-2 accident, the NRC staff is considering the possibility of including events beyond the design basis in the licensing review of LWRs. Such events might include consideration of degraded cores, partially melted cores and core melt itself. In

particular, the use of vent-filtered containment and a need to establish a new design basis for hydrogen production will be addressed. Similar specific problems, such as sodium fires (already being considered) and containment venting must be addressed for LMFBRs.

FINDINGS

In spite of differences in details, I find many similarities in the licensing approach for LWRs and LMFBRs. In particular the range of initiating events and their frequency of occurrence are similar, although their potential consequences may be an order of magnitude larger for the LMFBR.

Engineered safety features will be incorporated into each system so that they will conform to whatever design basis is ultimately adopted. If the design basis is intended to set a safety goal, and is based on risk-benefit or value-impact considerations, both the LWR and the LMFBR options can achieve the same level of safety. Moreover, some of the LWR safety research program, especially in the area of probabilistic risk assessment, could be utilized if and when the LMFBR option is ready for deployment.

The NRC staff's Long and Short-Term Lessons Learned Task Force Reports [8,9] have already been shaped into an Action Plan [22] which will be implemented over the next few years. Many of the more general changes required by that Action Plan will be directly or indirectly applicable to the LMFBR option. This applies both to the Design Basis Issue, as well as to the additional question discussed in Section VI.

RECOMMENDATIONS

In light of the DBA issues discussed in this Note, and the additional questions discussed in Section VI, the following are recommended:

1. The NRC is actively pursuing the possibility of developing quantitative risk criteria for LWRs. DOE should become an active participant in the dialogue which the NRC will undertake during these considerations.
2. The NRC Staff Action Plan includes some subjects whose resolution may impact directly on LMFBRs. In particular the following matters appear to be of special importance:
 - a) Siting policy
 - b) Degraded core accidents
 - c) Reliability engineering

The resolution of these issues may lead to a major departure from the single failure criterion, important new restrictions on siting, and requirements for design features to mitigate accidents involving severe core damage or core melt.

3. Other features receiving new attention in LWR safety review include a rethinking of the previously accepted differentiation between safety and non-safety systems. A new philosophic approach may be developed which could affect LMFBR design. Similarly, greater emphasis is being given to the potential use of dedicated shutdown heat removal systems.
4. A much more detailed analysis of system behavior for a wide range of complex transients in addition to DBA's is likely to become a requirement. This should be factored into a wide range of activities including the following:

- o engineering and operator training simulators
 - o emergency procedures prepared in a disciplined, optimal fashion
 - o systems for disturbance analysis and other improvements in helping the operator diagnose a complex event.
5. A design basis founded on risk criteria (i.e., frequency of occurrence and consequences in terms of health effects, applicable for both LWRs and LMFBRs) should be adopted. These risk criteria have been established only to a limited extent for the LWR design, and the NRC has committed itself to establishing a more complete set of safety criteria for LWRs before the end of this year.

Appendix

FREQUENCY AND CONSEQUENCES OF MODERATE, SMALL, AND UNLIKELY EVENTS:
DESCRIPTION AND EXAMPLES

This appendix describes by example the concept of moderate, small, and unlikely events as applied to both light water reactors and liquid metal fast breeder reactors. Much of this appendix is adapted from Reference 1. For the purpose of evaluation, light water reactor (LWR) accidents are usually categorized as [7]:

- 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.
- C. Highly unlikely accidents (potentially severe accidents of extremely low probability, postulated to establish performance requirements of engineered safety features and site acceptability).

Table A.1, taken from WASH-1250 [1,7], contains some examples of the events and postulated accidents 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 [10], 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 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 10 CFR 100 [3]. Furthermore, Category C accidents are termed Design

Table A.1

EXAMPLES OF POSTULATED REACTOR FACILITY ACCIDENTS BY CATEGORY

- A. Moderate 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^a
 - 9. Inadvertent depressurization of the primary coolant system.
- B. Infrequent Accidents of Small Probability (abnormal radioactive release possible, but not expected)^b
 - 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)^c
 - 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.

SOURCE: Ref. 7.

^aIt is of interest to note that the recent accident at Three Mile Island was apparently initiated by a loss of normal feedwater flow, compounded by valve failure and operator error.

^bMay exceed the guidelines of 10 CFR Part 20.

^cMay not exceed the guidelines of 10 CFR Part 100.

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 A.2 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 10CFR100.

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" [2].

The Reactor Safety Study, WASH-1400 [11] attempts to estimate the frequency and consequences of all accidents (including Class 9) that could occur in LWRs. In general, the results indicate that the frequency of the dominant contributor to risk (core melt) in the plants studied is about 5×10^{-5} per reactor year, with only about 2 percent of the core melt events resulting in early fatalities [11]. Table A.3, obtained from the draft version, indicates the range of frequencies considered. (It should be noted that postulated accidents 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 frequency of a core melt in the pressurized water reactor (PWR) studies, while they dominate the boiling water reactor (BWR) risk. The NRC staff has reviewed the extent to which Anticipated Transients Without Scram (ATWS)

* PWR = pressurized water reactor; BWR = boiling water reactor. The integer corresponds to a particular release category described in WASH-1400 [11].

Table A.2

REACTOR FACILITY CLASSIFICATION OF POSTULATED ACCIDENTS
AND OCCURRENCES

No. of Class	Description	Example
1	Trivial incidents	Small spills. Small leaks inside containment.
2	Miscellaneous 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. Mechanical 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 Safety Analysis Report	Reactivity transient. Rupture of primary piping. Decrease of flow or 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.

SOURCE: Ref. 7.

Table A.3

PROBABILITIES OF INDIVIDUAL
RELEASE CATEGORIES

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

SOURCE: Ref. 11.

^aFrequency.

^bAssumed not to result in core melt.

contribute to the overall frequency of core melt [2]. 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 conclusions are these:

1. The PWR studied in WASH-1400 is not representative of all PWRs (not necessarily of future designs).
2. The inclusion in the analysis of non-core melt sequences such as steam-generator tube failure.
3. The lack of the recirculation pump trip in some BWRs.
4. 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 [16] 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 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 several 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 calculations over those in WASH-1400 but cautioning 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 re-evaluates their position on ATWS acceptance criteria. In Section III, the originally proposed ATWS acceptance criteria and the re-evaluated 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

*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."

The importance of this statement is that some of the decisions made during the review of the Clinch River Breeder Reactor discussed below may require further review and modification.

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 can be defined 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 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 shutdown by any means available. 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) (shutdown on scram system) is assumed to fail.

Recently, there has been interest in core melt accidents initiating from the shutdown state. Two examples are:

- o The loss of heat sink with scram.
- 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 for

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

[†] This is applicable to loop-type LMFBRs, such as CRBR, but not to pool-type concepts.

the Clinch River Breeder Plant (CRBRP) [23]. Although the formal review has been suspended by the NRC, certain preliminary but 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 accommodate or mitigate CDAs were included. Subsequent to the initial docketing of the PSAR, an amendment 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 accommodating a range of events including those having an exceedingly low probability of occurrence" [23]. Included were plans to "incorporate features designed to mitigate consequences of accidents from loss of in-place (core) coolable geometry" [23].

Before suspending formal review of the PSAR for CRBRP, the NRC staff issued preliminary comments and guidance with respect to the PSAR for CRBRP [24]. 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 as shown [1] they parallel the proposed criteria for LWR-ATWS acceptance.*

* 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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