Thermal and Hydraulic Aspects of Nuclear Reactor Safety

Volume 2

LIQUID METAL FAST BREEDER REACTORS

Symposium on the Thermal and Hydraulic Aspects of Nuclear Reactor Safety

Volume 2:

LIQUID METAL FAST BREEDER REACTORS

presented at
THE WINTER ANNUAL MEETING OF
THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
ATLANTA, GEORGIA
NOVEMBER 27-DECEMBER 2, 1977

sponsored by
THE HEAT TRANSFER DIVISION, ASME

edited by
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The *Symposium on the Thermal and Hydraulic Aspects of Nuclear Reactor Safety* is comprised of two volumes:

Vol. 1 — Light Water Reactors
Vol. 2 — Liquid Metal Fast Breeder Reactors

Within the text of each paper, numbers in parentheses indicate equations, and numbers in square brackets denote bibliographical references, which are listed at the end of the paper.

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Library of Congress Catalog Card Number 77-088144

Associate Editors

Preface

[illegible] [illegible] was [illegible] by the pioneers in the field [illegible] [illegible] limits directive summary of the current state of knowledge of the thermohydraulic behavior of nuclear reactors as it relates to issues of public welfare and safety. A group of organizers was formed having broad expertise in the thermohydraulics of nuclear reactors. Through their efforts, authoritative summaries covering a broad range of reactor safety topics were obtained from recognized experts the world over. These edited summaries were combined into an archieval, two-volume, hard-bound set entitled, "Thermal and Hydraulic Aspects of Nuclear Reactor Safety." The contents of these two volumes were presented and discussed in a three-day symposium during the Winter Annual ASME Meeting, November 27 - December 2, 1977.

This volume, "Vol. 2. Liquid Metal Fast Breeder Reactors," contains an overview article dealing with the pertinent safety issues of LMFBR's including implications of alternate fuel cycles. The scenarios for differing reactor designs differ from each other and from light water reactors, and as such are considered separately. Furthermore, some aspects of the field, such as LMFBR fuel-coolant interactions, have not yet reached a consensus of opinion. Differing points of view were thus sought out and are delineated in the two summaries contained herein. Considerable work has been accomplished using in-pile experimental techniques and these, along with an exhaustive summary of heat transfer abnormalities due to blockages in fuel assemblies, are also included. The state-of-the-art of sodium boiling, along with the specific aspects of boiling due to flow coastdown, is summarized as is the status of work on natural convection in both pool- and loop-type LMFBR's. Finally, transition-phase phenomena such as melting, boiling, and freezing are reviewed and a comprehensive summary of post-accident heat removal has been included. We hope that the resulting volume will be of permanent value both to the student entering the field and to the practicing research or design engineer.

The sophistication and complexity of all topics, coupled with the type of ultra-low probability occurrences considered, attests to the high level of understanding currently enjoyed in the realm of conventional thermohydraulics. This understanding is reinforced by the unparalleled record of operational safety existing today in the nuclear industry, and underscores the success of the current engineering design approach, which seeks to minimize the probability of an accident occurring, and maximize the insensitivity of the system to any failures which should occur. No other power generation system expected to produce energy in large quantities this century can be expected to be as safe. Certainly, with these facts in hand, western European communities, Russia, and Japan have all recently committed themselves to increased emphasis on the breeder reactor. The organizers and editors of this symposium firmly believe in the further development and growth of nuclear power as a clean, safe, and reliable source of energy in the future.

We have been fortunate in enlisting as Associate Editors and Session Co-Chairmen individuals of considerable stature in the reactor safety field, some of whom have themselves contributed overview papers. In the light-water reactor field, Dr. Y. Y. Hsu, U. S. Nuclear Regulatory Commission, Silver Spring, Md., and Prof. R. T. Lahey, Rensselaer Polytechnic Institute, Troy, N. Y., have covered the Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR), respectively. Prof. Neil E. Todreas, Massachusetts Institute of Technology, served as an Associate Editor and Session Co-Chairman for the Liquid-Metal Fast Breeder Reactor (LMFBR) field. In addition, Prof. Virgil Schrock, University of California, Berkeley, and Dr. D. S. Rowe, Exxon Nuclear Fuels, Bellingham, Wash., served as Associate Editors and Session Co-Chairmen for the LWR field, and Prof. J. C. Chen, Lehigh University, for the LMFBR area. Prof. F. A. Kulacki, Ohio State University and Prof. A. A. Bishop, University of Pittsburgh, served in similar capacities for the session of contributed papers. We have also received valuable help and suggestions from a number of colleagues, including Dr. H. K. Fauske, Argonne National Laboratory, and the ASME editorial staff. To all of these individuals we are deeply indebted.

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Liquid Metal Fast Breeder Reactor Safety: An Overview Including Implications of Alternate Fuel Cycles

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ABSTRACT

The safety aspects of liquid metal fast breeder reactors (LMFBR) are discussed including relationships to light water reactors (LWR). Factors including neutronics, coolant properties, and core design are compared for the two cases. Keeping in mind the multi-barrier concept of protection embodied in the design of all reactors, the key safety issues of today are presented including the impact of alternate fuel cycles.

INTRODUCTION

The following questions related to public health and safety are generally acknowledged to be the key items in feasibility considerations on the fast breeder reactor:

Reactor Safety – Can large fast reactors be built and operated with an acceptable risk to the public?

Safeguards – Can adequate safeguards against diversion of fissionable materials be provided?

Waste Disposal – Can the entire fuel cycle be operated with acceptable environmental impact?

It is only with respect to hypothetical reactor accidents (item 1 above) that liquid metal fast breeder reactors (LMFBRs) differ significantly from light water reactors (LWRs). This is not because the probability of an accident is any greater for the LMFBR but because the accidents that require analysis are quite different in character, involving different physical phenomena and a separate body of supporting research. This is particularly true of the Hypothetical Core Disruptive Accidents (HCDAs) which presently dominate LMFBR safety discussions [1]. Considerable attention is also being given to questions of local (subassembly scale) faults, means of detceting such faults, and the possibility of propagation from subassembly to subassembly [2].

The basic protection to the public from fast reactors as from all other reactor types against the escape of radioactive material is through a multiplicity of barriers, e.g., pin cladding, subassembly structures, the reactor primary system, cleanup systems, and the secondary containment system. In addition to these engineered barriers, there may be other inherent or natural barriers or attenuators such as meteorological dispersion. It should be clear that the issue of public safety or environmental risk from fast reactors, as for thermal reactors, can only enter through the HCDA, e.g., all accidents of significance to the environment lead to a core melt resulting in breach of the engineered barriers either from mechanical energy release associated with the core disruption (leading to rapid release of radioactive materials) or from thermal melt-through due to continued decay heat generation (leading to radioactive release at a relatively later time). It is only with respect to potential accident energetics that LMFBRs differ significantly from LWRs. Some important differences between LMFBRs and LWRs are discussed below.

Fast Reactor Characteristics Affecting Safety (LMFBR vs LWR)

The general characteristics of an LMFBR relevant to safety are summarized below:

Neutronics

LMFBR's have a high energy neutron spectrum with a small prompt neutron lifetime of the order of $< 1\ \mu s$, compared to water reactor lifetimes of the order of a millisecond.

The neutron lifetime is not especially significant unless the excess reactivity becomes so large that the maintenance of the chain reaction is no longer governed by delayed neutrons. This amount of reactivity, defined as one dollar, is numerically equivalent to the delayed neutron fraction (i.e., $\Delta K/K = 0.0035 = \$1.00$). It is a firm requirement in the design of fast reactors that any means for rapidly introducing reactivity approaching a dollar be eliminated. Even if the reactivity should exceed \$1.00, the small prompt neutron lifetime has been shown to be not important because of the prompt negative feedback due to the Doppler effect.* In a fast reactor transient, the Doppler coefficient is one of the major contributors to the safety of the system. It is effective both in terms of turning mild transients around and minimizing damage to the core or, in the case of very severe transients with violent disassembly, in very much reducing the destructive energy release. Its importance comes about primarily from the fact that in ceramic-fueled fast reactors, the Doppler effect is the only mechanism that yields an immediate negative reactivity feedback.

Some years back, there was not even complete certainty as to the sign of the Doppler coefficient. As long as there was an even remote possibility that the Doppler coefficient might be positive and hence act as an autocatalytic agent in an excursion, the Doppler effect posed a most serious problem. Both experimental and theoretical work have long since totally removed that possibility [3, 4]. It is now definitely known that in any fast breeder reactor composition of interest, the negative ^{238}U Doppler coefficient will totally dominate the fissile component of the Doppler effect (which may or may not be positive, but in any event extremely small in magnitude compared to the ^{238}U component).

The neutron energy spectrum affects accident behavior in a fast reactor in several ways. First, in the energy spectrum typical of the LMFBR, neutron absorption and fission cross

*This coefficient derives from the resonance non-fission capture of neutrons; an increase in temperature of the fuel effectively broadens the resonances and allows more capture of neutrons, thus decreasing reactivity.

sections are small compared to those in a thermal reactor. Thus, self-shielding effects in fuel and absorber materials are not important nor are decoupling effects in large cores as pronounced. This is an advantage from a control standpoint since spatial power oscillations do not occur as in large thermal reactors.

While neutron cross sections are small in the fast spectrum their ratios are still spectrum dependent. Hence, one finds a variation in the fuel capture to fission ratio and a corresponding variation in neutron multiplication if the neutron energy spectrum is shifted. This phenomenon is of significance in loss-of-cooling accidents where the core sodium density may be reduced through boiling and vaporization. Sodium voiding causes the neutron spectrum to shift upward in energy because neutron collisions with sodium contribute to degradation of their energy. The resulting spectral hardening on voiding causes an increase in neutron multiplication and a rise in core power. In LWRs, expansion of the coolant tends to shut down the reactor, because the coolant also acts as a moderator. The effect is somewhat counteracted since neutron leakage is also enhanced by sodium loss. Thus one finds that the reactivity effect caused by sodium voiding is a function of core size and may vary from positive to negative from the center to the periphery of the core.* *This effect can be designed out of large systems by appropriate core design, but with the penalty of increasing the fuel inventory and thus increasing the doubling time.*

Coolant Properties

Although the LMFBR core operates at higher power densities than LWRs, its liquid metal coolant has excellent heat transfer properties and the system operates with the coolant well below its boiling point. System pressures typically vary from near atmospheric at the core exit to perhaps 200 psi at the pump discharge. This means that the system can be designed such that a break in the primary coolant boundary does not result in a loss of cooling to the reactor core as in the case of a LWR (LOCA) or from the primary reactor vessel.

In the event of loss of all electric power to the pumps, the system can be designed such that decay heat can be removed by natural circulation of the sodium coolant without fuel failures [5, 6]. Only if loss-of-heat sink or failure of plant-protection systems are postulated, can core melt be visualized. Here sodium has a strong chemical affinity for halogens, which represents a significant advantage over LWRs. Since the sodium is chemically highly reactive and large quantities of energy can be released by the chemical reaction of the coolant with air or water, the reactor system and steam generators must be designed to minimize the probabilities, extent, and consequences of such reactions.

Core Design

The compactness and high power density of the core (5 to 10 times that of an LWR) has the result that imbalances between heat production and heat removal can lead to rapid changes in core temperature. Indeed it is the protection against mismatches between power and flow, and the analysis of their consequences, which form the central issues in fast reactor safety.

The main characteristic of a fast reactor core which complicates safety analyses is the fact that the system is not in its more reactive configuration and the reactivity can be

*While this effect is significant for Pu^{239} fuel, for U^{233} based fuel the fuel capture to fission ratio is relatively insensitive to energy spectrum variations.

added by the relocation of materials within the core. As differentiated from a water reactor where the core is arranged in nearly the most reactive possible core configuration and changes in core constituents are likely to results in a loss of reactivity, in the fast reactor changes in core geometry can, at least theoretically, result in large reactivity additions and consequent core vaporization.* Therefore, the LMFBR design generally includes two independent, diverse, and functionally redundant reactor shutdown systems to insure that off-normal conditions requiring scram are reliably terminated. Despite this additional safety measure it has become customary for LMFBRs to consider HCDAs as part of the formal licensing process, although HCDAs are excluded from the spectrum of design basis accidents [1, 7].

The important differences between LWRs and LMFBRs are summarized in Table 1, illustrating both favorable and nonfavorable characteristics.

Table 1. Differences between Reactor Types

LWR	LMFBR
High Coolant Pressure (Loss of Pressure needs ECCS)	Low Coolant Pressure (Loss of Pressure; surrounding Na can in principle remove decay heat and absorb halogens)
Removal of Coolant Shuts Down Reactor	Removal of Coolant Increases Reactivity (Can be designed out)
Compaction of Core Decreases Reactivity	Compaction of Core Increases Reactivity
Power Density $\sim$100 W/cc	Power Density $\sim$500 W/cc (Increases the potential for overheating, but also increases the potential for fuel dispersal, e.g., prevents fuel compaction.)

Only the fuel recompaction and the recriticality issue is unique to the LMFBR and other fast reactors, but the LMFBR has inherent advantages in terms of decay heat removal and absorption capability of radioactive fission products by the liquid metal coolant as compared to LWRs.

HCDA INITIATORS, ACCIDENT PATHS AND KEY SAFETY ISSUES

The underlying principles of reactor safety in the United States have employed the conventional defense-in-depth philosophy by designing for (1) reliability of normal operation, (2) protective features to limit the consequences of potential malfunctions, and (3) additional public-safety margins for protection against unforeseen and unexpected circumstances. This approach has recently been discussed in terms of four lines of assurance against the consequences of a reactor malfunction: [8] (1) prevention of accidents, (2) limitation of core damage, (3) containment of accidents in the primary system, and (4) attenuation of radiological products.

*While the fast reactor has a large Pu inventory, it is the potential for Pu dispersal and release because of core vaporization rather than the inventory of Pu itself that is important.

The first line of assurance is to prevent any incident that would lead to fuel melting. This can be achieved to a high degree by reliability and quality assurance for components, plant-protection and shutdown heat-removal systems, and incorporation of design features promoting safety. Through this approach the probability of a serious accident that might have a significant impact on the public is reduced to a very low level. Many believe that the first line of assurance can be made sufficiently secure that no further considerations of CDAs are necessary. However, at the present time, a more balanced approach appears to be necessary for fast reactors which provides, in addition to the accident-prevention features, consequence-limiting features that would protect the public against the effects of an accident beyond the capacity of the normal protective systems [1, 8].

Early studies of CDA emphasized energetic excursions due to fuel collapse. For example, in the Bethe and Tait evaluation [9], no attempt was made to establish physically possible initial conditions; the analysis was performed for an assumed initial condition of a completely molten core in its original geometry, with the core assumed to be compressed by slumping of the top surface under the acceleration of gravity.

Also, even if a core meltdown occurs without an energetic excursion, the possibility of an energetic fuel-coolant interaction (vapor explosion) must be considered, as first discussed by Hicks and Menzies in 1965 [10]. Ever since the Bethe-Tait and Hicks-Menzies studies, the assessment of HCDAs, including recriticality and fuel-coolant interaction events, has been a major consideration in LMFBR safety analysis and development. This is because it is generally not considered practical to accommodate upper "theoretical" bounds. Furthermore, since the levels of energetics resulting from hydrodynamic disassembly are rather sensitive to small variations in the core average temperature [11] (the work is essentially proportional to fuel vapor pressure which is an exponential function of temperature) and hence to initial conditions like the driving ramp rate, it follows that it is desirable to be able to rule out energetic hydrodynamic disassembly conditions altogether [1].

Classes of possible initiating conditions have been identified as (1) those resulting in a reactivity insertion at a rate so great that the reactor plant-protection system would be unable to respond in time; (2) malfunctions or faults within the design basis of the reactor plant-protection system (these can be broken down into fuel-failure propagation, whole-core loss of flow, and transient overpower); and (3) malfunctions or faults leading to interruption of heat-removal capability even with shutdown, such as postulated severe pipe breaks or loss of heat sink. From a probabilistic point of view, it would appear to be appropriate to consider only the second and third classes of initiators since the first class (including control-rod ejection, gas-bubble intake, failure of core-support structure or core-restraint system, etc.) can be discounted on the grounds that they can be effectively precluded by design.

Depending on the particular initiator, the accident structure shown in Fig. 1 allows for an early termination path (accident path 1 in Fig. 1) where accidents are terminated by early negative-reactivity effects and only limited core damage [12]. It also identifies a general accident path (accident path 2 in Fig. 1) that leads to major core disruption and mild fuel dispersal without having an early energetic disassembly as the only exit path (accident path 3 in Fig. 1) [13]. These relatively new developments in accident analysis can be largely attributed to the development of detailed initiating-accident codes (SAS, MELT, etc.) [14, 15] that treat the coupling between core neutronics and specified material motions (multichannel treatment), including coolant, cladding, and fuel up to the point of loss-of-subassembly geometry. Various safety issues can be related to each acci-

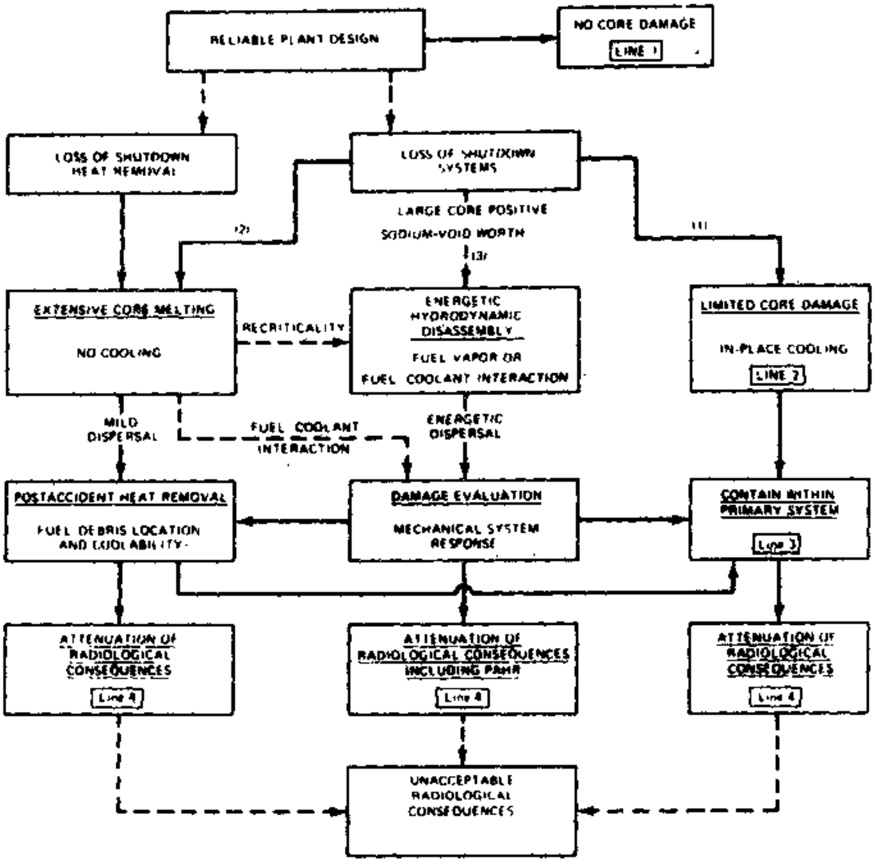

FIG. 1 SIMPLIFIED PATH STRUCTURES OF CDA INITIATORS, LINE OF ASSURANCE, KEY ACCIDENT ISSUES.

dent path as illustrated in Fig. 1. A recent evaluation of safety test facility needs led to consideration of 10 safety issues [16]. These are identified in Table 2 in which a relation to four lines of assurance is also indicated. Current status of these safety issues relative to the current mixed oxide plutonium fuel of the Fast Flux Test Facility and CRBRP as well as safety implications with alternate fuel cycles are summarized below.

ALTERNATE FUEL CYCLES

Since the goals of ERDA's LMFBR program have been significantly altered by the President's recent energy statement (April 20, 1977), and because of a determination to make the fuel cycle as proliferation and diversion resistant as possible, it would appear appropriate to include a brief discussion on alternate fuel cycles. (ERDA is currently redirecting the fast breeder program emphasizing studies of alternate fuel cycles over the next two years).

A variety of fast breeder cycles can be considered. All are variants of either the Pu-U^{238} cycle or the U^{233}-Th cycle or mixtures of the two. The mixtures can occur in

Table 2. Ten Key FBR Safety Issues and Their Relation to Lines of Assurance for Public Health and Safety

PREVENT ACCIDENTS – LOA 1

1. Fuel Behavior
2. Pin-to-pin Failure Propagation

LIMIT CORE DAMAGE – LOA 2

3. Subassembly-to-subassembly Fault Propagation
4. Extent of Core Damage from Whole-Core Accident Initiators

PRIMARY SYSTEM CONTAINMENT – LOA 3

5. Accident Energetics - Na Voiding
6. Accident Energetics - Recriticality
7. Accident Energetics - FCI
8. System Structural Response
9. Postaccident Heat Removal

ATTENUATE RADIOLOGICAL PRODUCTS – LOA 4

10. Radiological Consequences

various ways. The core can contain either or both the fissile isotopes (U^{233}, or Pu^{239}, or even U^{235} in a possible approach to equilibrium). The core can have either or both of the fertile isotopes (Th^{232} or U^{238}). The core can have either a uniform design (usually with two enrichment zones) or a heterogeneous design (with fissile and fertile regions interspersed). The blanket can consist of either U^{238} or Thorium. The core and blanket materials can be in the form of oxide, carbide, or metals. But the reduced breeding capability of all others relative to the Pu-U^{238} system may lead to an increased interest in the carbide and metal in order to offset the breeding loss.

Some of the more obvious alternate fuel cycles to receive serious consideration in fast breeder systems are listed in Table 3.

Table 3. Some Alternate Fuel Cycles

	CORE		BLANKET	
	FISSILE	*FERTILE*		
(a)	Pu	U^{238}	U	(the "pure" Pu-U^{238} cycle)
(b)	U^{233}	Th	Th	(the "pure" U^{233}-Th cycle)
(c)	Pu	Th	Th or U^{238}	
(d)	U^{233}	U^{238}	Th or U^{238}	

The following are some general considerations on safety characteristics.

(a) Delayed neutron fraction - the delayed neutron fraction of U^{233} is somewhat less than that of Pu^{239} - but one would not expect this to be of much significance.

(b) Doppler effect - generally not likely to be very much different for the various systems - not likely to be a major factor.

(c) Sodium void effect - the void coefficient is much smaller in U^{233}- fueled systems which may reduce or eliminate the incentive for heterogeneous cores. This is particularly true for the metal fuel where the U^{233}-Th systems hardly have a central positive sodium void region and it is clearly negative for the bulk of the core.

For other than the sodium void effect, we shall see that significant differences in safety characteristics to be much more dependent on whether the fuel is in the oxide, carbide, or metal form, than on whether the fuel is U^{233} or Pu^{239} or the fertile material is U^{238} or Thorium. A summary of these differences as well as the current status of the reference sodium cooled oxide system in the plutonium-uranium cycle relative to the key issues are summarized below.

CURRENT STATUS OF KEY SAFETY ISSUES, INCLUDING IMPACT BY ALTERNATE FUEL CYCLES

The experience accumulated in safety research indicates that it is principally the consideration of the core-disruptive accident that is of importance in safety assessment and licensing of commercial breeder reactors.

Thus key safety issues are found to be essentially the same for all fuel cycles and would generally include various aspects of fuel-failure propagation, accident energetics, postaccident heat removal and radiological-consequence assessment (see Table 2).

Each of these items is discussed below relative to expected differences resulting from various fuel types, that is, oxide, carbide or metallic fuel elements including both uranium- and thorium-based fuel cycles.

Fuel Behavior

The safety issue of fuel behavior covers the information needed to understand both transient and quasi-steady fuel performance up to and just beyond the limits of failure under off-normal steady state, off-normal transient and various transient conditions of intermediate severity. The current U.S. data base, established through TREAT and EBR-II irradiations, [17-19] can be substantially expanded to include more prototypic fuel types and a wider range of intermediate transient conditions. The basic behavior of the alternate fuels under steadystate and transient irradiation is generally not well-known, although ThO_2-UO_2 fuel is expected to behave quite similarly to UO_2-PuO_2 fuel. For carbide and metal fuels, key parameters which might relate to safety questions include differences in fission-gas release/retention; fuel swelling and fuel-cladding mechanical interaction; and compatibility with cladding material. The relatively high fission gas retention for both carbide and metallic fuels as well as the effects of sodium bond for the latter fuel type may significantly affect failure threshold and location. The extent to which these characteristics and differences in high-temperature thermophysical and mechanical properties may alter the initiating conditions for accidents and the subsequent scenarios needs to be reassessed.

Pin-to-pin Failure Propagation

The area of rapid pin-to-pin failure propagation is concerned with the occurrence of initially small local faults and the possibility of their rapid growth into more widespread serious faults.*

*For a review of blockages in LMFBR fuel assemblies see Ref. [20]

For the UO_2-PuO_2 fuel rapid pin-to-pin propagation is precluded and slow propagation (blockage propagation) is not expected, but it still remains to be finally verified [2]. The principal reasons for this conclusion are: 1) local faults (including fission-gas release, molten-fuel release, and localized boiling) have been shown to be isolated events (no rapid propagation and 2) slow blockage propagation appears nonmechanistic, particularly in wire-wrapped subassemblies. A similar claim can be made for ThO_2-UO_2 fuel. Evaluations of UC-PuC generally indicate the same favorable conclusions. For all high thermal conductivity fuel types (uranium- and thorium-based carbide and metal), direct release of molten fuel resulting from local faults appears unlikely, and if arbitrarily postulated, the potential for energetic interaction with the coolant appears as unlikely as that currently demonstrated for oxide fuel, e.g., rapid pin-to-pin propagation appears unlikely for all fuel types. The existence of eutectic alloys between the (Th, U) metallic fuel and stainless steel must be taken into account in the evaluation. Slow attack of cladding material may occur for $T > 725°C$ increasing the potential for slow failure propagation.

Subassembly-to-subassembly Fault Propagation

If proper detection is assumed, a full single subassembly meltdown would be a low probability event [2]. However, if a subassembly meltdown does occur and the reactor is not promptly scrammed, then subassembly failure propagation would be difficult to rule out with any high degree of assurance. If subassembly plugging should not occur, then the inherent dispersive nature of oxide fuels (uranium- as well as thorium-based fuels with steel as the working fuel) would prevent propagation by thermal melt-through [2]. The dispersive effect is illustrated in Figs. 2 and 3 for two different equilibrium conditions. [11]. If all the fuel is to remain within the active fuel zone [Fig. 2 (a)] the corresponding pressure to satisfy heat removal by upward vapor transport must be large [Fig. 2 (b)]. However, since complete plugging of the subassembly now appears unlikely, as discussed in the paper by Epstein [21] extended fuel dispersal may take place leaving behind a very small amount of liquid fuel in the active zone (Fig. 3). For a postulated single-subassembly disruption (Fig. 3 (a) with the void profile corresponding to nominal power level), this will decrease the radial heat flux by an order of magnitude. This heat flux can readily be removed by the normal flow conditions in the adjacent hexcans [2].

Also current experience suggests, that mechanical propagation resulting from energetic fuel-coolant interaction would be highly unlikely for the oxide-based fuel types (contact temperatures well below the spontaneous nucleation temperatures, as illustrated in Fig. 4). [22-24] For the high thermal conductivity fuels (both uranium- and thorium-based fuels), subassembly propagation would appear more likely both from thermal means (delay in fuel dispersal by vaporization (see Fig. 5) and increased tendency for freezing and plugging as compared with oxide fuels) and from mechanical means (contact temperatures can exceed the spontaneous nucleation temperature of sodium as illustrated in Fig. 6).* Effects of metallic fuel eutectic formation with steel may also increase the potential for propagation.

*Questions regarding the general validity of this *necessary* criterion (not sufficient) for the occurrence of large-scale vapor explosions have again been raised in the paper by Board and Caldarola [25]. However, it is difficult to understand why these authors continue to completely ignore addressing the two fundamental questions: 1) What is the cause for the extreme sensitivity for a go/no go situation relative to temperature? (1 or 2°C has been found in experiments with Freon 22-oil by Henry [24] as well as by Board [25]; this fact clearly mandates against a pure hydrodynamic model) and 2) What is the source that provides sufficient nucleation sites in a liquid-liquid system to cause explosive vapor growth? *The concept of spontaneous nucleation clearly provides answers to both questions* [22-24].